317
4/7/80

DR. 1022

UC-70

MASTER

ONWI-39

**AN ASSESSMENT OF LWR
SPENT FUEL
DISPOSAL OPTIONS**

**VOLUME 3
STUDY BASES AND
SYSTEM DESIGN CONSIDERATIONS
(APPENDICES)**

2/1

TECHNICAL REPORT

July 1979

ONWI
Office of Nuclear Waste Isolation
Battelle

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.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represent that its use would not infringe privately owned rights.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy ~~\$13.25~~; Microfiche ~~\$3.00~~

23-53

ONWI-39

**AN ASSESSMENT OF LWR
SPENT FUEL
DISPOSAL OPTIONS**

**VOLUME 3
STUDY BASES AND
SYSTEM DESIGN CONSIDERATIONS
(APPENDICES)**

TECHNICAL REPORT

July 1979

**Bechtel National, Inc.
Nuclear Fuel Operations
San Francisco, CA 94105**

DISCLAIMER

This report was prepared by Bechtel National Inc. under Subcontract E512-00100 with Battelle Project Management Division Office of Nuclear Waste Isolation under Contract No. EY-76-C-06-1830 with the U.S. Department of Energy. This subcontract was administered by the Battelle Office of Nuclear Waste Isolation.

FOREWORD

This report is the final product of a Spent Fuel Disposal Study (SFDS) conducted by Bechtel National, Inc., which was initiated under Subcontract 86Y-42514C with Union Carbide Corporation — Nuclear Division, Office of Waste Isolation (OWI), and was concluded under Subcontract E512-00100 with the Battelle Project Management Division, Office of Nuclear Waste Isolation (ONWI), a prime contractor of the U.S. Department of Energy (DOE). The change was made when DOE reassigned the responsibility for certain nuclear waste management activities from OWI to ONWI.

The SFDS is part of the DOE's National Waste Terminal Storage (NWTs) program. The principal objective of the NWTs program is to provide facilities in various deep geologic formations at multiple locations within the United States in which the safe disposal of radioactive material from commercial nuclear power plants can be accomplished.

The study (1) identifies several spent fuel waste forms as well as the high level waste (HLW) and other waste forms from fuel reprocessing, (2) defines facility and system design features to handle, transport, process, encapsulate, and dispose of these spent fuel and waste forms, and (3) provides a comparative assessment of each of the spent fuel/waste alternatives with respect to technical feasibility, safeguards, criticality, radiological impact, retrievability, and economics.

The report is in three volumes. Volume 1, containing the summary and conclusions, provides the reader with a general overview of the study, coupled with the specific conclusions developed from the comparative assessments of the study cases and study variations. Volume 2, in addition to introducing and defining the requirements for the study, contains the detailed evaluation of the study cases in each of six assessment areas, and describes and evaluates a number of study variations. Volume 3 (Appendices) provides a tabulation of the bases and assumptions used in the study as well as preconceptual design descriptions and cost estimates of the facilities and transportation systems necessary to implement the various study cases.

A peer review group was appointed by ONWI to periodically review and contribute expert opinions on the technical progress of the study. The peer review group consists of the following members:

W. M. Pardue, Chairman	Battelle — ONWI
S. C. Matthews	Battelle — ONWI
R. D. Widrig	Battelle — PNL
C. B. Bastin	Department of Energy, Headquarters
J. M. Morrison	Union Carbide Corporation — Nuclear Division
D. E. Ferguson	Oak Ridge National Laboratory
A. G. Croff	Oak Ridge National Laboratory
W. H. Pechin	Oak Ridge National Laboratory
M. J. Steindler	Argonne National Laboratory
H. J. Clark	Savannah River Laboratory
R. C. Lincoln	Sandia Laboratories
T. A. Sellers	Sandia Laboratories
G. H. Dyer	Bechtel National, Inc.

The contributions of the peer review group are gratefully acknowledged. While the opinions expressed during these reviews were carefully considered by Bechtel and ONWI in the development of this report, the conclusions reached do not necessarily reflect the positions of individual members of the group.

Volume 3

TABLE OF CONTENTS

Appendix		Page
	FOREWORD	iii
A	BASES AND ASSUMPTIONS	A-1
	Introduction	A-1
	A.1 Limitations and Exclusions	A-2
	A.2 AFRs	A-3
	A.3 Canisters/Packaging	A-4
	A.4	
	A.5 Criticality Design Basis	A-7
	A.6 Decommissioning	A-8
	A.7 Economics	A-9
	A.8 Fuel	A-11
	A.9	
	A.10	
	A.11 Processing/Encapsulation Facilities	A-15
	A.12 Process Selection and Assumptions	A-16
	A.13	
	A.14 Repository	A-18
	A.15 Safeguards	A-22
	A.16 Safety	A-22
	A.17 Thermal/Mechanical Properties	A-23
	A.18 Throughput	A-27
	A.19 Transportation	A-29
	A.20 Wastes, General	A-32
	References for Appendix A	A-33
B	AWAY-FROM-REACTOR STORAGE FACILITIES	B-1
	B.1 Approach	B-2
	B.1.1 Preconceptual Design Approach	B-2
	B.1.2 Cost Estimating Approach	B-2
	B.2 Preconceptual Design Description	B-5
	B.2.1 General Description	B-5
	B.2.2 Cask-Handling System	B-5
	B.2.3 Cask Flush and Cooldown	B-5
	B.2.4 Fuel Unloading-Loading Pools	B-6

Appendix		Page
	B.2.5 Cask Decontamination Area	B-6
	B.2.6 Fuel Storage Pools	B-6
	B.2.7 Pool Cooling System	B-6
	B.2.8 Radwaste Treatment	B-11
B.3	AFR Operational Considerations	B-12
	B.3.1 Carrier Receiving, Offloading, and Cask Cooldown	B-13
	B.3.2 Cask Unloading and Return Preparation	B-14
	B.3.3 Basket Moving	B-14
	B.3.4 Cask Loading and Shipping	B-14
	B.3.5 Pool Maintenance	B-15
	B.3.6 Waste Processing	B-15
	B.3.7 Accountability	B-15
	B.3.8 General Duties	B-15
	B.3.9 Overall Personnel Requirements	B-15
B.4	AFR Cost Estimates	B-17
	B.4.1 AFR Capital Cost Estimate	B-17
	B.4.2 AFR Operating Cost Estimate	B-17
	B.4.3 AFR Decommissioning Cost Estimate	B-17
	References for Appendix B	B-18
C	PROCESSING/ENCAPSULATION FACILITIES	C-1
C.1	Approach	C-2
	C.1.1 Preconceptual Design Approach	C-2
	C.1.2 Cost Estimating Approach	C-2
C.2	Reference Case	C-6
	C.2.1 Process Description	C-6
	C.2.2 Waste/Product Packages	C-22
	C.2.3 Facility Description	C-27
	C.2.4 Operational Considerations	C-33
	C.2.5 Cost Estimates	C-47
C.3	Case 1	C-50
	C.3.1 Process Description	C-50
	C.3.2 Spent Fuel/Waste Packaging	C-56
	C.3.3 Facility Description	C-60
	C.3.4 Operational Considerations	C-62
	C.3.5 Cost Estimates	C-74
C.4	Case 2	C-78

Appendix

		Page
	C.4.1 Process Description	C-78
	C.4.2 Spent Fuel/Waste Packaging	C-85
	C.4.3 Facility Description	C-87
	C.4.4 Operational Considerations	C-88
	C.4.5 Cost Estimates	C-99
C.5	Case 3	C-102
	C.5.1 Process Description	C-102
	C.5.2 Spent Fuel/Waste Packaging	C-107
	C.5.3 Facility Description	C-109
	C.5.4 Operational Considerations	C-110
	C.5.5 Cost Estimates	C-122
	References for Appendix C	C-126
D	GEOLOGIC REPOSITORIES	D-1
D.1	Approach	D-2
	D.1.1 Preconceptual Design Approach	D-2
	D.1.2 Cost Estimating Approach	D-3
D.2	Generic Repository Design Considerations	D-7
	D.2.1 Thermal Loading	D-7
	D.2.2 Site	D-7
	D.2.3 Surface Facilities	D-8
	D.2.4 Underground Facilities	D-19
	D.2.5 Ventilation	D-23
	D.2.6 Shafts	D-24
	D.2.7 Specialized Equipment	D-27
	D.2.8 Repository Validation	D-29
D.3	Reference Case	D-38
	D.3.1 Specific Repository Aspects	D-38
	D.3.2 Waste Emplacement	D-38
	D.3.3 Operational Considerations	D-44
	D.3.4 Repository Cost Estimates	D-54
D.4	Case 1	D-56
	D.4.1 Specific Repository Aspects	D-56
	D.4.2 Spent Fuel/Waste Emplacement	D-56
	D.4.3 Operational Considerations	D-57
	D.4.4 Repository Cost Estimates	D-65
D.5	Case 2	D-67
	D.5.1 Specific Repository Aspects	D-67
	D.5.2 Chopped Fuel/Waste Emplacement	D-67
	D.5.3 Operational Considerations	D-68

Appendix**Page**

	D.5.4	Repository Cost Estimates	D-75
D.6	Case 3		D-78
	D.6.1	Specific Repository Aspects	D-78
	D.6.2	Vitrified Fuel/Waste Emplacement	D-78
	D.6.3	Operational Considerations	D-81
	D.6.4	Repository Cost Estimates	D-86
		References for Appendix D	D-89
E	TRANSPORTATION SYSTEMS		E-1
E.1	Materials to be Transported		E-1
	E.1.1	Reactor Annual Discharge	E-2
	E.1.2	Away-From-Reactor Storage Facility (AFR) Annual Discharge	E-2
	E.1.3	Processing/Encapsulation (P/E) Facility Annual Discharge	E-2
E.2	Modes of Transportation		E-3
	E.2.1	Road Transportation	E-3
	E.2.2	Rail Transportation	E-5
	E.2.3	Water Transportation	E-9
	E.2.4	Intermodal Transportation	E-12
E.3	Transportation Regulations		E-16
	E.3.1	Federal Regulations	E-16
	E.3.2	State and Local Regulations	E-20
E.4	Transportation Logistics		E-31
	E.4.1	Distance Between Facilities	E-31
	E.4.2	Average Speed	E-32
	E.4.3	Shipments	E-33
E.5	Casks		E-35
	E.5.1	Available Casks	E-35
	E.5.2	Conceptual Casks	E-39
	E.5.3	Turnaround Times and Lease Rates	E-40
	E.5.4	Cask Fleet Requirements	E-44
E.6	Transportation Cost Estimates		E-45
	E.6.1	Cost Summary	E-45
	E.6.2	Transportation Cost Sensitivity	E-46
	E.6.3	Rail Cask Fleet as a Function of Distance	E-48
		References for Appendix E	E-53

F	TECHNICAL CONSIDERATIONS FOR SAFEGUARDS DESIGN	F-1
F.1	Safeguards Systems: General Considerations	F-1
F.1.1	Physical Protection System	F-2
F.1.2	Material Control and Accounting System	F-9
F.2	Safeguards Application to Fuel Cycle Facilities and Systems	F-13
F.2.1	Safeguards for a Transportation System	F-13
F.2.2	Safeguards for a P/E Facility	F-15
F.2.3	Safeguards for a Repository	F-19
F.2.4	Safeguards for an AFR	F-21
F.3	Sabotage and Theft	F-23
F.3.1	Adversaries	F-23
F.3.2	Threat	F-23
F.3.3	Fabrication of a Nuclear Explosive Device	F-24
F.3.4	Dispersal of Radioactivity	F-27
	References for Appendix F	F-29
G	TECHNICAL CONSIDERATIONS FOR LONG-TERM CRITICALITY	G-1
G.1	Characteristics of Fissile Material in the Waste	G-2
G.2	Critical Mass Data	G-9
G.3	Thermal Effects in a Repository	G-10
G.4	Geological Mechanisms and Factors Affecting Criticality Conditions in the Repository	G-11
G.4.1	Intrusion of Liquids	G-11
G.4.2	Canister Failure	G-13
G.4.3	Fuel Assembly Failure	G-13
G.4.4	Destruction of Waste Matrix	G-13
G.4.5	Fissile Material Separation and Transport	G-13
G.4.6	Fissile Material Reaccumulation	G-15
G.4.7	Time Factors	G-16
	References for Appendix G	G-17
H	TECHNICAL CONSIDERATIONS FOR THE ASSESSMENT OF RADIOLOGICAL IMPACT	H-1
H.1	Technical Issues	H-2
H.1.1	Operating Phase Exposure	H-2
H.1.2	Long-Term Risks	H-3
H.1.3	Regulatory Controls	H-4

Appendix		Page
	H.1.4 Repository Integrity and Waste Retrievability	H-6
	H.1.5 Radiological Impact Assessment Methodology	H-6
	H.1.6 Population Considerations	H-7
	H.1.7 Public Perception of Risks	H-7
H.2	Evaluation Criteria	H-8
	H.2.1 Near-Term Public Health and Safety and Environmental Impact	H-8
	H.2.2 Radiation Protection to Operators	H-8
	H.2.3 Long-Term Public Health and Safety and Environmental Impact	H-11
	H.2.4 Accident Potential	H-14
	H.2.5 Criticality Potential	H-16
	H.2.6 Retrievability Potential	H-16
H.3	Considerations Affecting Radiological Impact Assessments	H-17
	H.3.1 Waste Types and Inventories	H-17
	H.3.2 Radionuclides in Waste	H-17
	H.3.3 Radionuclide Release Mechanisms	H-18
	H.3.4 Biological Response	H-19
	H.3.5 Measurement of Radiological Impact	H-19
H.4	Radiological Impact Modeling	H-21
	H.4.1 Generalized Approach to Assessment of Radiological Impact	H-21
	H.4.2 Specific Radiological Impact Models	H-21
	References for Appendix H	H-27
I	TECHNICAL CONSIDERATIONS FOR CANISTER RETRIEVABILITY	I-1
I.1	Environmental Impact on Canisters	I-2
	I.1.1 Effect of Corrosion on Canisters	I-2
	I.1.2 Effect of Mechanical and Thermal Stresses on Canisters	I-4
I.2	Cost of Canister Materials	I-5
I.3	Facility Design Considerations	I-6
	I.3.1 Facilities for Retrievability	I-6
	I.3.2 Equipment for Retrievability	I-6
	I.3.3 Instrumentation	I-8

Appendix

Page

I.3.4	Ventilation	I-8
I.3.5	Radiological Protection	I-8
I.3.6	Sleeve-Plug Design	I-8
I.4	Retrievability Operational Considerations	I-10
I.4.1	Subsurface Operations	I-10
I.4.2	Surface Operations	I-11
	References for Appendix I	I-12

J ABBREVIATIONS AND DEFINITIONS

J-1

J.1 Abbreviations

J-1

J.2 Definitions

J-4

Volume 3

TABLES

Table	Page
A-1 Physical Characteristics of LWR Fuel Assemblies Charged to Reactor	A-14
B-1 Annual Material Flow at Each AFR	B-13
B-2 Total Direct Operating Staff for an AFR	B-16
B-3 Total Support and Overhead Personnel for an AFR	B-16
B-4 Total Overall Personnel Requirements for an AFR	B-16
B-5 Operating Cost Estimate for an AFR	B-17
C-1 Waste/Product Package Data, Reference Case	C-25
C-2 Annual Material Flow at Each P/E Facility, Reference Case	C-35
C-3 Total Operating Staff for Fuel Receiving and Storage	C-38
C-4 Total Operating Staff for Mechanical Processing	C-38
C-5 Total Operating Staff for Chemical Processing	C-39
C-6 Total Operating Staff for Conversion of Uranyl Nitrate to Uranium Hexafluoride	C-40
C-7 Total Operating Staff for the Conversion of Plutonium Nitrate to Plutonium Oxide	C-41
C-8 Total Operating Staff for the Solidification of HLW	C-42
C-9 Total Operating Staff for Volume Reduction of Fuel Cladding	C-43
C-10 Total Operating Staff for Volume Reduction of Combustible Solid Wastes and Fixation of Solid Wastes	C-45
C-11 Total Operating Staff for Offgas Treatment, Volatile Fission Products Recovery, and Packaging	C-46
C-12 Total Operating Staff Requirements for a Reference Case P/E Facility	C-46
C-13 Total Support Staff for Reference Case P/E Facility	C-47
C-14 Overall Total Staff Requirements for a Reference Case P/E Facility	C-47
C-15 Capital Cost Estimate for a Reference Case P/E Facility	C-48
C-16 Operating Cost Estimate for a Reference Case P/E Facility	C-49
C-17 Waste/Product Package Data, Case 1	C-59
C-18 Annual Material Flow at Each P/E Facility, Case 1	C-68
C-19 Total Operating Staff for Fuel Receiving	C-69
C-20 Total Operating Staff for Preparing Assemblies for Encapsulation	C-69
C-21 Total Operating Staff for Canister Loading and Capping	C-70
C-22 Total Operating Staff for Canister Inspection and Testing	C-71
C-23 Total Operating Staff for Canister Survey and Cask Loading	C-72

Table	Page
C-24 Total Operating Staff for Concentrating and Stabilizing Liquid Wastes	C-72
C-25 Total Operating Staff for Incinerating and Fixing All Solid Wastes	C-73
C-26 Total Operating Staff Requirements for a Case 1 P/E Facility	C-73
C-27 Total Support Staff for a Case 1 P/E Facility	C-74
C-28 Overall Total Staff Requirements for a Case 1 P/E Facility	C-74
C-29 Capital Cost Estimate for a Case 1 P/E Facility	C-75
C-30 Operating Cost Estimate for a Case 1 P/E Facility	C-77
C-31 Waste/Product Package Data, Case 2	C-86
C-32 Annual Material Flow at Each P/E Facility, Case 2	C-93
C-33 Total Operating Staff for Fuel Receiving	C-94
C-34 Total Operating Staff for Preparing Assembly for Chopping	C-94
C-35 Total Operating Staff for Mechanical Processing	C-95
C-36 Total Operating Staff for Voloxidation and Encapsulation	C-95
C-37 Total Operating Staff for Heat Treating and Canister Testing	C-96
C-38 Total Operating Staff for Canister Survey and Cask Loading	C-97
C-39 Total Operating Staff for Stabilizing Liquid Wastes	C-97
C-40 Total Operating Staff for Incinerating Combustible Solid Wastes and Fixing All Solid Wastes	C-98
C-41 Total Operating Staff Requirements for a Case 2 P/E Facility	C-98
C-42 Total Support Staff for a Case 2 P/E Facility	C-99
C-43 Overall Total Staff Requirements for a Case 2 P/E Facility	C-99
C-44 Capital Cost Estimate for a Case 2 P/E Facility	C-100
C-45 Operating Cost Estimate for a Case 2 P/E Facility	C-101
C-46 Waste/Product Package Data, Case 3	C-108
C-47 Annual Material Flow at Each P/E Facility, Case 3	C-115
C-48 Total Operating Staff for Fuel Receiving	C-115
C-49 Total Operating Staff for Mechanical Processing	C-116
C-50 Total Operating Staff for Chemical Processing	C-117
C-51 Total Operating Staff for Solidification of HLW	C-118
C-52 Total Operating Staff for Volume Reduction of Fuel Cladding	C-119
C-53 Total Operating Staff for Volume Reduction of Combustible Solid Wastes and Fixation of Noncombustible Solid Wastes	C-120
C-54 Total Operating Staff for Offgas Treatment, Recovery, and Packaging	C-121
C-55 Total Operating Staff for a Case 3 P/E Facility	C-121
C-56 Total Support Staff for a Case 3 P/E Facility	C-122
C-57 Overall Total Staff Requirements for a Case 3 P/E Facility	C-122
C-58 Capital Cost Estimate for a Case 3 P/E Facility	C-123
C-59 Operating Cost Estimate, Case 3 P/E Facility	C-125

Table	Page
D-1 SRENCO-SFDS Estimate Reconciliation for Total Generic Field Costs	D-4
D-2 SFDS Total Field Costs	D-4
D-3 Initial Room Mining Quantities	D-5
D-4 Annual Utilization of Repository Rooms	D-43
D-5 Operating Crew Requirements for Handling One Shipment of HLW or ILW Above Ground	D-44
D-6 Total Operating Staff for Handling All Shipments of HLW and ILW Above Ground	D-45
D-7 Operating Crew Requirements for Handling One Shipment of LLW Above Ground	D-46
D-8 Total Operating Staff for Handling All Shipments of LLW Above Ground	D-46
D-9 Operating Crew Requirements for Handling One Shipment of HLW or ILW Below Ground	D-47
D-10 Total Operating Staff for Handling All Shipments of HLW and ILW Below Ground	D-48
D-11 Total Operating Staff for Handling One Shipment of LLW Below Ground	D-48
D-12 Operating Equipment and Crew Requirements for Developing the Mine	D-50
D-13 Total Operating Staff for Mining Storage Rooms and Corridors	D-51
D-14 Total Operating Staff for Backfilling the Mine	D-51
D-15 Total Operating Staff for Managing the Surface Salt Stockpile	D-52
D-16 Total Staffing Requirements, Reference Case Repository	D-53
D-17 Capital Cost Estimate, Reference Case Repository	D-54
D-18 Sleeve and Plug Cost Estimate, Reference Case Repository	D-55
D-19 Annual Utilization of Repository Rooms	D-56
D-20 Total Operating Staff for Handling All Shipments of Spent Fuel and ILW Above Ground	D-58
D-21 Total Operating Staff for Handling All Shipments of LLW Above Ground	D-58
D-22 Total Operating Staff for Handling All Shipments of Spent Fuel and ILW Below Ground	D-61
D-23 Operating Equipment and Crew Requirements for Developing the Mine	D-62
D-24 Total Operating Staff for Mining Storage Rooms and Corridors	D-63
D-25 Total Operating Staff for Backfilling the Mine	D-63
D-26 Total Operating Staff for Managing the Surface Salt Stockpile	D-63
D-27 Total Staffing Requirements, Case 1 Repository	D-64
D-28 Capital Cost Estimate, Case 1 Repository	D-65
D-29 Sleeve and Plug Cost Estimate, Case 1 Repository	D-66

Table	Page
D-30 Annual Utilization of Repository Rooms	D-67
D-31 Total Operating Staff for Handling All Shipments of Chopped Fuel and ILW Above Ground	D-71
D-32 Total Operating Staff for Handling All Shipments of LLW Above Ground	D-71
D-33 Total Operating Staff for Handling All Shipments of Chopped Fuel and ILW Below Ground	D-72
D-34 Operating Equipment and Crew Requirements for Developing the Mine	D-73
D-35 Total Operating Staff for Mining Storage Rooms and Corridors	D-74
D-36 Total Operating Staff for Backfilling the Mine	D-74
D-37 Total Operating Staff for Managing the Surface Salt Stockpile	D-74
D-38 Total Staffing Requirements, Case 2 Repository	D-75
D-39 Capital Cost Estimate, Case 2 Repository	D-76
D-40 Sleeve and Plug Cost Estimate, Case 2 Repository	D-77
D-41 Annual Utilization of Repository Rooms	D-78
D-42 Total Operating Staff for Handling All Shipments of Vitrified Fuel and ILW Above Ground	D-82
D-43 Total Operating Staff for Handling All Shipments of LLW Above Ground	D-82
D-44 Total Operating Staff for Handling All Shipments of Vitrified Fuel and ILW Below Ground	D-83
D-45 Operating Equipment and Crew Requirements for Developing the Mine	D-84
D-46 Total Operating Staff for Mining Storage Rooms and Corridors	D-85
D-47 Total Operating Staff for Backfilling the Mine	D-85
D-48 Total Operating Staff for Managing the Surface Salt Stockpile	D-85
D-49 Total Staffing Requirements, Case 3 Repository	D-86
D-50 Capital Cost Estimate, Case 3 Repository	D-87
D-51 Sleeve and Plug Cost Estimate, Case 3 Repository	D-88
E-1 Materials Transported Annually from the Reactors, the AFRs, and the P/E Facilities — Reference Case, Case 1, Case 2, and Case 3	E-1
E-2 Weight Estimates for a Case/Car Packaging System	E-9
E-3 Areas Covered in ANSI N.552 Standard on Water Transportation	E-10
E-4 Maximum Activities for Normal Type A and B Quantities of Radioactive Material in a Single Container by Transport Group	E17
E-5 State and Municipal Legislation Affecting the Transportation of Radioactive Materials	E-21
E-6 Single and Tandem Axle Limits and Maximum Gross Vehicle Weight, by States	E-26

Table	Page
E-7 Average Distances Between Various Facilities	E-31
E-8 Time and Distance, One Driver vs. Two	E-33
E-9 Summary of Annual Material Shipments for SFDS Study Cases	E-34
E-10 Characteristics of LWR Spent Fuel Casks	E-36
E-11 Type B Packaging for Possible Use in the Transportation of Non-HLW	E-37
E-12 Data for Uranium Hexafluoride Packages	E-38
E-13 Conceptual Casks	E-39
E-14 Plutonium Package Description	E-40
E-15 Estimated Cask Turnaround Times	E-43
E-16 Cask Fleet Requirements	E-44
E-17 Transportation Cost Summary	E-46
 G-1 Waste Canister Data for the SFDS Study Cases	 G-2
G-2 Uranium and Transuranium Content of LWR Spent Fuel Assembly Following Removal From the Reactor	G-3
G-3 Computed Or Measured Minimum Critical Mass for Fissionable Nuclides	G-8
G-4 Leach Rates From HLW	G-14
G-5 Sorting of Individual Mechanisms Contributing to Criticality Potential Over Geologic Time	G-16
 H-1 Personnel Radiation Exposure Limits	 H-9
H-2 Radiation Exposure Limits for Various Repository Operating Zones	H-10
H-3 Total Source Strength — Reference Case	H-17
H-4 Nuclide Dose Conversion Factors	H-22
H-5 Input Data and Parameters for Ground Water Transport Model	H-24
H-6 Estimated Distribution Coefficients for Selected Radionuclides	H-25
 I-1 Stress Corrosion for 304L Stainless Steel Heater Cans in Brine Vapors	 I-3
I-2 Relative Cost of Materials	I-5

Volume 3

ILLUSTRATIONS

Figure	Page
A-1 Dose Rate from PWR Spent Fuel Assembly	
B-1 AFR Overall Block Diagram — Cases 1,2, and 3	B-7
B-2 AFR Building Arrangement — Cases 1, 2, and 3	B-9
 C-1 P/E Facility — Overall Block Diagram, Reference Case	 C-7
C-2 P/E Facility — HLW Fixation and Vessel Offgas Flow Diagram, Reference Case	C-11
C-3 P/E Facility — HLW Encapsulation Flow Diagram, Reference Case	C-15
C-4 P/E Facility — Volatile Fission Products Recovery Flow Diagram, Reference Case	C-17
C-5 P/E Facility — Cladding Hulls Encapsulation Flow Diagram, Reference Case	C-19
C-6 P/E Facility — Miscellaneous Radwaste Fixation Flow Diagram, Reference Case	C-23
C-7 P/E Facility — Plot Plan, Reference Case	C-29
C-8 P/E Facility — Building Arrangement Plan, Reference Case	C-31
C-9 P/E Facility — Overall Block Diagram, Case 1	C-51
C-10 P/E Facility — Spent Fuel Encapsulation Flow Diagram, Case 1	C-53
C-11 P/E Facility — Radwaste Treatment Process Flow Diagram, Case 1	C-57
C-12 P/E Facility — Plot Plan, Case 1	C-63
C-13 P/E Facility — Building Arrangement, Case 1	C-65
C-14 P/E Facility — Overall Block Diagram, Case 2	C-79
C-15 P/E Facility — Volatile Fission Products Recovery Flow Diagram, Case 2	C-81
C-16 P/E Facility — Chopped Fuel and End Fitting Encapsulation Flow Diagram, Case 2	C-83
C-17 P/E Facility — Plot Plan, Case 2	C-89
C-18 P/E Facility — Building Arrangement, Case 2	C-91
C-19 P/E Facility — Overall Block Diagram, Case 3	C-103
C-20 P/E Facility — Vitrified Fuel Encapsulation Flow Diagram, Case 3	C-105
C-21 P/E Facility — Plot Plan, Case 3	C-111
C-22 P/E Facility — Building Arrangement, Case 3	C-113

Figure	Page
D-1 Generic Repository — Plot Plan, All Cases	D-9
D-2 Generic Repository — Receiving Building (Waste/Spent Fuel— Arrangement Schematic, All Cases	D-13
D-3 Generic Repository — Canister Handling Flow Diagram, All Cases	D-15
D-4 Generic Repository — LLW Handling Flow Diagram, All Cases	D-17
D-5 Generic Repository — Underground Development Schematic, All Cases	D-21
D-6 Generic Repository — Shafts Configuration, All Cases	D-25
D-7 Generic Repository — Sequence of Operations for the Transfer Cask Transporter, All Cases	D-31
D-8 Generic Repository — Canister Storage Room Gantry Crane, All Cases	D-33
D-9 Generic Repository — Sequence of Operations for Cask and Canister in the Storage Room, All Cases	D-35
D-10 Geologic Repository — Waste Emplacement Schematic for the Retrievability Period for HLW/ILW Rooms, Reference Case	D-39
D-11 Geologic Repository — Waste Emplacement Schematic for the Retrievability Period for ILW Rooms Only, Reference Case	D-41
D-12 Geologic Repository — Spent Fuel/Waste Emplacement Schematic for the Retrievability Period, Case 1	D-59
D-13 Geologic Repository — Chopped Fuel/Waste Emplacement Schematic for the Retrievability Period, Case 2	D-69
D-14 Geologic Repository — Vitrified Fuel/Waste Emplacement Schematic for the Retrievability Period, Case 3	D-79
E-1 Road Transport Configuration	E-4
E-2 Design Features of a Rail Cask for Spent Fuel Shipment	E-6
E-3 AAR Rail Car Size Units	E-7
E-4 AAR Width Limits for Rail Cars	E-8
E-5 Roll-On/Roll-Off Movement of Spent Fuel Package	E-13
E-6 Direct Load Lift to Barge	E-14
E-7 80,000 lb GVW Limitations	E-28
E-8 73,280 lb GVW Limitations	E-29
E-9 Overpack and Canisters for the Plutonium Oxide Product	E-41
E-10 Handling of the Plutonium Oxide Overpack	E-42
E-11 Rail Freight Charges for Spent Fuel	E-47
E-12 Rail Cost Versus Distance for Spent Fuel (Case 1)	E-49
E-13 Rail Transport Cost Versus Average Rail Speed (Case 1)	E-50
E-14 Rail Transport Cost Versus Cask Lease Rate (Case 1)	E-51
E-15 Rail Cask Fleet Versus Distance (Case 1)	E-52

Figure	Page
F-1 Safeguards Systems Overview	F-3
G-1 Summary of Isotopic Changes in LWR Fuel Assembly During Burnup	G-5
G-2 Neutron Multiplication vs Burnup on Rods and Assemblies	G-6
G-3 Radioactive Production and Decay Chain of the Actinide Elements	G-7
I-1 Corrosion in Canister Materials	I-3
I-2 Retrieval Block Flow Diagram	I-7

•
•
•
•

•
•

Appendix A
BASES AND ASSUMPTIONS

TABLE OF CONTENTS

	Page
Introduction	A-1
A.1 Limitations and Exclusions	A-2
A.2 AFRs	A-3
A.3 Canisters/Packaging	A-4
A.4	
A.5 Criticality Design Basis	A-7
A.6 Decommissioning	A-8
A.7 Economics	A-9
A.8 Fuel	A-11
A.9	
A.10	
A.11 Processing/Encapsulation Facilities	A-15
A.12 Process Selection and Assumptions	A-16
A.13	
A.14 Repository	A-18
A.15 Safeguards	A-22
A.16 Safety	A-22
A.17 Thermal/Mechanical Properties	A-23
A.18 Throughput	A-27
A.19 Transportation	A-29
A.20 Wastes, General	A-32
References for Appendix A	A-33

•

.

•

.

•

•

Appendix A
BASES AND ASSUMPTIONS

INTRODUCTION

This appendix presents the bases and assumptions that govern the Spent Fuel Disposal Study. The values listed herein are not always maximum, minimum, or optimum values; rather, they represent reasonable assumptions from available data. The appendix is organized alphabetically by major categories, such as AFRs, repositories, etc. The information given in each category is presented in the order received, and there is only minimum reorganization within each category.

This appendix contains both front-end study bases and assumptions, which were "givens" or were established early in the study, and data derived during the course of the study, which are significant in making the technical assessments.

A.1

LIMITATIONS AND EXCLUSIONS

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
1	Quantity of spent fuel available to be processed and stored	5,000 MTHM/yr (steady-state flow) Ignores front-end backlog prior to startup and industry growth factors	X	X	X	X	Client direction
2	Safeguards	Domestic threats only	X	X	X	X	Project assumption
3	Industrial sabotage	Considering effects on public only	X	X	X	X	Project assumption
4	Facilities outside scope of study	Reactors, shallow land burial, and recycle facilities (fuel fabrication and enrichment)	X	X	X	X	Client direction
5	Waste from AFR	Assumes only non-TRU LLW, to be disposed of at shallow land burial		X	X	X	Client direction
6	Transport vehicles	Legal weight truck (73,250 lb) Unrestricted weight rail car (263,000 lb)	X X	X X	X X	X X	Project assumption
7	Cutoff date for data accumulation	February 1979	X	X	X	X	Project assumption
8	Licensability of facilities	Assumes current designs will be licensable	X	X	X	X	Project assumption
9	Ultimate useful cycles for maximum utilization of MOX fuel	Ignores ultimate liability to dispose of MOX fuel assemblies after about 3 to 5 cycles	X				Project assumption
10	Reactor type to utilize Pu recycle	Confined to LWR only	X				Project assumption
11	Optimization of facilities	No optimization of facilities or processes was included. Choice reflects best engineering judgment	X	X	X	X	Project assumption
12	Disposal of flow channels from BWR fuel assemblies	To shallow land burial directly from reactor pool	X	X	X	X	Project assumption
13	Waste from P/E facility storage pool	Small amounts of TRU-ILW from resin incineration	X	X	X	X	Project assumption

A.2 AFRs

Item	Description	Study Bases	Ref Case	Applicability			Remarks and References
				Case 1	Case 2	Case 3	
1	Ownership	U S Government		X	X	X	Project assumption
2	Throughput of individual facility	555 MTHM/yr (1.5 MTHM/day)		X	X	X	Project assumption
3	Storage capacity (each of 9 AFRs)	5 000 MTHM (fresh fuel weight)		X	X	X	Project assumption
4	Age of fuel						
	Incoming	1 year		X	X	X	Project direction
	Outbound	10 years		X	X	X	
	Variations	Sensitivity study on 5- and 10-yr-old incoming fuel		X			
5	Limit of radioactivity in pool	$< 2 \times 10^{-4} \text{ Ci/m}^3$		X	X	X	DPSTD-ISFS-78-7 ⁽¹⁾ (4/78)
6	Design conductivity in pool	$< 1 \text{ to } 2 \text{ } \mu\text{mho/cm}$		X	X	X	DPSTD-ISFS-78-7
7	Area provided for cask (each facility)						
	Receiving/shipping	2 rail casks		X	X	X	Project assumption
	Washdown/cooldown	2 rail casks		X	X	X	(rail casks shown to
	Decontamination	2 rail casks		X	X	X	give maximum area
	Unload/load pool	2 rail casks					commitment road
							casks accommodation
							also feasible)
8	Fuel storage baskets	4 PWRs or 9 BWRs/basket		X	X	X	DPSTD-ISFS-78-7
9	Water shielding	$\geq 13 \text{ ft}$		X	X	X	DPSTD-ISFS-78-7
10	Useful life	40 yr		X	X	X	Project assumption
11	Depth of pool	Enough to pass fuel in water over top of stored fuel (48 ft)		X	X	X	Project assumption
12	Fuel basket materials	Stainless steel Partial neutron absorption		X	X	X	DPSTD-ISFS-78-7
13	Location of AFRs	Each AFR separately and regionally located		X	X	X	Project assumption
14	Operating days	365 days/yr		X	X	X	Project assumption
15	Fuel unloading — cask to AFR	Wet		X	X	X	Project assumption
16	Fuel loading — AFR to cask	Wet		X	X	X	Project assumption

A.3 CANISTERS/PACKAGING

Item	Description	Study Bases	Ref. Case	Applicability			Remarks and References
				Case 1	Case 2	Case 3	
1	Encapsulated fuel assemblies						
	1 PWR/canister	14-in OD x 3/8-in wall x 15 ft, 6 in		X			Project assumption Applies to GE 8 x 8 assemblies and Westinghouse 17 x 17 assemblies
	3 BWRs/canister	16-in OD x 3/8-in wall x 15 ft, 6 in		X			
	Encapsul fuel-pin bundles						
	2 PWRs/canister	12 7/8 in OD x 3/8-in wall x 15 ft, 0 in		X			Project assumption Shorter cans possible with disassembly option
	6 BWRs/canister	12 7/8 in OD x 3/8-in wall x 15 ft, 0 in		X			
2	Compacted hulls	20-in OD x 3/8-in wall x 10 ft, 0 in	X			X	Project assumption (to accommodate 22-in overpack and stay within proposed acceptance criteria)
3	HLW, vitrified	12 7/8-in OD x 3/8-in wall x 10 ft, 0 in	X				See Subsection C 2
4	Fuel, chopped	12 7/8-in OD x 3/8-in wall x 10 ft, 0 in			X		Project assumption
5	Fuel vitrified	20-in OD x 3/8-in wall x 10 ft 0 in				X	Project assumption
6	TRU and non-TRU wastes						
	LLW	55-gal drum	X	X	X	X	Project assumption (80% fill nominal) Not to exceed weight limitations
	ILW	20-in OD x 3/8-in wall x 10 ft, 0 in	X	X	X	X	
7	End fittings	20-in OD x 3/8-in wall x 10 ft, 0 in	X	X	X	X	Project assumption
8	PuO ₂	AGNS-Sandia annular canister 6-in OD x 2-in ID x 12-in (8 kg) 4 canisters per PPP-IM Package (32 kg)	X				Subsection E 5 2
9	UF ₆	UF ₆ cylinder, Model 48X 49 25-in OD x 12 1/2 in	X				Subsection E 5 2
10	Overpacks, if required						
	HLW, vitrified	14-in OD x 3/8-in wall x 11-ft, 0-in	X				Project assumption 1 to 2% of total canister throughput to be overpacked
	Fuel, encapsulated PWR	16-in OD x 3/8-in wall x 16 ft, 6 in (not at P/E)		X			
	BWR	18-in OD x 3/8-in wall x 16 ft, 6-in (not at P/E)		X			
	Fuel-pin bundles PWR	14-in OD x 3/8-in wall x 16 ft, 0-in (not at P/E)		X			Additional corrosion allowance may have to be provided when more is known about corrosive environment
	BWR	14-in OD x 3/8-in wall x 16 ft 0-in (not at P/E)		X			

A.3 CANISTERS/PACKAGING (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
	Chopped fuel	14-in OD x 3/8-in wall x 11 ft, 0 in			X		
	Vitrified fuel	22-in OD x 3/8-in wall x 11 ft, 0 in				X	
	Hulls	22-in OD x 3/8-in wall x 11 ft, 0 in	X		X	X	
	End fittings	22-in OD x 3/8-in wall x 11 ft, 0 in	X	X	X	X	
	LLW (TRU and non-TRU)	83-gal drums (26-in dia x 35-1/2 in)	X	X	X	X	
	ILW (TRU and non-TRU)	22-in OD x 3/8-in wall x 11 ft, 0 in	X	X	X	X	
	UF ₆ , PuO ₂	None	X				
	Kr, Xe, CO ₂	None	X		X	X	
11	Materials	<u>Primary</u> <u>Overpack</u>					Project assumptions
	Fuel, encapsulated	CS (a) CS (not at P/E)		X	X	X	
	HLW	CS CS	X				
	LLW and ILW (TRU and non-TRU)	CS CS	X	X	X	X	
	Hulls	CS CS	X			X	For hulls, neutralize with caustic prior to loading
	PuO ₂	304L CS	X				
	UF ₆	CS —	X				
	End fittings	CS CS	X	X	X	X	
	Fuel, chopped	CS CS			X		
	Fuel, vitrified	CS CS				X	
12	Primary filling media						Project assumptions
	HLW, vitrified	Ambient air	X				Cans will have a leak-test capsule which will provide He in primary gas space as required
	Hulls and end fittings	Ambient air	X	X	X	X	
	TRU and non-TRU wastes	Ambient air	X	X	X	X	
	Fuel, encapsulated	He		X			
	PuO ₂	Ambient air	X				
	Fuel, vitrified	Ambient air				X	
	Fuel, chopped	He			X		
	UF ₆ , Kr-Xe-CO ₂	None	X		X	X	
13	Canister design criteria	Design and fabrication per ASME code (Section III, Division 1, Class 3)	X	X	X	X	RHO-CD-542 ⁽²⁾ If the canister is overpacked, the overpack must also meet the above requirements
14	Capacity						Project calculations (80% fill nominal not to exceed weight limitations) Assumes 2.48 overall factor for compaction of hulls Assumes careful orientation in end fittings packing
		MTHM Equiv per Canister					
		2.1					
	HLW, vitrified						
	Spent fuel						
	3 BWRs/canister	0.55			X		
	1 PWR/canister	0.46			X		
	Fuel, chopped	0.62				X	
	Fuel, vitrified	0.27					X
	Fuel-pin bundles						
	6 BWRs/canister	1.1			X		
	2 PWRs/canister	0.93			X		
	Hulls, compacted	2.2					X

(a) Carbon Steel

A.3

CANISTERS/PACKAGING (Cont'd)

Item	Description	Study Bases	Ref Case	Applicability			Remarks and References
				Case 1	Case 2	Case 3	
	End fittings (re- maining beyond in- fact fuel pins)	6 4 N A N A		X			
	End fittings	3 3 N A N A	X		X	X	
	PuO ₂ — primary (8 kg)	0 5	X				
	UF ₆	6 8 7 2 6 6	X				
15	Weight of empty/filled	<u>Empty/80% filled, lb</u>					TM-36/22 ⁽³⁾
	HLW, vitrified	600/1,720	X				
	PWR fuel	900/2,400		X			For 17 x 17 assemblies
	BWR fuel	1,000/2,800		X			For three 8 x 8 assemblies
	Hulls	1 000/4,200	X			X	
	LLW-TRU and non-TRU (55-gal conc)	60/840	X	X	X	X	
	End fittings	1,000/1,300	X	X	X	X	
	Fuel, chopped	600/2,550			X		
	Fuel, vitrified	1,000/3,880				X	
	UF ₆	4,500/25,500	X				
	ILW-TRU and non-TRU (20 in O D x 120 in)	1,000/3,300	X	X	X	X	
	Kr-Xe-CO ₂	N A	X		X	X	
16	Kr-Xe-CO ₂ (gas)	49-liter Linde Standard T-Size compressed gas cylinder (9 in O D x 55 in) 6 gas cylinders packed in a 22-in O D x 3/8 in wall x 10 ft 0 in	X		X	X	HEF assumption ^(a) DOT Spec 3A-2400 2,200 psig, 120°C CO ₂ is based on quantity in air (chopper and voloxidizer)
17	Tritiated water and barium iodate	55-gal drum (cement-fixed)	X		X	X	Project assumption

(a) Hot Experimental Facility of the Advanced Fuel Recycle Program⁽²²⁾

A.5

CRITICALITY DESIGN BASIS

Item	Description	Study Bases	Ref Case	Applicability			Remarks and References
				Case 1	Case 2	Case 3	
1	Storage pool (AFR)	Fissile material UO_2 Enrichment = 3.5% U-235 (no burnup) Physical form fuel assemblies		X	X	X	NRC licensing requirement
2	Surge pool (P/E facility)	UO_2 fuel assemblies (no burnup) Enrichment = 3.5% U-235	X	X	X	X	NRC licensing requirement
3	Process systems (P/E facility)	Uranium and plutonium compounds — solid and liquid Uranium = 3.5% U-235 96.5% U-238 Plutonium = 95% Pu-239 5% Pu-240 UO_2 fuel assemblies Enrichment = 3.5% U-235	X		X	X	NRC licensing requirement
4	Spent fuel (repository)	Canistered UO_2 fuel assemblies Enrichment = variable		X			Section 6
5	HLW (repository)	Vitrified actinides Np, Am, Cm, Pu, and U from UO_2 exposed to 33,000 MWD/MT at 30 MW/MT UO_2 powder and pellets Enrichment = variable Vitrified uranium oxides Enrichment = variable	X		X	X	NRC licensing requirement
6	Spent fuel cask	UO_2 fuel assemblies Enrichment = 3.5% U-235	X	X	X	X	NRC licensing requirement
7	Plutonium product container	PuO_2 powder Isotopic composition — Pu-239 = 95% Pu-240 = 5%	X				NRC licensing requirement
8	Uranium product container	UF_6 Enrichment = 3.5% U-235	X				NRC licensing requirement
9	High-level cask	Vitrified actinides Np, Am, Cm, Pu, and U from UO_2 exposed to 33,000 MWD/MT at 30 MW/MT Canistered UO_2 fuel assemblies Enrichment = 3.5% U-235 UO_2 powder and pellets Enrichment = 3.5% U-235 Vitrified UO_2 Enrichment = 3.5% U-235	X	X	X	X	NRC licensing requirement

A.6 DECOMMISSIONING

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
1	Extent of decommissioning	Remove surface facilities, backfill mine and shafts	X	X	X	X	Project assumption
2	Time to decommission						
	AFRs	3 years		X	X	X	TR W Study (3/78)
	P/E facilities	4 years	X	X	X	X	NUREG-0278 ^(a) (10/77)
	Repositories	7 years	X	X	X	X	TRW Study 3/78
3	Density of crushed salt used for backfill	80 lb/ft ³	X	X	X	X	Stearns-Roger
4	Practical limit on typical backfill techniques	80% of void filled (2 to 3 ft void near ceiling of rooms not filled)	X	X	X	X	Stearns-Roger

(a) Developed for Reference Case

A.7 ECONOMICS

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
1	Dollar basis	January 1979	X	X	X	X	Project assumption
2	Expression of costs	\$/MTHM as charged to reactor	X	X	X	X	Client direction
3	Availability of facilities	When needed	X	X	X	X	Project assumption
4	Recovered products and quantities per MTHM	Credit for recycle U = \$93/kg UF ₆ Pu = \$30/g PuO ₂	X X				Project calculations
5	Cost of enrichment (3.2%)	\$110/SWU	X				Project assumption
6	Cost of U ₃ O ₈	\$43/lb	X				Project assumption
7	Discount rate for present value calculations	10%	X	X	X	X	Project assumption
8	Period of analysis	Costs allocable to spent fuel received in first 40 years of operations	X	X	X	X	Project assumption
9	Capital cost contingency	20% of facility costs	X	X	X	X	Project assumption
10	Owner's costs (during construction)	30% of total estimated facility costs	X	X	X	X	Project assumption
11	Costs for decommissioning						
	AFR	10% of capital		X	X	X	TRW study (3/78)
	P/E facilities	9% of capital		X	X	X	TRW study (3/78)
	P/E facility	15% of capital	X				NUREG-0278 (10/77)
	Repository	33% of capital	X	X	X	X	TRW study (3/78) Note: Costs do not include land acquisition and interest costs during construction
12	Operating Lives	Years					Project assumptions and calculations
	AFR	40		X	X	X	
	P/E facilities	40	X	X	X	X	
	Repository	40	X	X	X	X	More than one repository may be required
13	R&D costs prior to the selection of the system	Not included	X	X	X	X	Project assumptions
14	Government administrative costs	Not included	X	X	X	X	Project assumptions

A.7

ECONOMICS (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
15	Operating costs P/E facilities	Census x \$24,000/yr	X	X	X	X	Project assumptions
		1% capital for operating supplies and main materials	X	X	X	X	
		0.1% capital for utilities costs	X	X	X	X	
		Canisters @ \$5,000 each	X	X	X	X	
	AFR	DOE ratio AFR/case 1 P/E facility		X	X	X	
		multiplied by Case 1 P/E facility from above					
	Repository	Stearns-Roger Report	X				
		Census ratio x Reference Case		X	X	X	
		Adjust for postretrieval period on census changes and sleeve credits	X	X	X	X	

A.8 FUEL

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
1	Exposure history	PWR = 33,000 MWD/MTHM at 30 MW/MTHM	X	X	X	X	OWI/TM-36/22 ⁽³⁾
	Total burnup and average specific power	BWR = 27,500 MWD/MTHM at 20.7 MW/MTHM	X	X	X	X	
2	Physical characteristics of fresh fuel	See Table A-1	X	X	X	X	
3	Radioactivity	@ 1 yr					Y/OWI/TM-34 ⁽⁴⁾
	UO ₂ fuel	2.136x10 ⁶ Ci/MTHM		X	X	X	ORIGEN Printout
		@ 10 yr					
		3.811x10 ⁵ Ci/MTHM		X	X	X	
	UO ₂ /MOX blend	@ 1 yr					
		2.301x10 ⁶ Ci/MTHM	X				
4	Heat generation	@ 1 yr					
	UO ₂ fuel	9.45 kW/MTHM		X	X	X	Y/OWI/TM-34
		@ 10 yr					
		1.12 kW/MTHM		X	X	X	ORIGEN Printout
	UO ₂ /MOX blend	@ 1 yr					
		10.73 kW/MTHM	X				
5	U, Pu content	@ 1 yr					
	U total MOX fuel	952.0 kg/MTHM	X				Y/OWI/TM-34 ORIGEN printout
	UO ₂ fuel	@ 1 yr					
		958.1 kg/MTHM		X	X	X	
		@ 10 yr					
		957.8 kg/MTHM		X	X	X	
	Pu total MOX fuel	@ 1 yr					
		14.4 kg/MTHM	X				
	UO ₂ fuel	@ 1 yr					
		9.1 kg/MTHM		X	X	X	
		@ 10 yr					
		8.7 kg/MTHM		X	X	X	
6	Assemblies per year	BWR/PWR					Project assumption
	By number of assemblies	60/40 (10,200 BWR/6,800 PWR)	X	X	X	X	
	By weight (HM)	37/63	X	X	X	X	
7	Quantity of heavy metal per assembly	MTHM (charged per assembly)					In fresh fuel
	BWR (UO ₂)	0.183		X	X	X	
	PWR (UO ₂)	0.461		X	X	X	
	BWR (MOX)	0.183	X				MOX fuel
	PWR (MOX)	0.461	X				Y/OWI/TM-34 ⁽⁴⁾
8	Isotopic Pu (spent fuel)	<u>Nonrecycle</u>					Y/OWI/TM-34
	238	1%	X	X	X	X	ORIGEN printout
	239	59%	(Pu re cycle)	(non re cycle)	(non-re cycle)	(non re cycle)	Recycle data based on first cycle
	240	24%					
	241	12%					
	242	5%					

A.8

FUEL (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
9	Active volume of the individual assembly						
	PWR (UO ₂)	2.8 ft ³ (0.08 m ³)		X	X	X	Y/OWI/TM-34
	BWR (UO ₂)	1.1 ft ³ (0.03 m ³)		X	X	X	Total volume occupied by claddings in active volume
	PWR (MOX)	same as above	X				
	BWR (MOX)	same as above	X				
10	<u>Pu recycle fuel</u>						
	Core makeup	25% of elements are MOX fuel	X				Y/OWI/TM-36/19 ⁽⁵⁾
	Pu fissile fresh	61.3%					TAD-ERDA-76-43
	fissile spent	55.4%					Subsection 3.1
	fissile Pu to replace fissile U	1.2 atoms of Pu replaces one U-235 atom					
11	Expected radiation levels from spent fuel @ 10 yr	(See also Figure A-1)	X	X	X	X	Y/OWI/TM-36/22 ⁽³⁾
	PWR, surface of canister	3.5 x 10 ⁺⁴ rem/hr, γ 3.0 x 10 ⁻¹ rem/hr, n					ORNL/TM-6008 ⁽⁶⁾
	PWR, 10 feet from center	1.8 x 10 ⁺³ rem/hr, γ 1.8 x 10 ⁻² rem/hr, n					For conservative design, ORNL used 2 x γ figures and 30 x n numbers
	BWR, surface of canister	2.4 x 10 ⁺⁴ rem/hr, γ 2.2 x 10 ⁻¹ rem/hr, n					
	BWR, 10 feet from center	1.3 x 10 ⁺³ rem/hr, γ 1.3 x 10 ⁻² rem/hr, n

Figure A-1
Dose Rate from PWR Spent Fuel Assembly

Table A-1
PHYSICAL CHARACTERISTICS OF LWR FUEL ASSEMBLIES
CHARGED TO REACTOR

	BWR⁽⁷⁾ GE (8x8)	PWR^(a) Westinghouse (17x17)
Overall assembly length, m	4.470	4.058
Cross section, cm	13.9 x 13.9	21.4 x 21.4
Fuel element length, m	4.064	3.84
Active fuel height, m	3.759	3.658
Fuel element O.D., cm	1.252	0.95
Fuel element array	8 x 8	17 x 17
Assembly total weight, kg	275.7	665
Uranium/assembly, kg	183.3	461
UO ₂ /assembly, kg	208.0	523
Zircaloy/assembly, kg	57.9 ^(b)	127 ^(c)
Hardware/assembly, kg	9.77 ^(d)	15 ^(e)
Total metal/assembly, kg	67.7	142
Nominal volume/assembly, m ³	0.0864 ^(f)	0.186 ^(f)

(a) Westinghouse

(b) Includes Zircaloy fuel-element spacers

(c) Includes Zircaloy control-rod guide thimbles

(d) Includes stainless steel tie-plates and Inconel springs

(e) Includes 10 kg stainless steel nozzles and Inconel-718 grids

(f) Based on overall outside dimension

Note: The physical characteristics above are also assumed to apply to the MOX fuel assembly.

A.11

PROCESSING/ENCAPSULATION FACILITIES

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
1	Ownership	U S Government	X	X	X	X	Project assumption
2	Age of fuel for processing/encapsulation	1 yr 10 yr	X	X	X	X	Y/OWI/TM-34 ⁽⁴⁾ Client direction
3	Throughput of individual facilities	1,500 MTHM/yr	X	X	X	X	Project assumption
4	Useful life for the P/E facility	40 yr	X	X	X	X	Project assumption
5	Waste loading — P/E facility to cask	Dry	X	X	X	X	Project assumption
6	Geographical location	Regionally located	X	X	X	X	Project assumption
7	Storage or surge capacity						Project assumption
	Uranyl nitrate	8 days (input to UF ₆)	X				
	UF ₆	25 days (product storage area)	X				
	Pu nitrate	10 days (input to conversion)	X				
	PuO ₂	30 days (product storage area)	X				
	TRU fixed	30 days	X		X	X	
	Hulls, end fittings	30 days (storage area/encapsulation cell)	X		X	X	
	HLLW	40 days (tanks — in cell)	X				
	HLW + Pu liquid	30 days (dissolved fuel, tanks)				X	
	HLW vitrified	9 yr (in pool)	X				
	Fuel, incoming	60 days (in pool)	X	X	X	X	
	encapsulated	30 days (in pool)		X			
	chopped	30 days (surface storage)			X		
	vitrified	30 days (surface storage)				X	
8	Area provided for cask						Project assumption
	Receiving	2 rail casks	X	X	X	X	Rail cask shown to give
	Decon testing	4 rail casks	X	X	X	X	maximum area
	Shipping	4 rail casks	X	X	X	X	commitment
9	Fuel storage baskets	4 PWRs or 9 BWRs/basket	X	X	X	X	DPSTD-ISFS-78-7 ⁽¹¹⁾
10	Basket material	Stainless steel, not poisoned (partial neutron adsorption)	X	X	X	X	Project assumption
11	Design limit for radioactivity in pool	$<2 \times 10^{-4}$ Ci/m ³	X	X	X	X	DPSTD-ISFS-78-7 (4/78)
12	Design maximum conductivity	<1 to 2μ mho/cm	X	X	X	X	DPSTD-ISFS-78-7 (4/78)
13	Water shielding above fuel in transit or storage	≥ 13 ft	X	X	X	X	DPSTD-ISFS 78-7 (4/78)
14	Depth of pool	Enough to pass fuel in water over top of stored fuel (48 ft)	X	X	X	X	Project assumption
15	Fuel unloading — cask to P/E facility	Wet	X	X	X	X	Project assumption
16	Operating days	Days/yr					Project assumption
	Fuel receiving and waste shipping	365	X	X	X	X	
	Processing	300	X	X	X	X	

A.12

PROCESS SELECTION AND ASSUMPTIONS

Item	Description	Study Bases	Ref Case	Applicability			Remarks and References
				Case 1	Case 2	Case 3	
1	End fittings removal and fuel chopping	Fuel disassembly machine systems	X	X	X	X	HEF ^(a) Interim Design Report, ORNL/AFRP-78/6
2	Volatile fission products recovery						ORNL/NUREG/TM-6 ^(b) and project assumption
	Species recovered	Tritium, iodine, Kr-Xe- ¹⁴ C	X		X	X	
	Tritium (from voloxidizer)	≈ 100% recovery as water along with moisture in air inleakage	X		X	X	
	Iodine (from voloxidizer and dissolver offgas)	≈ 100% recovery	X			X	
	Iodine (from voloxidizer)	≈ 65% recovery			X		
	Kr-Xe (from voloxidizer and dissolver offgas)	≈ 100% recovery	X			X	
	Kr-Xe (from voloxidizer)	≈ 50% recovery			X		
	Carbon-14 as ¹⁴ CO ₂ and mixed with Kr and Xe	≈ 100% recovery along with CO ₂ in air inleakage	X		X	X	
	Recovery process						ERDA-76-43
	Tritium	Oxidation, adsorption-desorption, and then fixed in cement	X		X	X	pp 13 21 to 13 25 and 14 12 to 14 14
	Iodine	Separated as iodic acid (iodox process), converted to barium iodate, and then fixed in cement	X		X	X	pp 13 17 to 13 19 and p 14 8
	Kr-Xe- ¹⁴ CO ₂	R-12 absorption	X		X	X	pp 13 6 to 13 9
	Assumptions						Project assumptions
	Air inleakage to chopper and voloxidizer	≈ 100 SCFM	X		X	X	
	Air inleakage to dissolver	≈ 100 SCFM	X			X	
	H ₂ O in cell air	40°F dew point	X	X	X	X	
	CO ₂ in cell air	≈ 0.03%	X	X	X	X	
3	Offgas treatment						ERDA-76-43
	NO _x	Aqueous scrubbing followed by NH ₃ catalytic reduction	X	X	X	X	pp 13 44 to 13 47
	Iodine	Silver zeolite adsorption and then fixed in cement	X	X	X	X	pp 13 15 to 13 16 and 14 8
	Ruthenium (voloxidation)	Silica gel adsorption	X		X	X	
	Ruthenium (from calcination)	Sugar addition for volatility suppression followed by silica gel adsorption	X			X	pp 13 59 to 13 63
	HCl combustion products	Conversion to NaCl in molten salt combustion unit	X	X	X	X	pp 13 49 to 13 50
	Particulates	Prefilters and HEPA filters	X	X	X	X	pp 13 64 to 13 67
4	Fuel dissolution	Barnwell Nuclear Fuel Plant (BNFP) process	X			X	BNFP final safety analysis report (FSAR)
5	Solvent extraction	BNFP process	X				BNFP FSAR
6	Hulls treatment						Project assumption
	Tube segment	Compacted in small can and then packaged 7 cans per canister, overall volume reduction ratio is 2.48	X			X	
	End fittings	Packed into canister without compaction	X	X	X	X	

A.12

PROCESS SELECTION AND ASSUMPTIONS (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
7	UF ₆ production	BNFP process: calcination, hydrofluorination, and fluorination	X				BNFP
8	PuO ₂ production	BNFP process: oxalate precipitation, drying, and calcination	X				BNFP
9	HLLW constituents	Aqueous raffinate from HA column combined with ILLW from solvent cleanup	X				BNFP and Y/OWI/TM-34 ⁽⁴⁾
10	HLW fixation	Concentration, spray calcination, and immediate vitrification in continuous ceramic melter with 2:1 glass to calcined waste ratio	X				ERDA-76-43 pp. 6.28 to 6.29 and pp. 6.43 to 6.47
11	Combustible TRU waste fixation	Burned in molten salt combustion unit; reduced volume to 1/40 of feed	X	X	X	X	ERDA-76-43 pp. 9.54 to 9.57
12	Fixation of TRU waste ash	Fixed in clay: 1 part ash in 1 part clay	X	X	X	X	ERDA-76-43 pp. 12.28 to 12.31
13	Acid recovery and excess water release						
	Nitric acid	Recovered by distillation for recycle (BNFP process)	X			X	BNFP, FSAR
	Water	Recovered from acid recovery overhead for recycle (BNFP process)	X			X	BNFP, FSAR
	Water	Recovered from waste concentration overhead		X	X		Project assumption
	Excess water release	Vaporized with ruthenium suppression, iodine adsorption and particulate removal	X	X	X	X	Project assumption
14	Dissolved spent fuel fixation	Concentration, spray calcination, and vitrification in continuous ceramic melter with 3:1 glass to calcined fuel ratio				X	Project assumption
15	Age of fuel processed at P/E facility						
	1 yr	3:1 UO ₂ to MOX Fuel	X				Y/OWI/TM-36/19 ⁽⁵⁾
	10 yr	UO ₂ fuel		X	X	X	Project direction
16	Encapsulation	See "Canisters/Packaging"					

(a) Not Experimental Facility of the Advanced Fuel Recycle Program⁽²²⁾

A.14 REPOSITORY

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
1	Storage media, primary	Salt	X	X	X	X	Other media will be discussed in study variations
2	Rate of receiving	5,000 MTHM equiv /yr	X	X	X	X	Project direction
3	Minimum age to repository	Years					Project assumption
	HLW	10	X				
	Fuel, encapsulated	10		X			
	Variations	5, 25		X			
	Fuel, chopped	10			X		
	Fuel, vitrified	10				X	
	Other waste	As soon as it is generated	X	X	X	X	
4	Ownership	U S government	X	X	X	X	Project direction
5	Retrievability	<u>Years</u>					Project direction
	HLW	5	X				
	Hulls/end fittings	5	X	X	X	X	
	Fuel, chopped	5			X		
	Fuel, vitrified	5				X	
	Fuel, encapsulated	5		X			
	Variations	Study sensitivity of 10-yr and 25-yr designs		X			
6	Area definitions		X	X	X	X	Stearns-Roger and Kaiser design
	Fenced area (above grd)	600 acres					
	Gross area (below grd)	2,000 acres					
	Central core (below grd)	130 acres					
	Main drifts (below grd)	170 acres					
7	Max temperature allowed						
	1% of volume of unit cell	250°C	X	X	X	X	Subsection D 2 1
	25% of volume of unit cell	200°C	X	X	X	X	
	Earth's surface	+ 1 °C	X	X	X	X	
	Outside restricted area	+ 1 °C	X	X	X	X	
8	Time required for retrieval	Same time as required to fill (up to 5 years)	X	X	X	X	Y/OWI/TM-44 ⁽⁹⁾
9	Sleeve requirements	For spent fuel, HLW and hulls/end fittings during retrieval period	X	X	X	X	Project assumption
10	Sleeve height	Full length of hole Segmented so as not to affect height of rooms	X	X	X	X	Project assumption
11	Pressure relief on sleeves	Overboring of hole shall prevent distortion, which could inhibit removal of canister during retrievability period	X	X	X	X	Project assumption
12	Surge storage capacity at receiving building		X	X	X	X	Subsections D 3 to D 6
	Remote handling	10 days (shipping cask receipts)					
	Contact handling	1 truck shipment					

A.14 REPOSITORY (Cont'd)

Item	Description	Study Bases			Applicability				Remarks and References
					Ref. Case	Case 1	Case 2	Case 3	
13	Sleeve size	O.D. in.	Wall Thickness in.	Length ft.					
	HLW	16	3/8	15	X				Project assumption Larger O.D. sleeves will be required for overpacked canisters (Subsection A.3)
	Hulls/end fittings	22	3/8	23.5	X	X	X	X	
	Fuel, canistered	18	3/8	20.5		X			
	Fuel, chopped	16	3/8	23.5			X		
	Fuel, vitrified	22	3/8	15				X	
	Material is carbon steel								
14	Maximum areal heat load	Load kW/acre	Emplace Age, yr.	Retrieval yr.					Project direction. Assembly blend of 60/40 BWR/PWR and 3/1 UO ₂ /MOX
	HLW, vitrified	150	10	5	X				3 BWRs or 1 PWR
	Fuel, encapsulated	60	10	5		X			3 BWRs or 1 PWR
	Variation	48	10	10		X			3 BWRs or 1 PWR
	Variation	36	10	25		X			3 BWRs or 1 PWR
	Variation	101	5	5		X			3 BWRs or 1 PWR
	Variation	41	25	5		X			3 BWRs or 1 PWR
	Fuel-pin bundles	60	10	5		X			3 BWRs or 1 PWR
	Fuel, chopped	60	10	5			X		Assembly blend 60/40 BWR/PWR
	Fuel, vitrified	60	10	5				X	Assembly blend 60/40 BWR/PWR
15	Time to fill,	Years							
	HLW, vitrified	39			X				
	Fuel, encapsulated	17				X			
	Variation	14				X			
	Variation	10	Approximate values			X			
	Variation	18				X			
	Variation	18				X			
	Fuel-pin bundles	17				X			
	Fuel, chopped	19					X		
	Fuel, vitrified	17						X	
16	Shafts								Stearns-Roger concept
	Man, materials	31 ft x 2,000 ft			X	X	X	X	
	Spent fuel, HLW, ILW, and LLW materials	28 ft x 2,000 ft			X	X	X	X	
	Supply vent	26 ft dia. x 2,000 ft			X	X	X	X	
	Exhaust vent (emplacement area)	22 ft dia. x 2,000 ft			X	X	X	X	
	Exhaust vent (mining area)	16 ft dia. x 2,000 ft			X	X	X	X	
17	Room closure during retrieval period	< 10 to 15% of original room opening			X	X	X	X	Y/OWI/TM-36/1 ⁽¹⁰⁾
18	Far field uplift over repository	< 4 to 5 ft			X	X	X	X	Y/OWI/TM-36/1

A.14

REPOSITORY (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
19	Storage wells	Hole Dia in Depth ft					Project assumption
	HLW, vitrified	28 16½	X				
	Hulls, end fittings, gas	26 25	X		X	X	
	ILW	24 25	X	X	X	X	
	Fuel, encapsulated	30 22		X			
	chopped	28 16½			X		
	vitrified	34 16½				X	
20	Shaft location	Central ingress, egress	X	X	X	X	Stearns-Roger shaft pillar concept
21	Depth of storage/disposal of all nuclear materials	2,000 ft	X	X	X	X	Adaptation of Stearns-Roger
22	Corridors	30 ft wide x 15 ft high	X	X	X	X	Stearns-Roger concept
23	Arrays of storage wells for heat producing, high-level wastes	Minimum Canister Spacings					Number of rooms are established by HLW and spent fuel canisters ILW will be emplaced in the same rooms (or in ILW-only rooms) Based on thermal loading
		No of Rows Per Room In a Row in Between Rows in					
	HLW, vitrified	1 56 N A	X				
	Fuel, encapsulated	2 64 64		X			
	Fuel-pin bundles	2 73 72		X			
	Fuel, chopped	2 56 56			X		
	Fuel, vitrified	2 76 66				X	
24	Minimum web thickness (between adjacent holes in row)	Equal to hole diameter	X	X	X	X	Relationship of hole diameter to web thickness from Y/OWI/TM-36/6 ⁽¹¹⁾ (confirmed also by Kaiser Eng) Actual values per SAI calculations
25	Closest approach to wall on outside row of holes	5.0 ft from center of hole	X	X	X	X	Y/OWI/SUB-77/22324 ⁽¹²⁾
26	Annulus around sleeve						
	HLW, spent fuel (high heat generating)	6-in air gap	X	X	X	X	Y/OWI/TM -36 ⁽¹⁰⁾ applied to all cases
	Hulls, end fittings (low heat generating)	Minimal	X	X	X	X	(indicates no temperature problem)
27	Storage/disposal room dimensions	Width ft Height ft Length ft					Project assumption
	HLW	20 19.5 1,000	X				
	ILW	20 17.5 1,000	X				
	Fuel, encapsulated	20 25 1,000		X			
	Fuel, chopped	20 19.5 1,000			X		
	Fuel, vitrified	20 19.5 1,000				X	

A.14 REPOSITORY (Cont'd)

Item	Description	Study Bases			Applicability				Remarks and References
					Ref Case	Case 1	Case 2	Case 3	
28	Pillar size	Width ft	Emplace Age yr	Retrieval Period yr					Project direction SAI calculations for SFDS (gives minimum extraction rates)
	HLW, vitrified	142	10	5	X				
	Fuel, encapsulated	130	10	5		X			
	Variation	167	10	10		X			
	Variation	230	10	25		X			
	Variation	130	5	5		X			
	Variation	132	25	5		X			
	Fuel-pin bundles	243				X			
	Fuel, chopped	179	10	5			X		
	Fuel, vitrified	49	10	5				X	
29	Continuous borer machine mining rate of excavation	Over 200 tons/hr (with conveyors)			X	X	X	X	Kaiser NWTS-2990 ⁽¹³⁾
30	Boring rate of storage wells	3 to 4 holes/crew per shift			X	X	X	X	Y/OWI/TM-36/8 ⁽¹⁴⁾
31	Operating days/yr	300			X	X	X	X	Project assumption
32	Waste emplacement during retrievability period								Subsections D 3, D 4, D 5, and D 6
	High level/high heat generating (HLW, spent fuel)	Packed in canisters, one canister per storage hole in salt, sleeve and plug			X	X	X	X	
	High level/low heat generating (hulls, end fittings, gases)	Packed in canisters, 2 tandem canisters per storage hole in salt, sleeve and plug			X	X	X	X	Case 1 applies to compact fuel pin option (only end fittings)
	ILW	Packaged in canisters, 2 tandem canisters per storage hole in salt, no sleeve or plug			X	X	X	X	
	LLW	Packed in drums, stacked in room (6 drums per pallet)			X	X	X	X	
33	Waste emplacement after retrievability period								Subsections D 3, D 4, D 5, and D 6
	High level/high heat generating (HLW, spent fuel)	Packed in canisters, one canister per storage hole in salt, no sleeve or plug			X	X	X	X	
	High level/low heat generating (hulls, end fittings, gases)	Packed in canisters, 2 tandem canisters per storage hole in salt, no sleeve or plug			X	X	X	X	Case 1 applies to compact fuel pin option (only end fittings)
34	Retrievable materials	HLW and spent fuel only			X	X	X	X	Project assumption Subsections D 3, D 4, D 5, and D 6
35	Operations	300 days, 24 hr/day			X	X	X	X	Project assumption
36	LLW emplacement	LLW is placed in temporary rooms and, after retrieval period, is moved into rooms that are being backfilled			X	X	X	X	Project assumption
37	Ventilation	Based on Kaiser design			X	X	X	X	EY-77-C-05-5366 ⁽¹⁵⁾

A.15 SAFEGUARDS

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
1	Physical security Transportation P/E facilities AFR Repository	NRC regulations and state-of-the-art guides	X	X	X	X	See Section 5
2	Material control and accounting Transportation P/E facilities AFR Repository	NRC regulations and state-of-the-art guides	X	X	X	X	See Section 5
3	Theft	To fabricate nuclear explosive devices	X	X	X	X	See Section 5
4	Sabotage	Consequence to public from dispersal of radioactivity	X	X	X	X	See Section 5

A.16 SAFETY

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
1	Applicable regulations and guides	Proposed for AFRs Standards for protection Siting waste facilities General radiation safety Transportation Packaging wastes Waste management	X	X	X	X	10 CFR 72 10 CFR 20 10 CFR 50 (App F) ERDA Manual Chapter 0524 49 CFR 171-179 10 CFR 71 ERDA Manual Chapter 0511
2	Emergency dose limit, whole body	100 rem	X	X	X	X	Same as Item #1
3	Restricted work zones (noncontinuous occupancy) Aboveground facilities Belowground facilities	0.6 mrem/hr 0.1 mrem/hr	X	X	X	X	Same as Item #1
4	Whole body dose limits Occasionally exposed workers Average population exposure Genetic limit to general population	5 rem/yr (total combined whole body and internal deposition) 0.5 rem/yr 0.17 rem/yr 0.17 rem/yr	X	X	X	X	Same as Item #1

A.17

THERMAL/MECHANICAL PROPERTIES

Item	Description	Study Bases	Ref. Case	Applicability			Remarks and References
				Case	Case	Case	
				1	2	3	
1	Materials	Density, lb/ft ³					Y/OWI/TM-36/19 ⁽⁵⁾
	HLW, vitrified	187	X				
	He	0.004		X			
	Stainless steel 304L	488	X	X	X	X	
	Carbon steel	490	X	X	X	X	
	Concrete	144	X	X	X	X	
	Salt	135	X	X	X	X	
	Air	0.07	X	X	X	X	
	UO ₂	684		X	X		
	Zr-clad	412	X	X	X	X	
	Inconel-718	524	X	X	X	X	
	Sand	98		X			Study variation
2	Materials	Spec Heat Btu/(lb-°F)	Thermal Conduct. Btu/(hr-ft-°F)				Presumed to be values at expected temperatures in repository environment. Y/OWI/ TM-36/19
	HLW, vitrified	0.20	0.7	X			
	He	1.24	0.12		X		
	Stainless steel 304L	0.11	9.5	X	X	X	
	Carbon steel	0.11	26	X	X	X	
	Concrete	0.2	0.54	X	X	X	
	Salt	0.22	See Item # 8	X	X	X	
	Sand	0.2	0.2				Study variation
	Air	0.24	See Item #9	X	X	X	
	UO ₂	See Item #6	See Item #6		X	X	
	Inconel-718	N.A.	6.5	X	X	X	Engineering Handbook
	Zr-clad	See Item #7	See Item #7	X	X	X	
3	Salt properties			X	X	X	New Mexico Salt Y/OWI/TM-36/14 ⁽¹⁶⁾ (Table 4-3) Reference does not identify whether permeability is derived from formation or sample
	Moisture	1.75%					
	Porosity	< 3%					
	Compressive strength	4,000 psi					
	Tensile strength	240 psi					
	Thermal expansion (linear)	22.2x10 ⁻⁶ /°F					
	Permeability (vert., horiz.)	10 ⁻¹⁸ ft/min					
4	HLW decay heat rate (@ 10 yr)	PWR	Watts/MTHM BWR Weighed Average				Assumes 63/37 PWR/ BWR mix by weight
	No recycle						
	Actinides	61	51				Y/OWI/TM-34 ⁽⁴⁾
	Fission products	970	785				Table A-7 and Table A-8
		1,031	836	959	X		
	U, Pu recycle						
	Actinides	1,442	1,071				Y/OWI/TM-34
	Fission products	829	677				Table A-17 and Table A-18
		2,271	1,748	2,077	X		
	3 parts UO ₂ /1 part MOX	1,341	1,064	1,240	X		Calculated
5	Spent fuel, decay heat rate (@10 yr)	1,189	996	1,118		X	Y/OWI/TM-34 Table A-3

A.17

THERMAL/MECHANICAL PROPERTIES (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
6	UO ₂	Spec Heat Btu/(lb-°F)	X	X	X	X	Y/OWI/TM-36/19
	@ 300°F	4.05					
	@1,000°F	2.37					
	@2,000°F	1.56					
	@3,000°F	1.33					
7	Zr-clad	Heat Capac Btu/(lb-°F)	X	X	X	X	Y/OWI/TM-36/19
	@ 70°F	0.070					
	@ 200°F	0.073					
	@ 400°F	0.077					
	@ 600°F	0.081					
	@ 800°F	0.085					
	@1,000°F	0.086					
	@1,200°F	0.086					
8	Salt	Thermal Conductivity Btu/(hr-ft-°F)	X	X	X	X	Y/OWI/TM-36/19 ⁽⁵⁾
	@ 32°F	3.53					
	@ 200°F	2.90					
	@ 212°F	2.43					
	@ 302°F	2.08					
	@ 392°F	1.80					
	@ 482°F	1.60					
	@ 572°F	1.44					
	@ 662°F	1.33					
	@ 752°F	1.20					
9	Air	Thermal Conductivity Btu/(hr-ft-°F)	X	X	X	X	Y/OWI/TM-36/19
	@ 32°F	0.014					
	@ 122°F	0.016					
	@ 212°F	0.018					
	@ 302°F	0.019					
	@ 392°F	0.021					
	@ 482°F	0.023					
	@ 572°F	0.024					
	@ 662°F	0.026					
	@ 752°F	0.028					
10	Ambient temperature of repository						Y/OWI/TM-36/19
	@2,000 ft	93°F	X	X	X	X	
11	Geothermal gradient deep formations	0.021°F/ft	X	X	X	X	Y/OWI/TM-36/19
12	Maximum allowable waste temperature during retrievability period	Temp. of Waste °C					Y/OWI/TM-42 (8/77)
	HLW, vitrified	500	X				
	Fuel, encapsulated	200 (clad)		X			RHO-ST-10
	pin bundles	200 (clad)		X			
	Fuel, chopped	700			X		
	Fuel, vitrified	500				X	

A.17

THERMAL/MECHANICAL PROPERTIES (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
13	Emissivity of HLW can	0.5 to 1.0 (steel)	X				Y/OWI/TM-36/19 Radiant heat transfer Calculations are not sensitive to this range of values
14	Emissivity of spent fuel radiating to canister	0.6		X			Y/OWI/TM-36/19
15	Far-field factors		X	X	X	X	Y/OWI/TM-36/19
	Solar flux	73.4 Btu/hr-ft ²					
	Wind velocity	10 mph					
	Air temperature	66°F					
	Surface emissivity	0.3					
	Sky temperature	45°F					
	Geothermal gradient	0.021 °F/ft					
	Thermal diffusivities	0.085 ft ² /hr					
16	Identification of confinement barriers	Canister/overpack/ sleeve/salt	X	X	X	X	Project assumption
17	Salt temperature limit before decrepitation begins	260°C to 320°C	X	X	X	X	Y/OWI/TM-36/19 ⁽¹⁰⁾
18	Area available for near field heat transfer in repository	Room, adjacent pillars, but not corridors, buttress pillars (if any), or receiving areas	X	X	X	X	Project assumption
19	Maximum heat loading permitted in canister vitrified HLW						Y/OWI/TM-36/19 ⁽⁵⁾
	(12-in. canister, 8-ft fill)	3.2 kW/canister (if the canister is the limiting factor) 3.6 kW/canister (if waste temperature is the limiting factor)	X				
20	Maximum loading permitted in other diam. of HLW (vitrified)	Loading kW/canister					
	4 inches	2.3	X			X	If the temperature of the canister is the limiting factor Y/OWI/ TM-36/19
	8 inches	2.8					
	16 inches	3.5					
	24 inches	3.7					
21	Maximum loading permitted for HLW (vitrified), with varied thickness of backfill around sleeve	Backfill in 0 4 6 8 12 18	Loading kW/canister 5.3 3.8 3.2 2.8 2.5 2.5	X		X	Y/OWI/TM-36/19
22	Decay heat rate expected from HLW canister of						Y/OWI/TM-36/22 ⁽³⁾
	PWR (3 UO ₂ /1 MOX)	2.8 kW	X				For 10 yr aged materials
	BWR (3 UO ₂ /1 MOX)	2.3 kW	X				
	Blended PWR/BWR (MOX)	2.6 kW	X				

A.17

THERMAL/MECHANICAL PROPERTIES (Cont'd)

Item	Description	Study Bases	Ref Case	Applicability			Remarks and References
				Case 1	Case 2	Case 3	
23	Decay heat rate (maximum from uncompacted hulls (@ 1 yr cooled)	240 W/MTHM approx 16 W/ft ³ 48% cladding 12% end fittings 40% hardware	X		X	X	For maximum conceptual impact use XN-FR-32 (converted to 1 yr cooled) which assumes 0.12% Co-59 in end fittings 0.20% Co-59 in hardware and 40 000 burnup at 40 kW/kg
24	Decay heat rates						
	ILW solids (compacted)	0.002 kW/m ³	X		X	X	
	TRU LLW solids (compacted)	0.001 kW/m ³ (given)	X	X	X		
25	Decay heat rate from spent fuel	kW/canister					Project direction
	Fuel, canistered			X			
	BWR	0.55 (3 BWR/canister)					
	PWR	0.55 (1 PWR/canister)					
	Fuel, chopped	0.68			X		
	Fuel, vitrified	0.23				X	
26	Corrosion rate in salt mine during retrievability period						Y/OWI/TM-36/22
	Stainless steel	0.001 mil/yr	X	X	X	X	
	Carbon steel	2 mils/yr	X	X	X	X	Requires water (as liquid) and O ₂
	Likelihood of stress	C/S is less likely to have a stress cracking problem	X	X	X	X	Only water vapor at retrieval temperatures
27	Volume of HLW after vitrification	3.0 ft ³ /MTHM	X				Y/OWI/TM-44 ^(a) Calcine is intermediate to glass
28	Volumes of HLLW ILLW	150 gal/MTHM 75 gal/MTHM	X X				BNWL-2169
29	Density of wastes						
	Noncombustible, misc						
	Radwaste	0.25 g/cm ³	X			X	Y/OWI/TM-36/2 ⁽¹⁷⁾
	Failed equipment	0.5 g/cm ³	X	X	X	X	Y/OWI/TM-36/2 ⁽¹⁷⁾
	HEPA filter media	0.7 g/cm ³	X	X	X	X	Y/OWI/TM-36/2 ⁽¹⁷⁾
	Clay fixed ash	2 g/cm ³	X			X	Y/OWI/TM-36/2 ⁽¹⁷⁾
30	Total length of both end fittings						Project assumption
	PWR	8 in		X ^(a)			Cuts made just beyond fuel tube ends
	BWR	16 in		X ^(a)			
	PWR	16 in	X		X	X	Cuts made 1 in beyond active fuel section
	BWR	28 in	X		X	X	

(a) Case 1 applies only to the end fittings of the compact fuel pin option

A.18 THROUGHPUT

Item	Description	Study Bases	Applicability				Remarks and References
			Ref Case	Case 1	Case 2	Case 3	
1	Fuel cycle rate of spent fuel generation by assembly	5,000 MTHM/yr @ steady state	X	X	X	X	Project direction
		10,200 BWR/6,800 PWR	X	X	X	X	
2	P/E facilities	5,000 MTHM/yr (all facilities)	X	X	X	X	Project direction
	UF ₆ out	7,000 MT/yr (740 canisters)	X				Calculated
	PuO ₂ out	80 MT/yr (2,500 packages, 32 kg each)	X				Calculated
	Waste out	See Items #4 and #5					
3	AFR, fuel	5,000 MTHM/yr (all facilities)		X	X	X	Project assumption
	Waste out	See Item #5					
4	To repository	Packages/yr					For 5,000 MTHM/yr system
	High level/high heat generating						Project assumption
	HLW, vitrified	2,380 (TRU) Canisters	X				
	Spent fuel, BWR	3,400 (TRU) Canisters		X			
	PWR	6,800 (TRU) Canisters		X			
	Chopped fuel	8,070 (TRU) Canisters			X		
	Vitrified fuel	18,520 (TRU) Canisters				X	
	Fuel-pin bundles	5,100 Canisters		X			
	High level/low heat generating						
	Hulls	2,300 (TRU) Canisters	X			X	
	End fittings	1,520 (TRU) Canisters	X		X	X	
	End Fittings	800 (TRU) Canisters		X			
	Kr-Xe-CO ₂	128 (non-TRU) Canisters	X			X	6 gas cylinders per 22 in x 10 ft canister
	Kr-Xe-CO ₂	58 (non-TRU) Canisters			X		
	Intermediate level/low heat generating trash (radwaste)						
	From P/E facility	7,090 (TRU) Canisters	X				
	From P/E UF ₆ facility	920 (non-TRU) Canisters	X				
	From P/E facility	2,320 (TRU) Canisters		X			
	From P/E facility	3,650 (TRU) Canisters			X		
	From P/E facility	5,610 (TRU) Canister				X	
	Failed equipment	540 (TRU) Canisters	X				
		250 (TRU) Canisters		X			
		250 (TRU) Canisters			X		
		400 (TRU) Canisters				X	
	Low-level/low heat generating						
		1 680 (TRU) 55-gal drums	X				
		550 (TRU) 55-gal drums		X			
		850 (TRU) 55-gal drums			X		
		1 230 (TRU) 55-gal drums				X	
	³ H, iodine, and silver zeolite	250 (non-TRU) 55-gal drums	X		X	X	No volume reduction

A.18 THROUGHPUT

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
5	To shallow land burial Low-level waste trash (radwaste)	Packages/yr					For 5,000 MTHM/yr system
	From P/E UF ₆ facility	13,000 (non-TRU) 55-gal drums	X				No volume reduction
	From P/E facility fuel pools	2,860 (non-TRU) 55-gal drums	X	X	X	X	Volume reduction
	From AFRs	6,200 (non-TRU) 55-gal drums		X	X	X	4:1 compaction applied to 80% of waste volume

A.19 TRANSPORTATION

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
1	Materials to be transported						
	Reactor to AFR	Spent fuel, 1 yr old		X	X	X	Appendix B
	Reactor to P/E facility	Spent fuel, 1 yr old	X				Appendix C
	AFR to P/E facility	Spent fuel, 10 yr old		X	X	X	Appendices B and C
	AFR to shallow land burial	Non-TRU LLW		X	X	X	Appendix B
	P/E to repository	Encapsulated fuel, 10 yr old		X			Appendix D
		Chopped fuel, 10 year old			X		Appendix D
		Vitrified fuel, 10 year old				X	Appendix D
		HLW, 10 yr old	X				Appendix D
		Hulls (including end fittings)	X	X ^(a)	X	X	Appendix D
		LLW (TRU)	X	X	X	X	Appendix D
		ILW (TRU and non-TRU)	X	X	X	X	Appendix D
		Misc. offgas products	X		X	X	Appendix D
	P/E to shallow land burial	Non-TRU LLW	X	X	X	X	Appendix C
	P/E to fuel fabrication	UF ₆ and PuO ₂	X				Appendix D
2	Mode of transportation						
	From reactor						
	Spent fuel	1 shipment/rail: 1 shipment/road	X	X	X	X	Appendix E Subsection 2
	From AFR						
	Spent fuel	Rail only		X	X	X	
	Non-TRU (LLW)	Road only		X	X	X	
	From P/E facility						
	Encapsulated fuel	Rail only		X			
	Chopped fuel	Rail only			X		
	Vitrified fuel	Rail only				X	
	HLW	Rail only	X				
	Hulls (including end fittings)	Rail only	X	X ^(a)	X	X	
	ILW (TRU, non-TRU)	Road only	X	X	X	X	
	LLW (TRU, non-TRU)	Road only	X	X	X	X	
	Kr-Xe-CO ₂	Rail only	X		X	X	
	UF ₆	Road only	X				
	PuO ₂	Road only	X				
	Failed equipment	Rail	X	X	X	X	
3	Logistics	Distance, mi					
	Reactor to AFR	400		X	X	X	Appendix E Subsection 4
	Reactor P/E facility	700	X				
	AFR to P/E facility	700		X	X	X	
	P/E to repository	1,100	X	X	X	X	
	P/E to fuel recycle	400	X				
	P/E to shallow land burial	700	X	X	X	X	
	AFR to shallow land burial	400		X	X	X	

A.19

TRANSPORTATION (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
4	Shipping casks						Project assumption (based on ORNL support work)
	Spent fuel (1 yr old) road	NAC-1 assemblies	1 PWR/2 BWR	X	X	X	
	Spent fuel (1 yr old) rail	Conceptual assemblies	8 PWR/19 BWR	X	X	X	
	Spent fuel (10 yr old) rail	Conceptual assemblies	14 PWR/34 BWR		X	X	
	Encap fuel rail	Conceptual assemblies	9 PWR/27 BWR	X			
	Chopped fuel rail	Conceptual (14 canisters/12 in. diameter)			X		
	Vitrified fuel rail	Conceptual (9 canisters/20 in. diameter)				X	
	HLW rail	Conceptual (12 canisters/12 in. diameter)	X				
	Hulls (end fittings and KR-Xe-CO ₂) rail	Conceptual (9 canisters/20 in. diam.)	X	X ^(a)	X	X	
	Failed equipment rail	Conceptual (9 canisters/20 in. diam.)	X	X	X	X	
	ILW (TRU, non-TRU) road	20 in.-O.D. canisters, 7 canisters/cask	X	X	X	X	
	LLW (TRU, non-TRU) road	55-gal. drums, 42 drums/shipment	X	X	X	X	
	UF ₆ container road	83-gal. drums, 33 drums/shipment	X	X	X	X	
	PuO ₂ road	48X, 10 tons/container, 2 containers/shipment	X				
		PTV (256 kg/shipment)	X				
5	Tariffs	Road	X	X	X	X	Current industry rates ⁽¹⁸⁾
		Rail	X	X	X	X	
6	Cask lease rates						
	Spent fuel	Road	\$900/day	X	X	X	These current industry rates may have to be adjusted to reflect public financing.
	Spent fuel, HLW, hulls, Kr-Xe-CO ₂ , etc.	Rail	\$3,600/day	X	X	X	
	TRU, non-TRU	Road	\$250/day	X	X	X	
	UF ₆ container	Road	—	X			
	PuO ₂	Road	\$3,000/day	X			
7	Cask turnaround time						
	Loading and unloading time	Road	24 hr	X	X	X	Current industry practice
		Rail	48 hr	X	X	X	
8	Transportation speed						
	Road	33 mph	X	X	X	X	Current industry experience
	Rail	5 mph					Low rail speed results from loading/unloading delays, shunting for priority service, etc.

(a) Case 1 applies only to the end fittings of the compact fuel pin option.

A.19
TRANSPORTATION (Cont'd)

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
9	Computer codes		X	X	X	X	
	Q GERT	A transportation logistics model					BNPL
	PACRAT	A radwaste shipping cask conceptual design					ORNL
	ORIGEN	A transportation/logistics model Isotopic generation and depletion code					ORNL/TM-6192 ⁽¹⁹⁾ ORNL

A.20

WASTES, GENERAL

Item	Description	Study Bases	Applicability				Remarks and References
			Ref. Case	Case 1	Case 2	Case 3	
1	Volatiles in HLW	0.1% iodine, bromine	X				Y/OWI/TM-34 ⁽⁴⁾
2	Actinides in HLW	0.5% Pu 0.5% U	X X				Y/OWI/TM-34
3	Pu in spent fuel						
	Fissile	6.56 kg/MTHM-no recycle		X	X	X	
	Total	9.4 kg/MTHM-no recycle		X	X	X	
4	Origin of non-TRU wastes	Reactors, reprocessing plant, spent fuel pools, and repository	X (front-end & UF ₆)	X	X	X	RHO-CD-237 ⁽²⁰⁾
5	Fraction of TRU wastes (other than hulls) that can be reduced by compaction	80% of TRU waste is compactable	X		X	X	Y/OWI/SUB-77/42513 ⁽²¹⁾ (2/78)
6	TRU nuclides	Np-237 Pu-238, 239, 240, 241 Am-241, 243 Cm-243, 244	X	X	X	X	Y/OWI/TM-36/2
7	Surface of canister dose rate from HLW						
	PWR, γ_n	8.8×10^{-4} rem/hr 3.8×10^{-1} rem/hr	X X				Y/OWI/TM-36/22
	BWR, γ_n	7.1×10^{-4} rem/hr 3.9×10^{-1} rem/hr	X X				
	10 ft from center of canister, dose rate from HLW						
	PWR, γ_n	4×10^{-3} rem/hr 1.7 rem/hr	X X				
	BWR, γ_n	3.2×10^{-3} rem/hr 1.9 rem/hr	X X				
8	Gamma spectrum 5 yr, aged HLW	Photons/sec 8.14 E 14 1.77 E 16 8.80 E 14 1.30 E 14 1.20 E 13 3.66 E 12 2.86 E 11 9.07 E 9 3.25 meV	X			X	Y/OWI/TM-36/22 ⁽³⁾ To obtain 10-yr photon flux decay, corrections are needed
9	Fuel content of clad waste	0.05% of spent fuel quantity except volatile fission products	X			X	Y/OWI/TM-34

REFERENCES FOR APPENDIX A

- (1) "Spent Fuel Handling and Storage Facility for the International Spent Fuel Storage Program — Including Supplement," Savannah River Laboratory, E.I. du Pont de Nemours & Co., DPSTD-ISDS-78-7(70), April 1978.
- (2) E. L. Moore, "Acceptance Criteria for Spent Fuel Packages to be Stored in a Federal Repository Located in a Salt Formation," Rockwell Hanford Operations, RHO-CD-542, August 1978.
- (3) "Nuclear Considerations for Repository Design," Science Applications, Inc., Y/OWI/TM-36/22, April 1978.
- (4) "Nuclear Waste Projections and Source-Term Data for FY 1977," Union Carbide Corporation, Office of Waste Isolation, Y/OWI/TM-34, May 1978.
- (5) "Thermal Analyses," Science Applications, Inc., Y/OWI/TM-36/19, April 1978.
- (6) "Projections of Spent Fuel to be Discharged by the U.S. Nuclear Power Industry," Union Carbide Corporation, Chemical Technology Division, ORNL/TM-6608, October 1977.
- (7) "Spent Fuel Handling and Packaging Program: Subtask 1.4, Centralized Fuel Data Preliminary Report," Hanford Engineering Development Laboratory, TC-1076, April 1978.
- (8) "Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle — Reprocessing Light-Water Reactor Fuel," Oak Ridge National Laboratory, ORNL/NUREG/TM-6, January 1977.
- (9) "Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations," Union Carbide Corporation, Nuclear Division, Y/OWI/TM-44, April 1978.
- (10) "Executive Summary, Volume 1," Science Applications, Inc., Y/OWI/TM-36/1, April 1978.
- (11) "Repository Preconceptual Design Studies: BPNL Waste Forms in Salt," Parsons-Brinckerhoff, Quade & Douglas, Inc., Y/OWI/TM-36/16, April 1978.
- (12) "Emplacement Hole Drill Evaluation and Specification Study, Volume I," Foster-Miller Associates Inc. Engineers, Y/OWI/SUB-77/22324, September 1977.
- (13) "Conceptual Design Report National Waste Terminal Storage Repository 2, Volume I — Conceptual Design Report, Parts 1 and 2," Kaiser Engineers, Oakland, California, NWTS-2990, August 1978.
- (14) "Repository Preconceptual Design Studies: Salt," Parsons-Brinckerhoff, Quade & Douglas, Inc., Y/OWI/TM-36/8, April 1978.
- (15) "Planning Study for National Waste Terminal Storage 25 Year Retrievability Special Study No. 1," Kaiser Engineers, EY-77-C-05-5366, October 1977.
- (16) "Repository Preconceptual Design Studies: BASALT," Parsons-Brinckerhoff, Quade & Douglas, Inc., Y/OWI/TM-36/14, April 1978.
- (17) "Commercial Waste Forms, Packaging and Projections for Preconceptual Repository Design Studies," Science Applications, Inc., Y/OWI/TM-36/2, April 1978.
- (18) "Specific Truckload Commodity Rates" and "Mileage Truckload Commodity Rates," Tri-State Motor Transit Company, June 1978.

- (19) "Logistics Models for the Transportation of Radioactive Waste and Spent Fuel," Union Carbide Corporation, ORNL/TM-6192, March 1978.
- (20) "Status Report Radioactive Waste Packaging Study," Rockwell International Corporation, Atomics International Division, RHO-CD-237, December 1977.
- (21) "Current Status and Future Considerations for a Transportation System for Spent Fuel and Radioactive Waste," Allied General Nuclear Services, Y/OWI/SUB-77/42513, February 1978.
- (22) "Hot Experimental Facility Interim Design Report," Oak Ridge National Laboratory and Bechtel National, Inc., ORNL/AFRP-78/6, 1978.

Appendix B

AWAY-FROM-REACTOR STORAGE FACILITIES

TABLE OF CONTENTS

	Page
B.1 Approach	B-2
B.1.1 Preconceptual Design Approach	B-2
B.1.2 Cost Estimating Approach	B-2
B.2 Preconceptual Design Description	B-5
B.2.1 General Description	B-5
B.2.2 Cask-Handling System	B-5
B.2.3 Cask Flush and Cooldown	B-5
B.2.4 Fuel Unloading-Loading Pools	B-6
B.2.5 Cask Decontamination Area	B-6
B.2.6 Fuel Storage Pools	B-6
B.2.7 Pool Cooling System	B-6
B.2.8 Radwaste Treatment	B-11
B.3 AFR Operational Considerations	B-12
B.3.1 Carrier Receiving, Offloading, and Cask Cooldown	B-13
B.3.2 Cask Unloading and Return Preparation	B-14
B.3.3 Basket Moving	B-14
B.3.4 Cask Loading and Shipping	B-14
B.3.5 Pool Maintenance	B-15
B.3.6 Waste Processing	B-15
B.3.7 Accountability	B-15
B.3.8 General Duties	B-15
B.3.9 Overall Personnel Requirements	B-15
B.4 AFR Cost Estimates	B-17
B.4.1 AFR Capital Cost Estimate	B-17
B.4.2 AFR Operating Cost Estimate	B-17
B.4.3 AFR Decommissioning Cost Estimate	B-17
References for Appendix B	B-18

Appendix B

AWAY-FROM-REACTOR STORAGE FACILITIES

This appendix contains the preconceptual design description and cost estimates for an away-from-reactor storage facility (AFR) required as part of the spent fuel disposal cases studied. The AFR is an interim surface storage facility in which spent fuel elements are held in underwater storage basins. The storage period begins when the fuel elements are removed from the reactor storage basins and ends when the fuel elements are transported to an encapsulation facility to be prepared for placement in the geologic repository. An AFR is required only in the three spent fuel cases, since the Reference Case spent fuel is reprocessed immediately after removal from the reactor storage basin.

The description and cost estimates given in this appendix, though based on preconceptual designs, are intended to be sufficiently definitive to provide a basis for the assessments and comparisons in Sections 4 through 10 of this report. In preparing this appendix, it was necessary to make a number of fairly arbitrary design and cost assumptions and design selections. In a few instances, where the design selections or assumptions were thought to have a significant impact on the outcome of the study, study variations (or sensitivity studies) were made. For example, time factor variations will influence the length of time spent fuel will be in storage at an AFR, although it will not influence conceptual design considerations. These time factor, and other, variations are identified and analyzed in Section 10.

The bases and assumptions regarding the AFRs are presented in Appendix A.

B.1 APPROACH

B.1.1 Preconceptual Design Approach

The scope of the Spent Fuel Disposal Study (SFDS) was limited to a preconceptual design effort, since the primary purpose of the study is to compare relative merits of several spent fuel disposal alternatives. The design effort involved studying resource materials in order to develop broad descriptions of processes and facilities in an AFR. Information from both government and industry sources were studied, including:

- Existing facilities
- Detailed design projects
- Conceptual design studies
- Preliminary engineering studies
- Preliminary design criteria
- Technical reports
- Miscellaneous data

The descriptions and definitions are presented in narrative form and supplemented by block diagrams and facility layouts, as required. The material, as presented in the following portions of this appendix, meets these objectives:

- Provides a basis, in terms of facilities configuration and process technology definition, for assessing and comparing the cases
- Furnishes information for developing order-of-magnitude capital and operating cost estimates
- Investigates the state of the art of equipment and process technology in order to assess the technical feasibility and to provide a basis for planning continuing work

With these objectives in mind, the design effort was limited to identifying the basic parameters, defining the process technology, outlining the principal features of an AFR, and identifying the major equipment and material flow. Design optimization was not within this scope of work. The basic goal was to arrive at a facility size and to incorporate the major process components and systems.

B.1.2 Cost Estimating Approach

The following cost estimates were prepared in connection with the AFR:

- Capital cost estimates
- Operating cost estimates
- Decommissioning cost estimates

The approach taken in estimating these components of AFR cost is discussed below; the actual cost estimates for these three items are given in Subsection B.4. Transportation cost estimates for the spent fuel disposal system are presented and discussed in Appendix E.

All estimated costs are in January 1979 dollars with no allowance for future escalation. Since the estimates are based upon preconceptual design data, they are order-of-magnitude estimates with expected accuracies of +75 percent, -25 percent.

B.1.2.1 AFR Capital Cost Estimate. The capital cost estimate for the AFR includes all costs associated with the design, procurement, and construction of the permanent facilities described in Subsection B.2 plus an allowance for the owner's costs during construction.

The AFR capital cost estimate is based upon the published work of the Savannah River Laboratory (SRL),⁽¹⁾ and is given by the following formula:

$$C = 85 + 15 R + 13 S$$

where,

C = the capital cost in 1978 millions of dollars

R = the receiving/shipping rate in multiples of
1,000 MTHM/yr

S = the storage capacity in multiples of 1,000 MTHM

Capital cost C is used here as Bechtel's "basic" capital cost and includes a 25 percent allowance for contingency, which is consistent with other SFDS estimates.

The SRL formula was then modified as follows:

- Land costs of \$5 million (included in SRL's formula) were deleted since they are part of the owner's costs
- An additional \$16 million was added to capitalize all of the baskets required for the SFDS facility. (SRL capitalized only \$11 million worth of baskets.)
- Escalation of 10 percent per annum was added to bring the costs to January 1979 cost levels
- The formula costs were then increased by 30 percent to incorporate all of the owner's costs. The owner's costs normally do not form a part of the architect-engineer's or contractor's scope of work, but are additional costs incurred by the owner within his own organization. The owner's costs include:
 - Permits and licenses
 - Land acquisition
 - Owner's engineering, operating, and accounting activities
 - Interest during construction
 - Insurance
 - Startup
 - Operating procedures and manuals
 - Equipment spares not permanently installed in the plant
 - Other spare parts and warehouse stock
 - Initial inventories, e.g., furniture, tools, and maintenance equipment
 - Catalysts and chemicals
 - Process royalties

These modifications result in the following formula for estimated AFR capital cost:

$$C' = 137 + 21 R + 18 S,$$

where C' is Bechtel's estimated cost in millions of January 1979 dollars.

Bechtel tested this modified formula by direct comparison with several existing Bechtel AFR capital cost estimates. The test involved escalating the Bechtel estimates E_x to January 1979 levels and comparing them with C' estimates determined from the above equation. The comparison in each case was made using the following equation:

$$\text{Variance \%} = \frac{E_x - C'}{C'} \times 100$$

The results were found to be within reasonable agreement, and as a result, the modified SRL formula was concluded to be an appropriate means of determining the required capital cost for an SFDS AFR.

The actual capital cost estimate developed for the SFDS AFR is given in Subsection B.4.1.

B.1.2.2 AFR Operating Cost Estimates. Annual operating cost estimates for an AFR were based on:

- Estimated labor costs
- An allowance for operating supplies and maintenance materials
- An allowance for utility costs
- Estimated waste container costs

The total labor force at an AFR was estimated to be made up of 75 percent operating, 10 percent supervisory, 10 percent engineering/technical, and 5 percent management personnel. Bureau of Labor statistics were used to establish wage rates for the lowest classification, and each higher grade was assumed to be salaried at 20 percent above the next lower grade. Fringe benefits of 30 percent were added to give an average rate of \$24,000/yr per census member.

The total census for each facility was estimated by Bechtel based on specific operations to be performed. For a detailed discussion, see Subsection B.3, "AFR Operational Considerations."

An allowance of 1 percent per year of the original total capital investment in the facility was made for operating supplies and maintenance materials. An allowance of 0.1 percent per year of the capital cost estimate was made for utility costs. An allowance of \$300,000/yr was made for waste containers and disposal costs at shallow land burial grounds. The cost estimates are provided in Subsection B.4.2.

B.1.2.3 AFR Decommissioning Cost Estimates. AFRs could be decommissioned by any of the following alternatives: protective storage, entombment, or decontamination and dismantling.

Protective storage may be the most attractive option for AFRs because the expected contamination due to long-lived transuranic (TRU) materials in the storage pool and pool water cleanup equipment is low. However, for continuous occupancy, surveillance would have to continue until the short-lived fission products have decayed to safe levels. Under this decommissioning mode, initial cost commitments are low, but surveillance costs are high for several years. Protective storage involves removal of all spent fuel, radioactive sources, and decontaminated components, such as pool cleanup filter and ion exchange equipment prior to the start of the surveillance period.

After an AFR has been prepared for protective storage, it can be converted into the entombment alternative by sealing all openings into the facility. Entombment has two significant advantages over protective storage: it has lower security costs and lower surveillance costs. Entombed facilities cannot be used for any other purpose.

The decommissioning mode selected for the SFDS is decontamination and dismantling of the AFR as recommended by C. D. Zerby (1/6/78 letter to the Department of Energy (DOE) "Spent Fuel Repository Cost Estimates"). In general, this approach leads to the greatest initial decommissioning costs, highest total personnel exposures to radioactive materials and radiation during decommissioning, and lowest long-term surveillance requirements.

The estimate of AFR decommissioning costs was taken as a percentage of the original facility capital cost, excluding the owner's costs. This percentage was taken from the TRW Economic Study.⁽²⁾ The resultant cost includes the dismantling and removal of all AFR facilities. The percentage and the estimated cost are provided in Subsection B.4.3.

B.2 PRECONCEPTUAL DESIGN DESCRIPTION

B.2.1 General Description

An AFR is designed to receive 1 year-old spent fuel from reactors and to store it for 9 additional years before processing and/or encapsulation operations begin. Each AFR has a storage capacity of 5,000 MTHM.

Nine regionally located AFRs with a total system throughput of 5,000 MTHM/yr have been provided. The receiving and shipping facilities of each AFR are designed to handle an equal share of the total system throughput, or 555 MTHM/yr.

The AFR design is based on the Spent Fuel Handling and Storage Facility designed by SRL for the International Spent Fuel Storage Program.⁽¹⁾ The SRL design permits 1-year-old LWR fuel to be stored at a throughput of 1,000 MTHM/year.

Figure B-1 is an overall block diagram that shows the major operational steps and the functional systems in the AFR; Figure B-2 shows the facility arrangement. The principal components of the facility are described below.

B.2.2. Cask-Handling System

An AFR receives irradiated fuel assemblies in shielded, gas-filled, or water-filled casks which have been transported by rail or truck. Upon arrival at an AFR, each cask and transport vehicle (trailer or rail car) is identified, inspected for shipping damage, and monitored for radiation levels. The cask holding area accommodates a gross vehicle weight of 350,000 pounds (e.g., NL Industries 10/24 cask and carrier). The casks are handled by a 125-ton crane. Radiation monitors, remote computer terminals, and cask vehicle transporters are also provided outside the cask holding area.

The cask holding area has a surge capacity for two NL 10/24 casks.

The cask preparation area is used to (1) offload the cask or to prepare the cask for shipping off site, and (2) store tools and equipment used in receiving and shipping operations such as lifting yokes, personnel barriers, and impact limiters.

After inspection of the cask and carrier, the overhead crane removes the shipping cask from the transport vehicle to one of two cask washdown stations. Road dirt is removed and the cask is transferred to the cask flush and cooldown area.

B.2.3 Cask Flush and Cooldown

At the cask flush and cooldown area, the cask primary coolant temperature and pressure are checked, the cask is vented into the offgas system, and the primary coolant is sampled and analyzed for gross beta/gamma radioactivity. If the cask requires cooldown prior to immersion in the fuel unloading pool, cooling water is circulated through the cask. If the cask is gas-cooled, a two-stage cooldown is used. In the first stage, steam is injected; in the second stage, water is employed to reduce the temperature of the fuel. Final coolant is sampled for radioactivity. If the coolant is contaminated, flush water is circulated through a filter-deionizer system to reduce the contamination to acceptable levels for the fuel unloading pools. The flush water can also be transferred to a holding tank for later processing. The cask flush and cooldown area is designed to handle two casks simultaneously. Major equipment in the area includes washing solution tanks, rinse water tanks, pumps, coolers, a sampling system, and heat exchangers.

B.2.4 Fuel Unloading-Loading Pools

After flushing cask cavity and cooling fuel to an acceptable temperature, the cask is moved to one of two fuel unloading-loading pools.

The cask is lowered into the water of the fuel unloading-loading pool and placed on an energy-absorbing pad at the bottom of the pool. The cask lids are removed and transferred to an underwater shelf in the pool, and individual fuel assemblies are removed from the cask using a 10-ton fuel unloading crane. Provisions have been made to can grossly failed fuel assemblies.

The fuel unloading-loading pools are capable of loading or unloading two casks simultaneously (one in each pool) in the vertical position. Each pool has its own supporting equipment such as fuel storage baskets, basket supporting racks, and cask lid storage shelf, and comes equipped with fuel-handling grapples for different fuels, instruments to detect leaking fuel, a cask rinse ring, an underwater vacuum system, underwater TV, and scrap containers. The depth of water is adequate to maintain shielding during the fuel unloading operation.

The fuel assemblies are transferred to a multiple-assembly storage basket located in the unloading-loading pool. The fuel baskets are designed to handle 4 pressurized water reactor (PWR) or 9 boiling water reactor (BWR) fuel elements per basket. The baskets maintain a 12-inch center-to-center spacing for BWR assemblies. The baskets are made of stainless steel to provide partial neutron absorption.

B.2.5. Cask Decontamination Area

Empty casks from the fuel unloading-loading pools are transferred to the cask decontamination area, where the casks are flushed with deionized water until cavity decontamination is acceptable. The exterior surface of the cask is washed until contamination meets U.S. Department of Transportation (DOT) limits for transport. Two casks can be decontaminated simultaneously, and sufficient room has been provided around each cask to allow access for a ring with high-pressure nozzles. The floor has a sump drain connected to a holding tank.

B.2.6 Fuel Storage Pools

When the fuel storage basket is full, the basket is moved from the unloading pool to the storage pool using the basket-handling crane. The main function of the storage pool is to cool and shield fuel during storage. The baskets are stored in fixtures fastened to the pool floor. These fixtures are designed to maintain the spacing of the baskets even during extreme disturbances by natural phenomena.

Each of the 10 storage pools has a capacity of 500 MTHM of spent fuel. The water in the pools at each AFR is 48 feet deep which is sufficient to provide 13 feet of shielding water over fuel in baskets being moved into storage position.

B.2.7 Pool Cooling System

The pool cooling system removes decay heat from the stored nuclear fuel. Some heat is also rejected by evaporation. If loss of cooling occurs in a single storage pool filled with spent fuel, the large heat sink will not begin to boil for 4 days. In an emergency, makeup water can be supplied from the plant water or emergency water supply by pumps using emergency power if necessary.

Figure B-1.
AFR
Overall Block Diagram
Cases 1, 2, and 3

1 2 3 4

5 6

SCALING RATES

SIZE E

ENTRANCE

PLAN

AREA DESCRIPTION

- 1 SPENT FUEL STORAGE POOL 500 MTU)
- 2 FUEL UNLOADING POOL
- 3 TEST AND DECONTAMINATION PIT
- 4 CASK FLUSH AND COOLDOWN PIT
- 5 CASK PREPARATION AREA
- 6 CASK HOLDING AREA
- 7 RR CAR AND TRUCK WASH STATION
- 8 FUEL TRANSFER CANAL
- 9 POOL WATER TREATMENT AND COOLING FACILITY
- 10 RADWASTE PROCESS AND SOLIDIFICATION FACILITY
- 11 RADWASTE TRANSPORT STATION
- 12 HVAC SUPPLY AND EXHAUST FACILITY
- 13 EMERGENCY UTILITY ROOM
- 14 PERSONNEL AND SECURITY AREA
- 15 ACCESS AISLE
- 16 AIR LOCK
- 17 TRANSPORTATION EQUIPMENT LAYDOWN AREA
- 18 CASK HANDLING CRANE
- 19 CASK UNLOADING CRANE
- 20 FUEL STORAGE POOL CRANE
- 21 CASK WASHDOWN

GRAPHIC SCALE

Figure B-2.
AFR
Building Arrangement
Cases 1, 2, and 3

1 2 3 4

5 6

B.2.8 Radwaste Treatment

Solid, liquid and gaseous radwastes are expected to be generated in this facility. Collection and treatment of wastes reduce the quantity of radioactive materials discharged from the facility to a level that is as low as reasonably achievable (ALARA).

Liquid radwastes are generated from cask cooldown operation, pool water treatment, ion exchange resin regeneration, decontamination operations, and offgas scrubbing. The combined liquid radwastes are concentrated, the overhead condensate is recycled, and the bottoms are fixed in cement for disposal. Excess condensate that cannot be recycled is vaporized, superheated, filtered, and released from the plant stack.

Solid radwastes incidental to the operation of this facility are shredded, compacted, and then packaged for disposal. Provisions have been made to subdivide noncompactible solid waste before it is packaged for disposal.

Offgases are generated in this facility from cask venting, liquid radwaste handling and treatment, and decontamination operations. All offgases are collectively scrubbed, heated, filtered, passed through an adsorption bed to remove small amounts of iodine, filtered, and then released to the atmosphere.

For the most part, all packaged radwastes from this facility are considered to be low-level, non-TRU waste, according to industrial practice, and are shipped to shallow land burial. It is recognized that limited amounts of low-level, TRU waste and intermediate-level, non-TRU waste are generated at the AFR. However, because of the relatively low quantities expected, these wastes are not quantified or considered further in the study.

B.3 AFR OPERATIONAL CONSIDERATIONS

Operational considerations were applied to develop staffing requirements and costs for the economic comparison of cases. This subsection briefly discusses the sequence of operations. The principal operations at AFRs are receiving, storage and maintenance, and shipping. The functions of the receiving operation are to:

- Receive casks containing spent fuel, cool down cask internals, and determine whether significant quantities of radioactive materials are leaching into the internal cask cooling water (before the cask is opened)
- Transfer the casks into 1 of 2 fuel unloading-loading pools
- Unload the spent fuel into a storage basket, reclose the cask and transfer it to a decontamination area, purge the pool water from the casks, and prepare casks for return to the transportation fleet
- Transfer fuel basket to 1 of 10 storage pools

The functions of the storage and maintenance operation are:

- Clean up and maintain the pool water at the proper temperature
- Process and handle locally generated wastes (including general area cleanup)
- Account for all fuel materials

The functions of the shipping operation are to:

- Transfer the fuel basket from 1 of 10 storage pools to the fuel unloading-loading pool
- Transfer the clean empty cask to the fuel unloading-loading pool
- Load the fuel from the basket into the cask
- Transfer the cask to the decontamination area and prepare for shipping
- Ship the cask

As indicated in Subsection B.2, each AFR should be capable of storing 5,000 MTHM equivalent of spent fuel, and AFRs should be widely distributed geographically to best meet local nuclear reactor fuel storage needs. For the steady state balanced system assumed for the study, each AFR is assumed to handle an equal share of the 5,000 MTHM total fuel moving in from reactors or leaving for the processing/encapsulations (P/E) facility each year. The inbound fuel is 1 year old and the outbound fuel is 10 years old.

Table B-1 shows material flow through an AFR.

Table B-1
ANNUAL MATERIAL FLOW AT EACH AFR

Material	Annual Number of Shipments	Total Units of Waste	Additional Packaging Requirements
Incoming			
Spent fuel	86 rail	688	None
from PWRs	63 truck	63	None
Spent fuel	52 rail	988	None
from BWRs	75 truck	150	None
	<u>276 total</u>	<u>1,889 assem.</u>	
Internal transfers			
PWRs	188 bask. in/out	751 assem.	None
BWRs	126 bask. in/out	1,138 assem.	None
Outgoing			
Spent fuel			
PWRs	54 rail	751 assem.	None
BWRs	33 rail	1,138 assem.	None
Non-TRU, LLW	14 truck	578 drums *	55-gal drums

* Assumes 4:1 volume reduction on 80% waste

B.3.1 Carrier Receiving, Offloading, and Cask Cooldown.

Receiving operations begin with the identification, inspection, and radiation survey of each cask and transport vehicle. The transport vehicle and cask are moved to the offloading area, and the cask is lifted from the carrier and transferred to the cask washdown and cooling area.

After external cask washdown or decontamination, the cask primary coolant temperature and pressure are checked. Normally, the cask is vented into the facility offgas system, and the primary coolant is sampled and analyzed for gross beta/gamma radioactivity. If the cavity is water-filled, coolant is purged to the waste processing system; if the cask is gas-filled, it is vented and then sent to cooldown.

In cask cooldown, the operators initiate each of the three-stage cooldown operations: the first stage uses high-temperature and high-pressure water or steam, the second stage uses low-pressure hot water, and the third stage uses water cooling. The operators also monitor the buildup of radioactivity in the coolant stream leaving the cask, and service the cleanup system as necessary to maintain acceptable radiation levels in the vicinity of the cleanup system and acceptable levels of dissolved and suspended activity in the pool water. Most of the bolts holding the outer lid of the cask in place are removed, and the cask is moved into one of the cask unloading-loading pools.

Because hoisting speed limits on the cask-handling cranes are set quite low, moving the heavy casks, in particular the rail casks, consumes a great deal of operating time.

The operating crew requirements to receive a single cask are as follows:

Receiving a truck cask	2 persons for 4.5 hours
Receiving a rail cask	3 persons for 6 hours

To establish total operating staffing and meet material flow requirements, time was allocated for each operator on each shift.

B.3.2 Cask Unloading and Return Preparation

Once the cask is lowered into an unloading pool, the outer and inner lids are removed. A fuel storage basket is brought into the cask unloading pool and lowered onto a holding fixture. The operator uses a long-handled grapple and the fuel-handling crane to remove one assembly at a time and transfer it from the cask into the basket. Accountability records are entered to show assembly number, basket number, and grid location in the basket for each fuel assembly.

When the cask is unloaded, and before it is removed, the basket is moved from the cask unloading pool to the storage pool in order to eliminate the danger of fuel rupture in case a cask is dropped. The cask lids are reinstalled, and the cask is hoisted from the pool and transferred to the test and decontamination area for preparation before it is returned to the reactors.

The operating crew requirements for unloading are as follows:

Unloading a truck cask	1 person for 4.5 hours
Decontaminating/shipping by truck cask	2 persons for 4 hours
Unloading a rail cask	1 person for 9 hours
Decontaminating/shipping by rail cask	3 persons for 8 hours

B.3.3 Basket Moving

When a storage basket is filled with 9 BWR or 4 PWR assemblies, the operator moves the loaded basket from the cask unloading pool to 1 of the 10 storage pools of 500 MTHM capacity. Inventory records are adjusted to show the pool number and grid location of each basket.

The operating crew requirements for moving baskets are as follows:

Placing a single fuel basket into the storage pool	2 persons for 2 hours
Temporarily removing a fuel basket and placing it in the storage pool	1 person for 1/2 hour

B.3.4 Cask Loading and Shipping

After 9 years of storage, the fuel is reloaded into casks for shipment to a P/E facility. Since segregation of fuel by reactor source is not necessary, offsite shipment can be programmed on a first-in/first-out basis by fuel type (PWR or BWR). Some additional operating efficiency should be possible during the loading of outbound fuel into casks, as compared with the unloading of inbound fuel, because in the former, less time needs to be devoted to monitoring the fuel condition; significantly leaking fuel will have been identified and canned during storage. The cask pool should be less contaminated when handling fuel bound for offsite shipment; hence the length of time it takes to decontaminate the casks before shipment will be reduced. The slight improvement in efficiency due to more efficient loadout has not been taken into account here.

The operating crew requirements for reloading casks are as follows:

Transferring the fuel basket to the cask unloading-loading pool	2 persons for 2 hours
Loading the truck cask	1 person for 4.5 hours
Decontaminating/shipping by truck cask	2 persons for 4 hours
Loading the rail cask	1 person for 4 hours
Decontaminating/shipping by rail cask	3 persons for 8 hours

B.3.5 Pool Maintenance

Cleanup and cooling of pool water involves some manual operation, such as changing the filter media or precoat, starting the resin regeneration operation, sampling and analyzing the pool water, and routing water flows from less contaminated to more contaminated pools, as much as practical.

The operating crew requirements for pool cleanup and cooling are as follows:

Cleaning and cooling the pool	3 persons for a total 9 hr/day (about 3 hr/shift)
----------------------------------	--

B.3.6 Waste Processing

Solid wastes are generated from normal cleanup operations and maintenance and decontamination of failed equipment. Liquid wastes originate from cask cooldown, flushing, and decontamination operations. Since the AFRs are considered to be primarily storage facilities, elaborate solid waste processing equipment has not been incorporated. Details of waste processing have been described in Subsection B.2.8.

The operating crew requirements for waste processing are as follows:

Preparing and packaging wastes	2 persons for 3 hr/day
--------------------------------	------------------------

B.3.7 Accountability

The unloading-loading and storage pools are item control areas which require continuing inventory control to maintain the identification and location of each fuel assembly. If inventories are kept current, periodic audits should not be necessary. Time has been allotted only to update the accountability records each time a fuel transfer is made.

The operating crew requirements for accountability are as follows:

Maintaining accountability records	2 persons for 1 hr/day equivalent
---------------------------------------	--------------------------------------

B.3.8 General Duties

Additional operating time is allowed for preparation of work areas, routine monitoring of personnel and equipment, movement in and out of potentially contaminated areas, personnel training, etc. An extra 20 percent of the labor force has been assumed to complete general duties.

B.3.9 Overall Personnel Requirements

One truck or one rail cask arrives each day, and two rail casks are loaded and shipped. The oper-

ating crews on each shift were determined by time allocation charts to meet the throughput rates. Total direct operating personnel requirements for an AFR are shown in Table B-2; total support and overhead personnel requirements for an AFR are shown in Table B-3.

Table B-2
TOTAL DIRECT OPERATING STAFF
FOR AN AFR

Personnel	Number Required
Shift workers (8 x 4 shifts)	32
Day workers (4)	4
Supervision (1 x 4) (1 on days)	4 1
Total	<hr/> 41

Table B-3
TOTAL SUPPORT AND OVERHEAD PERSONNEL
FOR AN AFR

Personnel	Number Required
Maintenance (0.5 x oper. pers.)	20 (some shift, some days)
Utilities (2/shift)	8 (rotating shifts)
Analytical (0.1 x oper. pers.)	4 (day shift)
Health and safety (0.1 x oper. pers.)	4 (rotating shifts)
Plant overhead (0.3 x oper. pers.)	12 (some shift, most days)
Total	<hr/> 48

Total overall personnel requirements for an AFR are shown in Table B-4.

Table B-4
TOTAL OVERALL PERSONNEL
REQUIREMENTS FOR AN AFR

Personnel	Number Required
Direct operating	41
Support and overhead	48
Total	<hr/> 89

In summary, the total overall operating complement for each AFR with receiving/shipping capacity of 555 MTHM equivalent per year and storage capacity of 5,000 MTHM has been estimated at 89 persons.

Subsection B.4.2 gives the estimated operating costs.

B.4 AFR COST ESTIMATES

This subsection presents the capital, operating, and decommissioning cost estimates for the AFR facilities. These estimates are based on the approach discussed in Subsection B.1 and the facility descriptions discussed in Subsection B.2.3. Each AFR has an annual throughput of 555 MTHM and a total storage capacity of 5,000 MTHM. A total of 9 independently located AFRs are required for the SFDS project for each of Cases 1, 2, and 3. The AFRs are identical in scope for each of these three cases.

B.4.1 AFR Capital Cost Estimate

As stated in Subsection B.1.2.1, the basic AFR estimate is developed using a modification of the SRL formula.⁽¹⁾

$$C' = 137 + 21 R + 18 S$$
$$C' = \$240 \text{ million}$$

This capital cost for each of 9 AFRs, each with a receiving/shipping rate of 555 MTHM/yr and a storage capacity of 5,000 MTHM, is in January 1979 dollars.

B.4.2 AFR Operating Cost Estimate

As stated in Subsection B.1.2.2, the estimate of the operating costs for an SFDS AFR is based on the costs of operating labor, the allowances for supplies and utility services, and the allowance for waste containers and disposal services at shallow land burial grounds. The estimate is given in Table B-5 and is \$5 million/yr for each AFR.

Table B-5
OPERATING COST ESTIMATE FOR AN AFR
(555 MTHM/yr)

Item	Cost \$ million/yr
Personnel	
89 persons x \$24,000/yr	2.1
Operating/maintenance supplies	
0.01 x \$240 million	2.4
Utilities	
0.001 x \$240 million	0.2
Canister and disposal allowance	0.3
Total	<hr/> 5.0

B.4.3 AFR Decommissioning Cost Estimate

As stated in Subsection B.1.2.3, the AFR decommissioning cost estimate was derived as a fraction of the AFR capital cost estimate. The fraction used was 10 percent of the total AFR capital costs, excluding the owner's costs, as suggested by the TRW study. ⁽²⁾ Thus, the decommissioning cost for each AFR (555 MTHM/yr throughput) is \$19 million.

REFERENCES FOR APPENDIX B

- (1) "Spent Fuel Handling and Storage Facility for the International Spent Fuel Storage Program," Savannah River Laboratory, DPSTD-ISDS-78-7(70), April 1978.
- (2) "NWTS Economic Study Part 2, Spent Fuel Management Pricing," prepared by TRW for UCC-OWI, March 1978.

Appendix C

PROCESSING/ENCAPSULATION FACILITIES

TABLE OF CONTENTS

		Page
C.1	Approach	C-2
	C.1.1 Preconceptual Design Approach	C-2
	C.1.2 Cost Estimating Approach	C-2
C.2	Reference Case	C-6
	C.2.1 Process Description	C-6
	C.2.2 Waste/Product Packages	C-22
	C.2.3 Facility Description	C-27
	C.2.4 Operational Considerations	C-33
	C.2.5 Cost Estimates	C-47
C.3	Case 1	C-50
	C.3.1 Process Description	C-50
	C.3.2 Spent Fuel/Waste Packaging	C-56
	C.3.3 Facility Description	C-60
	C.3.4 Operational Considerations	C-62
	C.3.5 Cost Estimates	C-74
C.4	Case 2	C-78
	C.4.1 Process Description	C-78
	C.4.2 Spent Fuel/Waste Packaging	C-85
	C.4.3 Facility Description	C-87
	C.4.4 Operational Considerations	C-88
	C.4.5 Cost Estimates	C-99
C.5	Case 3	C-102
	C.5.1 Process Description	C-102
	C.5.2 Spent Fuel/Waste Packaging	C-107
	C.5.3 Facility Description	C-109
	C.5.4 Operational Considerations	C-110
	C.5.5 Cost Estimates	C-122
	References for Appendix C	C-126

Appendix C

PROCESSING/ENCAPSULATION FACILITIES

This appendix contains the preconceptual design descriptions and cost estimates for the processing/encapsulation (P/E) facilities required in the various cases studied. The P/E facility varies from a nuclear fuel reprocessing plant in the Reference Case to a facility for the simple encapsulation of spent fuel elements in Case 1. In all cases, it is the step in the fuel cycle in which the spent fuel and/or various nuclear waste forms are prepared for ultimate disposal in geologic repositories, through appropriate processing and encapsulation operations.

The descriptions and cost estimates given in this appendix, though preconceptual in nature, are intended to be sufficiently definitive to provide a basis for the assessments and comparisons in Sections 4 through 10 of this report. In preparing this appendix, it was necessary to make a number of fairly arbitrary design and cost assumptions and design selections. In a few cases where the design assumptions or selections might have a significant impact on the outcome of the study, study variations (or sensitivity studies) were made. These variations are identified and analyzed in Section 10.

The bases and assumptions regarding the P/E facilities are presented in Appendix A.

C.1 APPROACH

C.1.1 Preconceptual Design Approach

In the Spent Fuel Disposal Study (SFDS), the general design approach and objectives for the P/E facilities are the same as the approach and objectives for the away-from-reactor storage facilities (AFRs). (See Subsection B.1.1.) Design descriptions of the P/E facilities for the Reference Case, Case 1, Case 2, and Case 3 are provided in Subsections C.2, C.3, C.4, and C.5, respectively, of this appendix. The Reference Case P/E facility draws heavily on the technology of the Barnwell Nuclear Fuel Plant (BNFP), while the Case 1 P/E facility is based upon Rockwell Hanford's conceptual design of a spent fuel encapsulation facility. The Case 2 and Case 3 designs draw from both these sources in developing their particular required capabilities.

The P/E facilities for all cases are sized for a handling, processing, and encapsulation capacity of 1,500 MTHM/yr. This value was selected because (1) it is the basic capacity of the BNFP; (2) it represents a reasonable plant capacity from the standpoint of financial investment and plant operation; (3) it is reasonable from the geographical standpoint since it serves approximately 50 nuclear power plants; and (4) for purposes of comparison, it is helpful to have the capacity of P/E facilities the same for all cases.

The P/E facilities for the four cases described below are designed to process the spent fuel to the degree required in each case, and to encapsulate the fuel and the generated waste forms for disposal in the geologic repository. The P/E facility will be located at an independent site and will be a self-sufficient plant. In the Reference Case, the plant is capable of processing 1-year-old spent fuel from light water reactors (LWRs). (A total of 25 percent of the fuel assemblies are assumed to be mixed oxide (MOX) and the balance are assumed to be uranium oxide.) In all other cases, the plant is capable of encapsulating 10-year-old uranium oxide, LWR spent fuel previously stored in an AFR.

C.1.2 Cost Estimating Approach

The following cost estimates were prepared in connection with the P/E facilities:

- Capital cost estimates
- Operating cost estimates
- Decommissioning cost estimates

The approach taken in estimating these components of P/E facility cost is discussed below; the actual estimates for these items are presented and discussed in Subsections C.2.5, C.3.5, C.4.5, and C.5.5. Transportation cost estimates for the spent fuel disposal system are presented and discussed in Appendix E.

All estimated costs are in January 1979 dollars, with no allowance for future escalation. Since the estimates are based upon preconceptual design data, they are order-of-magnitude estimates, with expected accuracies of + 75 percent, - 25 percent.

C.1.2.1 Capital Cost Estimates. The capital cost estimates for the P/E facilities have been separately developed for each of the four cases. These estimates include all costs associated with the design, procurement, and construction of the permanent facilities described in Subsections C.2.3, C.3.3, C.4.3 and C.5.3, together with allowances for:

- Contingency at 20 percent of the total field cost
- Architect-engineer services at 20 percent of sum of total field cost and contingency. Architect-engineer services include:
 - Management
 - Design
 - Engineering
 - Procurement
- The owner's costs at 30 percent of the sum of the field, contingency, and architect-engineer services costs. These costs normally are not part of the architect-engineer's or contractor's scope of work, but are additional costs incurred by the owner within his own organization. They include:
 - Land acquisition
 - Interest during construction
 - Equipment spares not permanently installed in the plant
 - Permits and licenses
 - Startup
 - Catalysts and chemicals
 - Initial inventories
 - Owner's engineering, operating, and accounting activities
 - Process royalties
 - Spare parts and warehouse stock
 - Operating procedures and manuals
 - Insurance

The P/E facility capital cost estimates for each case were prepared for each of the component process areas or subfacilities. The subfacility cost estimates are based on published work by others or on prior Bechtel estimating experience. The subfacility estimates were modified to meet the current SFDS design requirements. The sum of the subfacility estimates gives the total capital cost for a P/E facility capable of handling 1,500 MTHM/yr. Each case requires the equivalent of 3-1/3 P/E facilities to handle the total study capacity of 5,000 MTHM/yr.

The subfacility descriptions and their corresponding capital cost estimates are presented and discussed in the following subsections:

● Reference Case P/E facility capital cost estimate	C.2.5.1
● Case 1 P/E facility capital cost estimate	C.3.5.1
● Case 2 P/E facility capital cost estimate	C.4.5.1
● Case 3 P/E facility capital cost estimate	C.5.5.1

C.1.2.2 Operating Cost Estimates. Annual operating cost estimates for the P/E facilities were based on:

- Estimated labor costs
- An allowance for operating supplies and maintenance materials
- An allowance for utility costs
- Estimated waste container costs

The total labor force at a P/E facility was estimated to be made up of 75 percent operating, 10 percent supervisory, 10 percent engineering/technical and 5 percent management personnel. Bureau of Labor statistics were used to establish wage rates for the lowest classification, and each grade was assumed to be salaried at 20 percent above the next lower grade. Fringe benefits of 30 percent were added to give an average rate of \$24,000/yr per census member.

The total census for each facility was estimated by Bechtel on the basis of specific operations to be performed. For a detailed discussion, see "Operational Considerations" in Subsections C.2.4, C.3.4, C.4.4, and C.5.4.

An allowance of 1 percent per year of the original total capital investment in the facility was made for operating supplies and maintenance materials. An allowance of 0.1 percent per year of the capital cost estimate was made for utility costs.

Waste containers are a major contributor to annual operating costs. To allow for packaging costs for spent fuel, high level waste (HLW) glass hulls, and solid intermediate level waste (ILW), an average cost of \$5,000/canister was assumed. The spent fuel canisters are approximately 16 inches in diameter and 16 feet long, HLW glass canisters are 13 inches in diameter and 10 feet long, and hulls and ILW canisters are 20 inches in diameter and 10 feet long. Although the required quantities of each type of canister differ, the average cost assumed is considered to be appropriate and realistic for the purposes of the study.

C.1.2.3 Decommissioning Cost Estimates. The Reference Case P/E decommissioning cost estimate is based upon the BNFP facility. Battelle Pacific Northwest Laboratories (BPNL) has estimated the cost of dismantling the BNFP facility at \$58 million in 1975 dollars (see NUREG-0278, 10/77). This figure is based on an estimated BNFP capital cost of \$430 million.

Allied General Nuclear Services (AGNS) has estimated the capital cost of the BNFP facility to be \$480 million in mid-1976 dollars. This includes decommissioning of the uranium conversion (UF₆) facility, with an estimated capital cost of \$50 million which was not considered in NUREG-0278.

The relative decommissioning cost factor was developed by adjusting the NUREG dollars to mid-1976 using a 10 percent escalation allowance to place costs on a comparable basis. A \$50 million downward adjustment was applied to AGNS's capital cost figures to remove the impact of the uranium hexafluoride facility. Thus, decommissioning costs were related to capital costs by the following ratio:

$$\text{Decommissioning costs} = \frac{\$58 \text{ million} \times 1.10}{\$480 \text{ million} - \$50 \text{ million}} = 15\% \text{ of the capital costs for the Ref. Case}$$

Although the complexity of P/E facilities may differ significantly from one case to another, the cost of decommissioning the complex chemical/mechanical facility in Case 3 is expected to be comparable to the cost of decommissioning in the Reference Case facility, and thus for Case 3, the value 15 percent, derived above, was employed.

Decommissioning of the P/E facilities in Cases 1 and 2 is expected to be significantly less complex because processing operations in these facilities are primarily mechanical operations. The reference used for establishing decommissioning costs in Cases 1 and 2 was TRW's, "National Waste Terminal Storage Economic Study, Part 2, Spent Fuel Management Pricing," published in March 1978. The decommissioning cost factor for Cases 1 and 2 was taken at 10 percent of the total capital costs, less the owner's costs, as recommended in the TRW study.

The decommissioning cost estimates for each case are summarized and discussed in Subsections C.2.5.3, C.3.5.3, C.4.5.3, and C.5.5.3.

C.2 REFERENCE CASE

The P/E facility for the Reference Case is a fuel reprocessing plant designed to separate the uranium and plutonium in the light water reactor (LWR) spent fuel for recycle and to fix and encapsulate all of the waste from the facility for disposal in a geologic repository.

C.2.1 Process Description

The Reference Case process is described in Figure C-1. The reprocessing technology for the BNFP⁽¹⁾ has been chosen as the reference process, with the missing back-end process components added and the gaseous effluent control upgraded. The upgrading is not to retrofit the BNFP plant, but to incorporate the required facilities into the basic SFDS design. Subsection C.2.1.1 describes the BNFP process and identifies the process components to be added; Subsection C.2.1.2 describes the modifications to the BNFP design.

C.2.1.1 BNFP Process. Processes employed in the existing BNFP are discussed below.

Fuel unloading. The spent fuel in the cask, when received in the plant, is subjected to cask washdown (cask exterior wash) and cask cooldown operations. During cask cooldown, the gas in the cask is vented and then purged with plant air into the vessel offgas (VOG) system. Coolant at a controlled temperature and rate is circulated through the cask to cool the fuel. After cooldown, the spent fuel assemblies are unloaded, and the cask is decontaminated and shipped out for reload.

Fuel chopping, dissolution, and feed preparation. The fuel assemblies are chopped in the fuel chopper. The chopped fuel and the end fittings are received in a basket within a dissolver vessel, and the fuel is leached in the dissolver by a nitric acid solution. After leaching, the hulls and the hardware are rinsed, first with the dilute nitric acid and then with water, and finally removed in the basket to be dumped into a canister. Sand can be added to the canister in case of fire.

The solution from the dissolver is clarified to remove the solid particles, accounted (volume, density, and plutonium and uranium concentrations measured precisely), and prepared (chemically treated to adjust the valency of plutonium and uranium in the solution) for separation by solvent extraction.

Solvent extraction. A solvent extraction is a separation or purification process accomplished by selectively extracting (transferring) one or more dissolved species from one liquid into another. The two liquids are immiscible and form two layers, the aqueous layer and organic (30 percent tributyl phosphate, TBP, in purified kerosene) layer. The extraction process is normally carried out in a pulsed column, but other equipment can be used. The liquid layer that extracts the product from the feed is known as the extract. The liquid feed, when it has emerged from the extraction process, is known as the raffinate.

Solvent extraction is normally carried out in a multiple of cycles. In the cycle, the dissolved species in the aqueous feed solution are selectively transferred into an organic phase in the first stage, and then transferred back into a fresh aqueous phase in the second stage. Concentration adjustment by removing part of the aqueous solvent is normally required after the second stage and before the next cycle.

Co-decontamination. The first solvent extraction cycle is the co-decontamination cycle. Plutonium and uranium are extracted from the aqueous feed, and most of the fission products remain in the aqueous raffinate, which constitutes the main source of HLW. In the BNFP, this stage is carried out in a centrifugal contractor. The extract is scrubbed in a column to remove some additional residual fission products.

1 1 1 1

1 1

Partitioning. The scrubbed organic extract is treated in the partition cycle to separate the uranium from the plutonium. In the BNFP, this is done by contacting the organic phase extract from the codecontamination cycle with an aqueous solution in an electrocell.⁽²⁾ Plutonium is reduced from the tetravalent to the trivalent state at the cathode (sieve trays) and leaves with the aqueous bottom stream. Uranium, on the other hand, is oxidized to the hexavalent state at the anode and remains in the organic phase which emerges from the top of the electrocell.

Uranium purification. The uranium in the organic phase is stripped (back-extracted) into an aqueous phase. A second uranium purification solvent extraction cycle follows the first one. The aqueous uranium product from the second uranium cycle is concentrated and passed over a silica gel bed for final removal of zirconium and niobium.

Uranium trioxide to uranium hexafluoride conversion. The aqueous concentrate containing uranyl nitrate hexahydrate (UNH) is converted first into uranium trioxide, next into uranium dioxide, then into uranium tetrafluoride, and finally into uranium hexafluoride in a conversion facility. After purification, the uranium product is bottled into cylinders for shipment. The description of the cylinder is given in Subsection C.2.2.6.

Plutonium oxide production. The plutonium in the aqueous phase from the electrocell is purified in two additional solvent extraction cycles. The aqueous plutonium nitrate is concentrated and stored in the plant. Although the BNFP has a plutonium nitrate loading facility for shipping plutonium nitrate solution, shipping of this solution is no longer permissible.

A design for converting plutonium nitrate into plutonium oxide (PuO_2) has been completed.⁽³⁾ The conversion process first produces plutonium oxalate precipitate, which is then filtered, dried, and calcined to produce plutonium oxide. The oxide is then blended, sized, and packaged for shipment in the plutonium shipping package described in Subsection C.2.2.5.

Solvent treatment. A solvent treatment facility treats the used solvent from the solvent extraction process for reuse. The treatment process is a sodium carbonate wash, filtration, weak acid wash, and a final sodium carbonate wash. A recently developed solvent treatment process using macroreticular anion resin could be used instead of the weak acid wash step.

A burner is provided for the disposal of degenerated solvent. Offgases from the burner are cooled, scrubbed in a quench pot, and further processed in the plant offgas system.

Liquid radwaste. Liquid wastes are divided into three classifications: high-level liquid waste (HLLW), with an activity $> 1 \text{ Ci/l}$; low-level liquid waste (LLLW), with an activity $< 1 \text{ mCi/l}$; and intermediate-level liquid waste (ILLW), with an activity level between those of HLLW and LLLW.

The LLLW consists of:

- The aqueous raffinates and other miscellaneous streams that might contain uranium or plutonium
- Low-level aqueous wastes that are unlikely to contain any uranium, plutonium, or significant amounts of acid
- Aqueous wastes that are not likely to contain any significant radioactivity

The aqueous raffinates, etc. are collected for product recovery by solvent extraction and then concentrated. The concentrator bottoms are sent to the high level waste system and the overheads to an acid recovery system. The low-level aqueous wastes are combined and concentrated in the general-purpose (GP) concentrator. The concentrate is sent to an ILLW storage tank, and the overhead condensate is sent to an excess water vaporizer. Aqueous wastes that are not likely to contain significant radioactivity are concentrated in the service concentrator. The concentrate is sent to the ILLW storage tank, and the overhead is discharged to the atmosphere.

The ILLW is mainly the concentrates from the LLLW concentration described above. The accumulated ILLW is eventually sent to the HLLW calciner.

The HLLW is mainly the aqueous raffinate from the co-decontamination cycle. This waste is normally concentrated to about 150 gal/MTHM equivalent, but is stored in the HLLW storage tank at a lower concentration. The lower concentration is due to jet dilution during transfer.

Acid recovery. The overheads of the low activity waste (LAW) concentrator are fed to the acid fractionator, which produces 12 M nitric acid from the bottom and essentially water from the overhead. Both streams are recycled, and any excess water is vaporized to the atmosphere.

Offgas treatment. Dissolver offgases (DOG) are collected, cooled, and separated from condensate in the DOG knockout (KO) pot. Most of the DOG consists of purge air drawn into the shear chute to prevent the diffusion of acidic dissolver gases back to the shear. Dissolver reaction gases, air from solution transfer lifts, spargers, and air drawn into open-dissolver ports constitute the rest of the DOG.

Vented gases from the HLLW concentrator, general purpose concentrator, acid fractionator condenser, and plutonium offgas (POG) enter the DOG system downstream of the DOG KO pot. The combined stream is scrubbed by a mercuric nitrate solution for bulk removal of iodine in the No. 1 iodine scrubber. The gases then go to the nitrogen dioxide absorber for nitrogen dioxide and nitrogen oxide removal. The adsorbed gas forms nitric acid, which is recovered through the LAW concentrator. The treated combined offgas then enters the vessel offgas (VOG) system, which collects offgas from the DOG system, the cask vent gas, air lifts, spargers, the solvent burner, instrument purges, and the HLLW offgas. The combined offgases are sent to the No. 2 iodine scrubber for another bulk iodine removal, using mercuric nitrate solution. The gases are then heated, passed through the iodine adsorber (for nearly total iodine removal), filtered, and blown to the main stack.

C.2.1.2 Modifications to the BNFP Process. The modifications are designed to accomplish the following:

- Reprocessing 1-year-old spent fuel; HLLW calcination, vitrification, and encapsulation; and storing the encapsulated 1-year-old HLW until it is shipped to the repository after aging 10 years from the time of reactor discharge
- Upgrading the radioactive effluent control to meet the as low as reasonably achievable (ALARA) objective
- Processing other wastes to meet the acceptance criteria at the repository

Fuel processing age. Since the BNFP is designed to process spent fuel that has been cooled for only 160 days, the processing of 1-year-old fuel, as assumed for the SFDS, presents no problems.

HLLW storage facility. Prompt solidification of HLLW reduces the HLLW storage capacity from the 1-year (300,000 gallons) at the BNFP facility to a smaller volume equivalent to a 40-day process surge. A water basin to store the encapsulated 1-year-old HLW until it is 10 years old is provided.

HLLW calcination and vitrification. The HLLW is calcined in a spray calciner and vitrified in a continuous ceramic melter, as shown in Figure C-2. In this process, the denitrated HLLW is pumped to an internal-mixing, pneumatic, atomizing nozzle at the top of the heated spray calciner barrel having a wall temperature of 700°C. As they fall through the hot barrel, the atomized droplets (nominally < 70 microns in diameter) are flash-dried and calcined. The offgas is drawn through sintered stainless steel filters, and the finely divided product is mixed with glass frit and then fed into a continuous ceramic melter for vitrification. The melter operates at about 1,150°C. While moving along the melter, the mixture of calcine and glass frit is melted. The molten glass that emerges from the melter is collected in a canister, which is either air- or water-cooled to maintain a surface temperature of less

Figure C-2.
P/E Facility
HLW Fixation and
Vessel Offgas Diagram
Reference Case

1 2 3 4

5 6 7 8

than 200°C. After the contents are cooled, the canister is sealed, as described below in the subsection entitled "HLW encapsulation."

Offgas from the melting process is vented through the calciner offgas system. The combined calcination and melting offgas is scrubbed, condensed (the condensibles), and then sent via the NO_x reactor to the VOG system for further treatment and then released as described below in the subsection entitled "Volatile fission products recovery and offgas treatment."

HLW encapsulation. The encapsulation of HLW is closely related to the vitrification of HLW calcine. Figure C-3 shows the steps required in the process.

The empty canister is tested for leaks and transferred to the vitrification and encapsulation cell. The melt is then fed into the canister, during which time the canister is cooled. After the canister is filled and cooled, it is moved to the canister closure station, a cover is added and welded, and the weld is leak-checked by an appropriate method to ensure integrity. A repair option is provided to repair unsatisfactory canister closures by returning the canister to the welding section. Following testing, the canister is moved first to the decontamination station and then to the interim storage area for inspection; if it passes the inspection, it is transferred to the storage pool located outside the encapsulation cell. The encapsulated HLW is stored in the pool until the waste is ready for shipment at age 10 years. At that time, the canister is removed from the storage pool and sent to the inspection station, where the container is checked for radiation to ensure that leaks did not develop during storage. If damage is detected, an overpack facility is provided to overpack the canister. Upon passing inspection, the HLW package is moved through the transfer canal and placed in a cask. The cask is then removed from the loading pool, decontaminated, and checked to determine whether it meets Department of Transportation (DOT) requirements before it can be loaded onto a truck or rail car and shipped to a repository for disposal. For a description of the HLW canister and overpack, see Subsection C.2.2.1.

Volatile fission products recovery and offgas treatment. Existing standards for environmental protection do not require the recovery and treatment efforts described in the paragraph below. But the use of ALARA guidelines leads to the considerations discussed here regarding the upgrading of existing offgas treatment system in the BNFP design.

The main objective is to recover volatile fission products and thus reduce the amount of volatile fission products released from the offgas treatment facility to ALARA levels. Krypton-85, iodine-129, tritium, and carbon-14 are released during fuel preparation and dissolution; ruthenium, oxides of nitrogen, and particulates are released during solidification of waste solutions; and hydrogen chloride and particulates are released during combustion of solid wastes containing chlorinated hydrocarbons.⁽⁴⁾ To be effective, the recovery of these components is conducted as near to the point of release or generation as possible, as shown in Figure C-4.

Voloxidation. The voloxidation process promotes the release of gaseous fission products from oxide fuel by oxidizing uranium dioxide (UO₂) to uranium octoxide (U₃O₈) at 400°C to 500°C in air. This results in the disintegration of the fuel pellets. The disintegration provides an easier escape path for the trapped or bound gaseous fission products (i.e., tritium, iodine, krypton, and xenon) from the chopped fuel prior to dissolution. This process is currently being developed at the Oak Ridge National Laboratory (ORNL).⁽⁵⁾ Earlier reviews in 1973 indicated that the process was very effective in releasing tritium and only partially successful with respect to iodine and noble gases. For this study it is assumed that the process completely removes tritium. Because noble gases and iodine are only partially removed,⁽⁶⁾ they have to be recovered from other streams (e.g., the DOG stream) in the process.

Fuel chopping modifications. Excessive air dilution is not favorable for the recovery of the small amount of volatile fission products. The existing fuel shear at the BNFP will have to be replaced by one that has air seals, such as the fuel disassembly machine system developed for the Hot Experi-

mental Facility (HEF),⁽⁷⁾ so that the air rate to the voloxidizer can be reduced. In this machine, the end fitting is separated from the chopped fuel.

Chop-leach residue. Figure C-5 shows the flow diagram for the treatment and encapsulation of metallic waste, such as end fittings, cladding hulls, and other fuel assembly hardware. The end fittings are separated from the assembly before dissolution and are packaged without treatment, but other parts, mainly the cladding hulls from the dissolver, are leached, washed, neutralized, given another wash, and then submitted to a compaction process in order to minimize the space required at the repository. To compact the hulls and ensure that the pyrophoricity of material does not interfere with the process, the fines must be removed, deactivated separately, and then packaged with the compacted hulls.

The cladding hulls are compacted in a small container with a hydraulic press and a restraining die. These compacted hulls remain in the small container, and several of these containers are banded together and then encapsulated in a carbon steel canister. The canister is tested for leaks by a helium test procedure, and then decontaminated before it is transferred in the dry mode to a cask for shipment to the repository. If the canister leaks, it should be overpacked, leak-tested, and decontaminated before shipment. The end pieces, which were separated during the fuel chopping, are canned in carbon steel canisters and treated in the same way as the hulls canisters were treated. For a description of the canisters and overpacks, see Subsection C.2.2.2.

Tritium recovery. The offgas from the fuel chopper voloxidizer may contain partially oxidized tritium. This stream is passed through a HEPA filter to remove the entrained particulates, through a catalytic recombiner to assure the complete oxidation of the tritium, through a gas cooler to cool the gas stream, and finally through a zeolite bed to remove the tritiated water and the moisture that enters the system with the air. The remaining gas, tritium offgas (TOG), is combined with the offgas from the DOG NO_x absorber and passed through a silica gel bed for ruthenium removal prior to subsequent treatment.

Two-bank zeolite beds are alternated for moisture adsorption and regeneration. After a period of service, the adsorption bank is taken out of the flowstream and heated to drive out the adsorbed tritiated water with a stream of dry nitrogen. The nitrogen stream is chilled, and the water is collected in a tank for fixation by forming cement mix and pouring this mixture into a container. The nitrogen offgas from zeolite regeneration is combined with the voloxidizer offgas from the catalytic recombiner.

DOG treatment modification. The modification of the fuel chopper and the addition of the voloxidizer change the distribution of the air flow and the composition of the DOG. The process flow sheet is adjusted to fit the DOG treatment into the overall volatile fission products recovery scheme. The adjustment involves moving the nitrogen dioxide absorber in the BNFP ahead of the iodine scrubber (renamed NO_x absorber). This NO_x absorber is used for treating DOG only. Four other streams that enter the nitrogen dioxide absorber in the BNFP are directed to the VOG system. The offgas from the DOG NO_x absorber is combined with TOG, as described under "Tritium recovery." An iodine stripper is added to treat the bottoms of the DOG NO_x absorber, and, as a result, the iodine appears in the DOG stream.

New iodine recovery system. The existing iodine scrubber at the BNFP uses a mercuric nitrate/nitric acid scrubbing process. This process is ineffective for removing organic species because of the slow reaction rate of organic iodides and the problem of iodine bleedoff from the recirculating solution.⁽⁴⁾ The disposal of mercuric iodate is an additional problem. In the current BNFP design, the mercuric iodate is stored in the large ILW storage tank. Given the possibility of bleedoff mentioned earlier, the storage of mercuric iodate in the ILW storage tank may not be adequate.

The iodox scrubbing process is reported to be able to remove all iodine species, elemental or

1 2 3 4

5 6 7 8

1 2 3 4

5 6 7 8

2 4 6 8

10 12 14 16

organic, from either the DOG or the VOG, and oxidize them to nonvolatile iodate with a high retention factor.⁽⁴⁾ In the iodox process, concentrated nitric acid (22 M) is used to scrub the combined stream of TOG and DOG. The iodine-rich solution is evaporated to precipitate an anhydride of periodic acid, which is reacted with barium hydroxide to form barium iodate.⁽⁸⁾ The barium iodate is then mixed in the same cement mix used for tritium fixation. Nitric acid from the iodate evaporator is reconcentrated to 22 M in an extractive distillation column using 75 percent magnesium nitrate solution. Makeup nitric acid is also concentrated in this system. Offgas from the iodox absorber is passed through a silver zeolite bed to recover the residual iodine, filtered, and then sent to the krypton-xenon and carbon-14 recovery system.

Krypton-xenon and carbon-14 recovery. Krypton and xenon are the two noble gases formed by nuclear fission. Only krypton-85, which has a relatively long half-life (about 10 years), will be of concern at the time of fuel reprocessing. The major source of carbon-14 in spent fuel elements is from the approximately 25 ppm nitrogen originally present in the fuel.⁽⁴⁾ Almost all of these gases are released to the gas stream after voloxidation and fuel dissolution, and subsequently recovered from the combined TOG and DOG.

Currently, there are two processes⁽⁴⁾ for noble gas recovery: cryogenic distillation and liquid fluorocarbon absorption. The advantage of cryogenic distillation is that it isolates the radioactive krypton, allowing the krypton to be stored separately from the more abundant xenon; the advantage of liquid fluorocarbon absorption is that it can tolerate typical feed gas impurities. This latter process is presently used in the design of the HEF.

In the fluorocarbon absorption process, refrigerant-12 (R-12 or dichlorodifluoromethane) is used to selectively absorb the noble gases and carbon dioxide (containing carbon-14), leaving the nitrogen, oxygen, and argon in the offgas stream. This process is carried out at about 160 psia.

The feed gas to the absorption column flows through (1) the catox bed to convert the carbon-14-containing component into carbon dioxide, (2) the NO_x absorber to remove NO_x, and (3) the freeze dryer to remove the moisture before it enters the krypton absorber. In the absorber, krypton, xenon, and carbon dioxide are dissolved in the liquid R-12 and emerge from the bottom. The krypton stripper (operating at about 30 psia) is used to recover the dissolved components from the liquid R-12. The small amount of gaseous R-12 in the overhead product is removed by a zeolite R-12 recovery bed. The remaining overhead product is mainly a mixture of krypton-xenon and carbon dioxide. At this point, the mixture is compressed to about 2,200 psia and bottled for storage in a repository. R-12 from the bottom of the krypton stripper is purified by removing the degradation products, and the rest of the R-12 is recycled to the system. The small amount of R-12 degradation products⁽⁹⁾ is collected and stored in the P/E facility. Offgas from the krypton absorber is sent to the VOG system.

For a description of the compressed gas cylinders used to hold krypton, xenon, and carbon dioxide, see Subsection C.2.2.4.

NO_x and ruthenium emission control. The bulk recovery of NO_x is carried out as close to the point of generation as possible. The NO_x in the DOG is first recovered in the DOG NO_x absorber and then in the krypton NO_x absorber as described earlier; hence, the NO_x content in the treated DOG is very low. Other high-NO_x-content offgases, such as those from uranium calcination, HLLW calcination, waste incineration, and other processes, are first individually recovered and the lean gas fed to an NO_x reactor,⁽⁴⁾ where the NO_x in the gas is reduced by ammonia in a heated catalytic bed before entering the VOG system (see Figure C-2). This NO_x reactor is a new component added to the existing BNFP design. Ruthenium appears in the TOG, in the DOG, and in the various aqueous streams. A ruthenium absorber, described under "Tritium recovery," treats the TOG-DOG offgas. In the HLLW concentration, HLLW calcination, and acid recovery systems, sugar addition has been used to control the volatility of ruthenium in the BNFP design. The VOG system

is upgraded by replacing the No. 2 iodine scrubber in the BNFP by the VOG iodox adsorber. A ruthenium adsorber is added before the iodox adsorber; this reduces the amount of ruthenium and iodine released.

The excess water vaporization system will also be upgraded by adding a ruthenium trap and an iodine trap on the vaporizer feed stream for ruthenium and iodine removal prior to releasing the excess water to the atmosphere.

Miscellaneous radwaste fixation. Incidental to the operation and maintenance of the P/E facility, miscellaneous radwastes, called "trash", are generated. Although additional radwastes are not described in Subsections C.2.1.1. and C.2.1.2, they are listed under Item No. 5 of Appendix A-18, and are considered in the overall waste management system.

Combustible solid radwastes (such as contaminated paper, cloth, gloves, clothing, filters, and spent resins) and combustible liquid wastes (waste solvent, etc.) are reduced to ashes in an incinerator. The ash is then fixed, either in cement mix for nontransuranic (non-TRU) ashes or in aluminosilicate mineral (clay) for the TRU ashes. The flow sheet for this fixation process is shown in Figure C-6. The incinerator is the key piece of equipment. Descriptions of various pieces of equipment to incinerate combustible waste are available.⁽⁴⁾ Of these, the Molten Salt Combustion Unit developed by Atomics International (currently Rockwell International) is favored, for the following reasons:

- It has a wide variety of waste-handling capabilities, including metal tubing and glassware. This simplifies the processing of items such as filters, which have light metal frames
- It oxidizes wastes beneath the surface of the molten salt. Since acidic gases and particulate matter are trapped in the molten salt, the result is a substantial reduction in the corrosion of the offgas cleanup system

Combustible TRU or non-TRU wastes are sorted and batch-processed in the molten salt combustion unit. The ashes from the combustion unit are determined for TRU content and then properly fixed, as described earlier. The fixing of TRU ashes in clay is to immobilize the high-sodium ashes to the same order of leaching resistance as that of vitrified HLW.⁽⁴⁾ The level of activity is determined for proper selection of the packaging container (see Subsection C.2.2.3).

Low-level waste (LLW) in the form of solid or slurry generated in the uranium conversion and uranium hexafluoride production and noncombustible solid and slurry radwastes are fixed in cement mix and cast in containers. Failed equipment is encapsulated, as described in Subsection C.2.3.8.

C.2.2 Waste/Product Packages

In the Reference Case, the following types of waste containers are required for encapsulation and containment:

- HLW canisters
- Hulls canisters
- End fittings canisters
- TRU and non-TRU waste containers
- Plutonium oxide containers
- Uranium hexafluoride containers
- Gas cyclinders

1 2 3

4 5 6

Table C-1 gives the number, size, weight, decay heat rate, overpack size, and MTHM equivalent expected from a 1,500 MTHM equivalent/yr throughput facility.

Table C-1
WASTE/PRODUCT PACKAGE DATA
REFERENCE CASE
(1,500 MTHM/yr)

Content and Type of Package	No. per Year	O.D. x Length, in.	Gross Weight, lb	Decay Heat, kW	Overpack O.D. x Length, in.	MTHM Equiv. per Package	Remarks
HLW, vitrified, canister	714	12 x 120	1,720	2.6	14 x 132	2.1	80% fill
Hulls canister	690	20 x 120	4,200	0.46	22 x 132	2.2	
End fittings canister	456	20 x 120	1,300	0.10	22 x 132	3.3	
Kr-Xe-CO ₂ canister	38	22 x 120	—	—	—	39	6 gas cylinders per canister
ILW canister							
TRU and non-TRU wastes	2,403	20 x 120	3,300	—	22 x 132	0.6	
Failed equipment	162	20 x 120	—	—	22 x 132	9.3	
LLW (TRU) drum	504	55-gal drum	840	—	83-gal drums	3	
LLW (non-TRU) drum	4,833	55-gal drum	840	—	83-gal drums	0.3	
PuO ₂ container	750*	17.5 x 63.25	32 kg (net)	0.62	—	2	
UF ₆ cylinders	222	49 x 121	25,500	—	—	6.8	66% fill

*Four 6-in. x 2-in. I.D. x 12-in. long (8 kg net each) containers per package (PPP-1M type)

C.2.2.1 HLW Canister and Overpacks. The HLW canister and overpack must be compatible with storage concepts in the repository. The HLW canister and overpack configurations and parameters are based on the waste container studies by Rockwell Hanford Operations (RHO).⁽¹⁰⁾

For the SFDS, the HLW canister is to be fabricated from carbon steel pipes, 12 inches in (nominal) diameter, 10 feet long, with a wall thickness of 3/8 inch. An 11-foot-long carbon steel pipe, 14 inches in diameter, encases the canister as the overpack. Each canister contains approximately 6.3 ft³ of vitrified waste (80 percent filled) and generates 22 kilowatts of heat when vitrification of waste takes place 1 year after discharge from the reactor. After 9 years of storage (10 years old) in the P/E facility, the heat generation rate is about 2.6 kilowatts per canister.

A further discussion of other types of canister material and their behavior is found in Appendix I of this report.

C.2.2.2 Hulls, End Fittings, or Failed Equipment Canister. The hulls canister is fabricated from carbon steel pipe; it is 20 inches in diameter, 10 feet long, and has a 3/8 inch wall. An overpack capability is provided in case the canister is damaged before it is stored in the repository. The overpack is fabricated from carbon steel pipe, 22 inches in diameter and 11 feet in total length.

The end fittings canisters are identical to the hulls canisters and hold the ends of either three pressurized water reactor (PWR) or six boiling water reactor (BWR) assemblies per canister, which corresponds to a weighted average of 3.3 MTHM equivalent per can. The overpack is a carbon steel container identical to the hulls canister overpack.

Failed equipment, which is disassembled and reduced in size at the facility, is packaged in the carbon steel containers used for the hulls canisters.

C.2.2.3 TRU Waste and Non-TRU Waste Container. The packaging of these wastes depends upon the level of activity of the processed wastes. The low-level TRU and non-TRU wastes are sealed in 55-gallon carbon steel drums (DOT 17C or 17H). Depending on the corrosive character of the waste, some of the drums may require plastic liners or be stainless steel with 83-gallon carbon steel drum overpack. The ILW (TRU or non-TRU) is encapsulated in the same kind of 20-inch canister and with the same type of overpack as described in Subsection C.2.2.2.

Packaged low-level non-TRU wastes are shipped for shallow land burial. Packaged TRU wastes (LLW and ILW) and non-TRU ILW are shipped to the repository. Waste packages with less than a 10 mrem/hr dose rate are considered LLW, while those exceeding a 10 mrem/hr dose rate are considered ILW.

C.2.2.4 Gas Cylinders. The compressed gas cylinder used for the containment of krypton, xenon, and carbon dioxide is a modified Linde standard T-sized compressed gas cylinder (DOT Specification 3A-2400). It has a nominal volume of 50 liters, an outside diameter of 9 inches, and a height of 55 inches. The cylinder is modified by replacing the standard cylinder valve with a special valve designed for high-temperature service. The activity of the gas-filled cylinder is considered high-level. Each group of six cylinders is encapsulated in 22-inch canister.

C.2.2.5 Plutonium Oxide Package. The plutonium oxide powder is filled in a stainless steel annular container, 6 inches O.D., 2 inches I.D., and 12 inches long. This container is called the canister and holds 8 kilograms of plutonium oxide (PuO₂). Four canisters are inserted in the AGNS primary pressure vessel (PPV). The PPV is 53 inches long and made of a stainless steel pipe, 6 inches in diameter, that is capped at one end and fitted with a 400-pound pipe flange at the other end.

Another pressure vessel provides secondary containment for the product during shipment. The body of the vessel is stainless steel pipe, 7.5 inches in diameter, capped at one end and fitted on the other end with a concentric reducer, which expands to a 10-inch, 300-pound flange. Before shipping, the primary and secondary pressure vessel package is inserted in the AGNS-Sandia overpack, known as the Modified Overpack Model PPP-1M,⁽¹¹⁾ 17.5 inches O.D. and 63.25 inches long. The overpack offers gamma and neutron shielding, as well as fire and shock protection for the plutonium product during shipment.

C.2.2.6 Uranium Hexafluoride Container. The uranium hexafluoride product is shipped in the uranium hexafluoride cylinder, Model 48X, known as the "10-ton". The container, which meets DOT specifications, is a steel tank designed and fabricated in accordance with ASME boiler and pressure vessel code, Section VIII. It is nominally 39 inches in diameter and 121 inches long. For details about the canister, see Reference MLM-2228.⁽¹²⁾

C.2.3 Facility Description

The process in the P/E facility for the Reference Case has been developed from the existing design of the BNFP, as indicated in Subsection C.2.1. Several facilities have been added to the original plant in order to accommodate the new processes, which were not part of the BNFP original design; other facilities have been deleted or reduced in size, as required by the SFDS. A description of these facilities is presented below.

C.2.3.1 BNFP Facility versus SFDS Facility. The facility in the BNFP is described in the Final Safety Analysis Report, Docket 50-332, dated October 10, 1973.⁽¹⁾ The BNFP is a privately owned fuel reprocessing plant designed in accordance with Atomic Energy Commission regulations applicable at the time of its conception. The plant, which is not in operation at the present time, is located in Barnwell County, South Carolina.

Plant construction began in 1970 and was halted in 1975 as a result of the reversal in U.S. government policies. Because these policies gave the highest priority to nonproliferation, the reprocessing of nuclear fuel was deferred indefinitely. Since 1975, regulations for handling waste have been modified in order to safeguard and protect personnel and the public, and new processes for treatment of waste have been developed that require additional facilities not originally included in the BNFP. The SFDS includes these facilities.

The SFDS facility, shown in Figure C-7, is a plant located at a hypothetical site and provided with the equipment and auxiliary services necessary to meet all reprocessing objectives; namely, recycling the fuel values, isolating and treating the wastes for disposal, and furnishing the safeguards to satisfy the current federal regulations and to offer protection against radiation. Figure C-8 shows the building arrangement for the processing facility.

The following facilities at the SFDS plant have been expanded or added:

- Spent fuel receiving, encapsulated HLW storage, and shipping facilities
- Cladding hulls compaction and encapsulation
- HLLW calcination and vitrification
- Volatile fission products recovery
- Plutonium oxide production and encapsulation
- Miscellaneous radwaste fixation and packaging
- Failed equipment packaging

The following facilities at the SFDS have been deleted or reduced in scope:

- HLLW storage tankage
- Plutonium nitrate storage and shipping

C.2.3.2 Shipping, Receiving, and Storage Pool. The shipping and receiving areas are expanded to handle the arriving spent fuel by truck and rail, and the shipping of encapsulated HLW by rail only.

The storage pool is expanded and designed to store both fuel and HLW canisters with sufficient water depth to allow the movement of an assembly or canister over the stored material, and maintain the necessary shielding. The capacity of the pool is based on the following criteria:

- Surge storage space for fuel assemblies to support 60 days of plant operation (300 MTHM)
- 9 years of storage of encapsulated HLW (13,500 MTHM equivalent)

The storage pool has an area of 16,000 ft² to account for the storage of 600 BWR assemblies, 440 PWR assemblies, and 6,430 canisters of encapsulated HLW. The support systems for the storage pool (i.e., the water cooling system and the water treatment system) are also expanded accordingly.

C.2.3.3 Cladding Hulls Compaction and Encapsulation. The hulls encapsulation cell is located below the BNFP remote maintenance and scrap cell. In addition to the mechanical equipment required for processing the hulls, the cell has space for surge storage for 80 canisters (176 MTHM equivalent). After encapsulation, the canisters are passed through the transfer shaft into the hulls shipping area, and a bottom loading cask, for shipment to the repository.

C.2.3.4 HLLW Calcination, Vitrification, and Encapsulation. The SFDS assumption that HLLW is solidified immediately eliminates the need for long-term storage of this type of waste. In-plant surge storage capacity, however, is still required, and this is provided within the remote facility in the process building.

Calcination and vitrification take place at a remote facility located near the encapsulated waste storage pool. In this facility, the process is monitored from operating galleries situated at the perimeter of the cell, where the mechanical controls and viewing equipment are located. The canisters are sent to the storage pool via the transfer canal using a remotely operated, self-propelled vehicle. Once in the pool, the canisters are placed in modular storage racks.

Details of the calcination and vitrification processes are discussed in Subsection C.2.1.2.

C.2.3.5 Volatile Fission Product Recovery and Offgas Treatment. The addition of the voloxidation process and tritium recovery system requires the use of a larger remote process cell. The voloxidizer and its attendant gas recovery system need more cell and facility space. To provide this space, the process building was enlarged.

At the present time, this study has adopted the policy that recovered materials (solidified or bottled in cylinders) will be shipped to a repository or shallow land burial as soon as the waste is produced. Consequently, the amount of storage capacity provided in the plant is minimal.

C.2.3.6 Plutonium Oxide Production and Encapsulation. This facility was designed for the BNFP and is incorporated in the SFDS P/E facility with no change. The following principal areas are included in the facility: conversion feed, conversion contact, precipitation, and conversion dry cells; and canister filling, decontamination, and canister loadout cells.

1 2 3 4

5 6 7 8

C.2.3.7 Miscellaneous Waste Incineration, Fixation, and Encapsulation. Facilities are included for incineration and immobilization of combustible and noncombustible wastes. This facility is capable of handling either LLW or ILW. It is divided into two areas: one is designed to operate by contact methods; the other is designed to operate remotely. Treatment of solid combustible waste prior to incineration consists of sorting and assaying for contents.

Waste may arrive at this facility from other parts of the plant in wooden boxes, plastic bags, cardboard boxes, or drums. At the facility, combustible and noncombustible materials are separated and then processed. HEPA filters that are highly contaminated are handled in the remote area, where the filter media are cut out and the frames compacted and packaged. The combustible materials are passed through a shredder and then fed to the incinerator.

C.2.3.8 Failed Equipment Packaging. Pieces of equipment that are too large are disassembled or cut into smaller pieces suitable for packaging at the remote maintenance and scrap cell. Canister welding, testing, and shipment of the failed equipment are carried out in the same facilities as are used for encapsulating cladding hulls.

C.2.3.9 HLLW Storage. The 300,000-gallon stainless steel tanks of the existing BNFP are not necessary for the SFDS study. All support facilities, including the underground concrete vault and cooling facilities, are deleted.

C.2.3.10 Plutonium Nitrate Storage. The storage requirements for plutonium nitrate are reduced to 10 days of operating capacity.

C.2.4. Operational Considerations

The principal operations at P/E facilities for the Reference Case are:

- Receive casks of spent fuel, cool cask internals, unload the fuel into storage, and return casks to the transportation system
- Transfer individual spent fuel assemblies to the mechanical processing equipment, shear off end fittings close to the active fuel pellets, segregate end fittings in separate hulls containers, chop fuel into short lengths in preparation for acid leaching, and segregate fuel nozzles as waste
- Chemically process fuel materials to decontaminate uranium and plutonium from fission products, and separate plutonium from uranium
- Convert uranyl nitrate to uranium hexafluoride, in which form uranium can be returned to an enrichment facility
- Convert plutonium nitrate to plutonium oxide for recycling back to fuel fabrication (and return to the LWR fuel cycle)
- Solidify and vitrify HLLW to stabilize it for long-term storage in the salt repository
- Compact chopped fuel residues, package compacted residues, and package uncompacted end fittings of fuel assemblies
- Incinerate combustible solid wastes, fix ashes and non-combustible solid wastes in a less leachable form, and package them for shipment
- Process offgases, recover radioactive gases, and package the gases for disposal

The SFDS concept of reprocessing is based on the BNFP facility design, which can accommodate a throughput of 1,500 MTHM/yr of spent fuel. The fuel receiving and offsite shipping operations are continuously operated; the chemical processing and conversion facilities are assumed to be

on stream for 300 days per year to permit certain maintenance and repair work to be done. At any time, approximately 60 days' equivalent of spent fuel (300 MTHM) is in storage, along with up to 9 years' equivalent of vitrified HLW. Fuel enters mechanical processing at a cooling age of 1 year out of the reactor, and HLLW is calcined and vitrified essentially immediately upon formation. Since SFDS design concepts provide sufficient surge capacities to decouple the main processing steps from one another, an unscheduled shutdown in one processing area is presumed not to interrupt processing in any other area. For example, 8 days' storage of uranium nitrate product decouples the separations facility from the uranium conversion facility, and about 40 days' storage of HLLW is available to decouple separations from the waste solidifications facility. Additional surge capacity for product storage reduces the immediate impact of transportation interruptions.

The flow of fuel, product, and waste materials for the 1,500-MTHM equivalent throughput system is shown in Table C-2.

Table C-2
ANNUAL MATERIAL FLOW AT EACH P/E FACILITY
REFERENCE CASE
(1,500 MTHM/yr)

Material	Annual Number of Shipments	Total Units	Remarks
Incoming			
Spent PWRs	224 rail	1,792	14 PWR/cask
	248 truck	248	1 PWR/cask
Spent BWRs	148 rail	2,812	34 BWR/cask
	124 truck	248	2 BWR/cask
	<u>744 total</u>	<u>5,100 assem.</u>	
Internal transfers			
Fuel baskets			
PWR	510	2,040	4 PWR/basket
BWR	<u>340</u>	<u>3,060</u>	9 BWR/basket
	850 baskets	5,100 assem.	
Fuel assemblies			
PWR	2,040	2,040	
BWR	3,060	3,060	
Outgoing			
UF ₆ ^(a)	111 truck	222 containers	2 cont./shipment
PuO ₂ ^(b)	94 truck	750 canisters	8 large can./ship.
Hulls	127 rail	1,146 canisters	9 can./cask
HLW glass	60 rail	714 canisters	12 can./cask
Non-TRU LLW	114 truck	4,758 drums	42 drums/shipment
TRU-LLW	12 truck	504 drums	42 drums/shipment
TRU-ILW	344 truck	2,403 canisters	7 can./shipment
Radioactive gases	2 rail	11 canisters	9 can./cask
Failed equipment	23 rail	162 canisters	7 can./shipment

(a) 48X container.

(b) Annular canisters + PPP-1M package.

C.2.4.1 Fuel Receiving and Storage. Fuel receiving and storage includes the following:

- Receiving and cooldown of the cask
- Unloading the cask
- Moving fuel baskets
- Cleaning and cooling of the pool
- Processing wastes from the receiving area
- Accountability
- General duties
- Overall operating staff requirements

Receiving and cooldown of the cask. Receiving operations at a P/E facility begin with the removal of accumulated nonradioactive contamination that has collected during transportation from the reactors. The cask is then moved into the cask receiving area within the facility, lifted off the transport vehicle, and lowered into the cask cooldown pit if the cask cavity is gas-filled or into the test and decontamination pit if the cask cavity is water-filled. In cask cooldown, the operators initiate each of the three-stage cooldown operations: the first stage uses high-temperature and high-pressure water, the second stage uses low-pressure hot water, and the third stage uses water cooling. The operators also monitor the buildup of radioactivity in the coolant leaving the cask, and service the cleanup system, as necessary, to maintain acceptable radiation levels in the vicinity of the cleanup system and acceptable levels of dissolved and suspended activity in the pool water. Most of the bolts holding the outer lid of the cask in place are removed, and the cask is moved from either the test and decontamination pit or the cooldown pit into one of the cask unloading pools. Because hoisting limits on the cask-handling cranes are set quite low, moving the heavy casks, in particular the rail casks, consumes a great deal of operating time.

The operating crew requirements for receiving a single cask are as follows:

Receiving a truck cask	2 persons for 4.5 hours
Receiving a rail cask	3 persons for 6 hours

Unloading the cask. Once the cask is lowered into the unloading pool, the outer and inner heads are removed, and a fuel storage basket is brought from the storage pool into the cask unloading pool and lowered onto its holding fixture. The operator uses a long-handled grapple suspended from the fuel-handling crane to pick up one fuel assembly at a time and transfer it into the storage basket. Accountability records are entered to show assembly number, basket number, and grid location in the basket for each assembly. When the cask is unloaded, the basket containing the fuel assemblies is moved from the cask unloading pool in order to eliminate the possibility of a rupture in case a cask is hoisted from the pool and lowered into the test and decontamination pit for cleanup before it is returned to the reactors.

The operating requirements for unloading the cask are as follows:

Receiving a truck cask	2 persons for 4.5 hours
Receiving a rail cask	3 persons for 6 hours
Unloading a truck cask	1 person for 4.5 hours
Decontaminating/shipping truck cask	2 persons for 4 hours
Unloading a rail cask	1 person for 9 hours
Decontaminating/shipping rail cask	3 persons for 8 hours

Moving fuel baskets. When a storage basket has been filled with 9 BWR or 4 PWR assemblies, the operator moves the loaded basket from the cask unloading pool to the fuel and encapsulated

waste storage pool. Inventory records are adjusted to show pool location, basket number, and grid location of each basket in the storage pool.

The operating crew requirements for moving fuel baskets are as follows:

Transferring the fuel basket to the storage pool	2 persons for 2 hours
Temporarily removing the fuel basket from the cask unloading pool to permit cask movement	1 person for 1/2 hour

Cleaning and cooling of the pool. Cleaning and cooling of pool water requires some manual operations, such as changing the filter media or precoat, starting the resin regeneration operation, sampling and analyzing the pool water, and valving to direct water flows from lower to higher levels of contamination, as much as practical.

The operating crew requirements for pool cleanup and cooling are as follows:

Cleaning and cooling the pool	3 persons for a total of 8 hr/day (about 3 hr/shift)
-------------------------------	---

Processing wastes from the receiving area. Solid wastes accumulate from pool cleanup operations, from materials used in preparing areas for operating and maintenance work, and from failed equipment items. Liquid wastes accumulate from cask cooldown, cask flushing, and cask decontamination operations.

Liquid wastes are concentrated and stabilized in such a way that there are no free liquids that could leak into the environs around a shallow land burial site should the container be breached. Solid wastes are packaged without further processing.

The operating crew requirements for processing wastes from the receiving area are as follows:

Concentrating, packaging, and shipping liquid waste	1 person for 8 hr/day
Packaging and shipping solid waste	2 persons for 8 hr/day

Accountability. The unloading and storage pools are item control areas. Records must be kept to show current inventories and identify and locate each fuel assembly. Since current inventory records are expected to provide a satisfactory accountability, no time has been allotted to make periodic audits that randomly select certain fuel assemblies for verification and to make an annual physical inventory. Time is allocated only to adjust accountability records each time a fuel transfer is made.

The operating crew requirements for accountability are as follows:

Maintaining accountability records	2 persons for 8 hr/day (about 3 hr/shift)
------------------------------------	--

General duties. Additional operating time is consumed in preparing work areas, routinely monitoring personnel and equipment, preparing for entry or exiting from potentially contaminated areas, housekeeping, training, etc. To arrive at the total operating crew requirements, an additional 20 percent of the labor force for the preceding items has been assumed.

Overall operating staff requirements. The number of people required in the operating crew of each shift in order to achieve a throughput rate of 2 casks per day was determined using time allocation charts. The day workers work on a variety of jobs, some of which are not related to throughput.

Total operating staff requirements for fuel receiving and storage are shown in Table C-3.

Table C-3
TOTAL OPERATING STAFF FOR FUEL
RECEIVING AND STORAGE
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (7 x 4 shifts)	28
Day workers	4
Supervisors (1/shift + 1 day)	5
Total	<hr/> 37

C.2.4.2 Mechanical Processing. After the individual fuel assemblies have been selected for the next processing lot, fuel storage baskets (BWR or PWR) are moved to the fuel transfer canal. One assembly at a time is transferred under water to the remote process cell. In the remote process cell, the fuel assembly is loaded into the shear, and the leading end fitting is sheared from the assembly (about an inch outside the active portion of the assembly) and allowed to slide into a waste canister. The active length of fuel is sheared into short segments which fall into the voloxidizer; the trailing end fitting is then diverted into a waste canister. After volatile fission products are released from the fuel tube segments in passing through the voloxidizer, fuel materials and clad segments fall into the dissolver. Following the dissolution of the fuel pellets, the fuel assembly residues are monitored to be sure fuel materials are fully dissolved. Additional processing of residues includes volume reduction, passivation of zircaloy fines, and packaging for disposal. (See Subsection C.2.4.7 for further discussion.)

The operating crew requirements for mechanical processing are as follows:

Moving fuel to shear	1 person/shift
Shearing fuel	1 person/shift
Operating voloxidizer	1 person/shift
Handling hulls	1 person/shift
Performing general duties	1 person/shift
Supervision	1 person/shift

Total operating staffing requirements for mechanical processing are shown in Table C-4.

Table C-4
TOTAL OPERATING STAFF FOR
MECHANICAL PROCESSING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (5 x 4 shifts)	20
Day workers	0
Supervisors (1/shift)	4
Total	<hr/> 24

C.2.4.3 Chemical Processing. After dissolution of fuel materials, the main chemical processing steps are preparation, solvent extraction of uranium and plutonium, scrubbing of fission products from uranium and plutonium, separation of plutonium from uranium, and purification of uranium and plutonium product. Other operations involve processing of offgas released during chemical process (see Subsection C.2.4.9).

Staffing to support these remote chemical operations is determined from work loads anticipated by functional areas of the facility. Personnel in the senior licensed categories work in the central control room, where almost all the instrument and control functions can be performed. Included in this category are supervisors, lead operators, and control room operators, all of whom are qualified to start up, operate, and shut down the process system. Other supporting operators are stationed in the cold chemical preparation areas, and roving crews of operators are dispatched throughout the plant to monitor chemical processing, operate samplers, set manual valves and controls, and perform support functions for maintenance and safety.

The operating crew requirements for chemical processing are as follows:

Operating the control room	3 persons/shift
Preparing cold chemicals	2 persons/shift
	2 persons on day shift
Providing roving support	3 persons/shift
Performing general duties	2 persons/shift
Supervision	1 person/shift

The total operating staff requirements for chemical processing are shown in Table C-5.

Table C-5
TOTAL OPERATING STAFF FOR
CHEMICAL PROCESSING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (10 x 4 shifts)	40
Day workers	2
Supervisors (1/shift + 1 day)	5
Total	<hr/> 47

C.2.4.4 Conversion of Uranyl Nitrate to Uranium Hexafluoride. The main processing steps in the conversion of uranyl nitrate to uranium hexafluoride are calcination of uranium nitrate hexahydrate to uranium trioxide, reduction to uranium dioxide, hydrofluorination to uranium tetrafluoride, and fluorination to uranium hexafluoride. In general, the conversion plant is operated from a central control station, and roving operators collect samples and set manual controls and valves. Other operational duties include solid waste packaging, handling, and product shipment. The number of waste canisters to be handled is given in Appendix A-2. Most of the waste volume from uranium hexafluoride is in the non-TRU category. Approximately one product cylinder has to be loaded every other day and stored temporarily until a second canister is available to make up a transport vehicle load. A separate process area is provided that employs electrolytic cells to produce fluorine gas to be used in the fluorination step. General support to decontaminate work areas should be significantly less in the uranium hexafluoride plant than in the separations plant because potential contamination levels are much lower.

The operating crew requirements for the conversion of uranyl nitrate to uranium hexafluoride are as follows:

Operating the control station	3 persons/shift
Providing bulk chemicals	2 persons/shift
Providing roving support	2 persons/shift
Producing fluorine	2 persons/shift
Handling solid waste	2 persons on day shift
Loading and shipping product cylinders	2 persons on day shift
Providing general support (10 percent of the above)	1 person/shift
Supervision	2 persons/shift

Except for fluorine production, handling product cylinders, and providing bulk chemicals to certain work stations, the operating staff requirements do not directly depend on throughput.

Total operating staff requirements for the conversion of uranyl nitrate to uranium hexafluoride are shown in Table C-6.

Table C-6
TOTAL OPERATING STAFF FOR
CONVERSION OF URANYL NITRATE
TO URANIUM HEXAFLUORIDE
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (10 x 4 shifts)	40
Day workers	4
Supervision (2 x 4 shifts)	8
Total	52

C.2.4.5 Conversion of Plutonium Nitrate to Plutonium Oxide. Conversion of plutonium nitrate to plutonium oxide proceeds through the following processing operations: precipitation of plutonium oxalate, filtration, drying, and calcining to the oxide. All operations are conducted by remotely operated equipment located behind shielding walls. Some of the less contaminated items of equipment may be located in shielded glove boxes. Most of the operations needed to sustain continuous production are conducted from the central control room. As in the other reprocessing facilities, roving operators collect samples and adjust manually operated controls and valves located throughout the operating galleries in the facility. Product packaging and handling, waste assay, and waste handling in the facility usually place additional demands on security, accountability, and operations that are not required in the other areas of the fuel reprocessing complex. Special nuclear materials (SNM) have to be protected from diversion, and strict material balances have to be maintained. The inventory of plutonium in product and waste streams has to be accurately measured and controlled. If enough plutonium is present, wastes may have to be segregated, protected, and recycled for scrap recovery.

The operating crew requirements for the conversion of plutonium into nitrate to plutonium oxide are as follows:

Providing process control	3 persons/shift
Providing roving support	3 persons/shift
Preparing cold chemicals	1 person/shift
Packaging and handling the product	2 persons/shift
Handling solid wastes	2 persons/shift for 4 days
Providing bulk chemical service	2 persons for 5 days.
Assaying waste drums	2 persons for 5 days
Preparing product for shipment	2 persons for 5 days
Providing general support (+ 20 percent of above)	2 persons/shift
Supervision	2 persons/shift

Throughput influences the number of personnel required to make up cold chemicals, load out the product, provide bulk chemical service, and handle product shipment. Other work categories are not so directly influenced.

Total operating staff requirements for the conversion of plutonium nitrate to plutonium oxide are shown in Table C-7.

Table C-7
TOTAL OPERATING STAFF FOR
THE CONVERSION OF
PLUTONIUM NITRATE TO
PLUTONIUM OXIDE
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (11 x 4 shifts)	44
Day workers	4
Supervision (2 x 4 shifts)	8
Total	<hr/> 56

C.2.4.6 Solidification of HLW. HLLW from the uranium-plutonium separations operation is spray-dried and calcined, mixed with glass-forming materials, and melted to produce a borosilicate glass. The glass is packaged for disposal, and the waste canisters are stored in water until the waste decays sufficiently to permit offsite shipment. All these operations are carried out within heavily shielded cells; these remote mechanical operations are complex because of the nature of the processes described above. The integrity of the canister is verified by tests after filling, and provision has been made to overpack the primary canister in the event of a failure. The canister can be overpacked at any time during its 9-year storage period or prior to shipment to the repository. Approximately five HLW containers are produced every two days, and these containers fill a rail cask every 2-1/2 days (12 canisters). The HLW to be solidified is 1 year old. Shipments off site begin after 9 years' worth of filled waste cans has accumulated; the canisters of HLW glass are loaded dry into offsite shipping casks.

The operating crew requirements for the solidification of HLW are as follows:

Controlling solidification/ vitrification processing	3 persons/shift
Controlling mechanical processing	4 persons/shift
Providing cold chemicals and loading-in fresh canisters	2 persons/shift
Handling waste materials	2 persons on the day shift
Handling canisters (fresh)	2 persons on the day shift
Performing general duties (+ 20 percent of above)	2 persons/shift
Loading the cask	1 person on the day shift
Decontaminating the cask	3 persons on the day shift
Providing inventory control	1 person on the day shift
Supervision	2 persons/shift

Throughput of the HLW solidification facility directly influences mechanical processing, container supply operation, and loading and shipping operations.

Total operating staff requirements for the solidification of HLW are shown in Table C-8.

Table C-8
TOTAL OPERATING STAFF FOR
THE SOLIDIFICATION OF HLW
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (11 x 4 shifts)	44
Day workers	9
Supervision (2 x 4 shifts)	8
Total	<hr/> 61

C.2.4.7 Volume Reduction of Fuel Cladding. The end fitting sheared from the fuel-bearing portions of spent fuel are packaged without further processing in cans 20 inches in diameter and 10 feet long. If the fittings are carefully oriented in the filling operation, the container volume can be efficiently utilized.

No further fixation of those materials in an inert matrix is planned. The tube sections and metallic fines from the shearing operation are dumped from the dissolver basket onto a screen to segregate the fines from the larger tube sections. The tube sections are compacted from a density of about 1 g/cm³ to 3.7 g/cm³ in individual compact canisters, and seven of these canisters are bundled into close-packed arrays and loaded into a hulls container. Fines after the activation processing are poured on the compacted tube sections to fill the canister voids. The waste canisters then undergo welding, leak-testing, decontamination, and cask-loading. Mechanical processing

operations — most of them hands-on and remote-control — are viewed from operating galleries. The automatic welding and the compaction operations are manually initiated, and are followed by the automatic programmed operation.

About every 4 hours, a dissolver basket containing hulls from 1-MTHM-equivalent spent fuel is dumped, and fines are deactivated immediately. Tube sections are charged into a small compactor to fill the canister, the piston advances to crush the tubes, the piston retracts, a smaller charge of tube sections is added, and the cycle is repeated until the canisters are filled to the desired level. Seven cans are strapped together, grappled, and lowered into the larger 20-inch by 10-foot hulls container. Three such layers are required to fill the hulls container, and a canister is filled with compacted hulls every 10 hours. Approximately two canisters per day of end fittings and five canisters of compacted tube sections every two days accumulate from the hulls volume reduction operations at a 1,500 MTHM/yr reprocessing plant. Two casks per week have to be loaded for shipment to the geologic repository.

The operating crew requirements for the volume reduction of fuel cladding are as follows:

Dumping the basket and deactivating the fines	1 person/shift
Loading-out end fittings	1 person/shift
Operating the compactor	1 person/shift
Banding and packaging small canisters	1 person/shift
Performing welding and leak-testing	1 person/shift
Decontaminating canisters and loading-out the cask	2 persons/shift (for 5 days only)
Furnishing accountability records	1 person on day shift
Providing canisters, as needed	2 persons on day shift
Performing general duties	1 person/shift
Supervision	2 persons/shift

Almost all work in hulls volume reduction is directly related to throughput, except for some general work that would have to be performed even if the throughput were zero.

Total operating staff requirements for the volume reduction of fuel cladding are shown in Table C-9.

Table C-9
TOTAL OPERATING STAFF FOR
VOLUME REDUCTION OF FUEL CLADDING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (6 x 4 shifts) (2 x 3 shifts)	30
Day workers	3
Supervision (1 x 4 shifts) (1 on days)	5
Total	<hr/> 38

C.2.4.8 Volume Reduction of Combustible Solid Wastes and Fixation of Other Solid Wastes. Solid wastes accumulate from paper and plastics used in the preparation of work areas, and these wastes may be contaminated with radioactive materials. Small tools and failed equipment items, chemical processing precipitates, spent ion exchange resins, and spent ventilation filters also contribute significant quantities of solid waste materials. The SFDS assumes that 20 percent of the trash is not combustible and that the remaining 80 percent can be reduced in volume by a 40:1 ratio using molten salt combustion techniques. Fixation of ash from molten salt processing and fixation of noncombustible wastes results in a nominal 2:1 volume penalty. Process precipitates, such as those from offgas scrubbing in the uranium conversion facility, are assumed to be packaged without further fixation.

Table C-2 shows the expected quantity of solid wastes resulting from volume reduction and fixation. Non-TRU wastes that can be shipped to shallow land burial are shown at the bottom of the table.

Combustible wastes pass through shredding equipment and a molten salt incinerator before being blended with fixation materials and dumped into waste containers. Noncombustible wastes are blended with fixation materials and dumped into waste containers. Remote operations of all volume reduction and fixation systems are planned, and semi-remote or fully remote operations are involved in handling wastes inbound and outbound. At each facility, about 80 to 100 drums per day of waste have to be collected from their point of origin and delivered to waste processing.

The operating crew requirements for the volume reduction of combustible solid wastes and fixation of other solid wastes are as follows:

Collecting and delivering waste	4 persons/day for 5 days
Loading combustible and non-combustible feed hoppers	4 persons/shift
Operating process equipment	2 persons/shift
Providing containers for processing	1 person/shift
Providing bulk chemicals	2 persons/day for 5 days
Decontaminating and preparing drums for offsite shipments	2 persons/shift
Performing general duties (+ 20 percent of above)	2 persons/shift
Supervision	1 person/shift

Throughput directly determines the staffing requirements for solid waste volume reduction and fixation. Total operating staff requirements for volume reduction and fixation are shown in Table C-10.

Table C-10
TOTAL OPERATING STAFF FOR
VOLUME REDUCTION OF COMBUSTIBLE
SOLID WASTES AND FIXATION
OF SOLID WASTES
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (11 x 4 shifts)	44
Day workers	6
Supervision (1/shift + 1 day)	5
Total	55

C.2.4.9 Offgas Treatment, Volatile Fission Products Recovery, and Packaging. Offgases released during dissolution and chemical processing of fuel materials carry along volatile fission products such as tritium, iodine, krypton, xenon, ruthenium, and an activation product of carbon. Tritium is isolated as tritiated water and bound in the cement mix; iodine is isolated as barium iodate and also bound in the cement mix. Krypton, xenon, and carbon dioxide are recovered as pressurized gases in standard gas cylinders. Ruthenium is trapped on solid adsorbers, stripped, and recycled to HLW calcination. All of these operations are remote chemical operations up to the point of packaging, leak-testing, decontamination, and stripping operations. See Table C-2 for estimated production rates on waste containers and gas cylinders.

The operating crew requirements for offgas treatment, recovery, and packaging are as follows:

Operating the control room	2 persons/shift
Handling inbound containers	1 person on days for 5 days
Performing mechanical operations	2 persons/shift
Loading-out shipping waste packages	2 persons on days for 5 days
Providing bulk materials	2 persons on days for 5 days
Supervision	1 person/shift

Throughput influences all categories under this operation, except the control room functions. Total operating staff requirements for offgas treatment, recovery, and packaging are shown in Table C-11.

Table C-11
TOTAL OPERATING STAFF FOR
OFFGAS TREATMENT, VOLATILE
FISSION PRODUCTS RECOVERY, AND PACKAGING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (4 x 4 shifts)	16
Day workers	5
Supervision (1 x 4 shifts)	<u>4</u>
Total	25

Total direct operating personnel requirements for a P/E facility in the Reference Case are shown in Table C-12; total support and overhead personnel requirements for such a facility are shown in Table C-13.

Table C-12
TOTAL OPERATING STAFF
REQUIREMENTS FOR A
REFERENCE CASE P/E FACILITY
(1,500 MTHM/yr)

Personnel	Number Required
Fuel receiving	37
Mechanical processing	24
Chemical processing	47
Uranium conversion	52
Plutonium conversion	56
Waste solidification	61
Hulls voloxidation reduction	38
TRU volume reduction	55
Offgas recovery	<u>25</u>
Total	395

Table C-13
TOTAL SUPPORT STAFF FOR A
REFERENCE CASE P/E FACILITY
(1,500 MTHM/yr)

Personnel	Number Required
Maintenance (0.5 x oper.)	200 (some shift work, rest days)
Analytical (0.1 x oper.)	40 (shift work)
Health and safety (0.1 x oper.)	40 (shift work)
Utilities	30 (shift work)
Plant overhead (0.3 x oper.)	120 (some shift, mostly days)
Total	430

Total overall personnel requirements for a P/E facility in the Reference Case are shown in Table C-14.

Table C-14
OVERALL TOTAL STAFF
REQUIREMENTS FOR A
REFERENCE CASE P/E FACILITY
(1,500 MTHM/yr)

Personnel	Number Required
Operating	395
Support and overhead	430
Total	825

C.2.5 Cost Estimates

C.2.5.1 Capital Cost Estimate. The Reference Case capital cost estimate is based primarily on the BNFP⁽¹³⁾ process described in Subsection C.2.1. The estimate also draws upon the results of previous work Bechtel performed for the BPNL.⁽¹⁴⁾

The subfacility cost estimates are generic, and do not include any costs for abnormal site conditions. These estimates are broken down by direct manhours and material costs for the following elements:

- Site improvements
- Buildings and structures
- Major equipment
- Bulk materials

Contractors' indirect field costs at 85 percent of direct labor costs are included in the total field costs. Added to the total field costs are the costs for contingency, architect-engineer services, and the owner's cost as defined in Subsection B.1.2.1.

Table C-15 lists the capital costs by subfacility for a single 1,500 MTHM/yr P/E facility and also lists the reference that identifies the basic source of the subfacility estimate. For all subfacilities, the estimate basis is carefully reviewed for scope and date of estimate. Appropriate adjustments are made, as necessary, to properly reflect all required processes and facilities for the SFDS and to bring all costs to the required January 1979 level.

Table C-15
CAPITAL COST ESTIMATE FOR A
REFERENCE CASE P/E FACILITY
(1,500 MTHM/yr)^(a)

Item	Cost \$ million/yr ^(b)
Field costs (including 20% contingency)	
Fuel receiving and waste storage	153
Voloxidation ⁽¹⁴⁾	28
Separations (with surge storage)	219
Uranium hexafluoride	106
Plutonium oxide	107
HLLW	100
Hulls compaction and encapsulation	27
Radwaste incineration	18
Radwaste cement immobilization	28
Gas recovery	41
Total field costs	820
Architect-engineer services @ 20% of \$820	160
Owner's costs @ 30% of \$980	290
Total capital cost	1,270

(a) The 5,000 MTHM/yr SFDS economic study requires the equivalent of 3-1/3 of the above P/E facilities (see Subsection C.1.1).

(b) January 1979 dollars.

C.2.5.2 Operating Costs Estimate. The P/E facility in the Reference Case is basically the BNFP plant design with separation of uranium and plutonium, uranium conversion to uranium hexafluoride, plutonium conversion to plutonium oxide, and solidification of HLLW. Operating costs are estimated on the basis of Bechtel estimates of census, average burdened salaries, assumed canister prices, and allowances for operating supplies, maintenance materials, and utility costs.

Average salary figures lead to an annual average cost of \$24,000 per person for the total census as discussed in Subsection B.1.2.2.

For operating supplies and maintenance materials for all cases, 1 percent of the total capital investment is provided. Utility costs are assumed to be 0.1 percent of the total capital cost of a facility. Canister costs are assumed to have an average cost of \$5,000 each for HLW glass, hulls, and process ILW trash.

Table C-16 is a summary of the operating cost estimate for the Reference Case P/E facility.

Table C-16
OPERATING COST ESTIMATE FOR
A REFERENCE CASE P/E FACILITY
(1,500 MTHM/yr)

Item	Cost \$ million/yr
Personnel 825 persons x \$24,000/yr	19.8
Operating/maintenance supplies 0.01 x \$1,270 million	12.7
Utilities 0.001 x \$1,270 million	1.3
Canisters 4,440 canisters x \$5,000	<u>22.2</u>
Total	56.0

C.2.5.3 Decommissioning Cost Estimate. The Reference Case decommissioning costs are taken as 15 percent of the facility's total capital costs, less the owner's costs. This factor is 5 percent higher than other estimates for decommissioning simple fuel encapsulation facilities, largely because decontamination operations are much more complex and extensive. Processing equipment exposed to radioactive liquids are more difficult to decontaminate than surfaces simply contaminated with dry particulates, as in the spent fuel encapsulation facilities. Surfaces of equipment that become contaminated by liquids during plant operations are especially resistant to chemical decontamination techniques. Much of the equipment may have to be handled remotely for final size reduction and packaging operations if high levels of residual contamination persist. The decommissioning cost of the Reference Case P/E facility is taken as \$145 million.

C.3 CASE 1

The P/E facility for Case 1 is designed to package spent unprocessed fuel assemblies in steel canisters suitable for permanent storage in a salt repository.

The facility design and configuration and the encapsulation process are essentially the same as that of the Spent Fuel Receiving and Packaging Facility designed by RHO⁽¹⁵⁾ and Kaiser Engineers as part of the Department of Energy (DOE) Spent Fuel Handling and Packaging Program.⁽¹⁶⁾ Modifications have been made to reduce the process capacity from 3,300 to 1,500 MTHM/yr and to provide for product surge storage and shipping facilities. The description of the facility and the extent of these modifications are presented in Subsection C.3.3.

C.3.1 Process Description

The P/E facility overall block flow diagram is shown in Figure C-9. The facility is designed to encapsulate 10-year-old fuel assemblies that arrive in rail casks from an AFR.

On the basis of a 1,500 MTHM/yr capacity, the total number of assemblies processed in the facility is 5,100 per year, of which 40 percent are PWR and 60 percent BWR (see Appendix A). Processing takes place 300 days per year, in three 8-hour shifts per day, 7 days per week. The average throughput is either 7 PWR or 10 BWR assemblies per day.

C.3.1.1 Fuel Receiving and Offloading. The fuel receiving and offloading system can accept three rail casks. Each cask contains either 14 PWR assemblies or 34 BWR assemblies.

After cask unloading, the removal of fuel assemblies from the cask is done in a wet mode by immersing the cask in a cask unloading pool. The unloaded fuel assemblies are transferred underwater via a fuel transfer canal to the encapsulation cell. Leaky assemblies are sent to a separate, small pool for isolation and are encapsulated on a first-priority basis.

A spent fuel surge storage pool accommodates the fuel that cannot be processed immediately because of interruptions in the transportation system or by a temporary process shutdown. This pool has a capacity of 300 MTHM.

C.3.1.2 Fuel Encapsulation. The spent fuel encapsulation process is a semiautomatic, production-type, remotely operated system, in which the spent fuel assemblies are encapsulated in steel canisters. The flow diagram is shown in Figure C-10.

The spent fuel assemblies brought into the fuel transfer cell by a basket buggy are placed in a dryer, where heated air is blown around them and their holding basket. Then, the assemblies are removed from the dryer and transferred to the encapsulation cell by lowering them through a floor port and into a canister located under the roof of the encapsulation cell. The empty canister has the top lip sealed against the port so that contamination to the cell is avoided.

Empty canisters are introduced into the cell by a transfer buggy that travels in a tunnel under the cell. Canisters are handled (at the initial stage of encapsulation) by a transfer carousel which is programmed to perform the following operations:

- Receive, by pneumatic clamping, the incoming canister from the jib hoist system
- Swing the upright canister into position under the canister filling port
- Elevate and hold the canister in a sealed position against the port during filling
- Lower the canister from the filling port and swing it to the cap placement position
- Hold the canister while the manipulator places the cap

1 2 3 4

5 6 7 8

Figure C-10.
P/E Facility
Spent Fuel Encapsulation Flow Diagram
Case 1

Adapted from Rockwell Hanford
encapsulation concept⁽¹⁵⁾

1 1 1 1

1 1 1

- Swing and rotate the canister assembly into a horizontal position for closure welding
- Return to the empty canister receiving position to begin the next cycle

The welding operation is performed with the canister supported horizontally in turning rolls and positioned by adjustable pressure rolls. Welding by submerged arc welder is semiautomatic while the canister rotates at welding speed. After welding, the canister is rotated to the upright position and then transferred by crane to the weld test station.

All work stations have mechanized equipment to position the canister, to remotely manipulate the process equipment, and to transfer the canister to the next work station.

There are four identical postweld heat treat stations, each with an induction heat treat system designed to be applied to the cap weld area only. There is also a nondestructive weld inspection station in which the ultrasonic transducer equipment is rotated in an orbital path around the canister cap joint to search for discontinuity on the weld.

The helium pressurizing equipment is designed to evacuate the air and then backfill the canister with helium gas to a pressure of about 2 atmospheres. The canister is designed with a self-sealing valve, which retains the gas charge. After the canister is filled, the valve opening is closed with a mechanical sealing plug.

Two helium leak test stations, each with a mass spectrometer leak detector system, are provided. Either one can measure helium leakage from the canister.

After testing, the canisters are transferred to the radiation survey and external swipe sampling station, where a radiation survey of the canister is performed to detect the packages that contain unusual levels of activity. At this time, an operator using a master-slave manipulator obtains swipe samples to measure the canister external contamination.

Canisters that fail the nondestructive ultrasonic tests are transferred to the special function cell, which is provided for the repair of defective canisters and for the special handling of leaking or damaged fuel assemblies. Canisters with defective cap welds are opened and the assemblies removed and returned to the encapsulation cell via the water canal for repacking. The defective empty canisters are cut up and packaged for disposal as non-TRU LLW.

Canister packages that show excessive external contamination are subject to a decontamination bath and dried before returning them to the swipe station for retesting. Canisters that have no external contamination are transferred through a floor port into a waiting transfer buggy that carries them to the cask loading area.

C.3.1.3 Canister Loadout. The buggy moves the canisters laterally in a tunnel that leads to the canister loadout cell, where the canisters are removed from the buggy and loaded into the shipping cask, which was previously brought into the cell. The cask lid is placed, and the cask is moved to the inspection and test area, where the fasteners are applied by workers operating from elevated platforms. If the tests indicate external contamination, the cask is transferred to the decontamination area and is washed externally until all contamination is removed. The cask is then sent to the loading station. The shipping cask travels between the canister loadout cell and the cask loading station on an air pallet that supports the weight of the cask on a film of compressed air a few thousandths of an inch thick. The pallet is supplied with compressed air from its associated tugger, a small manually controlled, air-powered tractor. At the cask loading station, a 125-ton crane removes the cask from the air pallet platform, places it vertically on the rail car, and then moves it into a horizontal position.

If the cask meets DOT requirements, the tiedowns are applied and the rail car and cask are released for shipment to the repository.

C.3.1.4 Radwaste Treatment Process. Solid, liquid, and gaseous radwastes generated in the facility are treated for disposal in the repository or released to atmosphere to conform with the ALARA guidelines. Figure C-11 is the flow diagram of Radwaste Treatment Process for Case 1.

Liquid radwastes are collected from decontamination operation, pool water treatment ion exchange resin regeneration, the molten salt combustion unit offgas spray cooler, and the bottoms of the excess water vaporizer. These waste streams are collected in the radwaste feed tank and concentrated in the radwaste concentrator. The concentrate is fixed in cement mix. The overhead condensate from the radwaste concentrator is collected for recycle to process and pool water makeup. Excess condensate that cannot be recycled (abnormal case) is vaporized, filtered, and then released to the atmosphere through the stack.

Both TRU and non-TRU wastes are expected from this facility. The molten salt combustion unit burns all of the combustible radwastes, including waste resin from pool water treatment. This is the same unit that was described in the Reference Case. The treatment of these ashes and solid wastes discharged from the combustion unit is similar to that in the Reference Case and has been described in the subsection entitled "Miscellaneous radwaste fixation." The non-TRU LLW packages are sent to shallow land burial; the LLW and ILW TRU wastes are sent to the repository.

Offgases generated from the various operations in this facility, such as cask venting, water treating, decontamination, waste concentration, and combustion, are collected and treated together. First, a scrubber removes the soluble gases; then the scrubbed offgas is heated, filtered, and passed through an iodine adsorber to remove small amounts of iodine. Before the offgas is released to the stack, a final filter removes the residual particulates.

C.3.2. Spent Fuel/Waste Packaging

The design of canisters and containers for spent fuel and waste generated at nuclear facilities is currently part of the DOE-sponsored Commercial Waste Packaging Program, carried out by RHO.⁽¹⁷⁾ The program is designed to develop package acceptance criteria, container design criteria, prototype designs, and testing of prototype canisters. It is scheduled to be completed in FY-1982. Until the studies are completed, the preliminary designs for surface storage from the RHO spent fuel packaging facility are adopted by assuming that the canister design requirements for storage in a salt repository will not be significantly different from that for surface storage, except for canister material.

In Case 1, the following packages are required to encapsulate spent fuel and the waste generated at the facility:

- PWR spent fuel canisters
- BWR spent fuel canisters
- ILW waste canisters
- LLW waste containers
- Failed equipment canisters

Table C-17 gives the number size, weight, decay heat rate, overpack size, and MTHM equivalent of the packages expected from a 1,500 MTHM equivalent/yr throughput facility.

Table C-17
WASTE/PRODUCT PACKAGE DATA
CASE 1
(1,500 MTHM/yr)

Content and Type of Package	No. per Year	O.D. x Length, in.	Gross Weight, lb	Decay Heat, kW	Overpack O.D. x Length, in.	MTHM Equiv. per Package
Spent fuel canister						
BWR (3)	1,020	16 x 186	2,800	0.6	18 x 198	0.55
PWR (1)	2,040	14 x 186	2,400	0.5	16 x 198	0.46
ILW canister						
TRU waste	696	20 x 120	3,300	—	22 x 132	2.2
Failed equipment	75	20 x 120	—	—	22 x 132	20
LLW (TRU) drum	165	55-gal drum	840	—	83-gal drum	9
LLW (non-TRU) drum	858	55-gal drum	840	—	83-gal drum	1.7

C.3.2.1 PWR Spent Fuel Canister. The PWR spent fuel assemblies are packaged in canisters. The canister is a cylindrical carbon steel vessel fabricated from seamless steel pipe and forged seamless steel ellipsoidal caps, designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 2. The canister is fabricated from 14-inch O.D. pipe, with 3/8-inch wall and a length of 15-1/2 feet, exclusive of its lifting bail. The upper cap is equipped with a valve assembly used for evacuation of air and for the injection of helium gas.

The bottom design includes a skirt and flat bottom required to furnish base support during encapsulation and energy absorption in case of an accidental drop. A support cage inside the canister guides the assembly during insertion and furnishes lateral support during handling and transportation.

Because the encapsulation of fuel assemblies includes provisions for the detection of defective welds on canisters, and the consequent reencapsulation of the assemblies, no overpacks are required for the encapsulated fuel assemblies at the P/E facility. Overpacking might be required prior to disposal at the repository.

C.3.2.2 BWR Spent Fuel Canister. The BWR spent fuel assemblies are also packaged in canisters, three assemblies per canister. The canister is fabricated from 16-inch O.D. seamless carbon steel pipe with 3/8-inch wall and forged seamless steel ellipsoidal caps. The container is designed in accordance with the ASME Boiler and Pressure Vessel Code; the code requirements are the same as for the PWR canister. The total length, exclusive of the lifting bail, is 15-1/2 feet.

Inside the canister, a support cage guides the assemblies during insertion and provides lateral support for the stored assemblies during transit.

The BWR canister is designed with the same physical characteristics as the PWR canister at the top and bottom caps. Like the PWR canister, the BWR canister requires no overpacking.

C.3.2.3 ILW Waste Canisters. Solidified ILW is packaged in carbon steel canisters fabricated from 20-inch diameter, 3/8-inch wall pipe, 10 feet long. Overpacks, if required, are fabricated from 22-inch diameter carbon steel standard pipe, 11 feet long.

C.3.2.4 LLW Waste Containers. Solidified LLW (TRU or non-TRU) is packaged in 55-gallon carbon steel drums, DOT 17C or 17H. Any overpacking needed is in 83-gallon drums.

C.3.2.5 Failed Equipment Canister. Failed equipment, or scrap from failed canisters, is packaged in carbon steel canisters fabricated from 20-inch diameter, 3/8-inch wall pipe, 10 feet long. Overpacks for these containers are fabricated from 22-inch diameter carbon steel standard pipe, 11 feet long.

C.3.3 Facility Description

As indicated in the introductory description of Case 1, several modifications have been made to the RHO referenced facility because of the requirement that the encapsulation capacity be reduced, and because of the need of surge storage and shipping capability of the SFDS facility. The modifications are as follows:

- The fuel receiving, cask unloading, and fuel storage facilities have been reduced to one-half the size of the original facility
- The number of fuel encapsulation weld and test cells has been reduced from two each to one each

- Facilities for dry storage of canistered fuel have been added. These consist of a packaged spent fuel storage cell and its cooling system
- Facilities for loading canisters in a shipping cask, and the cask on a rail car are added
- A truck loading facility for shipment of solidified ILW and LLW waste is provided

The P/E facility is built on a hypothetical 150-acre site, located in the United States and centrally placed between the AFR and the repository.

Figure C-12 shows a plot plan for the facility. As required by Federal Regulation 10 CFR 73, the facilities are within a protected area and surrounded by physical barriers and an exclusion area.

All quality levels and standards of design used in the RHO facility for maintaining structural integrity and nuclear safety are adopted for the SFDS facility.

The following buildings are part of the P/E facility:

- Receiving and packaging building
- Administration building
- Security building
- Maintenance building
- Emergency generator building
- Boiler house
- Warehouse

Figure C-13 shows the building arrangement for the receiving and packaging building. A description of this building is given below.

C.3.3.1 Fuel Receiving and Cask Handling. There are three rail lines to handle the incoming casks. Each line has an area where the cask is prepared for unloading and where all impact absorbers and weather covers are removed. After these operations have been performed, the cask is moved to the offloading area for cask inspection, sampling, and monitoring for radioactivity.

Each cask offloading area is served by a 125-ton bridge crane, which raises the cask to the upright position, then transfers it to the wet receiving station. Here the cask atmosphere is sampled and the interior cavity cooled by flushing.

The cask unloading pool is used to remove fuel, under water, and to place the assemblies in baskets for transfer to the storage pool.

C.3.3.2 Surge Storage Pool. The surge storage pool connects to the cask unloading pool and to the encapsulation cell by means of a transfer canal. The fuels are kept in the basket during the surge storage period.

Storage pool and transfer canals have sufficient water depth so that the moving assemblies have at least 13 feet of shielding at all times. There are working corridors around the periphery of the pool and the transfer canals. There is also a gantry crane for handling the fuel baskets and an overhead bridge crane for routine maintenance tasks.

C.3.3.3 Encapsulation Cell. The encapsulation cell is designed to process the packaging and testing functions for a 10 MTHM/day rate. This represents a 100 percent overdesign capacity, which is required in order to allow for work interruption.

The operation of the cell is remotely controlled from operating galleries at the outside of the cell, with shielding windows located at intervals in the cell wall. The cell floors and the lower 2 feet of wall are lined with stainless steel; the rest of the walls and ceiling are coated with a sealer.

The cell is equipped with remotely operated jib hoists to move the canister through the process line in a semiautomatic mode. The overhead bridge crane serves as a standby tool for various support operations.

Other equipment consists of three transfer drawers for passing objects from the operating gallery into the cell, master-slave manipulators, a closed-circuit TV system, and other tools.

C.3.3.4 Special Function Cell. The special function cell is used for the repair of canisters and for handling damaged fuel assemblies. The facility is equipped to transfer the canister from the encapsulation cell to the special function cell. The transfer system includes a telescoping boom with a 5,000-pound hoisting capacity. Other equipment pertaining to this cell includes master-slave manipulators, canister cutting machines, and canister packaging equipment. Operations in the cell can be observed from the operating gallery through shielding windows.

C.3.3.5 Storage Cell. Canistered fuel is stored in the dry mode in a shielded cell which is connected to the encapsulation cell and the loadout facilities by means of an underground transfer tunnel. The cell has remotely operated equipment designed to put the canisters in their storage cubicles. Operations are controlled from the working gallery, with full view allowed by shielding windows. The cell has a stainless steel floor that extends 2 feet vertically into the wall, and is cooled by an independent ventilation system, located elsewhere in the facility.

C.3.3.6 Loadout Facilities. The loadout operations are described in Subsection C.3.1.3. The facilities for transferring the canister are shielded and remotely operated from the operating gallery, which has shielding windows for viewing.

Support facilities for loadout operations include personnel quarters and administrative operations for accountability and control of shipments.

C.3.3.7 Waste Treatment Area. The waste treatment area is dedicated to processing the solid, liquid, and gaseous radwaste generated at the facility, as described in Subsection C.3.1.4.

A remotely controlled console is used to blend the cement and waste in the proper ratios, and the blend is loaded into waste drums. The drums are transferred to a capping and inspection station, where an additional solidifying agent may be added to absorb excess water. The exterior of the drums are washed and decontaminated before shipping.

C.3.3.8 Waste Shipping Facility. Solidified waste is loaded into trucks in the waste shipping facility. Containers with activity levels higher than 10 mrem/hr are handled remotely in a cask and transferred from the cask into a shielded truck. Drums with activity less than 10 mrem/hr are handled by contact methods, but are also shipped in shielded trucks.

C.3.4 Operational Considerations

The principal operations at P/E facilities in Case 1 are:

- Receive casks of spent fuel, unload fuel into temporary storage, and return casks to the transportation system
- Move individual spent fuel assemblies from the fuel pool using a transfer fixture and place assemblies in the dryer
- Load one dried PWR or three dried BWR assemblies in a canister, weld its cap in place, and heat-treat the welded zone
- Inspect the quality of the canister closure, fill the canister with helium, and retest the canister for leaks

Figure C-12.
 P/E Facility
 Plot Plan
 Case 1
 Adapted from Rockwell Hanford
 conceptual design⁽¹⁵⁾

- Decontaminate the canisters, load the canisters in a cask, and ship the cask to the repository.
- Concentrate and stabilize the liquid wastes
- Incinerate the combustible solid wastes, fix the ashes and noncombustible solid wastes in a less leachable form, and package them for shipment

The SFDS concept of processing and encapsulation is basically the same as the Kaiser approach for spent fuel disposal. However, for the SFDS, the capacity per facility has been reduced; the SFDS assumes a facility throughput of 1,500 MTHM/yr and a system capacity of 5,000 MTHM/yr. The fuel receiving and shipping areas are capable of operating continuously; the P/E portion of the facility has been assumed to be operational only 300 days per year so as to permit time for maintenance and repair outages. At any time, as much as 300-MTHM-equivalent fuel can be accommodated in pool storage and 150-MTHM-equivalent fuel can be accommodated in a dry storage cell.

The annual material flow at each P/E facility for the 1,500 MTHM/yr throughput system is shown in Table C-18.

Table C-18
ANNUAL MATERIAL FLOW AT EACH P/E FACILITY
CASE 1
(1,500 MTHM/yr)

Material	Annual Number of Shipments	Total Units	Remarks
Incoming			
Spent PWRs	146	2,040	14 PWR/cask
Spent BWRs	90	3,060	34 BWR/cask
	<u>236 rail</u>	<u>5,100 assem.</u>	
Internal transfers			
PWR	510	2,040	4 PWR/basket
BWR	340	3,060	9 BWR/basket
	<u>850 baskets</u>	<u>5,100 assem.</u>	
Outgoing			
Spent fuel canister			
PWRs	227 rail	2,040 canisters	9 can./shipment
BWRs	114 rail	1,020 canisters	9 can./shipment
Non-TRU LLW	21 truck	858 drums	42 drums/shipment
TRU-LLW	4 truck	165 drums	42 drums/shipment
TRU-ILW	100 truck	696 canisters	7 can./shipment
Failed equipment	11 truck	75 canisters	7 can./shipment

C.3.4.1 Fuel Receiving. The fuel arrives in rail casks at the P/E facility after 10 years of cooling; therefore, the spent fuel will have cooled sufficiently so that water can be added to the cask cavity without further cooldown of cask internals. To be sure the fuel assemblies were not damaged during shipment, the cask is flushed and the flush is sampled and analyzed before the cask is lowered into the unloading pool. The operator transfers one assembly at a time from the cask into a fuel storage basket. The basket is moved from the pool before the empty cask is hoisted from the pool. The cask is decontaminated and prepared for return to the reactors. The total movement of fuel and wastes from the P/E facility is shown in Table C-18. On the average, five rail casks filled with unencapsulated fuel arrive each week from the AFRs, and seven rail casks per week are loaded with encapsulated spent fuel and shipped to the repository. Personnel allocations for each receiving operation are described in Subsection C.2.4.1.

The total operating staff for fuel receiving is shown in Table C-19.

Table C-19
TOTAL OPERATING STAFF FOR
FUEL RECEIVING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (5 x 4 shifts)	20
Day workers	4
Supervisors (1/shift + 1 day)	5
Total	<hr/> 29

C.3.4.2 Preparing Assemblies for Encapsulation. The operator moves a storage basket from its rack position to the transfer canal. One fuel assembly at a time is grappled, hoisted, and lowered into the transfer fixture. The transfer fixture drive is activated to move the assembly from the canal into the processing cell. An in-cell crane is used to move the assembly into one of three dryers. After a short time in the dryer, the fuel assembly is ready to be loaded into either a single-compartment PWR canister or a triple-compartment BWR canister. About six fuel assemblies per shift have to be transferred from the pool and dried in preparation for loading into a canister. Two to five canisters per shift have to be moved into the processing cell, depending upon the BWR/PWR mix.

The operating crew requirements for preparing assemblies for encapsulation are as follows:

Loading the transfer fixture (6 times)	1 person/shift
Loading/unloading the fixture and dryer (6 times)	1 person/shift
Transferring fresh canisters and canister caps into the cell (2 to 6 times)	1 person/shift
General duties (20 percent of above)	1 person/shift
Transferring canisters from the warehouse	1 person/shift

The total operating staff for preparing assemblies for encapsulation is shown in Table C-20.

Table C-20
TOTAL OPERATING STAFF FOR
PREPARING ASSEMBLIES FOR
ENCAPSULATION
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (3 x 4 shifts)	12
Day workers	2
Supervisors	0
Total	<hr/> 14

C.3.4.3 Canister Loading and Capping. The operator selects either a BWR canister or a PWR canister from the interim storage area in the cell and moves the canister to the carousel holding-rack for loading. One PWR or the first of three BWR fuel assemblies is loaded from the dryer into the canister. After the canister is filled with one PWR or three BWR assemblies, the carousel is rotated into the welding position. The canister cap is moved into position and clamped into place. Tack welds are made, the clamps retracted, and the continuous welding operation is started. On completion of the welding, the canister is moved to the heat treatment station for post-weld heat treatment.

The operating crew requirements for canister loading and capping are as follows:

Moving the canisters (2 to 6 per shift) to the loading carousel	1 person for 4 hours
Attaching the canister cap, tack welding, and continuous welding (2 to 6 units per shift)	1 person for 6 hours
Transferring canisters to heat treatment (2 to 6 units per shift)	1 persons for 4 hours
General duties	1 person for 2 hours
Supervision	1 person/shift

The total operating staff for canister loading and capping is shown in Table C-21.

Table C-21
TOTAL OPERATING STAFF FOR
CANISTER LOADING AND CAPPING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (3 x 4 shifts)	12
Day workers	0
Supervisors	0
Total	12

C.3.4.4 Canister Inspection and Testing. The operator moves the loaded canisters from the heat treatment station to the inspection station for ultrasonic examination. After examination, the operator moves the canister to the helium filling station where the canister is filled with helium. After helium filling, the canister is moved to the testing station for a leak test and then transferred to the decontamination station. Canisters of poor integrity are returned to the rework station where the canister caps are cut off and the fuel transferred to a new canister. The failed canister is chopped and the pieces dropped into a plain canister (without internal PWR or BWR spaces). The new canisters are cycled through the canister capping and welding section.

The operating crew requirements for canister inspection and testing are as follows:

Transferring the canister (2 to 6 per shift) to testing	1 person for 4 hours
Performing ultrasonic testing	1 person for 4 hours
Transferring the canister to the filling station	1 person for 4 hours
Filling the canister with helium	1 person for 4 hours
Testing the weld for leaks	1 person for 4 hours
Rejecting and transferring canisters back for rework	1 person for 1 hour (average)
Chopping the canister and packaging it for shipment (24 hr/can), based on 2 percent rejection rate or 1 can per week	

The total operating staff for canister inspection and testing is shown in Table C-22.

Table C-22
TOTAL OPERATING STAFF FOR
CANISTER INSPECTION
AND TESTING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (3 x 4 shifts)	12
Day workers	0
Supervisors (1 x 4 shifts)	0
Total	<hr/> 12

C.3.4.5 Canister Survey and Cask Loading. After decontamination, the canister is moved through the drying station, surveyed for surface contamination, and then loaded into a cask. When the cask is filled with nine canisters, it is closed and moved through decontamination, survey, and loading onto a railroad car.

The operating crew requirements for canister survey and cask loading are as follows:

Transferring the canister (2 to 6 per shift) to decontamination	1 persons for 4 hours
Transferring the canister to the dryer (2 to 6 per shift)	1 person for 4 hours
Moving the cask into position for loading (1 cask/day)	1 person for 2 hours
Collecting swipes of canisters and analyzing for contamination	1 person for 12 hours
Transferring the canister (2 to 6 per shift) to the cask	1 person for 4 hours
Installing the cask lids	1 person for 4 hours
Transferring and decontaminating the cask	2 persons for 4 hours

Loading the cask onto the transport vehicle	1 person for 4 hours
Completing the paperwork on the transfer	1 person for 2 hours
General duties (20 percent of the above)	1 person for 4 hours/shift

The total operating staff for the canister survey and cask loading is shown in Table C-23.

Table C-23
TOTAL OPERATING STAFF FOR
CANISTER SURVEY
AND CASK LOADING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (4 x 4 shifts)	16
Day workers	3
Supervisors (1 x 4 shifts)	4
Total	<hr/> 23

C.3.4.6 Concentrating and Stabilizing Liquid Wastes. Pool liquid wastes accumulate from the regeneration of pool cleanup resins and chemical decontaminants used in the canister cleanup. The wastes are concentrated by evaporation and stabilized in cement before shipment.

The operating crew requirements for concentration and stabilizing liquid wastes are as follows:

Operating the waste evaporator	2 persons for 8 hours
Blending wastes with the stabilizer and packaging the stabilized wastes	

The total staff for concentrating and stabilizing liquid wastes is shown in Table C-24.

Table C-24
TOTAL OPERATING STAFF FOR
CONCENTRATING AND
STABILIZING LIQUID WASTES
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers	0
Day workers	2
Supervisors	0
Total	<hr/> 2

C.3.4.7 Incinerating and Fixing All Solid Wastes. Solid wastes accumulate from areas prepared for contamination work from expendable items of protective apparel, small tools, and failed equipment. Molten salt combustion and clay or cement fixation are used and are expected to produce about 1 drum every 2 days and 3 canisters/day of solidified wastes to be shipped to the repository. In addition, about 3 drums/day are shipped to shallow land burial, as non-TRU wastes.

The operating crew requirements for incinerating and fixing all solid wastes are as follows:

Collecting and delivering waste	2 persons for 4 hr/day
Loading feed hoppers	1/2 person for 4 shifts
Operating process equipment	1/2 person for 4 shifts
Providing containers	1 person for 5 days
Providing bulk materials	1 person for 5 days
Preparing drums for shipment and general duties	1 person for 5 days

The total operating staff, plant operating staff, support staff, and overall staff for incinerating and fixing all solid wastes are shown in Tables C-25, C-26, C-27, and C-28, respectively.

Table C-25
TOTAL OPERATING STAFF FOR
INCINERATING AND FIXING
ALL SOLID WASTES
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (1 x 4 shifts)	4
Day workers	4
Supervisors	0
Total	<hr/> 8

Table C-26
TOTAL OPERATING STAFF REQUIREMENTS FOR A
CASE 1 P/E FACILITY
(1,500 MTHM/yr)

Operating Area	Number Required
Fuel receiving	29
Preparing assemblies for encapsulation	14
Canister loading and capping	12
Inspecting and helium testing of canisters	12
Decontaminating and shipping	23
Concentrating liquid wastes	2
Incinerating solid wastes	8
Total	<hr/> 100

Table C-27
TOTAL SUPPORT STAFF FOR A
CASE 1 P/E FACILITY
(1,500 MTHM/yr)

Support Area	Number Required
Maintenance (50 percent of operation)	50
Analytical (10 percent of operation)	10
Health and safety (10 percent of operation)	10
Utilities	8
Plant overhead (30 percent of operation)	30
Total	108

Table C-28
OVERALL TOTAL STAFF REQUIREMENTS FOR
A CASE 1 P/E FACILITY
(1,500 MTHM/yr)

Staff Type	Number Required
Operations	100
Support and overhead	108
Total	208

C.3.5 Cost Estimates

C.3.5.1 Capital Cost Estimate. The P/E facility capital cost estimate for Case 1 is developed from the Spent Fuel Receiving and Packaging Facility Conceptual Design Report, prepared by RHO and Kaiser Engineers in September 1978.⁽¹⁵⁾ The RHO Design Report is for a 3,300 MTHM/yr facility, as compared with a unit plant capacity of 1,500 MTHM/yr required for the SFDS. Therefore, each RHO subfacility was reviewed and adjusted for agreement within the required scope for the SFDS capacity.

The estimate for the Case 1 P/E facility follows the RHO Design Report subfacility breakdown at the field cost level. The architect-engineer services and owner's costs are then added to the total field costs. The capital cost estimate is presented in Table C-29.

Table C-29
CAPITAL COST ESTIMATE FOR A CASE 1 P/E FACILITY
(1,500 MTHM/yr)^(a)

Item	Cost \$ million/yr ^(b)
Field costs (including 20 percent contingency)	
Receiving and packaging building	
Fuel receiving and cask handling	28
Surge storage pool	20
Encapsulation	30
Storage cell	20
Waste treatment area	25
Loadout facilities and waste shipping	2
Building mechanical and electrical	25
Support areas	8
Vendor checkup and startup	2
Subtotal	160
Support facilities (including improvements to land)	37
Total field costs	197
Architect-engineer services @ 20 percent of 197	38
Owner's costs @ 30 percent of 235	70
Total capital cost	305

(a) The 5,000 MTHM/yr SFDS economic study requires the equivalent of 3-1/3 of the above P/E facilities.

(b) January 1979 dollars.

The required modifications to the RHO Design Report for the Case 1 P/E facility are discussed in detail in Subsection C.3.3. The major changes are in the receiving and packaging building (see Subsection C.3.3). To prepare the capital cost estimate for Case 1, the RHO subfacilities were reviewed and are modified as described below.

Improvement to land. Site development costs are reduced from the RHO estimate of \$5.9 million to reflect the deletion of 2,500 linear feet of railroad tracks and 30,000 ft² of paved road. The revised base of \$5.8 million is increased by \$0.3 million (10 percent per annum) to bring the basis from June 1978 to January 1979; a 20 percent contingency (\$1.2 million) is added to bring the total site development cost to \$7.3 million.

Fuel receiving and cask handling. Under the heading "Wet Receiving," the RHO Design Report includes preparation for offloading, wet receiving, and assembly unloading. The capacity of the RHO subfacility is reduced by 53 percent for the Case 1 subfacility. The RHO subfacility is valued at \$31.0 million. To reduce the cost, the following cost/capacity ratio is applied:

$$\text{\$ NEW} = \left[\frac{\text{Capacity New}}{\text{Capacity Old}} \right]^{0.5} \text{\$ OLD}$$

Using the five-tenths power equation lowers the cost by 30 percent to a value of \$22 million. Escalation from June 1978 to January 1979 at 10 percent per year adds \$1 million and the 20 percent contingency adds \$5 million. The total field cost of this subfacility is calculated to be \$28 million.

Surge storage pool. Under the heading "Lag Storage," the RHO Design Report includes a storage pool, a presentation pool, and interconnecting water-filled canals. Case 1 has a storage pool only which is 40 percent of the size of the RHO pool. Application of the five-tenths power equation yields a field cost for the SFDS pool of \$16.1 million. Escalation adds \$0.8 million (at 10 percent per year) and the 20 percent contingency adds \$3.1 million. The total field cost of this subfacility is calculated to be \$20 million.

Encapsulation. Under the heading "Packaging," the RHO Design Report includes three processing cells, two encapsulation cells, and one special function cell. Case 1 includes one encapsulation cell and one special function cell. For Case 1, the RHO subfacility is reduced 40 percent in size, resulting in a field cost of \$23.8 million. Since the encapsulation subfacility and the storage cell subfacility are very similar to the entire packaging subfacility in the RHO Design Report, the five-tenths cost/capacity factor is not applied. Escalation adds \$1.2 million (at 10 percent per annum) and the 20 percent contingency adds \$5 million. The total field cost of this subfacility is calculated to be \$30 million.

Storage cell. In Case 1, the storage cell is a shielded cell which stores canistered fuel in a dry mode. The cell is the same size as one of the encapsulation cells included in the RHO packaging subfacility.

For Case 1, the RHO subfacility is reduced in size by 60 percent to yield a field cost of \$15.9 million. Escalation adds \$0.8 million (at 10 percent per annum) and the 20 percent contingency adds \$3.3 million. The total field cost of this subfacility is calculated to be \$20 million.

Waste treatment area. The processing of solid, liquid, and gas radwaste takes place in this subfacility. The process is very similar to the radwaste incineration and cement immobilization subfacilities in the Reference Case P/E facility and is employed as the basis for Case 1 (\$27 million). Using the five-tenths cost/capacity equation lowers the cost by 5 percent and results in field cost, including escalation to January 1979, of \$21 million; the 20 percent contingency adds \$4 million. The total field cost of this subfacility is calculated to be \$25 million.

Load-out facilities and waste shipping. In the RHO Design Report, this subfacility is below ground, while in Case 1 it is above ground. This cost differential is assumed to be negligible. The RHO dollar value of \$1.6 million is adjusted to include escalation of \$0.1 million (at 10 percent per annum) and the 20 percent contingency adds \$0.3 million. The total field cost of this subfacility is calculated to be \$2 million.

Mechanical and electrical building. This subfacility is based on the RHO estimate (\$22 million) with the capacity reduced by 20 percent. Using the five-tenths cost/capacity equation lowers the cost by 11 percent and yields a field cost of \$20 million. Escalation added \$1 million (at 10 percent per annum) and the 20 percent contingency adds \$4 million. The total field cost of this subfacility is calculated to be \$25 million.

Support area: receiving and packaging building. The support area includes space for personnel, administration, equipment maintenance, storage, and supplies. The RHO estimate is \$6 million; escalation (at 10 percent per annum) adds \$0.3 million and the 20 percent contingency adds \$1.7 million. The total field cost for this subfacility is calculated to be \$8 million.

Vendor checkout and startup. An allowance of \$2 million is provided.

Support facilities. This subfacility includes the rail car and transport vehicle maintenance building, the general maintenance building, the stack and tunnel, instrumentation, communications system, the alarm system, utilities, etc. The RHO estimate is \$23.9 million; escalation (at 10 percent per annum) adds \$1.2 million and the 20 percent contingency adds \$4.9 million. The total field cost of these subfacilities is calculated to be \$30 million.

C.3.5.2 Operating Cost Estimate. The P/E facility for Case 1 essentially duplicates the front-end operations of the Reference Case P/E through the point where fuel is delivered into mechanical processing areas. Economies of staffing are predicted for fuel receiving operations in Case 1 because all fuel arrives by rail, whereas in the Reference Case fuel shipments are assumed to be half by truck and half by rail. However, additional personnel have to be provided in Case 1 in the mechanical processing areas to perform loading, helium filling, inspection, and decontamination operations on spent fuel canisters. The overall relative census requirements for Case 1 are estimated at about 25 percent of those for the Reference Case.

Average salary figures, cost factors for operating, maintenance supplies, utilities, and for unit canister costs are the same as for the Reference Case (Subsection C.2.5.2).

A summary of the Case 1 P/E facility operating cost estimate is shown in Table C-30.

Table C-30
OPERATING COST ESTIMATE FOR
A CASE 1 P/E FACILITY
(1,500 MTHM/yr)

Item	Cost \$ million/yr
Personnel 207 persons x \$24,000/yr	5.0
Operating/maintenance supplies 0.01 x \$305 million	3.1
Utilities 0.001 x \$305 million	0.3
Canisters 3,835 canisters x \$5,000	<u>19.2</u>
Total	27.6

C.3.5.3 Decommissioning Cost Estimate. Decommissioning of the Case 1 P/E facility is relatively free (compared with the Reference Case) of problems caused by residual radioactive materials or surface contamination. After removal of fuel-bearing materials, mechanical cleanup of loose particulates and some surface decontamination typical of storage pool decontamination are necessary. Beyond that initial phase, direct access to most equipment items should be possible. Demolition of structures should then be typical of any equivalent industrial structure. The decommissioning cost in Case 1 is taken as 10 percent of the total capital cost, less the owner's costs, to produce \$25 million/facility.

C.4 CASE 2

The P/E facility for Case 2 is designed to chop, voloxidize, and package spent fuel in steel canisters for emplacement in a salt repository. The following subsections describe the processing of LWR spent fuel and the treatment of the radioactive wastes incidental to the operation of the facility; the packaging of processed spent fuel and the treated radwastes; and the facilities required to support the above processes.

C.4.1 Process Description

The processing steps in the Case 2 P/E facility are shown in Figure C-14. As in Case 1, the facility is designed to process 10-year-old LWR spent fuel assemblies, received in rail casks from an AFR. The makeup of the fuel and processing capacity is identical to that described in Case 1 (see Subsection C.3.1).

After the spent fuel is received, the end fittings of the received spent fuel are cut off and encapsulated. The remaining portions of the fuel assemblies are then chopped and voloxidized. Upon voloxidation, the fuel fragments are collected, encapsulated in canisters, and then shipped to the repository.

C.4.1.1 Fuel Receiving, Offloading, and Handling. The fuel receiving, offloading, and surge storage system is the same as the one used in Case 1 and has been described in Subsection C.3.1.1. The unloaded spent fuel assemblies are transferred by the fuel conveyor from the transfer canal to the remote fuel cutting and voloxidation cell. In that cell, the fuel is force-dried and then sent to the fuel disassembly machine system (as mentioned in the Reference Case) for removal of end fittings and chopping of fuel assemblies. The chopped fuel is fed to the voloxidizer to release most of the volatile fission products. The voloxidized spent fuel is finally encapsulated in canisters and shipped to the repository.

C.4.1.2 Volatile Fission Products Recovery. Tritium, krypton, xenon, and iodine are partially released when the spent fuel is chopped, and additional amounts of these elements, as well as carbon-14 (as carbon dioxide), are released during voloxidation. These volatile fission products are recovered for isolation. The process flow sheet is shown in Figure C-15. The process is essentially the same as employed in the Reference Case, but without the dissolver offgas stream. The estimated percentage of the volatile fission product recovered from the spent fuel is:

$$\begin{array}{ll} \text{Tritium} \approx 100\% (4), (6) & \text{Iodine} \approx 65\% (6), (18) \\ \text{Kr-Xe} \approx 50\% (6), (18) & \text{Carbon-14} \approx 100\% \text{ (assumed)} \end{array}$$

The balance of volatile fission products remains in the processed spent fuel.

C.4.1.3 Processed Spent Fuel Encapsulation. Figure C-16 shows the mechanical flow of chopped fuel through the encapsulation step. The fuel encapsulation process is a semiautomatic production process. The voloxidized fuel is dropped into a vibrating feeder (which also serves as the surge storage of the voloxidized fuel) and then discharges from the feeder into canisters. The encapsulation process is similar to the one in Case 1 and consists of welding, ultrasonic testing, and heat-treating of the weld, helium filling, leak-testing, radiation survey, decontamination and drying, etc. Leaky canisters are overpacked, rather than reencapsulated, as is done in Case 1.

1 2 3 4

5 6 7 8

Figure C-15.
P/E Facility
Volatile Fission Products
Recovery Flow Diagram
Case 2

4 7 7 4

4 7 7 4

C.4.1.4 Canister Loadout. The loadout process for the encapsulated processed spent fuel and end fittings is the same as that for loadout of encapsulated spent fuel, as described in Case 1, Subsection C.3.1.3.

C.4.1.5 Radwaste Treatment Process. Both non-TRU and TRU wastes are expected to be produced by this facility. The treatment of the radwaste is shown in figure C-11 and is essentially as described in Case 1.

C.4.2 Spent Fuel/Waste Packaging

In Case 2, the following spent fuel/waste packages are produced:

- Canisters of chopped and voloxidized spent fuel
- Canisters of end fittings
- Canisters of ILW waste
- Drums of LLW waste
- Canisters of failed equipment

Table C-31 gives the number, size, weight, decay heat rate, overpack size, and MTHM equivalent of the packages expected from a 1,500 MTHM equivalent/yr facility.

C.4.2.1 Chopped and Voloxidized Fuel Canister. The chopped and voloxidized fuel is encapsulated in canisters fabricated from 12-inch O.D. carbon steel pipe, 10 feet in total length, with a 3/8-inch wall. The loading capacity is 0.62 MTHM equivalent of chopped fuel per canister. Damaged packages are overpacked in 14-inch-O.D. x 11-foot-long carbon steel canisters with 3/8-inch wall. These canisters are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 2, as stated in Case 1.

C.4.2.2 End Fittings Canister. The end fittings in this case are packaged in the same type of canister and overpack as those in the Reference Case. The sizes are shown in Table C-31. Each package contains about 3.3 MTHM equivalent of end fittings.

C.4.2.3 LLW and ILW Waste Packages. In addition to the TRU and non-TRU wastes generated in Case 1, the recovered tritium and iodine are fixed in cement and packaged in 55-gallon drums. The recovered krypton, xenon, and carbon dioxide gases (ILW waste), however, are contained in gas cylinders, and these are encapsulated in 22-inch canisters, as described in the Reference Case in Subsection C.2.2.4. The packaging of LLW and ILW has been described in Subsections C.3.2.3 and C.3.2.4.

C.4.2.4 Failed Equipment Canister. The failed equipment in this case is packaged in the same way as the failed equipment in Case 1. See Table C-31 for canister and overpack sizes.

Table C-31
WASTE/PRODUCT PACKAGE DATA
CASE 2
(1,500 MTHM/yr)

Content and Type of Package	No. per Year	O.D. x Length, in.	Gross Weight, lb	Decay Heat, kW	Overpack O.D. x Length, in.	MTHM Equiv. per Package	Remarks
Chopped fuel canister	2,421	12 x 120	2,550	0.68	14 x 132	0.62	
End fitting canister	456	20 x 120	1,300	0.10	22 x 132	3.3	
Kr-Xe-Co ₂ canister	17	22 x 120	—	—	—	86	6 gas cylinders per canister
ILW canister							
TRU waste	1,095	20 x 120	3,300	—	22 x 132	1.4	
Failed equipment	75	20 x 120	—	—	22 x 132	20	
LLW (TRU) drum	255	55-gal drum	840	—	83-gal drum	5.9	
LLW (non-TRU) drum	933	55-gal drum	840	—	83-gal drum	1.6	

C.4.3 Facility Description

The P/E facility is built on a hypothetical 150-acre site, located in the United States and centrally placed between the AFR and the repository. Figure C-17 shows the plot plan of the P/E facilities. This plot plan is located within a protected area surrounded by physical barriers and an exclusion area, and meets the same federal regulations as required in Case 1. The facility includes the following major buildings.

- Spent fuel receiving, processing, and packaging building
- Administration building
- Security building
- Maintenance building
- Emergency generator building
- Boiler house
- Warehouse

Since most operations are conducted within the spent fuel receiving, processing, and packaging building, only the installation within this building is described below. Figure C-18 shows the building arrangement.

C.4.3.1 Fuel Receiving and Cask Handling. This facility is designed to function in exactly the same way as the one in Case 1. For details, see the subsection entitled "Fuel Receiving and Cask Handling" in Subsection C.3.3.1.

C.4.3.2 Surge Storage Pool. This facility is designed to function in exactly the same way as the one in Case 1. For details, see the subsection entitled "Surge Storage Pool" in Subsection C.3.3.2.

C.4.3.3 Fuel Cutting and Voloxidation Cell. The spent fuel assembly conveyor, dryer, fuel disassembly machine system, and the voloxidizer are located in this cell. All of these are operated by remote control from the operating gallery. Shielded windows and viewing equipment are provided as required to aid in control of the operation, and remotely operated carts move canisters of end fittings and voloxidized spent fuel to the encapsulation cell. An overhead bridge crane is used for maintenance.

C.4.3.4 Encapsulation Cell. The design of this remotely operated cell is similar to the design used in Case 1. Provisions have been made to account for the following differences:

- The size and quantity of canisters
- The type of material in the canister — loose pieces and powder instead of fuel assemblies
- Overpacking leaky or unsatisfactory canisters rather than repacking them

C.4.3.5 Volatile Fission Products Recovery Cell. This cell houses the system to recover and package volatile fission products, such as titanium, iodine, krypton-xenon, and carbon-14.

C.4.3.6 Storage Cell. Surge storage of the encapsulated canisters is in the dry mode, as is the case for Case 1. For details, see the subsection entitled "Storage Cell" in Subsection C.3.3.5.

C.4.3.7 Loadout Facility. The loadout facility is the same as in Case 1. For details, see Subsection C.3.3.6.

C.4.3.8 Waste Treatment Area. This area is dedicated to the radwaste treatment process (Subsection C.4.1.5). The design is similar to that of Case 1. For details, see Subsection C.3.3.7.

C.4.3.9 Waste Shipping Facility. This facility is similar to that of Case 1. For details, see Subsection C.3.3.8.

C.4.4 Operational Considerations

Under the Case 2 option, spent fuel will be chopped, voloxidized, and packaged for emplacement in the repository. The principal operations at the P/E facility are:

- Receive rail casks containing spent fuel, unload fuel into temporary storage, and return casks to the transport system
- Move individual fuel assemblies from the pool, using a transfer fixture, and deliver them to a mechanical shear in the process cell
- Shear off the leading end fitting and let the fitting fall into a waste canister, chop the fuel bundle into short lengths and let the pieces fall into a feed hopper supplying the voloxidizer and push the trailing end fitting into a waste container
- Voloxidize the chopped fuel tube segments, transfer the materials leaving the voloxidizer in a separate waste canister, and temporarily attach a canister cap welding it in place
- Heat treat the welded area, inspect the canister closure ultrasonically, fill the canister with helium, and leaktest the closure
- Decontaminate the canister, load the cask, and ship them to a repository
- Concentrate, stabilize, and prepare the liquid wastes for shipment to a repository
- Incinerate the combustible solid wastes, fix the ashes and noncombustible solid wastes in a less leachable form, and package them all for shipment

The SFDS concepts for processing encapsulation are similar to Case 1 except that shearing, leaching, and voloxidation steps have been added. Those processing steps are basically the same as those selected for the Reference Case. At any time, as much as 300 MTHM-equivalent-fuel can be in pool storage and 150 MTHM-equivalent-fuel residue can be in the dry storage cell.

The flow of fuel, product, and waste materials for the 1,500 MTHM equivalent throughput system is shown in Table C-32.

C.4.4.1 Fuel Receiving. The fuel arrives in rail casks at the P/E facility after 10 years' cooling; therefore, the spent fuel has cooled sufficiently so that water can be added to the cask cavity without further cooldown of cask internals. To be sure the fuel assemblies were not damaged during shipment, the cask is flushed and the flush is sampled and analyzed before the cask is lowered into the unloading pool. The operator transfers one assembly at a time from the cask into a fuel storage basket. The basket is moved from the pool before the empty cask is hoisted from the pool. The spent fuel cask is decontaminated and prepared for return to the reactors. The total movement of fuel and wastes from the P/E facility is shown in Table C-32. On the average, five rail casks filled with unencapsulated fuel arrive each week from the AFRs and four rail casks per week are loaded with canisters of voloxidized, chopped spent fuel for shipment to a repository. Personnel allocations for each receiving operation have been described under Section C.2.4.

- (24) CONTROL ROOM
- (25) PERSONNEL AREA
- (26) ELEVATOR
- (27) STAIRWAY
- (28) CONTACT EQUIPMENT HANDLING AREA
- 29) COLD MAINTENANCE SHOP
- (30) EQUIPMENT LOADING AREA
- (31) ANALYTIC LABS
- (32) FILTER NICHE
- (33) REMOTE MAINTENANCE CELL
- (34) TRANSFER TUNNEL
- (35) CASK TEST AND DECON STATION
- (36) HVAC EQUIPMENT ROOM

AREA DESCRIPTION

- 1) SPENT FUEL RECEIVING STATION
- 2) CASK TEST AND COOLDOWN STATION
- 3) CASK DECON PIT
- 4) CASK UNLOADING POOL
- 5) SPENT FUEL STORAGE POOL
- 6) LEAKY SPENT FUEL STORAGE POOL
- 7) FUEL TRANSFER CANAL
- 8) FUEL CUTTING AND VOLOXIDATION CELL, REMOTE
- 9) ENCAPSULATION CELL, REMOTE
- 10) ENCAPSULATED SPENT FUEL TRANSFER TUNNEL
- 11) ENCAPSULATED SPENT FUEL DRY STORAGE ROOM
- 12) CASK LOADING CELL
- 13) ENCAPSULATED SPENT FUEL SHIPPING STATION
- 14) COOLING/VENTILATION FACILITY ROOM
- 15) VOLATILE FISSION PRODUCTS RECOVERY CELL
- 16) RADWASTE PROCESSING AREA
- 17) PACKAGED RADWASTE STORAGE ROOM
- 18) RADWASTE LOADING STATION
- 19) LOADING DOCK
- 20) NEW CANISTERS AND DRUMS STORAGE ROOM
- 21) CANISTER SUPPLY SHAFT
- 22) OPERATING GALLERY
- 23) POOL WATER TREATMENT AREA

Figure C-18.
P/E Facility
Building Arrangement
Case 2

1 2 3 4

1 2

Table C-32
ANNUAL MATERIAL FLOW AT EACH P/E FACILITY
CASE 2
(1,500 MTHM/yr)

Material	Annual Number of Shipments	Total Units	Remarks
Incoming			
Spent PWRs	146	2,040	14 PWR/cask
Spent BWRs	90	3,060	34 BWR/cask
	<u>236 rail</u>	<u>5,100 assem.</u>	
Internal transfers			
PWR	510	2,040	4 PWR/basket
BWR	340	3,060	9 BWR/basket
	<u>850 baskets</u>	<u>5,100 assem.</u>	
Outgoing			
Chopped spent fuel	173 rail	2,421 canisters	14 can./cask
Hulls	51 rail	456 canisters	9 can./cask
Non-TRU LLW	21 truck	858 drums	42 drums/shipment
TRU-LLW	6 truck	255 drums	42 drums/shipment
TRU-ILW	157 truck	1,095 canisters	7 can./shipment
Radioactive gases	1 rail	5 canisters	9 can./cask
Failed equipment	11 truck	75 canisters	7 can./shipment

The operating crew requirements for fuel receiving are as follows:

Unloading/loading rail casks	1 person for 9 hours
Decontaminating shipping casks	3 persons for 8 hours
Transferring fuel baskets to the storage pool	2 persons for 2 hours
Temporary removal of basket from pool before moving casks	1 person for 1/2 hour
Pool cooling and cleanup	3 persons for 8 hours/day
Accountability	2 persons for 8 hours/day
General duties (20 percent of above)	2 persons for 8 hours/day
Supervision	1 person shift for 1 day

The total operating staff for fuel receiving is shown in Table C-33.

Table C-33
TOTAL OPERATING STAFF FOR
FUEL RECEIVING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (5 x 4 shifts)	20
Day workers	4
Supervisors (1 /shift + 1 day)	5
Total	<hr/> 29

C.4.4.2 Preparing the Fuel Assembly for Chopping. The operator moves a basket from its rack position to the transfer canal. One fuel assembly at a time is grappled, hoisted, and lowered into the transfer fixture. The transfer fixture drive is activated to move the assembly from the canal into the shear located in the processing cell.

The operating crew requirements for preparing the assembly for chopping are as follows:

Loading the transfer fixture (6 times)	1 person for 8 hours
Transferring fresh canisters for hulls, end fittings and canister caps into the cell (2 to 6 times)	2 persons for 8 hours
General duties (20 percent of above)	1 person for 4 hours
Transferring canisters from warehouse	2 persons for 8 hours

The total operating staff for preparing the assembly for chopping is shown in Table C-34.

Table C-34
TOTAL OPERATING STAFF FOR
PREPARING ASSEMBLY FOR CHOPPING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (4 x 4 shifts)	16
Day workers	2
Supervisors	0
Total	<hr/> 18

C.4.4.3 Mechanical Processing. One fuel assembly at a time is fed into the shear where the end fittings are removed and diverted to a separate waste container. The remainder of the assembly is chopped and falls into the feed hopper upstream of the voloxidizer. The feed hopper fills in three hours and the end fittings container fills in 16 hours at a 1,500 MTHM throughput.

The operating crew requirements for mechanical processing are:

Shearing of fuel	1 person for 8 hours
Moving waste canisters into position (below shear and below voloxidizer)	1 person for 4 hours
General duties	1 person for 3 hours

The total operating staff for mechanical processing is shown in Table C-35.

Table C-35
TOTAL OPERATING STAFF FOR
MECHANICAL PROCESSING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (2 x 4 shifts)	8
Day workers	0
Supervisors (1 x 4 shifts)	4
Total	<hr/> 12

C.4.4.4 Voloxidation and Encapsulation. Offgases released during the heating of fuel materials contain a fraction of the volatile fission products of tritium, iodine, krypton, xenon, and an activation product of carbon contained in the fuel. Tritium is isolated as tritiated water combined with iodine as barium iodate and fixed in cement. Krypton, xenon, and carbon dioxide are isolated as pressurized gases in standard gas cylinders and then packaged in canisters (6 cylinders per canister). The remote operations include packaging, leak-testing, decontamination, and loadout of recovered gaseous fission products. (See Table C-32 for estimated production rates of various canisters.)

The operating crew requirements for voloxidation and encapsulation are as follows:

Operating the control room	2 persons/shift
Handling inbound containers	1 person on days for 5 days
Performing mechanical operations	2 persons/shift for 4 shifts
Loadout and shipping of waste packages	2 persons on days for 5 days
Providing bulk materials	2 persons on days for 5 days

The total operating staff for voloxidation and encapsulation is shown in Table C-36.

Table C-36
TOTAL OPERATING STAFF FOR
VOLOXIDATION AND ENCAPSULATION
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (4 x 4 shifts)	16
Day workers	5
Supervisors (1 x 4 shifts)	4
Total	<hr/> 25

C.4.4.5 Heat Treating and Canister Testing. After the waste has been encapsulated, the canister carousel rotates into a heat treatment position, then to an ultrasonic testing position. The canister is then transferred to a helium filling station, and a helium leak test is conducted after filling.

The operating crew requirements for heat treating and canister testing are as follows:

Rotating canisters to the heat treatment station positioning, heat treating, and ultrasonic testing	1 person for 8 hours/shift
Moving canisters to the testing station	1 person for 4 hours/shift
Filling canisters with helium, and performing helium leak testing	1 person for 8 hours/shift
General duties	1 person for 4 hours/shift

The total operating staff for heat treating and canister testing is shown in Table C-37.

Table C-37
TOTAL OPERATING STAFF FOR
HEAT TREATING AND CANISTER TESTING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (3 x 4 shifts)	12
Day workers	0
Supervisors	0
Total	12

C.4.4.6 Canister Survey and Cask Loading. After decontamination, the canister is moved through the drying station and then surveyed for surface contamination. After survey, the canister is loaded into a cask. When the cask is filled, the cask is closed and moved through decontamination survey and finally loaded on a railroad car for offsite shipment.

The operating crew requirements for canister survey and cask loading are as follows:

Transferring canister (3/shift to decontamination)	1 person for 3 hours
Transferring canister to dryer (3/shift)	1 person for 3 hours
Moving cask into position for loading (1 cask every 1-1/2 days)	1 person for 2 hours
Collecting swipes of canisters and analyzing for contamination	1 person for 9 hours
Transferring canisters (3/shift) to casks	1 person for 4 hours
Installing cask lids	1 person for 4 hours
Transferring and decontaminating the casks	2 persons for 4 hours
Loading casks on the transport vehicle	1 person for 4 hours
Completing paperwork on transfers	1 person for 4 hours
General duties (20 percent of above)	1 person for 4 hours/shift

The total operating staff for canister survey and cask loading is shown in Table C-38.

Table C-38
TOTAL OPERATING STAFF FOR
CANISTER SURVEY AND CASK LOADING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (3 x 4 shifts)	12
Day workers	3
Supervisors (1 x 4 shifts)	4
Total	<hr/> 19

C.4.4.7 Concentrating and Stabilizing the Liquid Wastes. Pool liquid wastes accumulate from regeneration of pool cleanup resins and chemical decontaminants used in canister cleanup. The wastes will be concentrated by evaporation and stabilized in concrete before shipment.

The operating crew requirements for concentrating and stabilizing the liquid wastes are as follows:

Operating waste evaporator blending wastes with stabilizer and packaging	2 persons for 8 hours
---	-----------------------

The total operating staff for concentrating and stabilizing the liquid wastes is shown in Table C-39.

Table C-39
TOTAL OPERATING STAFF FOR
STABILIZING LIQUID WASTES
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers	0
Day workers	2
Supervisors	0
Total	<hr/> 2

C.4.4.8 Incinerating Combustible Solid Wastes and Fixing All Solid Wastes. Solid wastes accumulate from areas prepared for decontamination work, from expendable items of protective apparel, and from small tools and failed equipment. Molten salt combustion and cement fixation are used and produce about 1 drum/day of solidified waste which is then loaded for shipment to the repository. In addition, about 3 drums/day of non-TRU waste is loaded for shipment to shallow land burial.

The operating crew requirements for incinerating combustible solid wastes and fixing all solid wastes are as follows:

Waste collection and delivery	2 persons/day for 2 hours/day
Loading feed hoppers	1/4 person for 4 shifts
Operating process equipment	1/4 person for 4 shifts
Providing containers	1/2 person for 5 days
Providing bulk materials	1/2 person for 5 days
Preparing drums for shipment and general duties	1/2 person for 5 days

The total operating staff for incorporating combustible solid waste and fixing all solid wastes is shown in Table C-40.

Table C-40
TOTAL OPERATING STAFF FOR
INCINERATING COMBUSTIBLE SOLID WASTES
AND FIXING ALL SOLID WASTES
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (1 x 4 shifts)	4
Day workers	2
Supervisors	0
Total	<u>6</u>

The overall personnel requirements per operating area for a P/E facility are shown in Table C-41.

Table C-41
TOTAL OPERATING STAFF REQUIREMENTS
FOR A CASE 2 P/E FACILITY
(1,500 MTHM/yr)

Operating Area	Number Required
Fuel receiving	29
Preparing fuel for chopping	18
Mechanical processing	12
Voloxidation and encapsulation	25
Heat treating and testing	12
Preparing waste for offsite shipment	19
Stabilizing liquid wastes	2
Incinerating solid wastes	6
Total	<u>123</u>

In addition, the total support personnel for the P/E facility are shown in Table C-42.

Table C-42
TOTAL SUPPORT STAFF FOR
A CASE 2 P/E FACILITY
(1,500 MTHM/yr)

Support Area	Number Required
Maintenance (50 percent of operations)	60
Analytical (10 percent of operations)	12
Health and safety (10 percent of operations)	12
Utilities	8
Plant overhead (30 percent of operations)	40
Total	<u>132</u>

The overall total personnel requirements for a 1,500 MTHM/yr P/E facility are shown in Table C-43.

Table C-43
OVERALL TOTAL STAFF REQUIREMENTS
FOR A CASE 2 P/E FACILITY
(1,500 MTHM/yr)

Personnel	Number Required
Operations	123
Support and overhead	<u>132</u>
Total	255

C.4.5 Cost Estimates

C.4.5.1 Capital Cost Estimate. The P/E facility capital cost estimate for Case 2 was developed from the Case 1 and the Reference Case capital cost estimates. The facility in Case 2 is designed to chop, voloxidize, and package spent fuel in steel canisters at the rate of 1,500 MTHM/yr.

The estimate for the Case 2 P/E facility follows the Case 1 subfacility breakdown at the total field cost level. The results of the Case 2 P/E facility capital cost estimate are presented in Table C-44.

Table C-44
CAPITAL COST ESTIMATE FOR A CASE 2 P/E FACILITY
(1,500 MTHM/yr)^(a)

Item	Cost \$ million/yr ^(b)
Field costs (including 20 percent contingency)	
Fuel receiving and cask handling	28
Surge storage pool	20
Fuel cutting and voloxidation cell	29
Encapsulation cell	31
Volatile fission products recovery cell	40
Storage cell	20
Loadout and waste shipping	2
Waste treatment area	25
Building mechanical and electrical	26
Support areas	8
Vendor checkup and startup	2
Subtotal	231
Support facilities (including improvements to land)	37
Total field costs	268
Architect-engineer services @ 20 percent of 268	54
Owner's costs @ 30 percent of 322	98
Total capital cost	420

(a) The 5,000 MTHM/yr SFDS economic study requires the equivalent of 3-1/3 of the above P/E facilities.

(b) January 1979 dollars.

Required modifications to the Reference Case and Case 1 facilities are discussed in detail in Subsection C.4.3. The impacts of these modifications on the cost of the Case 2 P/E facility are given below.

Improvement to land. A provision of \$7 million is made in land improvement — the same as for Case 1. This sum is included under the heading "Support Facilities".

Receiving, processing, and packaging building. The following sums were provided for operations within the receiving, processing, and packaging building:

Fuel receiving and cask handling. \$28 million, the same as for Case 1.

Surge storage pool. \$20 million, the same as for Case 1.

Fuel cutting and voloxidation cell. \$29 million, the same as for the Reference Case voloxidation cell.

Encapsulation cell. \$31 million. This cell is similar to the encapsulation cell in Case 1, but the capacity is increased by 10 percent in Case 2 for processing of the end fittings. To allow for the capacity increase, the five-tenths cost/capacity ratio formula was applied, which raised the estimated cost by 5 percent from the \$30 million for Case 1.

Volatile fission products recovery cell. \$40 million, the same as for the Reference Case gas recovery subfacility.

Storage cell. \$20 million, the same as for Case 1.

Loadout and waste shipping facility. \$2 million, the same as for Case 1.

Waste treatment area. \$25 million, the same as for Case 1.

Support areas. \$8 million, the same as for Case 1.

Support facilities. \$30 million, the same as for Case 1.

C.4.5.2 Operating Cost Estimate. The P/E facility for Case 2 essentially duplicates operations for Case 1, with additional processing operations for voloxidation and recovery of portions of the volatile radioactive gases. A careful review of all operating steps led to a census increase for Case 2 of about 23 percent over Case 1.

Average salary figures, cost factors for operating and maintenance supplies and for utilities, and unit canister costs are the same as for the Reference Case (Subsection C.2.5.2), and are shown in Table C-45.

Table C-45
OPERATING COST ESTIMATE FOR
A CASE 2 P/E FACILITY
(1,500 MTHM/yr)

Item	Cost \$ million/yr
Personnel	
255 persons x \$24,000/yr	6.1
Operating/maintenance supplies	
0.01 x \$420 million	4.2
Utilities	
0.001 x \$420 million	0.4
Canisters	
4.042 canisters x \$5,000	20.2
Total	30.9 (or 31.0)

C.4.5.3 Decommissioning Cost Estimate. Case 2 is considered to be a simple expansion of Case 1, where residual contamination is essentially by dry particulates. Thus, no extensive general use of chemical decontaminants should be required, although the fuel shearing, voloxidation, and storage pool cleanup equipment may have to be chemically decontaminated. However, most of the cost for decommissioning this P/E facility will be for demolition of the facilities. The decommissioning cost in Case 2 is set at 10 percent of the total capital costs, less the owner's costs, to produce \$35 million.

C.5 CASE 3

The P/E facility for Case 3 is designed to process spent fuel beyond the chopping and voloxidation steps carried out in Case 2 by additional processing steps (dissolution, concentration, calcination, and vitrification), before the final encapsulation. Process description, packaging of the spent fuel and the related wastes, and the installation provided in the facility are described below.

C.5.1 Process Description

Figure C-19 shows the overall block flow diagram of Case 3. The spent fuel receives additional treatments besides those described in Case 2. The chopped and voloxidized fuel is dissolved to separate the hulls from the fuel material. Dissolved spent fuel is concentrated, calcined, vitrified, and encapsulated in canisters. After encapsulation, inspection, and decontamination, the canisters are shipped to the repository for emplacement.

C.5.1.1 Fuel Receiving and Offloading. The fuel receiving and offloading steps are similar to those in Case 1 (see Subsection C.3.1.1 for details). The handling of the spent fuel up to the voloxidizer has been described for Case 2 in Subsection C.4.1.1.

C.5.1.2 Volatile Fission Products Recovery. Tritium, krypton, xenon, iodine, and carbon-14 are recovered from the voloxidizer offgas and from the dissolver offgas. The recovery flow sheet is the same as the one used in the Reference Case (Figure C-4), and has been described in great detail in the several subsections in Subsection C.2.1.2.

C.5.1.3 Fuel Dissolution and Hulls Treatment. The voloxidized fuel is dissolved in a nitric acid solution, and the hulls (the metallic wastes) are rinsed, neutralized, rinsed, compacted, and then encapsulated in the same way as described in the Reference Case in the subsection entitled "Chop-leach residue" in Subsection C.2.1.2.

C.5.1.4 Dissolved Spent Fuel Concentration, Calcination, and Vitrification. The dissolved spent fuel is concentrated, spray-calcined, vitrified, and encapsulated in a canister. The flow sheet for this concentration and fixation process is similar to that for HLW, as can be seen in part of Figure C-2. The vitrification of this high-uranium-content calcine has not been verified experimentally, and the calcine-to-glass-frit ratio for vitrification has not been established. The SFDS has assumed that a suitable glass frit is available and a homogeneous glass can be obtained with 3-to-1 frit-to-calcine weight ratio for the vitrification process.

A criticality analysis is required during the detailed design stage to establish the geometric constraint, if any, of all the equipment used in this system.

C.5.1.5 Encapsulation of Vitrified Spent Fuel. Figure C-20 shows the mechanical flow diagram of the encapsulation of vitrified spent fuel. The encapsulation steps are similar to those in the Reference Case and have been described in detail in "HLW encapsulation" in Subsection C.2.1.2.

C.5.1.6 Radwaste Treatment Process. The waste treatment process used in this case is similar to that of Case 1 (Figure C-11), including the offgas treatment portion. An acid recovery system is needed to recover the acid from the concentrator overhead by fractionation for reuse in the dissolution process.

1 1 1 1

1 1 1 1

1 2 3 4

5

6

C.5.2 Spent Fuel/Waste Packaging

In Case 3, the following types of spent fuel/waste packages are produced:

- Canisters of vitrified spent fuel
- Canisters of cladding hulls and end fittings
- Canisters of ILW waste
- Drums of LLW waste
- Canisters of failed equipment

Table C-46 gives the number, size, weight, decay heat rate, overpack size, and MTHM equivalent expected from a 1,500 MTHM equivalent/yr throughput facility.

C.5.2.1 Vitrified Spent Fuel Package. The vitrified spent fuel is encapsulated in canisters fabricated from 20-inch carbon steel pipe, 10 feet long, with a 3/8-inch wall. The loading capacity is 0.27 MTHM equivalent of vitrified spent fuel per canister. Overpacks for leaky canisters are made of 22-inch carbon steel pipe, 11 feet long, with a 3/8 inch wall. These canisters are designed and fabricated in accordance with the same codes as are described in Subsection C.3.2.1.

C.5.2.2 Cladding Hulls and End Fitting Packages. As in the Reference Case, both the cladding hulls and fittings are encapsulated in the same 20-inch-diameter canister and overpacked in 22-inch carbon steel canisters as described in the preceding section. The loading capacities for the hulls and the end fittings are 2.2 and 3.3 MTHM equivalent per canister, respectively.

C.5.2.3 LLW and ILW Waste Packages. The types of LLW and ILW generated by the operation and maintenance of the Case 3 facility are the same as those generated by the Case 2 facility; the only difference is the quantity. Descriptions of these waste packages can be found in Subsections C.3.2.3 and C.3.2.4.

C.5.2.4 Failed Equipment Packages. The failed equipment from this case is packaged in the same way as the failed equipment in Case 1. See Table C-46 for canister and overpack sizes.

Table C-46
WASTE/PRODUCT PACKAGE DATA
CASE 3
(1,500 MTHM/yr)

Content and Type of Package	No. per Year	O.D. x Length, in.	Gross Weight, lb	Decay Heat, kW	Overpack O.D. x Length, in.	MTHM Equiv. per Package	Remarks
Spent fuel vitrified canister	5,556	20 x 120	3,880	0.3	22 x 132	0.27	
Hulls canister	690	20 x 120	4,200	0.46	22 x 132	2.2	
End fittings canister	456	20 x 120	1,300	0.1	22 x 132	3.3	
Kr-Xe-CO ₂ canister	11	22 x 120	—	—	—	132	6 gas cylinders per canister
ILW canister							
Radwaste	1,683	20 x 120	3,300	—	22 x 132	0.9	
Failed equipment	120	20 x 120	—	—	22 x 132	12.5	
LLW (TRU) drum	369	55-gal drum	840	—	83-gal drum	4	
LLW (non-TRU) drum	880	55-gal drum	840	—	83-gal drum	1.7	

C.5.3 Facility Description

The hypothetical location of Case 3 facility is the same as that of Case 1. Figure C-21 shows the plot plan of the facility. All building protection barriers and rail and truck roads are identical to those in Case 2, except that the receiving, processing, and packaging building is expanded to accommodate the spent fuel dissolver and concentrator, the hulls compaction system, the spray calciner, the continuous ceramic melter, the acid recovery system and the associated auxiliary equipment.

Since most of the operations are conducted within the spent fuel receiving, processing, and packaging building, only the installations within this building are mentioned below. Figure C-22 shows the layout of the receiving, processing, and packaging building.

C.5.3.1 Fuel Receiving and Cask Handling. This facility is designed in the same way as the one in Case 1 (see Subsection C.3.3.1).

C.5.3.2 Surge Storage Pool. This surge pool is the same as the one used in Case 1 (see Subsection C.3.3.2).

C.5.3.3 Fuel Cutting and Voloxidation Cell. This cell is the same as the one used in Case 2 (see Subsection C.4.3.4).

C.5.3.4 Volatile Fission Products Recovery Cell. This cell is the same as the one used in Case 2 (see Subsection C.4.3.5).

C.5.3.5 Fuel Dissolution, Concentration, and Hulls Encapsulation Cell. This new cell is for the fuel dissolver, dissolved fuel concentrator, and hulls compaction and encapsulation operations. Because the operations are remotely controlled, all the remote operating provisions are included.

C.5.3.6 Spent Fuel Calcination, Vitrification, and Encapsulation Cell. This is another cell that is not included in Case 2 and requires remote operation of the equipment within the cell. All the equipment related to the calcination, vitrification, and encapsulation of the processed spent fuel is located in this cell.

C.5.3.7 Storage Cell. This is the surge storage of the encapsulated processed spent fuel, end fittings, and cladding hulls. The design is based on the same dry storage mode used in Case 1, and is large enough to provide the surge capacity, as specified in Item 7, Subsection A-11, of Appendix A.

C.5.3.8 Loadout Facility. This facility is designed in the same way as the one in Case 1; its shipping capacity, however, is greater.

C.5.3.9 Waste Treatment Area. This area is similar to the one in Case 1, but is expanded to accommodate acid recovery.

C.5.3.10 Waste Shipping Facility. This facility is similar to the one in Case 1; its capacity, however, is greater.

C.5.4 Operational Considerations

The principal operations for the P/E facility in Case 3 are:

- Receive casks of spent fuel, unload and transfer fuel into storage, and return casks to the transportation system
- Transfer individual spent fuel assemblies to the mechanical processing equipment, shear off end fittings close to the active fuel pellets, segregate end fittings in separate hull containers, and chop fuel into short lengths in preparation for acid leaching
- Dissolve chopped fuel materials in a nitric acid solution
- Solidify and vitrify the dissolved fuel materials in a stable form for long-term storage in salt repositories
- Compact chopped fuel residues, package compact residues, and package uncompacted end fittings of the fuel assemblies
- Incinerate combustible solid wastes, fix the ashes and noncombustible solid wastes in a "less leachable" form, and package them all for shipment
- Treat offgases, recover radioactive gases, and package them for shipment

The SFDS concept of partial processing of spent fuel for Case 3 is based on the BNFP facility design and is sized to accommodate a throughput of 1,500 MTHM/yr of spent fuel. The fuel receiving and offsite shipping operations are continuously operated; the chemical processes are assumed to be onstream for 300 days/yr to provide time for maintenance and repair of the facility. At that time, approximately 60 days' equivalent of spent fuel (300 MTHM) can be in storage along with about 30 days' equivalent of the waste in glass form. Fuel enters mechanical processing at a cooling time of 10 years out of the reactor (aged in an AFR for about 9 years), and the glass waste form is in essence produced immediately after dissolution of the fuel materials. About 30 days' storage of HLW is available to decouple the chemical processing from the waste solidification.

The flow of fuel, products, and waste materials is shown in Table C-47 for a 1,500 MTHM/yr throughput system.

C.5.4.1 Fuel Receiving. The fuel arrives in rail casks at the P/E facility after 10 years' cooling; therefore, the spent fuel is sufficiently cooled so that water can be added to the cask cavity without further cooldown of cask internals. To ensure that the fuel assemblies have not been damaged during shipment, the cask is flushed and the flush is sampled and analyzed before the cask is lowered into the unloading pool. The operator transfers one assembly at a time from the cask into a fuel storage basket. The basket is moved from the pool before the empty cask is hoisted from the pool. The cask is decontaminated and prepared for return to the reactors. The total movement of fuel and wastes from the P/E facility is shown in Table C-47. On the average, 5 rail casks filled with unencapsulated fuel arrive each week from the AFRs and 10 rail casks per week are loaded with vitrified spent fuel and shipped to the repository. Personnel allocations for each receiving operation have been described under Subsection C.2.4.

The total operating staff requirements for fuel receiving are shown in Table C-48.

Figure C-21.
P/E Facility
Plot Plan
Case 3

Adapted from Rockwell Hanford
conceptual design⁽¹⁵⁾

- 24 CONTROL ROOM
- 25 PERSONNEL AREA
- 26 ELEVATOR
- 27 STAIRWAY
- 28 CONTACT EQUIPMENT HANDLING AREA
- 29 COLD MAINTENANCE SHOP
- 30 EQUIPMENT LOADING AREA
- 31 ANALYTIC LABS
- 32 FILTER NICHE
- 33 REMOTE MAINTENANCE CELL
- 34 TRANSFER TUNNEL
- 35 SPENT FUEL FIXATION AND ENCAPSULATION CELL
- 36 CASK INSPECTION AND DECON CELL
- 37 HVAC EQUIPMENT ROOM

- AREA DESCRIPTION
- 1 SPENT FUEL RECEIVING STATION
 - 2 CASK TEST AND COOLDOWN STATION
 - 3 CASK DECON PIT
 - 4 CASK UNLOADING POOL
 - 5 SPENT FUEL STORAGE POOL
 - 6 LEAKY SPENT FUEL STORAGE POOL
 - 7 FUEL TRANSFER CANAL
 - 8 FUEL CUTTING AND VOLOXIDATION CELL, REMOTE
 - 9 HULL COMPACTING AND PACKAGING CELL, REMOTE
 - 10 ENCAPSULATED SPENT FUEL TRANSFER TUNNEL
 - 11 ENCAPSULATED SPENT FUEL DRY STORAGE ROOM
 - 12 CASK LOADING CELL
 - 13 ENCAPSULATED SPENT FUEL SHIPPING STATION
 - 14 COOLING/VENTILATION FACILITY ROOM
 - 15 VOLATILE FISSION PRODUCTS RECOVERY CELL
 - 16 RADWASTE PROCESSING AREA
 - 17 PACKAGED RADWASTE STORAGE ROOM
 - 18 RADWASTE LOADING STATION
 - 19 LOADING DOCK
 - 20 NEW CANISTERS AND DRUMS STORAGE ROOM
 - 21 CANISTER SUPPLY SHAFT
 - 22 OPERATING GALLERY
 - 23 POOL WATER TREATMENT AREA

Figure C-22.
P/E Facility
Building Arrangement
Case 3

10 5 0 10 20 30 40 50 FT
GRAPHIC SCALE

1 1 1

1 1 1

Table C-47
ANNUAL MATERIAL FLOW AT EACH P/E FACILITY
CASE 3
(1,500 MTHM/yr)

Material	Annual Number of Shipments	Total Units	Remarks
Incoming			
Spent PWRs	146	2,040	14 PWR/cask
Spent BWRs	90	3,060	34 BWR/cask
	<u>236 rail</u>	<u>5,100 assem.</u>	
Internal Transfers			
PWR	510	2,040	4 PWR/basket
BWR	340	3,060	9 BWR/basket
	<u>850 baskets</u>	<u>5,100 assem.</u>	
Outgoing			
Vitrified spent fuel	397 rail	5,556 canisters	14 can./shipment
Hulls	129 rail	1,146 canisters	9 can./shipment
Non-TRU LLW	21 truck	858 drums	42 drums/shipment
TRU-LLW	9 truck	369 drums	42 drums/shipment
TRU-ILW	240 truck	1,683 canisters	7 can./shipment
Radioactive gases	2 rail	11 canisters	9 can./cask
Failed equipment	17 truck	120 canisters	7 can./shipment

Table C-48
TOTAL OPERATING STAFF FOR
FUEL RECEIVING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (5 x 4 shifts)	20
Day workers	4
Supervisors (1 /shift + 1 day)	5
Total	<u>29</u>

C.5.4.2 Mechanical Processing. After the individual fuel assemblies are selected for the next dissolution batch, the appropriate fuel storage baskets (BWR or PWR) are moved to the fuel transfer canal. One assembly at a time is transferred under water to the remote process cell. In the remote process cell, the fuel assembly is loaded into the shear and the leading end fitting is sheared from the assembly and allowed to slide into a waste canister. The active length of fuel is sheared into short segments which fall into the voloxidizer. The trailing end fitting is then diverted into a waste canister. After release of volatile fission products from the fuel tube segments in passing through the voloxidizer, fuel materials and clad segments fall into the dissolver. Following dissolution of the fuel pellets, the fuel assembly residues are then monitored to be sure fuel materials are fully dissolved. Further processing of residues includes volume reduction, passivation of zircaloy fines, and packaging for disposal

The operating crew requirements for mechanical processing are as follows:

Moving the fuel to the shear	2 persons for 1 hour/assembly
Shearing the fuel	1 person for 8 hour/assembly
Operating the voloxidizer	1 person for 8 hours
Handling the hulls	1 person for 8 hours
Performing general duties (20 percent of above)	1 person for 8 hours
Supervision	1 person for 8 hours

The total operating staff requirements for mechanical processing are shown in Table C-49.

Table C-49
TOTAL OPERATING STAFF FOR
MECHANICAL PROCESSING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (5 x 4 shifts)	20
Day workers	0
Supervisors (1 x 4 shifts)	4
Total	<hr/> 24

C.5.4.3 Chemical Processing. After dissolution of fuel materials, the main chemical processing steps are the processing of offgases released during chemical processing and the solidification and vitrification of dissolved fuel pellet materials.

Staffing to support these remote chemical operations is determined by functional areas within the facility. In the central control room, where almost all the instrument and control functions are performed, the senior-licensed category of personnel will work. Included in this category are shift supervisors, lead operators, and control room operators, all of whom are qualified to start up, operate, and shut down the process system. Supporting operators are stationed in the cold chemical preparation areas, and roving crews of operators are dispatched throughout the plant to operate samplers and set manual valves and controls. Other operators perform support functions for maintenance and safety.

The operating crew requirements for chemical processing are as follows:

Operating the control room	2 persons for 8 hours
Preparing cold chemicals	1 person for 8 hours
	2 persons for 8 hr/day
Providing roving support	1 person for 8 hours
Providing general duties	1 person for 8 hours
Supervision	1 person for 8 hours
	+ 1 person 8 hr/day

The total crew requirements for chemical processing are shown in Table C-50.

Table C-50
TOTAL OPERATING STAFF FOR
CHEMICAL PROCESSING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (5 x 4 shifts)	20
Day workers	2
Supervisors (1 x 4 shifts + 1 day)	5
Total	27

C.5.4.4 Solidification of HLW. HLW wastes from fuel material dissolution are spray-dried and calcined, then mixed with glass-forming materials and melted to produce a borosilicate glass. The glass waste is packaged for disposal in canisters, which are then shipped offsite without further aging. All operations are carried out in heavily shielded cells. Remote mechanical handling operations are complex because of the nature of the remote calcine and glass furnace. Integrity of the canister must be verified before and after filling. A provision has been made to overpack the primary canister in case weld failure should occur. Approximately 19 HLW containers are produced each day. At 12 canisters per rail cask, 3 rail casks are required every 2 days. Canisters of HLW glass are loaded dry into offsite shipping casks.

The operating crew requirements for solidification of HLW are as follows:

Controlling the solidification and vitrification process	3 persons for 8 hours
Controlling mechanical processing	20 persons for 8 hours
Providing cold chemicals and loading fresh canisters	2 persons for 8 hours
Handling waste materials	2 persons for 8 hours
Handling fresh canisters	6 persons for 8 hours
Performing general duties (20 percent of above)	7 persons for 8 hours
Loading casks	2 persons for 8 hours
Decontaminating casks	3 persons for 8 hours
Supervision	2 persons for 8 hours + 1 person 8 hr/day

Throughput of the HLW solidification facility directly influences mechanical processing, container supply operations, and loading and shipping operations.

The overall operating crew requirements for solidification of HLW are shown in Table C-51.

Table C-51
TOTAL OPERATING STAFF FOR
SOLIDIFICATION OF HLW
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (40 x 4 shifts)	160
Day workers	0
Supervisors (2 x 4 shifts + 1 day)	9
Total	169

C.5.4.5 Volume Reduction of Fuel Cladding. The end fittings sheared from the fuel bearing portions of spent fuel are packaged without further processing in 20-inch diameter by 10-foot cans. By careful orientation of the end fittings in the packaging operation, reasonably optimized container volumes are achieved. No fixation of these materials (in concrete or other matrices) is dumped from the dissolver basket into a mechanism to segregate the fines from the tube sections. The tube sections are compacted from a density of about 1 g/cm³ to 3.7 g/cm³ in individual compact canisters. Seven of these cans are bundled into close-packed arrays and loaded three arrays high in a hulls container. Fines from the deactivation processing are poured onto the compacted tube sections. The waste canisters then pass through welding, leak testing, decontamination, and cask loading operations. Mechanical processing operations are viewed from operating galleries, and most operations are hands-on, remotely controlled operations. The automatic welding and the compaction operations are manually initiated operations, followed by automatic, programmed operations.

About every four hours a dissolver basket containing hulls from 1 MTHM equivalent spent fuel is dumped; fines are deactivated immediately. Sufficient tube segments are charged into a small compactor canister, the piston advances to crush the tubes, the piston retracts, and a smaller charge of tube segments is added. The cycle is repeated until the small canisters are filled to the desired level. Seven cans are strapped together, grappled, and lowered into the larger 20-inch by 10-foot hulls container. Three such layers are loaded into the large can. A small canister is filled with compacted hulls every 1-1/2 hours. Approximately 2 canisters per day of end fittings and 1 canister of compacted tube sections every 1-1/2 days accumulate from the hulls volume reduction operations for a 1,500 MTHM/yr reprocessing plant.

Two casks per week are loaded for shipment to the geologic repository.

The operating crew requirements for volume reduction of fuel cladding are as follows:

Operating the basket dumping and fines deactivation	1 person for 8 hours
Loadout of end fittings	1 person for 8 hours
Operating the compactor	1 person for 8 hours
Banding and packaging small canisters	1 person for 8 hours
Performing welding and leak testing	1 person for 8 hours
Decontaminating canisters and loading out of casks	2 persons for 8 hr/day
Providing accountability records	1 person for 8 hr/day
Providing canisters as needed	2 persons for 8 hr/day
Performing general duties (20 percent of above)	1 person for 8 hours
Supervision	2 persons for 8 hours

Almost all work in hulls volume reduction is directly related to throughput. An exception would be some general work that would have to be performed even if throughput were zero.

The total operating personnel requirements for volume reduction of fuel cladding are shown in Table C-52.

Table C-52
TOTAL OPERATING STAFF FOR
VOLUME REDUCTION OF FUEL CLADDING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (6 x 4 shifts)	24
Day workers	6
Supervisors (2 x 4 shifts)	8
Total	<hr/> 38

C.5.4.6 Volume Reduction of Combustible Solid Wastes and Fixation of Non-combustible Solid Wastes. Solid wastes accumulate from paper and plastics, etc., used in the preparation of work areas and from small tools and failed equipment items, spent ion exchange resins, and spent ventilation filters. The SFDS assumes that 20 percent of the trash is not combustible and that the remaining 80 percent can be reduced in volume by a 40:1 ratio, using molten salt combustion techniques. Fixation of ash from molten salt processing and fixation of noncombustible wastes result in a 2:1 volume penalty.

Table C-47 shows the expected quantity of solid wastes resulting from volume reduction and fixation. Non-TRU wastes, which can be shipped to shallow land burial, are also shown in the table.

Combustible wastes pass through shredding equipment and a molten salt incinerator before being blended with fixation materials and poured into waste drums. Noncombustible wastes are blended with fixation materials and poured into waste drums. Remote operation of all volume reduction and fixation systems is planned and semi-remote or fully remote operations are involved in handling wastes inbound and outbound.

At each facility, about 50 drums per day of waste are collected from point of origin of the waste and delivered to waste processing.

The operating crew requirements for volume reduction of combustible solid wastes and fixation of noncombustible solid wastes are as follows:

Waste collection and delivery	2 persons for 8 hr/day
Loading of combustible and noncombustible feed hoppers	2 persons for 8 hours
Operating the process equipment	2 persons for 8 hours
Providing containers for processing	1 person for 8 hours
Providing bulk chemicals	2 persons for 8 hr/day
Decontaminating and preparing drums for offsite shipments	2 persons for 8 hours
Performing general duties (20 percent of above)	2 persons for 8 hours
Supervision	1 person for 8 hours + 1 person for 8 hr/day

Throughput directly determines the staffing requirements for volume reduction of combustible solid wastes. The overall operating crew requirements are shown in Table C-53.

Table C-53
TOTAL OPERATING STAFF FOR
VOLUME REDUCTION OF COMBUSTIBLE
SOLID WASTES AND FIXATION OF
NONCOMBUSTIBLE SOLID WASTES
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (8 x 4 shifts)	36
Day workers	4
Supervision (1 x 4 shifts + 1 day)	5
Total	<hr/> 45

C.5.4.7 Offgas Treatment, Recovery, and Packaging. Offgases released during dissolution and chemical processing of fuel materials carry along volatile fission products of tritium, iodine, krypton, xenon, ruthenium, and an activation product of carbon. Tritium and iodine are isolated in a concrete matrix in 55-gallon drums; and krypton, xenon, and carbon-14 are isolated as pressurized gases in standard gas cylinders. All of these operations are remote chemical operations through packaging, leak testing, decontamination, and stripping operations. See Table C-46 for the 55-gallon waste drum and 200-scf gas cylinder estimated production rates.

The operating crew requirements for offgas treatment, recovery, and packaging are as follows:

Operating the control room	2 persons for 8 hours
Handling inbound containers	1 person for 8 hr/day
Performing mechanical operations	2 persons for 8 hours
Loading-out of waste packages	2 persons for 8 hr/day
Providing bulk materials	2 persons for 8 hr/day

Throughput influences all categories under this operation except the control room functions. The total personnel requirements for offgas treatment, recovery, and packaging are shown in Table C-54.

Table C-54
TOTAL OPERATING STAFF FOR
OFFGAS TREATMENT, RECOVERY,
AND PACKAGING
(1,500 MTHM/yr)

Personnel	Number Required
Shift workers (4 x 4 shifts)	16
Day workers	5
Supervision (1 x 4 shifts)	4
Total	<hr/> 25

The composite personnel requirements for a P/E facility for Case 3 are shown in Table C-55; total support and overhead personnel requirements for such a facility are shown in Table C-56.

Table C-55
TOTAL OPERATING STAFF FOR
A CASE 3 P/E FACILITY
(1,500 MTHM/yr)

Personnel	Number Required
Fuel receiving	29
Mechanical processing	24
Chemical processing	27
Waste solidification	169
Hulls volume reduction	38
TRU volume reduction	45
Offgas recovery	25
Total	<hr/> 357

Table C-56
TOTAL SUPPORT STAFF FOR
A CASE 3 P/E FACILITY
(1,500 MTHM/yr)

Personnel	Number Required
Maintenance (50 percent of operations)	178
Analytical (10 percent of operations)	36
Health and safety (10 percent of operations)	36
Utilities	30
Plant overhead (30 percent of operations)	110
Total	390

Total overall personnel requirements for a P/E facility in Case 3 are shown in Table C-57.

Table C-57
OVERALL TOTAL STAFF REQUIREMENTS
FOR A CASE 3 P/E FACILITY
(1,500 MTHM/yr)

Personnel	Number Required
Operating	357
Support and overhead	390
Total	747

C.5.5 Cost Estimates

C.5.5.1 Case 3 P/E Facility Capital Cost Estimate. The P/E facility capital cost estimate for Case 3 was developed from the Case 1 and Reference Case capital cost estimates. The Case 3 P/E facility is designed to process the spent fuel beyond the chopping and voloxidation steps carried out in Case 2 by additional processing steps before final encapsulation at the rate of 1,500 MTHM/yr.

The estimate for the Case 3 P/E facility follows the Case 1 subfacility breakdown at the total field cost level. The results of the Case 3 estimate are presented in Table C-58.

Table C-58
CAPITAL COST ESTIMATE FOR A CASE 3 P/E FACILITY
(1,500 MTHM/yr)^(a)

Item	Cost \$ million/yr ^(b)
Field costs (including 20 percent contingency)	
Receiving, processing and packaging building	
Fuel receiving and cask handling	28
Surge storage pool	20
Fuel cutting and voloxidation cell	29
Volatile fission products recovery cell	40
Fuel dissolution, concentration, and hulls encapsulation cell	50
Spent fuel calcination, vitrification, and encapsulation cell	282
Storage cell	20
Loadout and waste shipping facility	3
Waste treatment area	25
Support areas	9
Building mechanical and electrical	27
Vendor checkup and startup	2
Subtotal	535
Support facilities (including improvements to land)	40
Total field costs (including 20 percent contingency)	575
Architect-engineer services @ 20 percent of 575	115
Owner's cost @ 30 percent of 690	210
Total capital cost	900

(a) The 5,000 MTHM/yr SFDS economic study requires the equivalent of 3-1/3 of the above P/E facilities.

(b) January 1979 dollars.

The required modifications to the Reference Case and Case 1 facilities for Case 3 are discussed in detail in Subsection C.5.3, and their impacts on the cost are given below:

Improvement to land. A provision of \$7 million is made in improvement to land — the same as for Case 1. This sum is included under the heading, "Support Facilities".

Receiving, processing, and packaging building. The following sums were provided for operations within the receiving, processing, and packaging building:

Fuel receiving and cask handling. \$28 million, the same as for Case 1.

Surge storage pool. \$20 million, the same as for Case 1.

Fuel cutting and voloxidation cell. \$29 million, the same as for the Reference Case voloxidation cell.

Volatile fission products recovery cell. \$40 million, the same as for the Reference Case gas recovery subfacility.

Fuel dissolution concentration and hulls encapsulation cell. To develop the cost of this subfacility, three sources were used: the encapsulation cell included in Case 2 (\$31 million); the hulls compaction cell included in the Reference Case (\$14 million); and the dissolution subfacility from AGNS BNFP⁽¹⁴⁾ escalated to 1979 pricing. (The total dissolution subfacility field cost is \$5 million.) The total field costs for this subfacility were estimated at \$50 million.

Spent fuel calcination, vitrification and encapsulation cell. The Reference Case subfacility was used as a base (\$100 million). The throughput capacity in Case 3 is 8 times greater than that for the Reference Case subfacility. To cost the increased capacity, the 0.5 cost/capacity ratio formula described in Subsection C.3.5.1 was applied to determine the estimated cost of \$282 million.

Storage cell. \$20 million, the same as for Case 1.

Loadout and waste shipping. The Case 1 subfacility estimate of \$2 million was the basis for this subfacility estimate. In Case 3 the throughput is 2 times greater than in Case 1. The cost/capacity factor ratio described in Subsection C.3.5.1 was applied to develop the estimated cost of \$3 million.

Waste treatment area. \$25 million, the same as for Case 1.

Support areas. The support areas in the receiving, processing and packaging building are assumed to be 20 percent larger than those for Case 1 subfacilities (\$8 million). The 0.5 cost/capacity formula described in Subsection C.3.5.1 was applied to develop the estimated cost of \$9 million.

Building mechanical and electrical. This subfacility is assumed to be 20 percent larger than the Case 1 subfacility (\$25 million). The 0.5 cost/capacity formula described in Subsection C.3.5.1 was applied to develop the estimated cost of \$27 million.

Vendor checkup and startup. \$2 million, the same as for Case 1.

Support facilities. This subfacility is assumed to be 20 percent larger than the Case 1 subfacility (30 million). The 0.5 cost/capacity formula was applied to develop the total field cost of \$33 million.

C.5.5.2 Operating Cost Estimate. The P/E facility for Case 3 requires about 90 percent of the operating complement of the Reference Case. Case 3 requires fewer personnel in chemical processing operations because Case 3 involves only fuel dissolution but not separations, purification, and conversion operations required in the Reference Case. Liquid waste solidification, however, requires significantly greater numbers of people in Case 3 because more of the high-level glass canisters are required (about 18,500 canisters/yr vs. 2,400 canisters/yr).

Average salary figures, cost factors for operating and maintenance supplies and utilities, and unit canister costs are the same as for the Reference Case (Subsection C.2.5.2) and are shown in Table C-59.

Table C-59
OPERATING COST ESTIMATE CASE
3 P/E FACILITY
(1,500 MTHM/yr)

Item	Cost \$ million/yr
Personnel	
746 persons x \$24,000/yr	17.9
Operating maintenance supplies	
0.01 x \$900 million	9.0
Utilities	
0.001 x \$900 million	0.9
Canisters	
8,526 canisters x \$5,000	42.6
Total	<hr/> 70.4
	(or 70.0)

C.5.5.3 Decommissioning Cost Estimate. The Case 3 P/E facility is essentially as complex as the Reference Case P/E facility. Radioactive liquids will have contaminated facilities and equipment in many operations, and these will have to be removed and packaged for disposal while still significantly contaminated. Thus, decontamination and dismantling costs are higher for Case 3 than for Cases 1 and 2, and the same cost factor is used to calculate decommissioning costs as is used for the Reference Case.

The decommissioning cost in Case 3 is set at 15 percent of the total capital cost, less the owner's costs, to produce \$100 million.

REFERENCES FOR APPENDIX C

- (1) "Final Safety Analysis Report," Barnwell Nuclear Fuel Plant, Separation Facility, Docket 50-382, Allied-Gulf Nuclear Services, Barnwell, South Carolina, October 1973.
- (2) A. F. Cermak, J. H. Gray, E. W. Murbach, J. C. Neace, and R. C. Spannburgh, "Development of the Electropulse Column for the BNFP."
- (3) "Preliminary Safety Analysis Report," Barnwell Nuclear Fuel Plant, Division of Addendum No. 7, Plutonium Product Facility, July 1974.
- (4) "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," Vol. 2, ERDA-76-43, May 1976.
- (5) W. S. Groenier, "Current Positions and Status of Voloxidation Process at ORNL," ORNL/CF-77/67, April 1977.
- (6) B.C. Finney et al., "Correlation of Radioactive Waste Treatment Costs and Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle — Reprocessing LWR Fuel," ORNL/NUREG/TM-6, January 1977.
- (7) "Hot Experimental Facility Interim Design Report," Oak Ridge National Laboratory and Bechtel National, Incorporated, ORNL/AFRP-78/6, 1978.
- (8) W. E. Clark et al., "Fixation of Radioiodine with Portland Cement. I. Preliminary Scoping Studies," ORNL-TM-5064, December 1975.
- (9) J. R. Merriman et al., "Removal of Krypton-85 from Reprocessing Plant Off-Gas by Selective Absorption," K-L-620I, September 1972.
- (10) E. L. Moore, "Interim Report: Commercial Waste Packaging Studies, Task 2, High Level Waste Package Acceptance Criteria," Rockwell International, RHO-ST-10, March 1978.
- (11) "Plutonium Transportation Vehicle Concepts Descriptions and Analysis," Allied-General Nuclear Services, NMS-15, December 1977.
- (12) "Certification of ERDA Contractors' Packaging with Respect to Compliance with DOT Specification 7A, Performance Requirements, Phase II Summary Report," MLM-2288, June 1975.
- (13) "Nuclear Fuel Cycle Closure Alternates," Allied-General Nuclear Services, April 1976.
- (14) "Bechtel Inc. Capital Cost Estimates and Construction Impacts," prepared for Battelle Pacific Northwest Laboratories, Contract No. B-05516-A-E, 1977.
- (15) "Spent Fuel Receiving and Packaging Facility, Conceptual Design Report," Rockwell International, RHO-CD-506, Draft, September 1978.
- (16) "Spent Fuel Handling and Packaging Program—Program Plan," Rockwell International, RHO-CD-444, July 1978.
- (17) "Commercial Waste Packaging Program Plan," Rockwell International, RHO-CD-387, May 1978.
- (18) R. A. Lorenz et al., "Fission Product Source Terms for the LWR Loss-of-Coolant Accident: Summary Report," NUREG/CR-0091, ORNL/NUREG/TM-206.

Appendix D
GEOLOGIC REPOSITORIES

TABLE OF CONTENTS

	Page
D.1 Approach	D-2
D.1.1 Preconceptual Design Approach	D-2
D.1.2 Cost Estimating Approach	D-3
D.2 Generic Repository Design Considerations	D-7
D.2.1 Thermal Loading	D-7
D.2.2 Site	D-7
D.2.3 Surface Facilities	D-8
D.2.4 Underground Facilities	D-19
D.2.5 Ventilation	D-23
D.2.6 Shafts	D-24
D.2.7 Specialized Equipment	D-27
D.2.8 Repository Validation	D-29
D.3 Reference Case	D-38
D.3.1 Specific Repository Aspects	D-38
D.3.2 Waste Emplacement	D-38
D.3.3 Operational Considerations	D-44
D.3.4 Repository Cost Estimates	D-54
D.4 Case 1	D-56
D.4.1 Specific Repository Aspects	D-56
D.4.2 Spent Fuel/Waste Emplacement	D-56
D.4.3 Operational Considerations	D-57
D.4.4 Repository Cost Estimates	D-65
D.5 Case 2	D-67
D.5.1 Specific Repository Aspects	D-67
D.5.2 Chopped Fuel/Waste Emplacement	D-67
D.5.3 Operational Considerations	D-68
D.5.4 Repository Cost Estimates	D-75
D.6 Case 3	D-78
D.6.1 Specific Repository Aspects	D-78
D.6.2 Vitrified Fuel/Waste Emplacement	D-78
D.6.3 Operational Considerations	D-81
D.6.4 Repository Cost Estimates	D-86
References for Appendix D	D-89

• • • • •

• • • • •

Appendix D

GEOLOGIC REPOSITORIES

This appendix contains the preconceptual design descriptions and cost estimates for the geologic repositories required in the various cases studied. The geologic repository is similar for all cases, differing primarily in the type, form, and quantity of nuclear material handled and in the emplacement geometry. Repository emplacement is expected to be the final step in the fuel cycle.

The descriptions and cost estimates given in this appendix, though based on preconceptual designs, are sufficiently definitive to provide a basis for assessments and comparisons in Sections 4 through 10 of this report. In a few instances, where the design selections or assumptions were thought to have a significant impact on the outcome of the study, study variations (or sensitivity studies) were made. These variations are identified and analyzed in Section 10.

The bases and assumptions for the entire study are presented in Appendix A.

D.1 APPROACH

D.1.1 Preconceptual Design Approach

Contractually, the depth of design detail required in the Spent Fuel Disposal Study (SFDS) is limited to a preconceptual design effort. In this design effort, existing resource materials are studied in order to prepare general process and facility descriptions that could be adapted to spent fuel disposal systems. Since repository technology and facilities are under development, the types of information studied are limited to:

- Conceptual design studies and projects
- Preliminary engineering studies
- Preliminary design criteria
- Technical reports
- Miscellaneous data

The descriptions and definitions developed for SFDS repositories are in the form of descriptive text, block diagrams, flow diagrams, and preconceptual design layouts of facilities. This material, which is presented in this appendix, is in sufficient detail to meet the following objectives:

- To furnish a basis, in terms of facilities configuration and process technology definition, for assessing and comparing cases
- To furnish data for developing order-of-magnitude capital and operating cost estimates
- To investigate the equipment and process technology state of the art in order to assess the technical feasibility and to provide a basis for planning continuing work

The approach used for the preconceptual design studies is to identify the basic geologic disposal technology and to outline the principal features of the aboveground and underground facilities. The facilities are described briefly; no attempt is made to optimize systems or facilities. The basic goal is to arrive at a facility size and to identify major components and systems. For the underground mine development, a thermal analysis is performed to determine minimum extraction ratios and the loading geometry of spent fuel and other heat-producing waste in the repository.

For the preconceptual design of the repository, two major sources are employed: Stearns-Roger Engineering Company (SRENCO)^{(1),(2),(3)} and Kaiser Engineers,^{(4),(5),(6)} who were the design contractors for the National Waste Terminal Storage (NWTS) Repositories No. 1 and No. 2, respectively.

The specific approach taken for repositories is to describe first the generic characteristics of a repository capable of handling either high level waste (HLW), spent fuel, or other forms of waste. This description, given in Subsection D.2, identifies those features common to all SFDS cases. The generic repository description includes site, surface facilities, shafts, underground development, and special equipment. Generally, the SRENCO conceptual design is used, with some support from the Kaiser conceptual design and other sources.

Following the generic repository description, each case is analyzed to identify those features unique to that case from the point of view of waste form, waste emplacement, and special operational considerations. To obtain the complete picture of a repository in a given case, the generic description should be combined with the special requirements of that case.

It should be noted that, although NWTs Repositories No. 1 and No. 2 are in "dome" salt and bedded-salt formations, respectively, the SFDS considers only generic-salt formations. In all cases, HLW and spent fuel entering the repository are aged 10 years from the time of removal from the reactor. In addition to HLW and spent fuel, the repository receives cladding hulls, transuranic (TRU) waste, and non-TRU intermediate level waste (ILW), as applicable to the various cases. All non-TRU, low-level waste (LLW) generated in the repository operation are also disposed of in the repository.

D.1.2 Cost Estimating Approach

Several cost estimates are prepared in connection with the geologic repository. These estimates are:

- Capital cost estimates
- Operating cost estimates
- Decommissioning cost estimates

The approach taken in estimating these components of repository cost is discussed below. The actual cost estimates for the four cases are presented and discussed in Subsections D.3.4, D.4.4, D.5.4, and D.6.4. The transportation cost estimates are presented and discussed in Appendix E.

All estimated costs are in January 1979 dollars, with no allowance for future escalation, and as order-of-magnitude estimates, have expected accuracies of +75 percent and -25 percent.

D.1.2.1 Geologic Repository Capital Cost Estimates. The capital cost of the repository includes all engineering, procurement, and construction costs, and the owner's costs during construction. The estimates are based on the SRENCO NWTs Repository No. 1, which is the source for the preconceptual design (Subsection D.1.1). The SFDS design is basically generic; however, for each case, there are a few unique features, and these features are reflected in the cost estimates.

Two types of incremental cost adjustments are made to the SRENCO estimate:

- Scope changes to comply with the SFDS design requirements
- Deletion of nongeneric site costs

The SFDS design consolidates the HLW/spent fuel and the LLW handling facilities into a single unit. All separate aboveground and belowground LLW facilities (including the shaft) are deleted, and a wing is added to the original HLW/spent fuel receiving building. Caissons for the building foundations are also deleted. Table D-1 lists all the generic adjustments that must be made to the SRENCO field cost. Table D-2 shows the additions that must be made to this adjusted SRENCO field cost to arrive at the total field cost for each of the four cases.

Table D-1
SRENCO-SFDS ESTIMATE RECONCILIATION
FOR TOTAL GENERIC FIELD COSTS
(5,000 MTHM/yr)

Item	Cost \$ million
SRENCO field cost (not including architect-engineer or contingency cost, in 1978 \$)	643
Changes to SRENCO cost	
Delete land cost	- 23
Delete LLW facilities and mine development	- 93
Add LLW wing to waste building	+ 6
Reduce salt storage cost	- 10
Delete building caissons	- 50
Delete cost estimated for 26 initial rooms	- 20
Adjusted SRENCO field cost, in 1978 \$	<u>\$449</u>
Add escalation of 10%	+ 45
Adjusted SRENCO field cost (January 1979 \$)	<u>\$494</u>

Table D-2
SFDS TOTAL FIELD COSTS
(January 1979 Dollars)

Item	Ref. Case	Case 1	Case 2	Case 3
Adjusted SRENCO field cost	494	494	494	494
Cost of mining initial rooms	6	16	6	20
Cost of second waste shaft and facility	-	-	-	146
Cost contingency of 20%	<u>100</u>	<u>100</u>	<u>100</u>	<u>130</u>
Total field cost	600	610	600	790

Bechtel's estimate of the initial development cost is based on the required number of rooms for each of the four cases; this number differs from the SRENCO-determined number of rooms, which is 26. For each SFDS case, the initial mining cost is a function of the number of completed rooms required at the time of initial repository operation. These rooms serve as a repository of one year's wastes and as a buffer to be used at the start of operations. These initial room mining quantities are shown in Table D-3.

Table D-3
INITIAL ROOM MINING QUANTITIES

Item	Ref. Case	Case 1	Case 2	Case 3
Salt excavation (millions of tons)	0.5	1.2	0.7	1.9
Rooms (each)	13	30	21	62

The total field costs are listed in the capital cost estimates for each case (see Subsections D.3.4, D.4.4, D.5.4, and D.6.4) and are broken down into the following categories:

- Surface facilities
- Shafts and hoists
- Underground facilities
- Initial underground development

Each of these categories is subdivided into significant cost items. In addition to total field costs, separate costs are shown for contingency, architect-engineer services, and the owner's costs.

D.1.2.2 Geologic Repository Operating Cost Estimates. Annual operating cost estimates for the geologic repositories are based upon two periods of operation: the retrievability period and the postretrievability period. During the retrievability period (the first 5 years of operations), all salt excavated from the repository is transported to the surface for interim storage. Each storage well is sleeved to provide additional protection for the emplaced waste canister, and plugged for personnel protection. In the ensuing postretrievability period, only excess salt not currently needed to backfill rooms and branch corridors is transported to the surface for interim storage. Sleeving of storage wells is not required in the postretrievability period of operations.

The postretrievability period operating cost estimated for the Reference Case is based on the SRENCO estimate for the NWTS, adjusted for differences in annual throughput. The postretrievability period operating cost estimates for Cases 1, 2, and 3 are obtained by multiplying the adjusted Reference Case operating cost estimate by the ratio of operating census for each case to the operating census for the Reference Case.

The retrievability period operating cost estimates are based on the postretrievability period operating cost estimates for each case. Adjustments are made to accommodate reduced staffing requirements and the added costs of providing sleeves and plugs. Bechtel's estimated costs for upper and lower sleeves and the concrete plugs are based on standard materials of construction. Costs for quality assurance are not included in this estimate.

The operating cost estimates are presented in Subsections D.3.4.2, D.4.4.2, D.5.4.2, and D.6.4.2.

D.1.2.3 Geologic Repository Decommissioning Cost Estimates. Decommissioning of the repository includes backfilling the main drifts, the central core area, and the shafts, and dismantling the surface facilities. The differences in decommissioning costs between cases are assumed to be relative to the total capital cost of those facilities. All cases entail receipt of waste, inspection, handling, onsite transportation, and emplacement operations. When the last waste has been emplaced, the rooms and branch corridors will have been backfilled.

Decommissioning of the unbackfilled subsurface areas involves exactly the same volumes and complexity in all cases. The surface facilities vary in size depending on waste throughput rates; however, those differences have already been reflected in capital cost differences. Individual decommissioning cost estimates are developed by using a proportion of total capital cost in each case.

TRW's "National Waste Terminal Storage Economic Study, Part 2, Spent Fuel Management Pricing," published in March 1978, gives repository decommissioning costs as 33 percent of the total capital cost.

TRW also estimates that the decommissioning cost factor for the surface facilities would be about 10 percent of total capital costs. Thus, the underground decommissioning cost factor would have to be significantly above 33 percent, assuming the capital costs of facilities above and below ground were about equal.

For the SFDS, the repository decommissioning costs are estimated at 33 percent of the total capital costs, less the owner's costs. Estimates for each case are summarized in Subsections D.3.4.3, D.4.4.3, D.5.4.3, and D.6.4.3.

D.2 GENERIC REPOSITORY DESIGN CONSIDERATIONS

In this subsection, the repository design considerations common to all four SFDS cases are discussed. These considerations are as follows:

- Thermal loading
- Site
- Surface facilities
- Underground facilities
- Ventilation
- Shafts
- Specialized equipment
- Validation

D.2.1 Thermal Loading

Thermal loading of a salt repository affects the following:

- Stability of the waste materials and integrity of the canister and overpack barriers in the very-near-field region
- Migration of brine from minute cavities, decrepitation of salt, closure of rooms, and failure of the overlying water barriers in the near-field region surrounding the canisters
- Upheaval of the earth's surface and transfer of heat to aquifers in the far-field regions

Decay heat diffuses slowly through the geologic media after being released from stored spent fuel, HLW, and ILW, such as hulls. The heat transfer from the waste can is by conduction through a salt backfill or by radiation transfer across an air gap to the surrounding medium. Convection in the gaps between various components of the canisters, the canister wall, the overpack, the sleeve, and the rock are neglected.

Computer programs have been developed to evaluate the general heat transfer behavior from radioactive materials and thermal diffusion through various geologic media, including salt. The SFDS utilized the services of Science Application, Inc. (SAI), which has performed these types of analyses for similar projects. SAI determined the minimum extraction ratio that would provide maximum thermal loading in the repository. The SAI report of this work⁽⁷⁾ represents the basis for the preconceptual repository thermal design.

D.2.2. Site

The hypothetical site requirements include a 200-acre inner controlled area which surrounds the surface facilities and a 600-acre outer fenced area. These two areas overlie the underground storage site. A plot plan for the generic repository is shown in Figure D-1.

The inner controlled area is enclosed by a double fence with an unoccupied isolation zone of 25 feet between each fence. The double security fence is penetrated by guarded entries for plant personnel, visitors, and truck and rail shipments of nuclear waste.

Since this is a generic hypothetical site, it is assumed that it is on gently rolling rural terrain. The site requires major grading to ensure that floods can be diverted away from the mine shafts and surrounding structures. The grading covers approximately a 40-acre strip of area built up around the shafts. The remainder of the site is graded away from this higher area down to the outer fence boundaries.

Improvements to the site include demolition, clearing, and stripping in areas of construction, including the access road, access rail spur, pipeline, power line, and salt pile. Drainage is required for the access road, access rail spur, and salt pile.

D.2.3 Surface Facilities

The major surface facilities handle the incoming encapsulated spent fuel, HLW, ILW, and LLW, and prepare these materials for transfer to the underground facilities. These functions include:

- Unloading of shipping casks and cargo carrier arriving by rail or truck
- Transfer of waste packages to inspection stations
- Decontamination of waste packages
- Overpacking of waste packages, as required
- Transfer of waste packages to underground facilities
- Radwaste treatment and handling of onsite-generated wastes
- General support for the repository

The buildings and surface facilities that serve the underground development areas include the following:

- Receiving building (waste and spent fuel)
- Underground operations building
- Compressor and chiller building
- Waste shaft ventilation building
- Man and material shaft ventilation building
- Ventilation supply filter building
- Development exhaust filter building
- Repository exhaust filter building
- Radwaste building
- Administrative building
- Mined salt surface stockpile

If the loss of confinement in a particular surface building can have a significant impact on the health and safety of workers or nearby populations, that surface building is designed for appropriate seismic and tornado loads, normal live loads, and normal dead loads, in accordance with appropriate regulations and codes.

D.2.3.1 Receiving Building (Waste/Spent Fuel). The receiving building houses all facilities and equipment to receive, handle, decontaminate, temporarily store, and overpack canisters requiring remote handling, and prepare them for transfer to the underground storage. This building also has a separate wing in which LLW drums are received and prepared for underground storage by contact handling.

Figure D-1.
Generic Repository — Plot Plan
All Cases

Adapted from Stearns-Roger
conceptual design⁽³⁾

The receiving building contains the following areas:

- The shipping cask handling area
- The canister handling area
- The transfer cask handling area
- The canister surge storage area
- The LLW handling area
- Miscellaneous auxiliary and support areas such as HVAC and equipment maintenance areas and the central control room

To ensure a safe working environment, radiation monitoring and alarm systems are provided throughout the building.

A simplified plan view of the facility is shown in Figure D-2; a flow diagram, indicating the processing and handling operations, is shown in Figure D-3.

Shipping cask handling area. This area contains facilities for receiving, cleaning, unloading, and decontaminating the shipping cask, and reloading the empty cask onto the rail cars. Adequate rail spurs are provided for temporary storage of rail cars, and the receiving area contains three rail car bays.

The shipping casks arrive by rail; the incoming loaded rail cars, after inspection, are moved by the yard engine to the yard storage area or, if they need cleaning, into the shipping cask cleaning area. Here, the rail cars and the shipping casks are steam-cleaned and decontaminated. The clean, loaded cars may be transferred to the outside yard storage for interim storage, or moved by car pullers directly into the shipping cask handling area.

One of the overhead bridge cranes lifts the shipping cask off the railroad car and places it on a pallet which is moved by a surface effect (air pallet) vehicle. The loaded shipping cask is moved by the surface effect vehicle into the entry vestibule, where the cask lid fasteners are removed. After this operation, the shipping cask is transferred into the lid removal vestibule, a section of the building that provides confinement of radioactive materials. Here, the lid is removed and taken into storage by a job crane. The shipping cask is transported to the unloading cubicle under the unloading port of the hot cell, where the cask is mated and sealed to the hot cell port to prevent contamination during canister unloading. After unloading, the empty cask and cask lid are reassembled, decontaminated, returned to the receiving section, and placed on an outgoing railroad car.

Canister handling area. The canisters are handled within remote hot cells, with all activities observed and controlled from operating corridors.

After the loaded shipping cask and unloading port have been mated, the port plug and then the shipping cask plug are removed and placed in storage racks.

The canisters, still in the shipping cask, are visually inspected for damage and then lifted into the hot cell. After a visual verification that the shipping cask is empty, the cask and hot cell port plugs are reinserted, and the empty shipping cask is disengaged from the hot cell and returned to the offloading area for shipping out.

In the hot cell, the canisters are weighed and monitored for surface contamination, and input data are fed into an information processing system used for purposes of accountability (the accountability system). The canisters are also tested and inspected for visible damage, radiation leakage, and temperature measurement. The damaged canisters are separated for an overpacking operation, after which they join the stream of the clean, undamaged canisters.

Transfer cask handling area. In this area, canisters are loaded into the transfer cask and moved to the waste shaft hoist cage. The transfer cask is moved into the loading cubicle, where it is mated and sealed with the hot cell loading port. The loading port hatch is then opened, and one canister is hoisted with the cask winch into the transfer cask. Then the loading port and cask hatches are closed, and the transfer cask is moved by a surface effect vehicle to the waste shaft hoist cage for lowering into the underground facilities.

Canister surge storage area. A storage cell is provided to accommodate, when and as needed, the number of canisters expected during a 10-day shipping period. The surge storage cell receives canisters from the hot cell after they are unloaded from the shipping cask, and is equipped with a special decay heat removal cooling system which is activated only when the surge storage cell is in use.

LLW handling area. The LLW is received and handled in a separate section of the receiving building. This section is furnished with confinement barriers and unloading docks for incoming trucks. Such operations as inspecting, testing, monitoring, overpacking, decontamination, weighing, and storage areas are provided.

The LLW processing and handling operations are shown in Figure D-4.

The LLW packaged in drums arrives by truck. After inspection, the shipment is unloaded from the truck, and the truck is cleaned and decontaminated, as required, before it is returned to the shipper. The cargo carrier is inspected, and decontaminated, if necessary, then moved to a confined area where the LLW drums are removed. After internal inspection for contamination or damage, the cleaned cargo carrier is returned to the truck loading dock to be transported back to the shipper.

The LLW drums are surveyed and weighed, and all input data are fed into the accountability system. The drums are then tested and inspected for visible damage and radiation level. After overpacking and decontamination, the damaged drums join the stream of the clean, undamaged drums. The palletized drums are taken to a temporary storage area before being moved to the waste shaft hoist cage and lowered into the underground facilities. The temporary storage area size is based on the number of palletized drums received with one truck shipment.

D.2.3.2 Underground Operations Building. This building houses (1) the support operations for underground facility personnel, their equipment, and supplies, (2) a maintenance and repair area for servicing all noncontaminated underground equipment, and (3) a warehouse which provides storage space for equipment and supplies.

D.2.3.3 Compressor and Chiller Building. This building houses the compressors and water chillers for the air conditioning and ventilation systems.

The piping systems serving the head frames and service buildings include potable water, instrument air, compressed air, chilled water, and steam condensate.

D.2.3.4 Waste Shaft Ventilation Building. This building, located next to the waste shaft collar, supplies the head frame and the shaft with conditioned air and houses the intake and exhaust air fans, filter, air conditioning, and electrical equipment.

D.2.3.5 Man and Material Shaft Ventilation Building. This building, located adjacent to the man and material shaft head frame, houses the equipment (e.g., fans, filters, air conditioning and electrical equipment) needed to supply the head frame and shaft with conditioned ventilation air.

D.2.3.6 Ventilation Supply Filter Building. This building contains the equipment needed to supply conditioned ventilation air to the underground areas.

Figure D-2.
Generic Repository — Receiving Building
(Waste/Spent Fuel)
Arrangement Schematic
All Cases

x s r t

g r z

D.2.3.7 Development Exhaust Filter Building. This building contains the facilities and equipment needed to filter and exhaust the air from underground areas during mine development operations.

D.2.3.8 Repository Exhaust Filter Building. This building contains the facilities and equipment needed to filter ventilation exhaust air from underground repository areas.

D.2.3.9 Radwaste Building. This building houses facilities for processing the liquid and solid radioactive waste generated on site. The working environment in the liquid radwaste areas is monitored for alpha- and beta-emitting particulates. Area radiation monitors operate continuously in each room or area in which radwastes are handled. All controls for operation of the radwaste treatment systems are located in a control room. After treatment and encapsulation, the drums are conveyed to the LLW area of the receiving building in preparation for lowering to the underground repository area.

D.2.4 Underground Facilities

D.2.4.1 Design Considerations. The following are primary criteria for underground development of the repository:

- The salt pillar between rooms must provide an adequate heat sink and also maintain the structural integrity of the mine
- The average rate of delivery of waste to the repository is 5,000 MTHM/yr
- The areal heat loading density figure assumes a simultaneous, instantaneous loading of the entire mine
- The underground development area is nominally 2,000 acres
- Space allocation for shielding has to be considered in conceptual designs
- Shafts terminate in a central area
- Wastes are always emplaced first in the outermost available areas of the repository

D.2.4.2 Mine Layout. Room dimensions for the repository are based on the above criteria and on the study bases and assumptions shown in Appendix A, Subsections A-14 and A-17. An underground development schematic, illustrating the central core area, shafts, corridors, and storage rooms, is shown in Figure D-5.

The preconceptual configuration for the mine contains storage rooms connected at right angles to branch corridors. Entrance to these rooms can be from either end. Main corridors extend in two directions from the central core area and afford access to each branch corridor. The main and branch corridors are approximately 30 feet wide and 15 feet high, and each storage room is about 20 feet wide, 15 to 25 feet high, and 1,000 feet long.

Mine layout provides storage areas for spent fuel, HLW, ILW, and LLW. Specific layout requirements and waste emplacement for the various SFDS cases are discussed later in this appendix. Thermal and physical constraints ultimately determine the geometry used to emplace waste packages.

D.2.4.3 Initial Development. Five main corridors are needed to accommodate mined salt conveyors, mining equipment, lift trucks, personnel, and air supply and exhaust flow during mining operations. Support facilities for mining operations are located in the central area. This support area includes personnel accommodations, shops, tool room, storage areas, maintenance areas, and skip loading.

The following is a tentative sequence for the initial development of the repository after the mine and materials shaft has been installed (see Subsection D.3):

- Complete the service area
- Install the mined material handling system
- Complete the main line conveyor system
- Complete the ventilation supply shaft and exhaust shafts
- Put the mining machines in operation
- Complete the waste shaft
- Mine the corridors in the central area
- Mine the main corridors to the branch corridors
- Begin mining the storage rooms

D.2.4.4 Storage Room Excavation. The 20-foot-wide storage rooms are separated by pillar widths that are set by the extraction ratios for each specific case. The height of the room takes into account the movement of the ceiling and floor from thermally induced salt creep. The mine closure rate depends on the properties of the salt and on the depth and geometry of the repository. On the basis of the overall requirements for a generic repository, a room displacement of approximately 10 percent for a 5-year period is considered to be a design constraint.

As mining continues past the initial development stage, excavation production rates peak and remain constant for the 5-year retrievability period. All salt excavated is hoisted to the surface for stockpiling. No backfilling is scheduled during the first 5 years of emplacement. Mining machines are operated on a round-the-clock basis.

The sequence of completion of room preparation is such that waste emplacement does not occur when mining activities are taking place in that branch corridor.

D.2.4.5 Terminal Storage Wells. Storage wells are drilled three rows to a room, and the waste is emplaced using the equipment and procedures described in Subsection D.2.7. In the Reference Case, HLW packages are stored in one of three rows to comply with the areal heat loading requirements; in Cases 1, 2, and 3, spent fuel packages are stored in two of the three rows and ILW packages are stored in the third row. The spacing of storage wells varies with the heat load and the type of waste to be stored. Storage wells vary in depth from 16-1/2 to 25 feet. Minimum well spacing is set such that the thickness of the undisturbed salt web between the wells is equal to the diameter of the wells. A detailed description of waste emplacement, which includes the ILW and LLW, is found in the "Waste Emplacement" subsection for each of the cases studied.

D.2.4.6 Underground Operation. The underground operation covers the transfer, handling, and emplacement of spent fuel and/or waste canisters and LLW drums, after these canisters and drums have been prepared in the receiving building.

Canisters. At the waste shaft collar station, a surface effect vehicle places the transfer cask, which is loaded with one canister, into the hoist cage. The cask is then lowered to the underground station where it is removed from the hoist cage and transported to the crane area by a surface effect vehicle. Next, a crane lifts the cask onto a transporter vehicle, which hauls it to the storage room entrance (see Figure D-7), at which point a gantry crane picks up the cask and transfers it to the terminal storage well (see Figure D-9). The gantry crane then removes the shield plug from the well and positions the cask over this well. The canister is emplaced using a captive hoist, and the shield plug is reinserted in the well. (If an ILW canister is being emplaced, a shield plug is not employed; instead, crushed salt is charged into the well from a salt hopper mounted on

NOTES

1. FOR WASTE DISPOSAL ROOMS AND PILLARS DETAILS SEE FIGURES D-10 THRU D-14.
2. TOTAL UNDERGROUND DEVELOPMENT AREA IS 2000 ACRES.
3. CENTRAL PILLAR AREA CONTAINS ALL MINE SUPPORT FACILITIES.

SUMMARY TABLE

CASE NO.	TOTAL * CANISTERS PER YEAR	PILLAR WIDTH (FT.)	ROOMS PER YEAR	ROOM/UNIT CELL AREA (ACRES)	AREA FILLED PER YEAR (ACRES)	TOTAL NO. OF ROOMS	REPOSITORY USEFUL LIFE (YEARS)	REMARKS
REFERENCE	14,824	142	11	3.8	43	429	39	EVERY FIFTH PILLAR IS EXCAVATED FOR ILW ONLY, RESULTING IN 6 FT. WIDE PILLARS
1	12,796	130	28	3.5	99	476	17	
2	13,520	179	19	4.7	90	361	19	
3	28,438	49	60	1.6	97.5	1050	17.5	

* INCLUDES ALL TYPES OF WASTE.

Figure D-5.
Generic Repository — Underground
Development Schematic
All Cases

the crane.) The empty cask is now moved by gantry crane to the storage room entrance, returned by transporter vehicle to the waste area, and hoisted to the surface to receive another canister. The active cycle is then repeated.

LLW drums. The LLW drums, six drums to a pallet, are moved by forklift truck into the waste shaft hoist cage and lowered to the underground station. There, the palletized drums are loaded on a pallet transporter and taken through the mine corridors to the final storage room.

D.2.5 Ventilation

Ventilation of the underground area in this study is an adaptation of conceptual approaches developed by SRENCO for NWTS Repository No. 1^{(1),(2),(3)} and by Kaiser Engineers for NWTS Repository No. 2^{(4),(5),(6)}. Essentially, the shaft arrangement and central core are based on the NWTS Repository No. 1 concept, while the air distribution and control are based on NWTS Repository No. 2 concept. A composite schematic of the repository, which also includes the ventilation system, is shown in Figure D-5.

The purpose of mine ventilation is to minimize the exposure of personnel to airborne radon decay products, dust from mining operations, radioactive particulates entrained from and by waste containers, and toxic gases released from internal combustion engines. In addition, sufficient conditioned air must be supplied for life support and personnel comfort.

D.2.5.1 Supply Air System. A single aboveground source, situated in the ventilation supply filter building, supplies the repository with conditioned air, which is distributed, as required, to the development areas and to the waste emplacement areas. The supply air for the central pillar area is separately provided from the man and materials filter building, via the man and materials shaft.

Air leaving the blowers flows through the ventilation supply shaft and discharges into the central core area of the mine through air supply corridors. Bulkheads installed across these corridors and across the rooms interrupt the flow of air into unoccupied areas, thereby diverting the air to occupied areas. About half of the supply air flow is directed into active mining areas, the remaining air flow is directed into waste emplacement areas.

Since the waste storage depends only on heat dissipation through the salt media, ventilation of rooms is not required after waste emplacement crews vacate an area.

D.2.5.2 Exhaust Air System. The air exhaust is handled by two independent systems: the development exhaust system and the repository exhaust system.

Development exhaust system. This system serves those areas where active mining operations are taking place. The air is returned via the development exhaust shaft to the development exhaust filter building, where, after rough filtration, it is exhausted to the atmosphere through the ventilation stack. If particulate monitors indicate the presence of radioactive contamination, the air is passed through standby HEPA filters before being exhausted to the atmosphere.

Exhaust air from active mining areas is drawn by portable ventilation blowers into ducting and discharged into nearby exhaust corridors. During the initial pass with the mining machine into a branch corridor or room, an exhaust duct is located close to the mining machine to capture dust generated during mining operations. When a second pass is made, ventilation air is drawn through the corridors and rooms by a portable fan.

Exhaust air from active backfilling areas is drawn by portable ventilation blowers and ducting located close to the spreader conveyor (between the discharge point and personnel-occupied areas). Portable ducting that terminates in exhaust ducts is used in active mining operations.

Repository exhaust system. This system serves those areas where active waste emplacement is occurring. Exhaust air leaving corridors or rooms in the waste emplacement area is drawn through the repository exhaust shaft by blowers located in the repository exhaust filter building. Prior to discharge to the atmosphere, air passes through successive stages of moisture separators, two stages of roughing filters, first-stage HEPA filter (possible charcoal beds in Case 1), and final HEPA filters.

As waste transport vehicles and gantry cranes move waste containers into the area, exhaust air from active waste emplacement areas exits from the far end of corridors and rooms. Thus, fresh air enters at the near end of corridors and rooms and acts to sweep contamination away from operating personnel. After the storage room has been filled with waste, bulkheads are installed to isolate that room and redistribute supply and exhaust air flows into the next active storage room.

D.2.5.3 Monitoring Exhaust Air. Monitors in both exhaust streams collect particulate samples for radioactivity analyses. Other monitors (in the Reference Case, Case 2, and Case 3) detect any leakage of radioactive noble gases (e.g., ^{85}Kr), which are kept in cylinders and stored in the repository. Appropriate alarm or action functions are integrated into all monitoring systems.

D.2.6 Shafts

Five shafts are needed to develop and operate the repository:

- Canistered waste shaft
- Man and material (M&M) shaft
- Ventilation supply shaft
- Development exhaust shaft
- Repository exhaust shaft

A schematic drawing of each of these shafts is presented in Figure D-6; the positions of these shafts with respect to the underground facilities can be seen in Figure D-5. All five shafts are located over a central core area of the repository and, for the most part, are arranged on a common center line. The shafts serve the minus-2,000-foot level of the repository and are lined with concrete from the surface level to a depth of 100 feet into the salt formation.

The canistered waste shaft, the M&M shaft, and the development exhaust shaft are equipped with a hoisting apparatus and leave a concrete plug at the shaft bottom. Shaft headframe structures are used above those shafts to support hoists and related equipment.

During construction, a grout curtain is erected around the perimeter of each shaft to prevent excessive amounts of water from entering the shaft excavation. Once the excavation has fully penetrated the aquifer, a permanent steel-lined concrete hydrostatic liner is grouted in place, making the shaft watertight.

D.2.6.1 Canistered Waste Shaft. The canistered waste shaft provides a means for transporting canisters from the canistered waste receiving building to the underground storage level. The shaft is divided vertically into three compartments: two waste transport compartments, and a manway and utility compartment.

A transfer cask containing packaged waste is placed in the shaft cage in one of the two waste compartments. Depending on the specific repository requirements described in Subsections D.3 through D.6, the shaft accommodates HLW, spent fuel, and LLW and ILW containers. The shaft penetrates below the minus-2,000-foot level to accommodate hoist equipment and a sump. The two waste cages operate in balance, each acting as a counterweight to the other.

Figure D-6.
Generic Repository - Shafts Configuration
All Cases
From Stearns-Roger
conceptual design⁽³⁾

The headframe structure in the canister waste receiving building supports the cages and hoisting equipment. The hoist operator's console associated with this structure is enclosed and has a controlled atmosphere.

D.2.6.2 Man and Materials (M&M) Shaft. The M&M shaft is used to (1) transport all personnel, nonradioactive material, and communications and utility lines between the surface facilities and underground development areas, and to (2) provide a manway for emergency exits of all personnel. The shaft is divided vertically into an M&M compartment, two skip compartments, and a manway and utility compartment.

The shaft has a skip loading station and a shaft cleanup station below the minus 2,000-foot level. The headframe structure associated with this shaft supports a skip hoist, cage hoist, conveyances, and equipment, and uses two hoisting systems, one for the M&M cage and one for the skips.

D.2.6.3 Ventilation Supply Shaft. Ventilation air supplied from the ventilation building is passed through the ventilation shaft to the underground areas. A headframe structure supports a cage hoist, cage and counterweight, and associated equipment.

D.2.6.4 Development Exhaust Shaft. Ventilated air from underground development areas exhausts through the development exhaust shaft into filtration systems located in the development exhaust filter building. This shaft is sized to accommodate anticipated peak air flow rates during mining operations. No headframe structure is required. In this present concept, all exhaust air from underground development is filtered through prefilters and final filters before being discharged to the atmosphere.

D.2.6.5 Repository Exhaust Shaft. Exhaust air from the waste repository is directed through this shaft into the repository exhaust filter building, where it is filtered through both roughing and HEPA filters.

D.2.7 Specialized Equipment

D.2.7.1 Surface Equipment. Remotely and locally operated material handling equipment, presently well known in the nuclear industry, is used in the surface facilities. Such equipment includes cranes, forklifts, remote manual manipulators, remote powered manipulators, transfer fixtures, remote welding apparatus, etc. Conveyors and grading equipment are employed to distribute and groom the salt stockpile.

D.2.7.2 Underground Development Equipment. A full complement of mining and waste emplacement equipment is used in the mine. The most important pieces of major equipment are as follows:

- Continuous borer
- Conveyor belt system
- Drilling rig for storage wells
- Sleeve and well plug emplacement equipment
- LLW drum handling equipment
- Transfer cask
- Transporter
- Gantry crane

Continuous borer. A continuous borer excavates corridors and storage rooms, and a belt conveyor system carries the salt to a skip hoist for transfer to the surface. The continuous borer mines an 11-foot-high and an 18-1/2-foot-wide swath with two rotating cutting heads and trimming chains, at an advancing speed up to 18 in./min (200 tons/hr of salt). Plows on rotating units move the salt taken from the salt face to the throat of the mining machine conveyor. The continuous borer mining machine also drills holes in the floor for the supports of the expandable belt conveyors.

Conveyor belt system. The salt from the continuous borer discharges onto a belt conveyor mounted on a maneuverable bridge system. This belt conveyor is connected to an expandable conveyor, which, as it changes length, releases belting from a multiple takeup belt storage. The expandable conveyor system consists of a series of interlocked 36-inch wide belt conveyors which can be extended to the farthest point of the mine.

The expandable belt system transports the mined salt from the terminal storage room through branch corridors to the fixed-length conveyor belt located in the main corridor, which, in turn, carries the salt to surge bins in the mine shaft area. A skip hoist moves salt from the surge bins to the surface for distribution by belt conveyor and stacker-reclaimer in open storage piles. When the continuous borer completes a terminal storage room, the operators dismantle and transfer the expandable belt conveyor to the site of the next room.

After a 5-year retrievability period, the terminal storage rooms containing emplaced waste are backfilled. In addition to using salt from rooms newly opened, additional salt, as needed, is returned from the surface storage piles by way of the skip hoist to the mine receiving bin and carried to the storage rooms by means of the expandable belt conveyor system. From then on, over the remaining operating life of the repository, the new excavated rooms are backfilled immediately after they are filled with waste.

Drilling rig for storage wells. The burial wells in the terminal storage room floor are bored by a mobile drill. To remove the salt cuttings, conventional load-haul dump equipment is used to feed the expandable belt conveyor located in the branch corridors. A rotary drill is mounted on a traveling bridge supported on rails laid next to the side walls of the storage room. The drill frame can be braced against the room ceiling during drilling operation. The rotary drill has telescoping augers and a tilting drill frame for traveling or maintenance.

Sleeve and well plug emplacement equipment. Metal sleeves, which facilitate canister retrieval, are inserted in each terminal storage well to provide a secondary barrier for the canisters emplaced during the first 5 years of repository operation. The sleeve is recessed in the well to accommodate a concrete shield plug above the waste canister. A rail-mounted mobile crane emplaces sleeves and well shield plugs. The vehicle has four-wheel drive and four-wheel steering to enhance maneuverability.

LLW drum handling equipment. The LLW drums arrive from the waste shaft already palletized in a six-pack array. Forklift trucks and a transporter transfer pallets to the LLW storage rooms, where palletized drums are stored one drum high.

Transfer cask. An onsite transfer cask, which provides both gamma and neutron shielding, receives one canister from the hot cell at a time for delivery to the storage wells located in the underground terminal storage room. The outer carbon steel shell is coated for easier decontamination; the interior cylinder in contact with the canister is stainless steel. The cask is of a bottom-load/bottom-unload design with a double sliding door, and is surmounted by an integral captive winch that hoists canisters into and lowers canisters out of the cask.

A bridge crane places the transfer cask over the discharge port in the ceiling of the hot cell. After filling the cask with one canister, the crane places the cask onto an air pallet, which is then moved through an isolation corridor to the waste shaft hoist cage. At the underground level, the cask is

resupported by the air pallet and the operator transports the cask through an airlock to the cask transfer area. A crane then hoists the cask, with a yoke, and places it on the cask transporter.

Transporter. The transporter, illustrated in Figure D-7, carries the transfer cask from the mine shaft area to the corridor outside the storage room. This vehicle travels on rubber tires and consists of a standard prime mover and a specially designed trailing unit with articulated connection to facilitate steering and make possible a short turning radius. The operator's cab is air conditioned and shielded against radiation.

At the transfer cask loading area, a crane lowers the cask, suspended in a vertical position, onto the transporter, which then tilts the cask about the trunnions to approximately 15 degrees above the horizontal. This nearly horizontal position during transport lowers the head room requirements, while keeping the captive hoist rope in tension to minimize canister shifting in the cask. The transporter deposits the transfer cask on the floor outside the storage room and takes back an empty cask to the mine shaft area.

Gantry crane. Because of the large number of holes in the storage room floor, it is felt that a gantry crane, illustrated in Figure D-8, provides the most economical operation. This crane picks up the transfer cask and moves it to the canister storage position, and a secondary hoist on the crane removes the shield plug at the top of the storage well. Next, the main gantry hoist moves the cask over the well, the shield doors open, and the captive hoist lowers the canister into the well. The secondary hoist then moves the shielding plug housing and shield plug in place as the cask is moved away from the well, and the operator lowers the shield plug in place from his shielded control cab. The cab has been designed to reduce scattered radiation emitted through the slight gap between the cask and shield plug housing as the latter is being moved over the storage well. Figure D-9 illustrates the above sequence of operations.

D.2.8 Repository Validation

A validation program is conducted during the initial years of repository operation. Until the repository site has been shown to be adequate, all HLW and spent fuel is maintained in a retrievable mode. For the purpose of the SFDS, this validation period lasts 5 years. The effect of 10- or 25-year retrievability is also examined in a study variation.

Following the emplacement of the heat-producing wastes in the repository wells, the temperature of the sleeves and the surrounding medium rises, salt begins to creep toward the sleeves lining the storage wells and reduces the volume of excavated rooms and corridors, and moisture may be drawn from the salt medium toward the waste heat sources.

Monitoring to ensure that actual repository behavior conforms to predicted behavior is the principal means of achieving repository validation. In general, validation of a repository can only attempt to compare empirically determined short-range changes with predicted short-range changes anticipated during the preconceptual design. Long-range changes have to be related to short-range changes through previously acquired knowledge or by multidisciplinary ongoing research and development.

D.2.8.1 Validation Criteria. Probably the most important effects to be evaluated are the changes in temperature values and distribution during the retrievability period. The effects of heat transfer from the waste container across an air gap to the sleeve, through the sleeve, across a second air gap into the salt medium, and outward into the salt pillar should be monitored. Special consideration should be given to the rate of temperature increase in the salt web between successive canisters in adjacent positions (within rows and between rows). Comparisons should be made with predictions made by SAI and others in order to find out whether ultimate near-field and far-field temperatures were conservatively predicted.

The life expectancy of canister and sleeve materials in the repository environment should be ascertained. Both the rate and nature of the corrosion should be determined empirically; factors that could contribute to corrosion, such as migration of moisture toward these heated surfaces, should also be analyzed.

During repository validation it should be ascertained that flowing water is excluded from the mine. Ultimately, exploratory bore holes and all shafts penetrating the formation must be sealed against water intrusion.

D.2.8.2 Monitoring Program. In this program, the following items are monitored:

- Temperature
- Corrosion
- Water intrusion

Temperature. Temperature changes in the storage wells and nearby salt have to be measured during the retrievability period. To do this, one installs selected instrumentation in randomly located, high-level waste (or spent fuel) canisters, storage wells, rooms, and nearby salt pillars. The primary canister should be chosen to provide an upper limit heat source for the wastes currently being received, and storage wells and nearby pillars and rooms should be instrumented during sleeve emplacement operations in anticipation of receiving an instrumented canister. A sufficient quantity of thermowells should be inserted in the salt medium to permit lateral and vertical temperature distribution to be determined. The following temperatures are measured: lateral temperatures out to the pillar centerline in the plane of the waste centerline, and about half way through the pillar at an elevation of the midplane of the room; vertical temperatures from the floor of the room to about 20 feet below the waste canister. The number of canisters to be instrumented should be determined; in the Reference Case, if 1 percent of the vitrified HLW canisters are selected, about two instrumented canisters per month will have to be emplaced.

Secondary evidence of effects of waste heat and room closure should be documented. Preferably, a significant segment of a selected storage room should be loaded with similar high-heat waste so that local room closure rates may be related to areal heat loading. Mine structural engineers should be consulted to determine the location of this uniformly loaded section. Single thermocouples on each storage well are probably adequate for estimating the individual canister heat load. Room closure rates for other storage rooms should be roughly related to the total estimated heat loading taken from shipping data on received waste.

Corrosion. Corrosion of confinement barriers is measured in three ways:

- Corrosion samples are inserted into small, open-ended wells near instrumented waste containers; these samples are then withdrawn, weighed, and then undergo a metallographic analysis
- A small fraction of the actual waste sleeves are unloaded, and the sleeves are removed and destructively investigated in the underground shielded cell. This investigation includes a measurement of the deformation of the canister from salt creep (especially for the 25-year retrieval option) and external corrosion
- A small fraction of containers of wastes containing fixed liquid wastes (uncalcined) is destructively investigated to ascertain waste form stability and internal corrosion within the canister

The moisture content in the nearby salt should be measured periodically, either directly or by probe sampling through an open-ended well, if such arrangement can provide representative moisture values.

1 GANTRY CRANE LOADING TRANSPORTER VEHICLE WITH FULL CASK AT UNDERGROUND SHAFT STATION

2 CASK POSITIONED IN TRANSPORTER. LIFTING ARMS ENGAGE TRUNNIONS ON CASK.

3 CASK LOADED IN TRANSPORTER. GANTRY CRANE RETURNS TO SHAFT STATION.

4 ACTIVATE CASK TILTING CYLINDERS.

5 CASK TILTED TO TRANSPORTING POSITION. LOWER LIFTING ARMS.

6 CASK IN TRANSPORT POSITION.

7 TRANSPORTING CASK TO ACTIVE STORAGE ROOM.

8 ARRIVAL OF TRANSPORTER AT STORAGE ROOM. LIFTING ARMS ARE RAISED. TILTING CYLINDERS ACTIVATED TO POSITION CASK UPRIGHT.

9 CASK IN UPRIGHT POSITION. SPOT CASK AT UNLOADING STATION. PREPARE TO LOWER CASK TO FLOOR.

10 CASK POSITIONED. LOWER LIFTING ARMS TO PLACE CASK ON DRIFT FLOOR.

11 LIFTING ARMS DISENGAGED FROM CASK TRUNNIONS.

12 TRANSPORTER LEAVES FULL CASK. TRANSPORTER MANEUVERED TO LOAD EMPTY CASK; RETURNS TO SHAFT STATION UNDERGROUND

Figure D-7.
Generic Repository
Sequence of Operations for
the Transfer Cask Transporter
All Cases

From Stearns-Roger
conceptual design⁽³⁾

Figure D-8.
Generic Repository — Canister
Storage Room Gantry Crane
All Cases

**From Stearns-Roger
conceptual design⁽³⁾**

Figure D-9.
Generic Repository – Sequence
Operations for Cask and
Canister in the Storage Room
All Cases

From Stearns-Roger
conceptual design⁽³⁾

Water intrusion. Water intrusion should be continuously measured in shaft sumps, main corridors, etc. during the mining, emplacement, and retrievability periods. This monitoring is performed using typical liquid level instrumentation and typical liquid transfer equipment. The maximum rate of allowable intrusion will probably be specified in a technical specification which defines proper operating conditions and has been negotiated between regulatory and operating personnel. A normal operating limit and a limit prior to mine backfill or decommissioning phases will also probably be specified. Validation of compliance will require proof of water intrusion rates.

D.2.8.3 Adequacy of Validation. If temperature values or distribution indicates greater thermal impact than is predicted from emplaced heat loadings, the thermal calculations have to be updated, and revised projections of ultimate near- and far-field effects have to be made. If the analysis indicates that the final temperature values are undesirable, some of the waste containers can be relocated quite easily. Empirical data can also be used to recalculate future, less compact, emplacement geometries and pillar widths to reduce further potential uncertainty.

If corrosion rates appear to be excessive, it signifies a shorter life for the local confinement barriers, which, in turn, may affect the overall environmental impact. Since the most significant barrier is the storage medium itself, significant changes in the life of local barriers may have little influence. If calculated overall release rates based on empirical data indicate that the area surrounding the mine may be compromised because of excessive corrosion rates, the wastes in question can be relocated into more corrosion-resistant materials. During the retrievability period, measured corrosion rates should show that sleeves are no longer required and that the primary canister is a sufficient local barrier. If that conclusion is not valid, sleeves will have to be used during the postretrievability period also.

Water intrusion limits have to meet technical specification limits or corrective action will have to be taken to further seal penetrations into the repository.

D.3 REFERENCE CASE

D.3.1 Specific Repository Aspects

In the Reference Case, the repository receives, handles, and processes packages containing the various types of radioactive material identified in Table C-1. The site, surface facilities, shafts, equipment, underground development, and operations at the repositories are generally as described in Subsection D.2. The only significant departure from the generic repository is in the receiving building and the underground development. Specific requirements for these two areas for the Reference Case are described below.

D.3.1.1 Receiving Building. The receiving building has three hot cells connected to one waste shaft. In these facilities, 14,788 canisters (6,238 of HLW and 8,550 of ILW) are processed and handled each year. For the SFDS, the LLW building, shown in the Stearns-Roger Engineering Co. reference design,⁽¹⁾ was deleted. The LLW is handled through the receiving building, as discussed in Subsection D.2.3.

Surge capacity is provided to account for 10 days of shipping cask receipts. This represents the in-transit time for the case of a repository shutdown and is based on an average distance between the P/E facility and the repository of 1,100 miles and on a mean train speed of 5 mph. If these figures are used, the required surge capacity for the Reference Case is 405 canisters, stored in an area of approximately 2,000 ft.².

D.3.1.2 Underground Development. Of the 2,000-acre total mined repository area, approximately 1,700 acres are available for storage rooms, including the adjacent pillars and branch corridors. The unit cell for each storage room, including appropriate portions of adjacent corridors and pillars, is 1,030 feet long and 162 feet wide. The 142-foot pillar width is based upon the heat dissipation requirement of vitrified HLW. The area of each unit cell is 3.8 acres.

Rooms 20 feet wide and 1,000 feet long are excavated into every fifth pillar to accommodate ILW. ILW does not produce enough heat to enter into repository thermal considerations.

In the Reference Case, 11 unit cells, or about 43 acres, are filled every year. Therefore, the repository has a 39-year (1,700/43) useful life.

Assuming the storage room closes in by 10 percent during the retrievability period (5 years), a 19-1/2-foot room height will leave the necessary head room for the retrieval of canisters into a 15-foot-tall transfer cask. The two rooms to be filled each year with ILW could be 17-1/2 feet high since no retrieval of that waste is required.

D.3.2 Waste Emplacement

In the Reference Case, about 13 rooms per year are filled with canistered waste. Each storage room has three rows of storage holes (wells), as illustrated in Figure D-10. Pillar widths for vitrified HLW are set by thermal considerations. Rooms for storage of only ILW are inserted in pillars between HLW storage rooms, since the heat dissipation is not adversely influenced. This is shown in Figure D-11. Table D-4 summarizes the way the storage space, shown in Figures D-10 and D-11, is utilized.

VITRIFIED HLW

HULLS, END FITTINGS
AND GASES (HLW)

ILW

ROW	CANISTER CONTENT	NO. OF HOLES IN ROOM	NO. OF ROOMS FILLED PER YEAR	CANISTER 3/8" WALL THK		HOLE		SLEEVE 3/8" WALL THK	PLUG	COMPARATIVE LEVEL OF HEAT	REMARKS
				(A) OD"	(B) LENGTH'	(D) DIA."	(H) LENGTH'				
X	ILW	210	11.3	20	10 1/2	24	25	NONE	NONE	LOW	CANISTER LENGTH INCLUDES LIFTING FIXTURE (TYP.)
Y	VITRIFIED HLW	211	11.3	12 3/4	10 1/2	28	16 1/2	16	5 1/2	HIGH	
Z	HULLS, END FITTINGS AND GASES (HLW)	210	9.2	20	10 1/2	26	25	22	3 1/2	LOW	
	ILW		2.1			24		NONE	NONE		

Figure D-10.
Geologic Repository
Waste Emplacement Schematic
for the Retrieval Period
for HLW/ILW Rooms
Reference Case

ROW	CANISTER CONTENT	NO. OF HOLES IN ROOM	NO. OF ROOMS FILLED PER YEAR	CANISTER <small>3/8" WALL THK</small>		HOLE		COMPARATIVE LEVEL OF HEAT	REMARKS
				(A) OD"	(B) LENGTH'	(D) DIA"	(H) LENGTH'		
X	ILW	246	2	20	10 1/2	24	25	LOW	CANISTER LENGTH INCLUDES LIFTING FIXTURE (TYP.)
Y	ILW	245	2	20	10 1/2	24	25	LOW	
Z	ILW	246	2	20	10 1/2	24	25	LOW	

Figure D-11.
Geologic Repository
Waste Emplacement Schematic
for the Retrieability Period
for ILW Rooms Only
Reference Case

Table D-4
ANNUAL UTILIZATION OF REPOSITORY ROOMS

Approximate Number of Rooms	Waste Materials Emplaced		
	One Outside Row	Center Row	Other Outside Row
9	ILW	HLW vitrified	HLW, low-heat- producing (hulls, end fittings, ⁸⁵ Kr)
2	ILW	HLW vitrified	ILW
2	ILW	ILW	ILW

D.3.2.1 Emplacement-Retrievability Period. During the retrievability period (first 5 years), the storage wells for HLW are equipped with steel sleeves to facilitate the retrieval operation. Each sleeve has a flange at the top end which supports a concrete shielding plug with a recessed lifting fixture.

The diameter of each storage well for high-heat-producing waste is 12 inches larger than the steel sleeve O.D. and provides an annular space around the sleeve. The air gap should prevent damage to the sleeve from the creep effect of salt due to the heat generated during the retrievability period. Low-heat-producing wastes require less air gap.

ILW canisters are stacked two-high (piggyback) in the storage wells, without steel sleeves or concrete plugs. Since this type of waste is not retrievable, the well will be filled with crushed salt after the canister is in place. The wells are also large enough to accept overpacked canisters, if necessary.

The sleeves identified for vitrified HLW in Figure D-10 are designed to accommodate the basic single-pack canister. Since the SFDS assumes that less than 2 percent of the canisters will require overpacking, no further sleeve design perturbations are made. Sleeve allowance for overpacked canisters can be readily handled in a repository (detailed) design effort.

The LLW, packed in 55-gallon drums, is stored six drums to a pallet in separate rooms. When the retrievability period ends, the drums are relocated in the HLW/ILW-filled rooms, as described below in Subsection D.3.2.2.

D.3.2.2 Emplacement-Postretrievability Period. Once the retrievability period is over, storage wells for HLW can be of smaller diameter, since steel sleeves and concrete plugs are no longer required. After canister emplacement, the storage wells are filled with crushed salt. The depth of the wells will be increased slightly to provide salt shielding above the canister equivalent to that provided by the concrete plugs.

Prior to the start of the backfill operation, some of the LLW stored in separate rooms is moved to the HLW/ILW-filled rooms and placed on the floor in a single layer.

D.3.3 Operational Considerations

The operational considerations presented here are based on the material developed in Subsections D.2.3, D.2.4, and D.2.7. Additional information and assumptions are included, as needed, for arriving at operating crew requirements.

The principal operations at the repository for the Reference Case are:

- Receiving, handling, processing, and transporting radioactive waste packages (HLW, ILW, and LLW) above ground
- Handling, transporting, and emplacing radioactive waste packages below ground
- Mining repository corridors and storage rooms
- Backfilling repository corridors and storage rooms
- Managing the surface salt stockpile

D.3.3.1 Aboveground Operations. The aboveground operations taking place in the receiving building include the steps described in the flow diagram for canistered HLW (Figure D-3) and in the flow diagram for LLW (Figure D-4).

Handling HLW and ILW. An average of six casks per day are received at the facility by rail and truck (see Table E-9), transporting HLW, hulls, and ILW canisters. As shown in Table E-9, 6,238 canisters of HLW and 8,550 canisters of ILW are handled per year in the remote operation portion of the receiving building. It is assumed that the facility operates on a 300 day/yr, 24 hr/day schedule, and that three hot cells are used for processing canisters.

The operating crew requirements for handling one shipment of HLW or ILW above ground are given in Table D-5.

Table D-5
OPERATING CREW REQUIREMENTS FOR HANDLING
ONE SHIPMENT OF HLW OR ILW ABOVE GROUND

Item	People	Crew Time hr	Operations per Shipment	Total Manhours
Receive cask	3	4	1	12
Offload cask	2	1	1	2
Mate to cell	2	1.5	1	3
Unload cask	1	8	1	8
Overpack (1%)	(1)	(8)	—	—
Load onsite cask	1	1	9-12	10
Transfer to hoist	2	0.5	9-12	10
Initiate descent	1	0.5	9-12	5
Prepare cask for shipment	3	2	1	6
General duties (20% of above)	—	—	—	9
Total manhours per shipment				65

The total operating staff for handling all shipments of HLW and ILW above ground is given in Table D-6.

Table D-6
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF
HLW AND ILW ABOVE GROUND

Personnel	Number Required
Shift workers (17 x 4 shifts)	68
Day shift	0
Supervisor (1/shift, 1 on days)	5
Total	<hr/> 73

Handling LLW. The LLW is received by truck, with one shipment arriving approximately every week (40 shipments annually on a 365-day schedule). As shown in Table E-9, there are 1,680 LLW (TRU) drums per year sent to the underground storage rooms; a small fraction of these drums originates on site.

The LLW drums are handled in the contact operation portion of the receiving building. For the purpose of this study, all LLW drums have been assumed to be below a 10 mrem/hr dose rate. During detailed design, the LLW-HLW interface dose rate would have to be more accurately determined from time and motion estimates, the waste source term, distance to operators, and information on interposed shielding materials. By balancing those many factors, the dose rate limit from LLW might be set much higher than 10 mrem/hr without exceeding as low as reasonably achievable (ALARA) guidelines.

The operating crew requirements for handling one shipment of LLW above ground are given in Table D-7.

Table D-7
OPERATING CREW REQUIREMENTS FOR HANDLING
ONE SHIPMENT OF LLW ABOVE GROUND

Item	People	Crew Time hr	Operations per Shipment	Total Manhours
Receive transport	2	2	1	4
Decon. carrier	2	4	1	8
Offload carrier	2	1	1	2
Move surface effect vehicle to unloading	2	1	1	2
Unload carrier	1	2	14	2
Decon. canister	1	1/4	14	3.5
Overpack (1%)	1	1	0.14	0.1
Weigh, final survey	1	1/2	14	7
Pelletize drums	1	1/2	2.5	1.3
Move to hoist	1	1/4	2.5	0.6
Initiate transfer to mine	1	1/4	—	0.25
General duties (20% of above)	—	—	—	6
Total manhours per shipment				<hr/> 37

The operating crew handles LLW above and below ground. The total operating staff for handling all shipments of LLW above ground is given in Table D-8.

Table D-8
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF LLW
ABOVE GROUND

Personnel	Number Required
Shift workers	0
Day shift	3
Supervisor	<hr/> 0
Total	3

D.3.3.2 Belowground Operations. The operations taking place below ground are divided into five categories: handling HLW and ILW, handling LLW, mining storage rooms and corridors, back-filling storage rooms and corridors, and managing the surface salt stockpile.

Handling HLW and ILW. This operation covers the activities required for moving the transfer cask from the bottom of the waste shaft to the storage rooms and for emplacing the cask in the storage wells. As shown in Table E-9, a total of 14,824 HLW and ILW canisters per year are emplaced in the repository. Thus, over the anticipated operating schedule of 300 days/yr and three shifts per day (effective work time, 6 hours per shift), 50 transfer casks per day are to be handled. Because this schedule requires 22 minutes of turnaround time for the casks, emplacement operations have to take place in two storage rooms simultaneously to satisfy the demands of logistics. It is, however, assumed that 22 minutes is ample time for the hoist descent and ascent in the shaft.

The operating crew requirements for handling one shipment of HLW and ILW below ground are given in Table D-9.

Table D-9
OPERATING CREW REQUIREMENTS FOR HANDLING
ONE SHIPMENT OF HLW OR ILW BELOW GROUND

Item	People	Crew Time hr	Operations per Shipment (Approx.)	Total Manhours
Cask from hoist to transporter	0.25	2	10	5
Drive to storage room	0.25	2	10	5
Uncover well	0.1	1	10	1
Transport to well using gantry (@ 80 fpm, 250 avg. travel) round trip	0.1	1	10	1
Unload cask	0.25	2	10	5
Install plug	0.25	2	10	5
Return cask to transport	0.05	1	10	0.5
Return cask to hoist	0.25	2	10	5
General duties (20% of above)				5.5
Total manhours per shipment				33

The total operating staff for handling all shipments of HLW and ILW below ground is given in Table D-10.

Table D-10
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF
HLW AND ILW BELOW GROUND

Personnel	Number Required
Shift workers (9 x 4 shifts)	36
Day shift	0
Supervisor	5
Total	<u>41</u>

Handling LLW. LLW drums are transported in a palletized mode to an empty room and placed on the floor. Prior to backfilling operations, the drums are transferred to HLW/ILW rooms. The number of drums is indicated in Subsection D.3.3.1.

The total operating staff for handling one shipment of LLW below ground is given in Table D-11.

Table D-11
TOTAL OPERATING STAFF FOR
HANDLING ONE SHIPMENT OF LLW BELOW GROUND

Item	People	Crew Time hr	Operations per Shipment	Total Manhours
Receive and unload transport	2	1/4 per pallet	3	1.5
Deposit in room and return	2	1/2	1	1
Unload and store	2	1/4	3	1.5
Total manhours per transporter load				<u>4</u>

Note: These manhours have already been included in Table D-8; one crew handles LLW above and below ground.

Mining storage rooms and corridors. The SFDS requires specific retrievability capabilities for a specified period during initial repository operation. During that period, the control area and all main drifts will have been excavated, and all excavated salt from corridors, rooms, and storage wells will have been hoisted to the surface and added to the stockpile. Ordinarily, after retrievability requirements have been dispensed with, the salt excavated in one area is used to backfill rooms that have already been filled with wastes in other areas. Then, only excess salt has to be taken to the surface stockpile. Each year, 11 rooms and their supply corridors have to be mined. In one of the rooms, there is temporary storage of LLW drums prior to moving them to filled HLW/ILW rooms. Therefore, no special rooms for LLW are provided.

During the retrievability period the equipment required to excavate and transport about 500,000 tons of salt per year consists of one continuous mining machine, a set of continuous conveyors, a skip hoist in its own shaft space, miscellaneous rock-handling equipment at the mining level, and additional transportation and handling equipment at the surface. To hoist this quantity of rock salt, an 80 ton/hr skip hoist is furnished and operated on a continuous basis for 300 days/yr. Some other equipment may be of greater capacity and operate on a somewhat reduced schedule. Two passes are needed to mine the corridors and rooms.

The limiting factor in the mining operation is the capacity of the skip hoist and its feed system, which transport the excavated salt from the mined areas to the surface for storage until the salt is needed for backfilling below ground or until it can be shipped off site as excess material. The mining operation is assumed to be on stream 300 days/yr, and a hoist capacity of 80 tons is provided which requires an 85 percent efficiency of operation factor. The hopper feeding the machine has sufficient capacity to permit 8-hr/day operations. Each day, about 1,700 tons of salt are mined and transported to the surface as the mining operation advances about 125 feet. To cut and trim 4,000 feet of the two-pass branch corridors (some full-height, some half-height) takes about 75 days. In contrast, the 12 rooms that are only 1,000 feet long require about 19 days of excavation each, for a total of 228 days. Smaller equipment is used to trim walls and floors and transport salt material to the conveyor. Periodically, setup crews move and extend the ventilation and conveyor systems, as needed, to support the underground operations.

As soon as a room excavation is completed, the drilling crew enters, drills, inserts sleeves, and plugs the holes. A total of 12 rooms, containing about 8,600 storage wells, are excavated yearly; this comes to approximately 30 wells per day or 10 wells per shift.

Equipment operators perform routine cleanup along conveyor routes, and others are stationed at the skip hoist area to operate the system.

The SFDS has estimated operating crew requirements on the basis of nominal crew requirements per handling system and estimated number of handling systems. Since the study is concerned only with developing relative census factors between the Reference Case and spent fuel cases rather than absolute census figures, this approach is satisfactory. More extensive analysis of absolute census figures has been provided in NWTs Repository No. 1 and Repository No. 2 reports, prepared by Stearns-Roger and Kaiser Engineering.

The operating equipment and crew requirements for developing the mine are given in Table D-12. Note that this also includes data for backfilling the mine and managing the surface salt stockpile, as well as for mining rooms and corridors.

Table D-12
OPERATING EQUIPMENT AND CREW
REQUIREMENTS FOR DEVELOPING THE MINE

Item	Retrievability Period	Postretrievability Period
Mining rigs	1	1
Duty	16 hr x 5 days	16 hr x 5 days
People (14/crew)	31	31
Drilling rigs	1	2
Duty	24 hr x 7 days	24 hr x 7 days
People (4/rig)	37	37
Skip hoist rigs	1	1
Duty	8 hr x 5 days	1 day/wk
People (4/rig)	6 (2 supervisors)	5 (1 supervisor)
Sleeving rigs	1	—
Duty	16 hr x 5 days	—
People (4/rig + warehouse to mine)	14	—
Backfill rigs	—	1
Duty	—	16 hr/day
People	—	27 (3 supervisors)
Stockpile, people	7 (1 supervisor)	3 (1 supervisor)

The total operating staff for mining storage rooms and corridors is given in Table D-13.

Table D-13
TOTAL OPERATING STAFF FOR
MINING STORAGE ROOMS AND CORRIDORS

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers (variety of shift schedules)	71	60
Day workers	4	4
Supervisors (11 on shift, 2 on days)	13	9
Total	<u>88</u>	<u>73</u>

Backfilling storage rooms and corridors. Following the retrievability period, rooms and branch corridors are backfilled with crushed salt returned from the surface stockpile or transferred from areas being excavated at that time. For the reference case, backfilling equipment is assumed to be operated 16 hr/day, 5 days/wk. The total operating staff for backfilling the mine is given in Table D-14.

Table D-14
TOTAL OPERATING STAFF FOR
BACKFILLING THE MINE

Personnel	Number Required
Shift workers (12 x 2 shifts)	24
Day workers	0
Supervisors (2 on shift, 1 on days)	3
Total	<u>27</u>

Managing the surface salt stockpile. Labor is also expended to build, control, and monitor the surface salt stockpile. The spreader/conveyor needs to be relocated periodically, and general maintenance is required in connection with erosion, runoff control, and other environmental sampling. The total operating staff for managing the surface salt stockpile is given in Table D-15.

Table D-15
TOTAL OPERATING STAFF FOR
MANAGING THE SURFACE SALT STOCKPILE

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers	0	0
Day shift	6	2
Supervisors	1	1
Total	<u>7</u>	<u>3</u>

D.3.3.3 Total Staffing Requirements. The total staffing requirements for a Reference Case repository, combining both above and below ground operations, are shown in Table D-16.

Table D-16
TOTAL STAFFING REQUIREMENTS
REFERENCE CASE REPOSITORY

Personnel	Number Required	
	Retrieval. Period	Postretrieval. Period
Operating personnel		
Handling LLW above ground	3	3
Handling LLW below ground		
Handling HLW above ground	73	73
Handling HLW below ground	41	41
Mining rooms, etc.	88	73
Backfilling rooms, etc.	0	27
Managing the surface salt stockpile	7	3
Subtotal	212	220
Support and overhead personnel		
Maintenance (50% of operations)	106	110
Analytical	8	8
Health and safety (10% of operations)	21	22
Utility	30	30
Plant overhead (30% of operations)	64	66
Subtotal	229	236
Total	441	456

D.3.4 Repository Cost Estimates

D.3.4.1 Repository Capital Cost Estimate. The capital cost estimate for the Reference Case geologic repository is developed in accordance with Subsection D.1.2.1 and the Reference Case scope in Subsection D.3.1. The estimate is detailed in Table D-17. The total estimated capital cost is \$940 million.

Table D-17
CAPITAL COST ESTIMATE
REFERENCE CASE REPOSITORY

Item	Cost \$ million ^(a)
Field costs	
Surface facilities	
Buildings with equipment and bulk materials	230
Other structures	16
Outside utilities	43
Ponds	5
Salt storage	17
Site development with offsite improvements	53
Subtotal	<u>364</u>
Shafts and hoists	
Headframes with towers and equipment	30
Shafts and linings	48
Subtotal	<u>78</u>
Underground facilities	
Structures, equipment, and bulk materials	18
Initial underground development	
Mine and mining equipment	67
Mining — core and main drifts	66
Initial mine development	7
Subtotal	<u>140</u>
Total field cost (including 20% contingency)	600
Architect-engineer services @ 20% of \$600	120
Owner's cost @ 30% of \$720	220
Total facility capital cost	<u>940</u>

(a) Cost in January 1979 dollars.

D.3.4.2 Repository Operating Cost Estimates. The basis for the Reference Case operating cost estimates is the SRENCO NWTS Repository No. 1 estimate, which comes to \$50.6 million per year in 1978 dollars.

To obtain the estimated postretrievability period operating cost, one adds \$5 million and \$2.4 million to \$50.6 million. The \$5 million represents escalation at 10 percent per annum to January 1979 dollars; the \$2.4 million represents an adjustment to account for the fact that the SFDS repository is larger (14,800 waste canisters per year versus SRENCO's 13,800 waste canisters per year). Thus, the total operating cost per year for the Reference Case postretrievability period is estimated to be \$58 million in January 1979 dollars.

To obtain the estimated retrievability operating cost, one adds \$28 million and subtracts a little less than \$3 million from \$58 million. The \$28 million represents added sleeving and plugging costs, which are tabulated in Table D-18; the \$3 million represents a reduction in operating staff. Thus, the total operating cost per year for the Reference Case retrievability period is estimated to be about \$84 million in January 1979 dollars.

Table D-18
SLEEVE AND PLUG COST ESTIMATE
REFERENCE CASE REPOSITORY

Item	Dimension	Quantity	Unit Cost	Total Cost \$ million
Sleeves ^(a)				
Top Section	24"Ø x5'	2,380	2,135	5.1
	30"Ø x5'	1,930	2,895	5.6
Lower section	16"Ø 15'	2,380	2,300	5.5
	22"Ø x23.5'	1,930	5,700	11.0
Plugs	Full size	4,310	200	0.9
	Top section			
Total annual cost				28.0

(a) For cost estimating purposes only, these sleeves are assumed to be fabricated in two sections.

D.3.4.3 Repository Decommissioning Cost Estimate. The decommissioning cost factor used is 33 percent of the total capital cost, less the owner's cost. Thus, for the Reference Case, the decommissioning cost estimate is \$230 million.

D.4 CASE 1

D.4.1 Specific Repository Aspects

In Case 1, the repository receives, handles, and processes packages containing the various types of radioactive material identified in Table C-17. The site, surface facilities, shafts, equipment, underground development, and operations at the repository are generally as described in Subsection D.2. The only significant departure from the generic repository is in the receiving building and the underground development. Specific requirements for these two areas for Case 1 are described below.

D.4.1.1 Receiving Building. The receiving building has three hot cells connected to one waste shaft. In these facilities, 12,750 canisters (10,200 canisters of HLW and 2,570 of ILW) are processed and handled each year. For the SFDS, the LLW building, shown in the Stearns-Roger Engineering Company reference design,⁽¹⁾ was deleted. The LLW is handled through the receiving building, as discussed in Subsection D.2.3.

Surge capacity is provided, to account for 10 days of shipping cask receipts or 349 canisters, stored in an area of approximately 1,700 ft².

D.4.1.2 Underground Development. From the 2,000-acre total mine repository area, approximately 1,700 acres are available for storage rooms, including the adjacent pillars and branch corridors. The unit cell for each storage room, including appropriate portions of adjacent corridors and pillars, is 1,030 feet long and 150 feet wide. The 130-foot pillar width is based upon the heat dissipation requirements of spent fuel. The total area for each unit cell is 3.5 acres.

In Case 1, 28 unit cells, or about 99 acres, are filled per year. Therefore, the repository has a 17-year useful life.

Assuming the storage room closes in by 10 percent during the retrievability period (5 years), a 25-foot room height will leave necessary head room for the emplacement and retrieval of canisters into a 20-foot, 6-inch-long transfer cask.

D.4.2 Spent Fuel/Waste Emplacement

In Case 1, about 28 storage rooms per year are necessary for spent fuel and waste canister emplacement, as shown in Figure D-12. Table D-19 summarizes the way the storage space is utilized.

Table D-19
ANNUAL UTILIZATION OF REPOSITORY ROOMS

Approximate Number of Rooms	Waste Materials Emplaced		
	One Outside Row	Center Row	Other Outside Row
7	Spent fuel	ILW	Spent fuel
21	Spent fuel	None provided	Spent fuel

D.4.2.1 Emplacement-Retrievability Period. During the retrievability period (first 5 years), the storage wells for spent fuel are equipped with steel sleeves to facilitate retrieval operations. Each sleeve has a flange at the top end which supports a concrete shielding plug with a recessed lifting fixture.

The diameter of each storage well for high-heat-producing spent fuel is 12 inches larger than the sleeve O.D. and provides an annular space around the sleeve. This air gap should prevent damage to the sleeve from the creep effect of salt due to heat generated during the retrievability period. Low-heat-producing wastes require less air gap.

ILW canisters are stacked two-high (piggyback) in the storage wells, without steel sleeves or concrete plugs. Since this type of waste is not retrievable, the well will be filled with crushed salt after the canister is in place. The wells are also large enough to accept overpacked canisters.

The sleeves identified for spent fuel in Figure D-12 are designed to accommodate the basic single-pack canister. Since the SFDS assumes that less than 2 percent of the canisters will require overpacking, no further sleeve design perturbations are made. Sleeve allowance for overpacked canisters can be readily handled in a repository (detailed) design effort.

The LLW packed in 55-gallon drums is stored six drums to a pallet in separate rooms. When the retrievability period ends, the drums are relocated in the spent fuel/ILW-filled rooms, as described below in Subsection D.4.2.2.

D.4.2.2 Emplacement-Postretrievability Period. Once the retrievability period is over, the wells can be of smaller diameter for spent fuel canisters, since sleeves and concrete plugs are no longer required. After canister emplacement, the wells are filled with crushed salt. The depth of the wells will be increased slightly to provide salt shielding above the canister equivalent to that provided by concrete plugs.

Prior to the start of the backfill operation, some of the LLW stored in separate rooms is moved to the spent fuel/ILW-filled rooms and placed on the floor in a single layer.

D.4.3 Operational Considerations

The operational considerations presented here are based on the material developed in Subsections D.2.3, D.2.4, and D.2.7. Additional information and assumptions are included, as needed, for arriving at operating crew requirements.

The principal operations at the repository for Case 1 are:

- Receiving, handling, processing, and transporting radioactive waste packages (spent fuel, ILW, and LLW) above ground
- Handling, transporting, and emplacing radioactive waste packages below ground
- Mining repository corridors and storage rooms
- Backfilling repository corridors and storage rooms
- Managing the surface salt stockpile

D.4.3.1 Aboveground Operations. The aboveground operations taking place in the receiving building include the steps described in the flow diagram for canistered spent fuel (Figure D-3) and in the flow diagram for LLW (Figure D-4).

Handling spent fuel and ILW. An average of five casks per day are received at the facility by rail and truck (see Table E-9), transporting spent fuel and ILW canisters. As shown in Table E-9, 10,200 canisters of spent fuel and 2,570 canisters of ILW are handled per year in the remote operation portion of the receiving building. It is assumed that the facility operates 300 days/yr in three shifts of 6 hours' effective work time, and that three hot cells are used for processing canisters.

The operating crew requirements for handling one shipment of spent fuel or ILW above ground are the same as in the Reference Case and are given in Table D-5. The total operating staff for handling all shipments of spent fuel and ILW above ground is given in Table D-20.

Table D-20
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF SPENT FUEL
AND ILW ABOVE GROUND

Personnel	Number Required
Shift workers (14 x 4 shifts)	56
Day workers	0
Supervisors (4 on shift, 1 on days)	5
Total	<u>61</u>

Handling LLW. The LLW is received by truck, with one shipment arriving approximately every 4 weeks (13 shipments annually on a 365-day schedule). As shown in Table E-9, there are 550 LLW (TRU) drums per year sent to the underground storage rooms; a small fraction of these drums originates on site.

The operating crew requirements for handling one shipment of LLW above ground are the same as in the Reference Case and are given in Table D-7. The total operating staff for handling all shipments of LLW above ground is given in Table D-21.

Table D-21
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF
LLW ABOVE GROUND

Personnel	Number Required
Shift workers	0
Day shift	2
Supervisors	0
Total	<u>2</u>

D.4.3.2 Belowground Operations. The operations taking place below ground are divided into five categories: handling spent fuel and ILW, handling LLW, mining storage rooms and corridors, backfilling storage rooms and corridors, and managing the surface salt stockpile.

Handling spent fuel and ILW. This operation covers the activities required for moving the transfer cask from the bottom of the waste shaft to the storage rooms and for emplacing the cask in the storage wells. As shown in Table E-9, a total of 12,770 spent fuel and ILW canisters per year are emplaced in the repository. Thus, over the anticipated operating schedule of 300 days/yr and three shifts per day (effective work time, 6 hours per shift), 43 transfer casks per day are to be handled. It is assumed that the 25 minutes of cask turnaround time required by this schedule is ample time for the hoist descent and ascent in the shaft.

The operating crew requirements for handling one shipment of spent fuel and ILW below ground are the same as in the Reference Case and are given in Table D-9. The total operating staff for handling all shipments of spent fuel and ILW below ground is given in Table D-22.

Table D-22
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF SPENT FUEL
AND ILW BELOW GROUND

Personnel	Number Required
Shift workers (7 x 4 shifts)	28
Day shift	0
Supervisors (4 on shift, 1 on days)	5
Total	<hr/> 33

Handling LLW. Handling operations are described in Subsection D.3.3.2 (Reference Case). Operating personnel requirements are the same as those for aboveground handling; in other words, one crew handles both aboveground and belowground operations.

Mining storage rooms and corridors. Since the basic principles of mining storage rooms have already been discussed in Subsection D.3.3.2 (Reference Case), only those details specific to Case 1 are discussed here. The total salt mined is about 1.2×10^6 tons/yr, from 28 rooms, about 11,500 storage wells, and the appropriate number of branch corridors.

The operating equipment and crew requirements for developing the mine are given in Table D-23. Note that this also includes data for backfilling the mine and managing the surface salt stockpile, as well as for mining rooms and corridors. The total operating staff for mining storage rooms and corridors is given in Table D-24.

Table D-23
OPERATING EQUIPMENT AND CREW
REQUIREMENTS FOR DEVELOPING THE MINE

Item	Retrievability Period	Postretrievability Period
Mining rigs	2	2
Duty	24 hr x 7 days	24 hr x 7 days
	16 hr x 5 days	16 hr x 5 days
People (14/crew)	91 (7 supervisors)	91 (7 supervisors)
Drilling rigs	3	3
Duty	24 hr x 7 days	24 hr x 7 days
People (4/rig)	53 (5 supervisors)	53 (5 supervisors)
Skip hoist rigs	1	1
Duty	24 hr x 7 days	2 days/wk
People (4/rig)	21 (5 supervisors)	5 (1 supervisor)
Sleeving rigs	1	
Duty	24 hr x 7 days	
People (4/rig + warehouse to mine)	27 (5 supervisors)	
Backfill rigs	—	2
Duty	—	24 hr x 7 days
	—	16 hr x 5 days
People	—	79 (7 supervisors)
Stockpile, people	13 (1 supervisor)	4 (1 supervisor)

Table D-24
TOTAL OPERATING STAFF FOR
MINING STORAGE ROOMS AND CORRIDORS

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers (variety of shift schedules)	170	132
Day workers	0	4
Supervisors (4 on days, rest on shift)	22	13
Total	<u>192</u>	<u>149</u>

Backfilling storage rooms and corridors. Two backfilling rigs are required for salt handling and distribution in the postretrievability period. One of the rigs has to be in continuous service for 300 days/yr; the second rig has to be in service 16 hr/day, 5 days/wk. The total operating staff for backfilling the mine is given in Table D-25.

Table D-25
TOTAL OPERATING STAFF FOR
BACKFILLING THE MINE

Personnel	Number Required
Shift workers	72
Day workers	0
Supervisors	<u>7</u>
Total	79

Managing the surface salt stockpile. Labor is also expended to distribute, groom, and monitor the surface salt stockpile. The total operating staff for managing the surface salt stockpile is given in Table D-26.

Table D-26
TOTAL OPERATING STAFF FOR MANAGING
THE SURFACE SALT STOCKPILE

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers	0	0
Day workers	12	3
Supervisors	<u>1</u>	<u>1</u>
Total	13	4

D.4.3.3 Total Staffing Requirements. The total staffing requirements for a Case 1 repository, combining both above and below ground operations, are shown in Table D-27.

Table D-27
TOTAL STAFFING REQUIREMENTS
CASE 1 REPOSITORY

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Operating personnel		
Handling LLW above ground	2	2
Handling LLW below ground		
Handling HLW above ground	61	61
Handling HLW below ground	33	33
Mining rooms, etc.	192	149
Backfilling rooms, etc.	0	79
Managing the surface salt stockpile	13	4
Subtotal	<u>301</u>	<u>328</u>
Support and overhead personnel		
Maintenance (50% of operations)	150	164
Analytical	8	8
Health and safety (10% of operations)	30	33
Utility	30	30
Plant overhead (30% of operations)	90	99
Subtotal	<u>308</u>	<u>334</u>
Total	<u>609</u>	<u>662</u>

D.4.4 Repository Cost Estimates

D.4.4.1 Repository Capital Cost Estimate. The capital cost estimate for the Case 1 repository is developed in accordance with Subsection D.1.2.1 and the Case 1 scope in Subsection D.4.1. The estimate is detailed in Table D-28. The total estimated capital cost is \$950 million.

Table D-28
CAPITAL COST ESTIMATE
CASE 1 REPOSITORY

Item	Cost \$ million ^(a)
Field costs	
Surface facilities	
Buildings with equipment and bulk materials	230
Other structures	16
Outside utilities	43
Ponds	5
Salt storage	17
Site development with offsite improvements	53
Subtotal	<u>364</u>
Shafts and hoists	
Headframes with towers and equipment	30
Shafts and linings	48
Subtotal	<u>78</u>
Underground facilities	
Structures, equipment and bulk materials	18
Initial underground development	
Mine and mining equipment	67
Mining — core and main drifts	66
Initial mine development	7
Subtotal	<u>140</u>
Total field cost (including 20% contingency)	<u>600</u>
Architect-engineer services @ 20% of \$600	120
Owner's cost @ 30% of \$720	220
Total facility capital cost	<u>950</u>

(a) Cost in January 1979 dollars.

D.4.4.2 Repository Operating Cost Estimates. The basis for the Case 1 operating cost estimates is the estimate for the Reference Case (see Subsection D.3.4.2), which comes to \$58 million per year in January 1979 dollars.

To obtain the estimated postretrievability period operating cost, one adds \$26 million to the \$58 million. The \$26 million represents the increase in operating cost due to the increase in the number of personnel required in Case 1. A larger number of mining personnel are needed because the mining requirements for Case 1 are greater than the mining requirements of the Reference Case. Thus, the total operating cost per year for the Case 1 postretrievability period is estimated to be \$84 million in January 1979 dollars.

To obtain the estimated retrievability period operating cost, one adds \$60 million and subtracts \$7 million from the \$84 million. The \$60 million represents added sleeving and plugging cost, which are tabulated in Table D-29; the \$7 million represents a reduction in operating staff. Thus, the total operating cost per year for the Case 1 retrievability period is estimated to be about \$137 million in January 1979 dollars.

Table D-29
SLEEVE AND PLUG COST ESTIMATE
CASE 1 REPOSITORY

Item	Dimension	Quantity	Unit Cost	Total Cost \$ million
Sleeves ^(a)				
Top section	20"Ø x5'	10,200	2,500	25.5
Lower section	18"Ø x20.5'	10,200	3,150	32.1
Plugs	Full size	10,200	200	2.4
	Top section			
Total annual cost				60.0

(a) For cost estimating purposes only, the sleeves are assumed to be fabricated in two sections.

D.4.4.3 Repository Decommissioning Cost Estimate. The decommissioning cost factor used is 33 percent of the total capital cost, less the owner's cost. Thus, for Case 1, the decommissioning cost estimate is \$235 million.

D.5 CASE 2

D.5.1 Specific Repository Aspects

In Case 2, the repository receives, handles, and processes packages containing the various types of radioactive material identified in Table C-31. The site, surface facilities, shafts, equipment, underground development and operations are generally as described in Subsection D.2. The only significant departure from the generic repository is in the receiving building and the underground development. Specific requirements for these two areas for Case 2 are described below.

D.5.1.1 Receiving Building. The receiving building has three hot cells connected to one waste shaft. In these facilities, 13,460 canisters (9,610 canisters of HLW and 3,850 of ILW) are processed and handled each year. For the SFDS, the LLW building, shown in the Stearns-Roger Engineering Co.⁽¹⁾ reference design, was deleted. The LLW is handled through receiving building, as discussed in Subsection D.2.3.

Surge capacity is provided to account for 10 days of shipping cask receipts. This represents the in-transit time for the case of a repository shutdown and is based on an average distance between the P/E facility and the repository of 1,100 miles and on a mean train speed of 5 mph. If these figures are used, the required surge capacity for Case 2 is 370 canisters, stored in an area of approximately 1,800 ft².

D.5.1.2 Underground Development. Of the 2,000-acre total mined repository area, approximately 1,700 acres are available for storage rooms, including the adjacent pillars and branch corridors. The unit cell for each storage room, including appropriate portions of corridors and pillars, is 1,030 feet long and 199 feet wide. The 179-foot pillar width is based upon the heat dissipation requirement of chopped fuel. The total area for each unit cell is 4.7 acres.

In case 2, 19 unit cells, or about 90 acres, are filled per year. Therefore, the repository has a 19-year (1,700/90) useful life.

Assuming the storage room closes in by 10 percent during the retrievability period, a 19½-foot room height will leave the necessary head room for emplacement and retrieval of canisters into a 17½-foot-long transfer cask.

D.5.2 Chopped Fuel/Waste Emplacement

In Case 2, about 19 storage rooms per year are filled with canistered waste, as shown in Figure D-13. Table D-30 summarizes the way the storage space, shown in Figure D-13, is utilized.

Table D-30
ANNUAL UTILIZATION OF REPOSITORY ROOMS

Approximate Number of Rooms	Waste Materials Emplaced		
	One Outside Row	Center Row	Other Outside Row
4	Chopped fuel	Low heat HLW	Chopped fuel
9	Chopped fuel	ILW	Chopped fuel
6	Chopped fuel	None provided	Chopped fuel

D.5.2.1 Emplacement-Retrievability Period. During the retrievability period (first 5 years), the storage wells for chopped fuel and HLW are equipped with steel sleeves to facilitate the retrieval operation. Each sleeve has a flange at the top end which supports a concrete shielding plug with a recessed lifting fixture.

The diameter of each storage well for high-heat-producing chopped fuel is 12 inches larger than the steel sleeve O.D. and provides a 6-inch annular space around the sleeve. The air gap should prevent damage to the sleeve from the creep effect of salt due to heat generated during the retrievability period. Low-heat-producing wastes require less air gap.

ILW canisters are stacked two-high (piggyback) in the storage wells without steel sleeves or concrete plugs. Since this type of waste is not retrievable, the well will be filled with crushed salt after the canister is in place. The wells are also large enough to accept overpacked canisters, if necessary.

The sleeves identified for chopped fuel in Figure D-13 are designed to accommodate the basic single-pack canister. Since the SFDS assumes that less than 2 percent of the canisters will require overpacking, no further sleeve design perturbations are made. Sleeve allowance for overpacked canisters can be readily handled in a repository (detailed) design effort.

The LLW, packed in 55-gallon drums, is stored six drums to a pallet in separate rooms. When the retrievability period ends, the drums are relocated in the chopped fuel/ILW-filled rooms, as described below in Subsection D.5.2.2.

D.5.2.2 Emplacement-Postretrievability Period. Once the retrievability period is over, storage wells can be of smaller diameter, since steel sleeves and concrete plugs are no longer required. After canister emplacement, the storage wells are filled with crushed salt. The depth of the wells will be increased slightly to provide salt shielding above the canister equivalent to that provided by the concrete plugs.

Prior to the start of the backfill operation, some of the LLW stored in separate rooms is moved to the chopped fuel/ILW-filled rooms and placed on the floor in a single layer.

D.5.3 Operational Considerations

The operational considerations presented here are based on the material developed in Subsections D.2.3, D.2.4, and D.2.7. Additional information and assumptions are included, as needed, for arriving at operating crew requirements.

The principal operations at the repository for Case 2 are:

- Receiving, handling, processing, and transporting radioactive waste packages (chopped fuel, ILW, and LLW) above ground
- Handling, transporting, and emplacing radioactive waste packages below ground
- Mining repository corridors and storage rooms
- Backfilling repository corridors and storage rooms
- Managing the surface salt stockpile

D.5.3.1 Aboveground Operations. The aboveground operations taking place in the receiving building include the steps described in the flow diagram for chopped fuel (Figure D-3) and in the flow diagram (Figure D-4) for LLW.

ROW	CANISTER CONTENT	NO. OF HOLES IN ROOM	NO. OF ROOMS FILLED PER YEAR	CANISTER ^{3/8"} WALL THK		HOLE		SLEEVE ^{3/8"} WALL THK		PLUG	COMPARATIVE LEVEL OF HEAT	REMARKS
				(A) OD"	(B) LENGTH'	(D) DIA"	(H) LENGTH'	(C) OD"	(L) LENGTH'			
X	CHOPPED FUEL	211	19.1	12 ³ / ₄	10 ¹ / ₂	28	16 ¹ / ₂	16	5 ¹ / ₂	HIGH	CANISTER LENGTH INCLUDES LIFTING FIXTURE (TYP.)	
Y	HULLS, END FITTINGS AND GASES (HLW)	210	3.7	20	10 ¹ / ₂	26	25	22	3 ¹ / ₂	LOW		
	ILW		9.3			24		NONE	NONE			
Z	CHOPPED FUEL	211	19.1	12 ³ / ₄	10 ¹ / ₂	28	16 ¹ / ₂	16	5 ¹ / ₂	HIGH		

Figure D-13.
Geologic Repository
Chopped Fuel/Waste Emplacement
Schematic for the Retrievability Period
Case 2

Handling chopped fuel and ILW. An average of four casks per day are received at the facility by rail and truck (see Table E-9) transporting chopped fuel, hulls, and ILW canisters. As shown in Table E-9, 9,605 canisters of chopped fuel and 3,850 canisters of ILW are handled per year in the remote operation portion of the receiving building. It is assumed that the facility operates on a 300 day/yr schedule in three shifts of 6 hours' effective work time, and that three hot cells are used for processing canisters.

The operating crew requirements for handling one shipment of chopped fuel and ILW above ground are the same as in the Reference Case and are given in Table D-5. The total operating staff for handling all shipments of chopped fuel and ILW above ground is given in Table D-31.

Table D-31
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF CHOPPED
FUEL AND ILW ABOVE GROUND

Personnel	Number Required
Shift workers (15 x 4 shifts)	60
Workers	0
Supervisors (4 on shift, 1 on days)	5
Total	<u>65</u>

Handling LLW. The LLW is received by truck, with one shipment arriving approximately every 3 weeks (20 shipments annually on a 365-day schedule). As shown in Table E-9, there are 850 LLW (TRU) drums per year sent to the underground storage rooms; a small fraction of these drums originates on site.

The operating crew requirements for handling one shipment of LLW above ground are the same as in the Reference Case given in Table D-7. The total operating staff for handling all shipments of LLW above ground is given in Table D-32.

Table D-32
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF
LLW ABOVE GROUND

Personnel	Number Required
Shift workers	0
Day shift	2
Supervisors	0
Total	<u>2</u>

D.5.3.2 Belowground Operations. The operations taking place below ground are divided into five categories: handling chopped fuel, hulls, and ILW; handling LLW; mining storage rooms and corridors; backfilling storage rooms and corridors; and managing the surface salt stockpile.

Handling chopped fuel and ILW. This operation covers the activities required for moving the transfer cask from the bottom of the waste shaft to the storage rooms and for emplacing the casks in the storage wells.

As shown in Table E-9, a total of 13,455 chopped fuel/hulls, etc., and ILW canisters per year are emplaced in the repository. Thus, over the anticipated operating schedule of 300 days/yr and three shifts per day (effective work time, 6 hours per shift), 45 transfer casks per day are to be handled. It is assumed that the 24 minutes of turnaround time required by this schedule is ample time for the hoist descent and ascent in the shaft.

The operating crew requirements for handling one shipment of chopped fuel and ILW below ground are the same as in the Reference Case and are given in Table D-9. The total operating staff for handling all shipments of chopped fuel and ILW below ground is given in Table D-33.

Table D-33
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF CHOPPED
FUEL AND ILW BELOW GROUND

Personnel	Number Required
Shift workers (8 x 4 shifts)	32
Day shift	0
Supervisors (1 x 4 shifts, 1 day)	5
Total	<hr/> 37

Handling LLW. Handling operations are described in Subsection D.3.3.2 (Reference Case). Operating personnel requirements are the same as those for aboveground operations; in other words, one crew handles both aboveground and belowground operations.

Mining storage rooms and corridors. Since the basic principles of mining storage rooms have already been discussed in Subsection D.3.3.2 (Reference Case), only those details specific to Case 2 are discussed here. The total salt mined is about 0.8×10^6 tons/yr, from 19 rooms, about 10,800 storage wells, and the appropriate number of branch corridors.

The operating equipment and crew requirements for developing the mine are given in Table D-34. Note that this also includes data for backfilling the mine and managing the surface salt stockpile, as well as for mining rooms and corridors. The total operating staff for mining storage rooms and corridors is given in Table D-35.

Table D-34
OPERATING EQUIPMENT AND CREW
REQUIREMENTS FOR DEVELOPING THE MINE

Item	Retrievability Period	Postretrievability Period
Mining rigs	1	1
Duty	24 hr x 5 days 16 hr x 5 days	24 hr x 5 days
People (14/crew)	46 (4 supervisors)	46 (4 supervisors)
Drilling rigs	3	3
Duty	24 hr x 7 days	24 hr x 7 days
People (4/rig)	53 (5 supervisors)	53 (5 supervisors)
Skip hoist rigs	1	1
Duty	24 hr x 5 days	1.5 days/wk
People (4/rig)	22 (5 supervisors)	5 (1 supervisor)
Sleeving rigs	1	—
Duty	24 hr x 5 days	—
People (4/rig + warehouse to mine)	22 (5 supervisors)	—
Backfill rigs	—	1
Duty	—	24 hr x 5 days
People	—	44 (4 supervisors)
Stockpile, people	9 (1 supervisor)	3 (1 supervisor)

Table D-35
TOTAL OPERATING STAFF FOR
MINING STORAGE ROOMS AND CORRIDORS

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers (variety of shift schedules)	124	90
Day workers	0	4
Supervisors (4 on days, rest on shift)	19	10
Total	<u>143</u>	<u>104</u>

Backfilling storage rooms and corridors. One backfilling rig is required for salt handling and distribution in the postretrievability period. The rig has to be in service for 300 days/yr, 24 hr/day, and 5 days/week. The total operating staff for backfilling the mine is given in Table D-36.

Table D-36
TOTAL OPERATING STAFF FOR
BACKFILLING THE MINE

Personnel	Number Required
Shift workers (10 x 4 shifts)	40
Day workers	0
Supervisors	<u>4</u>
Total	<u>44</u>

Managing the surface salt stockpile. Labor is also expended to distribute, groom, and monitor the surface salt stockpile. The total operating staff for managing the surface salt stockpile is given in Table D-37.

Table D-37
TOTAL OPERATING STAFF FOR
MANAGING THE SURFACE SALT STOCKPILE

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers	0	0
Day workers	8	2
Supervisors	<u>1</u>	<u>1</u>
Total	<u>9</u>	<u>3</u>

D.5.3.3 Total Staffing Requirements. The total staffing requirements for a Case 2 repository, combining both above and below ground operations, are shown in Table D-38.

Table D-38
TOTAL STAFFING REQUIREMENTS
CASE 2 REPOSITORY

Personnel	Number Required	
	Retrieval Period	Postretrieval Period
Operating personnel		
Handling LLW above ground	2	2
Handling LLW below ground		
Handling HLW above ground	65	65
Handling HLW below ground	37	37
Mining rooms, etc.	132	104
Backfilling rooms, etc.	0	44
Managing the surface salt stockpile	9	3
Subtotal	<u>245</u>	<u>255</u>
Support and overhead personnel		
Maintenance (50% of operations)	122	128
Analytical	8	8
Health and safety (10% of operations)	25	26
Utility	30	30
Plant overhead (30% of operations)	74	77
Subtotal	<u>259</u>	<u>269</u>
Total	<u>504</u>	<u>524</u>

D.5.4 Repository Cost Estimates

D.5.4.1 Repository Capital Cost Estimate. The capital cost estimate for the Case 2 repository is developed in accordance with Subsection D.1.2.1 and the Case 2 scope in Subsection D.5.1. The estimate is detailed in Table D-39. The total estimated capital cost is \$940 million.

Table D-39
CAPITAL COST ESTIMATE
CASE 2 REPOSITORY

Item	Cost \$ million (a)
Field costs	
Surface facilities	
Buildings with equipment and bulk materials	230
Other structures	16
Outside utilities	43
Ponds	5
Salt storage	17
Site development with offsite improvements	53
Subtotal	<u>364</u>
Shafts and hoists	
Headframes with towers and equipment	30
Shafts and linings	48
Subtotal	<u>78</u>
Underground facilities	
Structures, equipment, and bulk materials	18
Initial underground development	
Mine and mining equipment	67
Mining — core and main drifts	66
Initial mine development	7
Subtotal	<u>140</u>
Total field costs (including 20% contingency)	600
Architect-engineer services @ 20% of \$600	120
Owner's cost @ 30% of \$720	220
Total facility capital cost	<u>940</u>

(a) Cost in January 1979 dollars.

D.5.4.2 Repository Operating Cost Estimates. The basis for the Case 2 operating cost estimates is the estimate for the Reference Case (see Subsection D.3.4.2), which comes to \$58 million per year in January 1979 dollars.

To obtain the estimated postretrievability period operating cost, one adds \$9 million to the \$58 million. The \$9 million represents two factors: the increase in the total number of rooms to be excavated and the decrease in the number of canisters to be handled. Thus, the total operating cost per year for the Case 2 postretrievability period is estimated to be \$67 million in January 1979 dollars.

To obtain the estimated retrievability period operating cost, one adds \$45 million and subtracts \$4 million from the \$67 million. The \$45 million represents added sleeving and plugging costs, which are tabulated in Table D-40; the \$4 million represents a reduction in operating staff. Thus, the total operating cost per year for the Case 2 retrievability period is estimated to be \$108 million.

D.5.4.3 Repository Decommissioning Cost Estimate. The decommissioning cost factor used is 33 percent of the total capital cost, less the owner's cost. Thus, for Case 2, the decommissioning cost estimate is \$230 million.

Table D-40
SLEEVE AND PLUG COST ESTIMATE
CASE 2 REPOSITORY

Item	Dimension	Quantity	Unit Cost	Total Cost \$ million
Sleeves ^(a)				
Top section	24"Ø x 5'	8,070	2,135	17.2
	30"Ø x 5'	770	2,895	2.2
Lower section	16"Ø x 15'	8,070	2,300	18.6
	22"Ø x 23.5'	770	5,700	4.4
Plugs	Full size	8,840	200	1.6
	Top section			
Total annual cost				44.0

(a) For cost estimating purposes only, the sleeves are assumed to be fabricated in two sections.

D.6 CASE 3

D.6.1 Specific Repository Aspects

In Case 3, the repository receives, handles, and processes packages containing the various types of radioactive material identified in Table C-46. The site, surface facilities, shafts, equipment, underground development, and operations are generally as described in Subsection D.2. The only significant departure from the generic repository is in the receiving building and the underground development. Specific requirements for these two areas for Case 3 are described below.

D.6.1.1 Receiving Building. The receiving building has six hot cells connected to two waste shafts. In these facilities, 28,390 canisters (22,380 canisters of HLW and 6,010 of ILW) are processed and handled each year. For the SFDS generic design, the LLW building, shown in the Stearns-Roger Engineering Co.⁽¹⁾ reference design, was deleted. However, a second receiving building is required at this location, to account for three additional hot cells. Also, the deleted LLW shaft must be replaced by an additional waste shaft in this case. The LLW is handled through the receiving building, as discussed in Subsection D.2.3.

Surge capacity is provided to account for 10 days of rail cask receipts. This represents the in-transit time for the case of a repository shutdown and is based on an average distance between the P/E facility and the repository of 1,100 miles and on a mean train speed of 5 mph. If these figures are used, the required surge capacity for Case 3 is 778 canisters stored in an area of approximately 3,800 ft².

D.6.1.2 Underground Development. Of the 2,000-acre total mined repository area, approximately 1,700 acres are available for storage rooms, including the adjacent pillars and branch corridors. The unit cell for each storage room, including appropriate portions of corridors and pillars, is 1,030 feet long and 69 feet wide. The 49-foot pillar width is based upon the heat dissipation requirement of vitrified fuel. The total area for each unit cell is 1.6 acres.

In Case 3, 60 unit cells, or about 97.5 acres, are filled per year. Therefore, the repository has a 17.5-year (1,700/97.5) useful life.

Assuming the storage room closes in by 10 percent during the retrievability period (5 years), a 19-1/2-foot room height will leave the necessary head room for emplacement and retrieval of canisters into a 17-1/2-foot-long transfer cask.

D.6.2 Vitrified Fuel/Waste Emplacement

In Case 3, about 60 storage rooms per year are necessary for vitrified fuel and waste emplacement, as shown in Figure D-14. Table D-41 summarizes the way the storage space, shown in Figure D-14, is utilized.

Table D-41
ANNUAL UTILIZATION OF REPOSITORY ROOMS

Approximate Number of Rooms	Waste Materials Emplaced		
	One Outside Row	Center Row	Other Outside Row
12	Vitrified fuel	Nonheat HLW	Vitrified fuel
20	Vitrified fuel	ILW	Vitrified fuel
28	Vitrified fuel	None provided	Vitrified fuel

VITRIFIED FUEL

HULLS, END FITTINGS
AND GASES (HLW)

ILW

ROW	CANISTER CONTENT	NO. OF HOLES IN ROOM	NO. OF ROOMS FILLED PER YEAR	CANISTER $\frac{3}{8}$ " WALL THK		HOLE		SLEEVE $\frac{3}{8}$ " WALL THK		PLUG		COMPARATIVE LEVEL OF HEAT	REMARKS
				(A) OD"	(B) LENGTH'	(D) DIA"	(H) LENGTH'	(C) OD"	(L) LENGTH	(L) LENGTH	(L) LENGTH		
X	VITRIFIED FUEL	155	60	20	10 $\frac{1}{2}$	34	16 $\frac{1}{2}$	22	5'-6"			HIGH	CANISTER LENGTH INCLUDES LIFTING FIXTURE (TYP.)
Y	HULLS, END FITTINGS AND GASES (HLW)	154	12.5	20	10 $\frac{1}{2}$	26	25	22	3'-6"			LOW	
	ILW		19.5			24		NONE	NONE				
Z	VITRIFIED FUEL	155	60	20	10 $\frac{1}{2}$	34	16 $\frac{1}{2}$	22	5'-6"			HIGH	

Figure D-14.
Geologic Repository
Vitrified Fuel/Waste Emplacement
Schematic for the Retrieval Period
Case 3

D.6.2.1 Emplacement-Retrievability Period. During the retrievability period (first 5 years) the storage wells for vitrified fuel and HLW are equipped with steel sleeves. Each sleeve has a flange at the top end which supports a concrete shielding plug with a recessed lifting fixture.

The diameter of each well for vitrified fuel is 12 inches larger than the steel sleeve O.D. to provide a 6-inch annular space around the sleeve. The air gap should prevent damage to the sleeve from the creep effect of salt due to heat generated during the retrievability period. Low-heat-producing wastes require less air gap.

ILW canisters are stacked two-high (piggyback) in the storage wells without steel sleeves or concrete plugs. Since this type of waste is not retrievable, the well will be filled with crushed salt after the canister is in place. The wells are also large enough to accept overpacked canisters, if necessary.

The sleeves identified for vitrified fuel in Figure D-14 are designed to accommodate the basic single-pack canister. Since the SFDS assumes that less than 2 percent of the canisters will require overpacking, no further sleeve design perturbations are made. Sleeve allowance for overpacked canisters can be readily handled in a repository (detailed) design effort.

The LLW, packed in 55-gallon drums, is stored six drums to a pallet in separate rooms. When the retrievability period ends, the drums are relocated in the vitrified fuel/ILW-filled rooms, as described below in Subsection D.6.2.2.

D.6.2.2 Emplacement-Postretrievability Period. Once the retrievability period is over, the wells can be of smaller diameter, since steel sleeves and concrete plugs are no longer required. After canister emplacement, the storage wells are filled with crushed salt. The depth of the wells will be increased slightly to provide salt shielding above the canister equivalent to that provided by the concrete plugs.

Prior to the start of the backfill operation, some of the LLW stored in separate rooms is moved to the vitrified fuel/ILW-filled rooms and placed on the floor in a single layer.

D.6.3 Operational Considerations

The operational considerations presented here are based on the material developed in Subsections D.2.3, D.2.4, and D.2.7. Additional information and assumptions are included, as needed, for arriving at operating crew requirements.

The principal operations at the repository for Case 3 are:

- Receiving, handling, processing, and transporting radioactive waste packages (vitrified fuel, ILW, and LLW) above ground
- Handling, transporting, and emplacing radioactive waste packages below ground
- Mining repository corridors and storage rooms
- Backfilling repository corridors and storage rooms
- Manage the surface salt stockpile

D.6.3.1 Aboveground Operations. The aboveground operations taking place in the receiving building include the steps described in the flow diagram for vitrified fuel (Figure D-3) and in the flow diagram for LLW (Figure D-4).

Handling vitrified fuel and ILW. An average of 11 casks per day are received at the facility by rail and truck (see Table E-9), transporting vitrified fuel, hulls, etc. and ILW canisters. As shown in Table E-9, 22,378 canisters of vitrified fuel and 6,010 canisters of ILW are handled per year in the remote operation portion of the receiving building. It is assumed that the facility operates on a 300 day/yr schedule in three shifts of 6 hours' effective work time and that three hot cells are used for processing canisters.

The operating crew requirements for handling one shipment of vitrified fuel and ILW above ground are the same as in the Reference Case and are given in Table D-5. The total operating staff for handling all shipments of vitrified fuel and ILW above ground is given in Table D-42.

Table D-42
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS
OF VITRIFIED FUEL AND
ILW ABOVE GROUND

Personnel	Number Required
Shift workers (31 x 4 shifts)	124
Day workers	0
Supervisors (8 on shift, 1 on days)	9
Total	133

Handling LLW. The LLW is received by truck, with one shipment arriving approximately every 2 weeks (29 shipments annually on a 365-day schedule). As shown in Table E-9, there are 1,230 LLW (TRU) drums per year sent to the underground storage rooms; a small fraction of these drums originates on site.

The operating crew requirements for handling one shipment of LLW above ground are the same as in the Reference Case and are given in Table D-7. The total operating staff for handling all shipments of LLW above ground is given in Table D-43.

Table D-43
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF
LLW ABOVE GROUND

Personnel	Number Required
Shift workers	0
Day shift	2
Supervisors	0
Total	2

D.6.3.2 Belowground Operations. The operations taking place below ground are divided into five categories: handling vitrified fuel hulls and ILW, handling LLW, mining storage rooms and corridors, backfilling storage rooms and corridors, and managing the surface salt stockpile.

Handling vitrified fuel and ILW. This operation covers the activities required for moving the transfer cask from the bottom of the waste shaft to the storage rooms for the actual emplacement in the storage wells.*

As shown in Table E-9, a total of 28,388 vitrified fuel/hulls, etc. and ILW canisters per year are emplaced in the repository. Thus, over the anticipated operating schedule of 300 days/yr and three shifts per day (effective work time, 6 hours per shift), 47 transfer casks per day are to be handled. It is assumed that the 23 minutes of turnaround time required by this schedule is ample time for the hoist descent and ascent in the shaft.

The operating crew requirements for handling one shipment of vitrified fuel and ILW below ground are the same as in the Reference Case and are given in Table D-9. The total operating staff for handling all shipments of vitrified fuel and ILW below ground is given in Table D-44.

Table D-44
TOTAL OPERATING STAFF FOR
HANDLING ALL SHIPMENTS OF VITRIFIED
FUEL AND ILW BELOW GROUND

Personnel	Number Required
Shift workers (16 x 4 shifts)	64
Day shift	0
Supervisors (1 x 4 shifts, 1 day)	5
Total	<hr/> 69

Handling LLW. Handling operations are described in Subsection D.3.3.2 (Reference Case). Operating personnel requirements are the same as those for aboveground operations; in other words, one crew handles both aboveground and belowground operations.

Mining storage rooms and corridors. Since the basic principles of mining storage rooms have already been discussed in Subsection D.3.3.2 (Reference Case), only those details specific to Case 3 are discussed here. The total salt mined is about 1.9×10^6 tons/yr, from 60 rooms, about 23,500 storage wells, and the appropriate number of branch corridors.

The operating equipment and crew requirements for developing the mine are given in Table D-45. Note that this also includes data for backfilling the mine and managing the surface salt stockpile, as well as for mining rooms and corridors. The total operating staff for mining storage rooms and corridors is given in Table D-46.

Backfilling storage rooms and corridors. Three backfilling rigs are required for salt handling and distribution in the postretrievability period. Two of the rigs have to be in continuous service for 300 days/yr. The third rig has to be in service 16 hr/day, 5 days/wk. The total operating staff for backfilling the mine is given in Table D-47.

*Note that in Case 3 there are two receiving buildings and two waste shafts. All operations/operators are duplicated except for LLW, which is handled in only one facility.

Table D-45
OPERATING EQUIPMENT AND CREW
REQUIREMENTS FOR DEVELOPING THE MINE

Item	Retrievability Period	Postretrievability Period
Mining rigs	3	3
Duty	24 hr x 7 days	24 hr x 7 days
	24 hr x 7 days	24 hr x 7 days
	16 hr x 5 days	16 hr x 5 days
People	151	151
(14/crew)	(11 supervisors)	(11 supervisors)
Drilling rigs	5	5
Duty	24 hr x 7 days	24 hr x 7 days
People	89	89
(4/rig)	(9 supervisors)	(9 supervisors)
Skip hoist rigs	2	1
Duty	24 hr x 7 days	4 days/wk
	8 hr x 5 days	—
People	26	5
(4/rig)	(6 supervisors)	(1 supervisor)
Sleeving rigs	2	—
Duty	24 hr x 7 days	—
	24 hr x 5 days	—
People	45	—
(4/rig + warehouse to mine)	(8 supervisors)	—
Backfill rigs	—	3
Duty	—	24 hr x 7 days
	—	24 hr x 7 days
	—	16 hr x 5 days
People	—	131
		(11 supervisors)
Stockpile, people	13	6
	(1 supervisor)	(1 supervisor)

Table D-46
TOTAL OPERATING STAFF FOR
MINING STORAGE ROOMS AND CORRIDORS

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers (variety of shift schedules)	277	224
Day workers	0	4
Supervisors (4 on days, rest on shift)	34	21
Total	311	245

Table D-47
TOTAL OPERATING STAFF FOR
BACKFILLING THE MINE

Personnel	Number Required
Shift workers (12 x 10 shifts)	120
Day workers	0
Supervisors (1 x 10 shifts, 1 on days)	11
Total	131

Managing the surface salt stockpile. Labor is also expended to distribute, groom, and monitor the surface salt stockpile. The total operating staff for managing the surface salt stockpile is given in Table D-48.

Table D-48
TOTAL OPERATING STAFF FOR
MANAGING THE SURFACE SALT STOCKPILE

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Shift workers	0	0
Day workers	12	5
Supervisors	1	1
Total	13	6

D.6.3.3 Total Staffing Requirements. The total staffing requirements for a Case 3 repository, combining both above and below ground operations, are shown in Table D-49.

Table D-49
TOTAL STAFFING REQUIREMENTS
CASE 3 REPOSITORY

Personnel	Number Required	
	Retrievab. Period	Postretrievab. Period
Operating personnel		
Handling LLW above ground	2	2
Handling LLW below ground		
Handling HLW above ground	133	133
Handling HLW below ground	69	69
Mining operations	311	245
Backfilling operations	0	131
Managing the surface salt stockpile	13	6
Subtotal	<u>528</u>	<u>586</u>
Support and overhead personnel		
Maintenance (50% of operations)	264	295
Analytical	8	8
Health and safety (10% of operations)	53	59
Utility	30	30
Plant overhead (30% of operations)	158	177
Subtotal	<u>513</u>	<u>569</u>
Total	1,041	1,155

D.6.4 Repository Cost Estimates

D.6.4.1 Repository Capital Cost Estimate. The capital cost estimate for the Case 3 repository is developed in accordance with Subsection D.1.2.1 and the Case 3 scope in Subsection D.6.1. Additional adjustments have been made to the Case 3 estimate to add the cost of a second waste receiving building, shaft, and underground facilities, with associated utilities. The estimate is detailed in Table D-50. The total estimated capital cost is \$1,240 million.

Table D-50
CAPITAL COST ESTIMATE
CASE 3 REPOSITORY

Item	Cost \$ million ^(a)
Field costs	
Surface facilities	
Buildings with equipment and buil materials	370
Other structures	16
Outside utilities	43
Ponds	5
Salt storage	17
Site development with offsite improvements	53
Subtotal	<u>504</u>
Shafts and hoists	
Headframes with towers and equipment	44
Shafts and linings	60
Subtotal	<u>104</u>
Underground facilities	
Structures, equipment, and bulk materials	26
Initial underground development	
Mine and mining equipment	67
Mining — core and main drifts	66
Initial mine development	23
Subtotal	<u>156</u>
Total field costs (including 20% contingency)	<u>790</u>
Architect-engineer services @ 20% of \$600	160
Owner's cost @ 30% of \$720	<u>290</u>
Total facility capital cost	<u>1,240</u>

(a) Cost in January 1979 dollars.

D.6.4.2 Repository Operating Cost Estimates. The basis for the Case 3 operating cost estimates is the estimate for the Reference Case (see Subsection D.3.4.2), which comes to \$58 million per year in January 1979 dollars.

To obtain the estimated postretrievability period operating cost, one adds \$89 million to the \$58 million. The \$89 million represents the increase in the total number of personnel required for the greatly increased amount of excavation. Thus, the total operating cost per year for the Case 3 postretrievability period is estimated to be \$147 million in January 1979 dollars.

To obtain the estimated retrievability period operating cost, one adds \$151 million and subtracts \$15 million from the \$147 million. The \$151 million represents added sleeving and plugging costs, which are tabulated in Table D-51; the \$15 million represents a reduction in operating staff. Thus, the total operating cost per year for the Case 3 retrievability period is estimated to be \$283 million.

Table D-51
SLEEVE AND PLUG COST ESTIMATE
CASE 3 REPOSITORY

Item	Dimension	Quantity	Unit Cost	Total Cost \$ million
Sleeves ^(a)				
Top section	30"Ø x5'	20,450	2,895	59.2
Lower section	22"Ø x15'	18,520	4,160	77.0
	22"Ø x23.5'	1,930	5,700	11.0
Plugs	Full size	20,450	200	3.8
	Top section			
Total annual cost				151.0

(a) For cost estimating purposes only, the sleeves are assumed to be fabricated in two sections.

D.6.4.3 Repository Decommissioning Cost Estimate. The decommissioning cost factor used is 33 percent of the total capital cost, less the owner's cost. Thus, for Case 3, the decommissioning cost estimate is \$305 million.

REFERENCES FOR APPENDIX D

- (1) "NWTS Repository No. 1 Conceptual Design Report," Vol. I, Stearns-Roger Engineering Co., EY-77-C-05-5367, August 1978.
- (2) "NWTS Repository No. 1 Conceptual Design Report," Vol. III, Stearns-Roger Engineering Co., EY-77-C-05-5367, August 1978.
- (3) "NWTS Repository No. I Conceptual Design Report," Vol. II, Stearns-Roger Engineering Co., DOE Contract No. EY-77-C-05-5367, November 1978.
- (4) "Conceptual Design Report National Waste Terminal Storage Repository No. 2," Vol. I, Parts 1 and 2, Kaiser Engineers, Oakland, California, NWTS-2990, August 1978.
- (5) "Conceptual Design Report National Waste Terminal Storage Repository No. 2," Vol. II, Kaiser Engineers, Oakland, California, August 1978.
- (6) "Conceptual Design Report National Waste Terminal Storage Repository No. 2," Vol. III, Kaiser Engineers, Oakland, California, NWTS-2991, August 1978.
- (7) "Storage Room Design Parameters for the ONWI Spent Fuel Disposal Study: Nine Case Studies in Salt," Science Applications, Inc., Oak Ridge, Tennessee, ONWI/78/SUB-E513-00413-1, March 23, 1979.

Appendix E
TRANSPORTATION SYSTEMS

TABLE OF CONTENTS

	Page
E.1 Materials to be Transported	E-1
E.1.1 Reactor Annual Discharge	E-2
E.1.2 Away-From-Reactor Storage Facility (AFR) Annual Discharge	E-2
E.1.3 Processing/Encapsulation (P/E) Facility Annual Discharge	E-2
E.2 Modes of Transportation	E-3
E.2.1 Road Transportation	E-3
E.2.2 Rail Transportation	E-5
E.2.3 Water Transportation	E-9
E.2.4 Intermodal Transportation	E-12
E.3 Transportation Regulations	E-16
E.3.1 Federal Regulations	E-16
E.3.2 State and Local Regulations	E-20
E.4 Transportation Logistics	E-31
E.4.1 Distance Between Facilities	E-31
E.4.2 Average Speed	E-32
E.4.3 Shipments	E-33
E.5 Casks	E-35
E.5.1 Available Casks	E-35
E.5.2 Conceptual Casks	E-39
E.5.3 Turnaround Times and Lease Rates	E-40
E.5.4 Cask Fleet Requirements	E-44
E.6 Transportation Cost Estimates	E-45
E.6.1 Cost Summary	E-45
E.6.2 Transportation Cost Sensitivity	E-46
E.6.3 Rail Cask Fleet as a Function of Distance	E-48
References for Appendix E	E-53

Appendix E

TRANSPORTATION SYSTEMS

This appendix discusses the materials to be transported, the modes of transportation, the transport regulations, the transport logistics, the types of packages, and the cost for all of the transportation steps. The assumptions for the mode of transport of various radioactive materials are consistent with present (actual or proposed) industrial practices.

E.1 MATERIALS TO BE TRANSPORTED

Table E-1 lists the weight or the number of packages of various radioactive materials to be transported, the places from which the material is discharged, and the age of the material.

Table E-1
MATERIALS TRANSPORTED ANNUALLY FROM THE REACTORS
THE AFRs, AND THE P/E FACILITIES
REFERENCE CASE, CASE 1, CASE 2, AND CASE 3

Place of Discharge	Materials	Age, yr	Ref. Case	Case 1	Case 2	Case 3
Reactors	Spent fuel, MTHM	1	5,000	5,000	5,000	5,000
AFRs	Spent fuel, MTHM	10	—	5,000	5,000	5,000
	Non-TRU LLW, drums	1 to 10	—	6,200	6,200	6,200
P/E Facilities	Encapsulated spent fuel, canisters	10	—	10,200	8,070	18,520
	UF ₆ , 48X cylinder	1	740	—	—	—
	PuO ₂ , PPP-1M	1	2,500	—	—	—
	Hulls, canisters	(a)	3,820	—	1,520	3,820
	Kr-Xe-CO ₂	(a)	128	—	58	128
	Vitrified HLW, canisters	10	2,380	—	—	—
	Non-TRU LLW, drums	(a)	16,110	2,860	3,110	3,110
	Non-TRU ILW, canisters	(a)	920	—	—	—
	TRU LLW, drums	(a)	1,680	550	850	1,230
	TRU ILW, canisters	(a)	7,630	2,570	3,900	6,010

(a) 10 years old for Cases 1 to 3; 1 year old for Reference Case.

E.1.1 Reactor Annual Discharge

Spent fuel is stored on site in reactor storage pools for a period of 1 year prior to shipment. The spent fuel discharge rate from the reactors is assumed to be 5,000 MTHM/yr for all cases studied. That quantity of spent fuel consists of 6,800 pressurized water reactor (PWR) fuel assemblies (3,130 MTHM) and 10,200 boiling water reactor (BWR) fuel assemblies (1,870 MTHM). This assumption is consistent with published projections for 1990, with a basis for a BWR/PWR fuel ratio of 60/40 by number of assemblies and 37/63 by weight of heavy metal.⁽¹⁾

Spent fuel is shipped from the reactor by both rail and road transport. Road transport is necessary since 27 percent of the reactors presently in operation are not accessible to rail transport.⁽²⁾

E.1.2 Away-From-Reactor Storage Facility (AFR) Annual Discharge

The quantity of 10-year-old spent fuel discharged from the AFRs for Cases 1 through 3 is given in Table E-1. To make maximum use of rail transportation for heavy haul, all spent fuel is assumed to be shipped from the AFRs by rail.

A total of 6,200 55-gal drums of nontransuranic (non-TRU) low level waste (LLW) is generated at the AFRs per year, and is assumed to be shipped by truck to shallow land burial.

E.1.3 Processing/Encapsulation (P/E) Facility Annual Discharge

Spent fuel is received and promptly processed and encapsulated in the P/E facility 10 years after discharge from the reactors in every case except the Reference Case. In the Reference Case, spent fuel is received and fully processed 1 year after discharge from the reactors. In the Reference Case, 7,000 MT of uranium hexafluoride (UF_6), 80 MT of plutonium oxide (PuO_2), and 2,380 canisters of vitrified HLW are produced. The high level waste (HLW) is transported by rail; uranium hexafluoride and plutonium oxide are transported by road. (PuO_2 transportation is discussed further in Subsection E.5.2.) In Case 1, which involves simple encapsulation of spent fuel, 10,200 canisters of encapsulated spent fuel assemblies are generated; 8,070 canisters of chopped and voloxidized spent fuel are generated in Case 2; and 18,520 canisters of vitrified spent fuel material are produced in Case 3.

Cladding hulls and/or end fittings are discharged in every case except Case 1. In the Reference Case and Case 3, a total of 3,820 canisters are discharged, which includes both the cladding hulls and end fittings. In Case 2, end fittings are discharged in 1,520 canisters and cladding remains with the fuel materials. All cladding hulls are transported by rail. Hulls-type canisters are also used to contain krypton, xenon, and carbon dioxide gas cylinders. The Reference Case requires 38 canisters, as does Case 3; Case 2 requires 15 canisters.

Other waste forms to be considered for transportation include TRU and non-TRU wastes. All LLW, TRU and non-TRU, are packed in 55-gallon drums. However, the non-TRU LLW is shipped to shallow land burial, while the TRU-LLW and non-TRU ILW are shipped to the repository. These forms of waste are assumed to be transported by road.

E.2 MODES OF TRANSPORTATION

The ability to safely transport spent nuclear fuel and radioactive waste materials has been successfully demonstrated in the United States. Road transport is the most widely used mode for the routine movement of lower level radioactive materials. Higher level shipments, such as spent fuel, have been sent by rail, but the railroads are becoming increasingly reluctant to transport this material. Water transport of spent nuclear fuel in the United States has been limited to the Brookhaven Laboratory experience. On the other hand, water transport of spent fuel from country to country is rather common. Air transport of radioactive materials is generally for lighter packaging and is not applicable to this study.

E.2.1 Road Transportation

The national highway and road systems offer easy truck access to every reactor site, and therefore the radioactive waste and spent fuel from these reactors can be shipped by truck. Physically, legal weight trucks and casks should have no difficulty in reaching any reactor sites within the United States. Overweight truck casks can be handled at most reactor sites, but are restricted by state regulations and may require selective routings between the reactor site and the ultimate destination in order to comply with overweight permit stipulations.

Trucks can be used in intermodal transport of overweight casks between a reactor site and the nearest offsite railroad or dock. Some utilities are considering this arrangement for future spent fuel shipment, but road weight limitations and physical distances between the reactor site and the nearest rail or dock facility can seriously inhibit consideration of shipments of this type.

Currently, transportation of radioactive waste is mainly by truck. Trucks have several advantages over other modes of transportation. They are relatively inexpensive and can operate on existing public roads from point to point in the continental United States more easily and quickly than railroads or barges. Truck transportation is very flexible with regard to scheduling, handling, routing, and turnaround time. Since published tariff rates are available, shipment costs can be forecast accurately.

The availability of necessary equipment lends further flexibility to the truck transport mode. The trucks themselves present few problems; tractors and trailers are available in substantial numbers, and lead times for delivery of new equipment, when required, are relatively short. The biggest constraint on truck transport is the availability of the highly specialized shipping casks required for movement of spent nuclear fuel and radioactive waste. Future needs will have to be anticipated and appropriate lead times will have to be provided as expansion of the current cask fleet takes place, in order to meet the growth requirements outlined in Subsection E.5.4. Compliance with the weight limitation applicable to legal weight truck shipments tends to reduce the flexibility, and hence the desirability, of the truck transport mode.

For the purpose of this study, only legal weight truck shipments are considered. Figure E-1 shows a road transport configuration for a spent fuel shipment. Truck shipments of uranium hexafluoride and LLW are common and will continue to be the most likely form of transportation. Plutonium oxide shipments in any large quantity require special transportation concepts, such as the Allied General Nuclear Services (AGNS)/Sandia design.⁽¹³⁾

Truck transportation is furnished by trucking firms or motor carriers that haul freight owned by another party. These common carriers provide transportation services to the public in general. In the nuclear industry, a shipper normally uses a common carrier for transporting shipments of radioactive materials.

Figure E-1.
Road Transport Configuration

Interstate common carriers are subject to the regulations of the Interstate Commerce Commission (ICC). To operate interstate, a common carrier must have its specific certified (or authorized) by the ICC. These certified carriers are also known as regulated or authorized carriers and serve the truck transport industry.

Several motor carriers are involved in transporting nuclear materials. The scope of services performed by each carrier is in strict compliance with its individual ICC-approved operating authority. Normally, an operating authority is issued as a 5-year renewable type.

Some of the specialized carriers have voluntarily initiated personnel indoctrination and training programs, acquired radiation detection and monitoring equipment, and established health physics programs. Some concern exists that in the authorization process, large heavy-haul trucking firms, with little demonstrated competency in handling nuclear material, may receive operating authorities for nuclear shipments primarily on the basis of their trucking experience.

Intrastate operating authority for point-to-point hauling within a single state is obtained by carrier directly from the state involved.

E.2.2 Rail Transportation

Railroads are the primary means of transporting freight commodities in the United States. With few exceptions, the railroads serve the nation as common carriers that provide services to all shippers. These services are offered under published tariff and are subject to control by the ICC.

Figure E-2 shows some of the design features of a rail cask for spent fuel shipment. Existing and proposed rail cask and car packaging systems range in weight from 255,000 to 350,000 pounds, with even larger cask systems being planned. However, a check of the weight limitations for U.S. railroads shows 263,000 pounds to be the most common limitation for the major railroads, with some routes going up to 315,000 pounds for four-axle cars. One railroad lists 300,000 pounds as its maximum weight, even for a six-axle car.

One reference⁽³⁾ indicates that the weight limit for unrestricted traffic in most railroads is a gross weight of 263,000 pounds, and in some cases, individual track limits may be as low as 177,000 pounds. Therefore, although total weights in excess of 263,000 pounds can be moved, interchanges are limited and require separate negotiation between the shipper and the railroad. The various problems engendered between the carriers of nuclear material and the railroads seem to lead to the general conclusion that nuclear packaging systems should be designed for unlimited interchange to avoid negotiation on size and weight clearances.

The SFDS uses a gross weight of 263,000 pounds as a design basis. If current rail cars (hopper cars) with a weight range of 57,000 to 63,000 pounds are employed, a cask/car packaging system with an empty cask of about 70 tons is compatible with the 263,000-pound weight limit.

In addition to the weight limitation, dimensional limitations exist for unlimited interchange, as shown in Figures E-3 and E-4 (based on Association of American Railroads (AAR) Plate B specifications⁽³⁾). A key factor is that longer overall car lengths can be achieved only by using narrower car widths. Weight estimates for a cask/car packaging system are given in Table E-2.

Figure E-2.
Design Features of a Rail Cask for Spent Fuel Shipment

Figure E-3.
AAR Rail Car Size Units⁽⁴⁾

Figure E-4.
AAR Width Limits for Rail Cars⁽⁴⁾

Table E-2
WEIGHT ESTIMATES FOR
A CASK/CAR PACKAGING SYSTEM

Item	Weight lb x 10 ³
Rail car	57 to 204
Appurtenances	4 to 7
Loaded cask	155 to 204
Totals	216 to 273
Maximum weight (for unlimited use)	263

Transporting spent fuel and radioactive waste materials via rail, instead of truck, greatly reduces the total number of required shipments. For example, 10 spent fuel assemblies of the PWR type (containing about 4.5 metric tons of uranium) can be transported in one modern rail cask, whereas only 1 such assembly can be shipped in a truck cask. In addition to this potential 10:1 advantage in number of required shipments, the overall operating expense, such as loading and unloading operations and radiation exposure at the reactor and the fuel storage facilities, are also reduced. The nuclear industry will have a more efficient and effective transportation system if rail service is the basis of this system. Thus far, the railroad industry has not shown a great deal of willingness to transport nuclear materials.

E.2.3 Water Transportation

This mode of transport for radioactive wastes and spent fuel has not been developed for commercial use, mainly because the nuclear industry until now has had little need for water transport services. As a result, little operational experience has been acquired. The only ongoing efforts at this time are the results of American National Standards Institute (ANSI) Standard N.552 (see Table E-3) and a Japanese program which has evolved from that country's need to ship spent fuel to British Nuclear Fuels Limited.

Table E-3
AREAS COVERED IN ANSI N.552 STANDARD ON
WATER TRANSPORTATION

Number	Subject
I	Scope
II	Definitions
III	Packaging <ul style="list-style-type: none"> NRC/DOT/USCG Regulation General Description Special Considerations
IV	Vessels <ul style="list-style-type: none"> General Hull Structure Stability Criteria for Vessels to be Utilized in Transporting Spent Fuel Tiedown Devices Between Package and Hull Structure
V	Licenses, Procedure, and Records <ul style="list-style-type: none"> Licenses and Documents Procedures Records
VI	Operations <ul style="list-style-type: none"> Transportation Vessel Manning Loading/Unloading Cranes and Rigging Docking Facilities Roll-On/Roll-Off
VII	Radiation Limits, Monitoring, and Control <ul style="list-style-type: none"> General Requirements Monitoring Radiation Levels Radiological Emergency Procedures
VIII	Recovery and Salvage

The lack of industry experience is probably the most severe problem facing potential users of water transport. In addition, the transport industry is cautious about attempting anything new in the present climate of environmental concern.

The navigable waterways of the United States cover a total length of approximately 29,000 miles. Of this length, about 16,000 miles are usable by vessels requiring operating depths of 9 feet or more. These deep waters are generally uninterrupted by shallows and encompass most of the Atlantic and Pacific seaboards, the Gulf Coast, the Mississippi River system, and the Columbia-Snake River system in the State of Washington. In general, navigable waterways do not exist west of a north-south line defined by the Kansas-Missouri border, and east of the Pacific coastal waters. Also, deep navigable waterways are not generally found between the slopes of the Appalachian Mountains and the Atlantic coastal regions.

The U.S. waterway system is a federally controlled commercial thoroughfare maintained by the U.S. Army Corps of Engineers and policed by the U.S. Coast Guard. Users of the waterways must comply with Coast Guard regulations, which are listed in 49 CFR 176.

The primary constraint in water transport is the degree of regulatory control of the carriage vehicle (whether barge or motor vessel) and hence the cost of complying with such regulations versus the economics of shipment. The Coast Guard, under whose control water transport of radioactive material falls, is likely to require a technique that assures sufficient safety and control of the load under worst-case sea and weather conditions.

Water transportation of radioactive waste has received considerable attention from potential users. Possible applications include land-based reactors, offshore reactors, and overseas reactors. Domestic experience has been limited to reviews of requirements, consideration of problems, and the initiation of an ANSI standard for water transport of irradiated nuclear fuel.

A considerable quantity of spent fuel has been shipped via water transport (over 100 shipments) in Europe, and the trend is increasing. A special motor ship for shipping spent fuel from Japan to Britain for reprocessing has been ordered by British Nuclear Fuels Limited (BNFL).

Science Applications Incorporated (SAI) has conducted studies on water transport of spent fuel.⁽⁵⁾ These studies, performed for the Environmental Protection Agency (EPA) and the Atomic Industrial Forum (AIF), have focused on safety aspects of shipment within the territorial waters of the United States. Battelle Memorial Institute (BMI) has performed a similar study and has also examined intercontinental ocean travel.⁽⁶⁾ The SAI and BMI reports have concluded that there is little environmental risk, and that casks sunk in shallow coastal waters will not rupture from the pressure. Casks sunk in considerably deeper water are also not expected to rupture, but some leakage of seawater past the closure head seals may occur. The leakage will stop when the internal cask pressure and the ocean's hydrostatic pressure equalize. BMI researchers have concluded that only small amounts of radioactivity (less than the background activity of seawater) will be released, and this will be due to molecular diffusion. SAI researchers have concluded that shipping casks will survive any impact due to collisions between ships or a ship colliding with a fixed structure.

A review of the above data shows that technical assessment of water transport is premature. A discussion of the commercial aspects, such as quantities of shipment waste, mode of operations, number of casks, transport rate, and even cost, is meaningless at this time.

E.2.4 Intermodal Transportation

The current planning for moving spent fuel from commercial nuclear power plants has been limited to single-mode (either truck or rail) transport only. In the comparatively few spent fuel shipments made in the United States, only single-mode transport has been used. Recent evaluations made on facility handling requirements, operator radiation exposure level, and public acceptance of radioactive materials transport have indicated that a large percentage of reactor plants (perhaps 75 to 90 percent) either desire or may be required to ship spent fuel via rail casks. Since many of these plants are situated from 2 to 40 miles from the nearest rail spur, development of private railroad trackage by the electric utility is as a rule economically prohibitive and in some cases infeasible. A possible alternative is to utilize two or more modes of transport. For example, since over 100 nuclear plants are located on navigable waterways, the spent fuel package may be transported by a barge or motor ship to an intermediate rail spur if direct delivery were not possible. The alternative is to "heavy-haul" the spent fuel cask by a transporter directly to a nearby rail spur. This option of using intermediate (heavy-haul transporter or barge) mode of transport is commonly referred to as intermodal transport (IMT).

Research reactor fuel from Brookhaven National Laboratory (BNL) on Long Island (N.Y.) has been trucked, ferried over water to Connecticut, and then transported via truck to the Savannah River Plant in South Carolina. This intermodal operation was begun in January 1976, and is the first IMT of its type in the United States.

The BNFL plant at Windscale in the British Isles receives spent fuel from all over the world. A sea voyage is mandatory. Spent fuel has been transported to this facility by road, rail, and sea for over 9 years. The choice of transport method is dictated largely by the geography of the reactor. Casks from reactor sites are sent to Windscale by a number of different methods, including the following IMT methods: (1) by direct loading to a charter ship at the site; (2) by road transport to the nearest port for loading to a chartered ship. The transfer from road or rail to sea transport requires the use of an expensive heavy lift crane and careful surveillance of the tiedown after the transfer. Overweight truck casks weighing 38 tons (similar to the TN 8/9 in the United States) have been used for truck-ship-truck movements.

The number of water shipments made to BNFL from the European continent through 1976 is given below.

- Charter ship cask movements — 38
- Rail ferry cask movements — 36
- Roll-on/roll-off ferry movements — 45

In Europe, casks of spent fuel are frequently heavy-hauled by truck to a rail spur and then to the reprocessing plant located in La Hague in France.

Direct movement of rail cars onto a barge — roll-on/roll-off ferry movement, as shown in Figure E-5 — has been practiced for over 75 years. For such a movement various techniques such as stationary dock, ballasting of the barge to the proper height, single-point hinging of the barge to the dock, hinged transfer bridges, and self-ballasting floats have been used. In all cases, a sheltered area is required to minimize wind and wave action. In the case of separating the cask from the rail car, shown in Figure E-6, there is a need for costly equipment and facility improvement. In both cases, the labor cost of a special handling crew and the extra lease cost of the cask accrued while the cask is being onloaded/offloaded are in addition to the costs normally associated with single mode transport. The European experience with IMT indicates that the onloading and offloading sites require special costly modifications.

Figure E-5.
Roll-On/Roll-Off Movement of Spent Fuel Package⁽¹⁴⁾

Figure E-6.
Direct Load Lift to Barge⁽¹⁴⁾

At present, to make IMT economically feasible, it is necessary to reduce the number of utility operators and operation time, to relax facility receipt limitations, and to reduce operator radiation exposure. The problems with railroads, coupled with local highway restrictions for truck transport, make the formation of rail-truck IMT unattractive. Hence, there has been considerable interest in developing IMT involving a water transport alternative.

The European experience indicates that IMT is both technically feasible and warranted. As noted previously, both the British and the Japanese feel sufficiently strong about the need for water shipments that they have ordered a ship specifically for this purpose.

The only other U.S. participation in IMT studies has been by owners and designers of reprocessing plants, who have been involved out of necessity. The Allied General Nuclear Services (AGNS) plant is located in an area of South Carolina where IMT appears to be viable. AGNS has also conducted studies and discussions with various electric utilities concerning IMT matters.

The results of both evaluation of European experience and studies conducted within the United States indicate two things:

- Intermodal transport of nuclear materials eventually may be required within the United States
- There are no major technical obstacles to IMT at this time. The equipment does exist

E.3 TRANSPORTATION REGULATIONS

E.3.1 Federal Regulations

The transportation or packaging for transport of radioactive material requires appropriate licenses to be issued. Applicants for a license to package or to transport radioactive material show by a combination of analysis and experiment that the proposed package or transport vehicle satisfies all requirements set forth in the appropriate code of federal regulations. The application describes proposed controls or precautions to be used in the loading, unloading, handling, and transportation of radioactive material, and procedures to be followed in the event of an accident delay or shipment. Inspection and accountability procedures also have to be described.

The following federal regulations are applicable to the transport of radioactive materials within the context of the Spent Fuel Disposal Study (SFDS).

- 49 CFR 171: General information, regulations, and definitions
- 49 CFR 172: Hazardous materials table and communications regulations
- 49 CFR 173: General requirements for shipping and packaging
- 49 CFR 174: Carriage by rail
- 49 CFR 177: Carriage by public highway
- 49 CFR 178: Shipping container specifications
- 10 CFR 71: Nuclear Regulatory Commission (NRC) regulations governing packaging and shipment of radioactive materials
- 10 CFR 75: NRC regulations for protection of special nuclear material in transit

E.3.1.1 Transport Group. Radioactive materials are classified for transportation purposes into one of seven transport groups according to their radiotoxicity if released to the environment. Transport Group I is the most restrictive and consists of highly toxic material, such as plutonium; Transport Group VII is the least restrictive and consists of materials with relatively low hazard potential, such as tritium gas. The classification of various isotopes in the seven transport groups is specified in Appendix C of 10 CFR 71.

E.3.1.2 Quantity Designations. A shipment of radioactive material in an individual container is classified as Type A, Type B, or large quantity of radioactive material, depending on the amount of radioactivity it contains. The maximum activities for normal-form* Type A and Type B quantities of radioactive material by transport group are shown in Table E-4.

If a shipment contains materials from more than one transport group, the quantity type is usually determined by the most restrictive group present (see 10 CFR 71.4).

*Normal-form material is everything that does not meet the special-form criteria outlined in Subsection E.3.1.4

Table E-4
MAXIMUM ACTIVITIES FOR NORMAL TYPE A AND B QUANTITIES
OF RADIOACTIVE MATERIAL IN A SINGLE CONTAINER
BY TRANSPORT GROUP

Transport Group	Type A Quantity, Ci	Type B Quantity, Ci
I	0.001	20
II	0.05	20
III	3	200
IV	20	200
V	20	5,000
VI and VII	1,000	50,000
Special Form	20	5,000

NOTE: Any quantity greater than Type B is defined as a large quantity of radioactive material. An exception is ^{252}Cf , for which the Special Form, Type A quantity limit is 2 curies.

E.3.1.3 Low-Specific-Activity Materials. Some shipments of material with a relatively low hazard potential are exempt from certain of the requirements in the regulations, even though they may contain Type A or B quantities of radioactive materials. Low-specific-activity materials include uranium or thorium ores; unirradiated, depleted, or natural uranium; unirradiated thorium; aqueous solutions of tritium oxide containing less than 5 mCi/ml; and materials having uniformly distributed activity in which concentration per gram does not exceed the following:

- 0.1 microcurie of Group I materials
- 5.0 microcuries of Group II radionuclides
- 300 microcuries of Group III and IV materials

Externally contaminated nonradioactive materials may be considered low-specific-activity provided contamination is not readily dispersible and the average surface contamination per square centimeter does not exceed the following:

- 0.1 microcurie for Group I radionuclides
- 1.0 microcuries for other materials

E.3.1.4 Special-Form Material. Radioactive material in special form may be transported as if it belonged in Transport Group V, except for ^{252}Cf in Type A quantities, as indicated in Table E-4. Special-form material is radioactive material, either in solid form or encapsulated, that meets one or more of the following criteria:

- Has no dimension less than 0.5 millimeter or one dimension greater than 5 millimeters
- Does not sublime, melt, or ignite in air at 538°C
- Remains nondispersible after immersion for 1 week in water at 20°C or air at 30°C

Special-form material in a capsule must remain intact after completion of a test series which may include a 9.1-meter drop, impact of a 1.4-kilogram steel rod dropped from a height of 1 meter, heating to 800°C for 10 minutes, and immersion in water for 24 hours.

E.3.1.5 Fissile Classification. A shipment containing fissile materials (^{233}U , ^{235}U , ^{238}Pu , ^{239}Pu , or ^{241}Pu) is classified according to the amount of control that must be exercised to prevent nuclear criticality in shipment.

- Fissile Class I packages may be shipped in unlimited numbers in any arrangement. They do not pose any danger of nuclear criticality
- Fissile Class II packages may be shipped in any arrangement, but the total number limited to an aggregate transport index total of 50. For purposes of nuclear criticality safety, individual packages are classified to have transport indices between 0.1 and 10, depending upon chemical form and total weight of fissile material
- Fissile Class III packages must be controlled in transport by special arrangements between shipper and carrier to ensure nuclear criticality safety

Certain shipments of fissile materials are exempt from the normal requirements for packages containing fissile material. These packages meet one or more of the following criteria (see 10 CFR 71.9):

- Contain less than 15 grams of fissile material
- Contain thorium or uranium with less than 0.72 percent of fissile material
- Contain less than 350 grams of fissile material with a maximum of 5 grams in any 0.028 m^3 (1 ft^3) of the package

E.3.1.6 Special Nuclear Material (SNM). A shipment of radioactive material is designated SNM if it contains 5,000 grams or more of ^{235}U (contained in uranium enriched to 20 percent or more), or 2,000 grams or more of ^{233}U or plutonium, or any combination of the three, exceeding 5,000 grams, as computed by the formula.

$$\text{grams SNM} = \text{grams } ^{235}\text{U} + 2.5 (\text{grams } ^{233}\text{U} + \text{grams Pu})$$

Regulation 10 CFR 75 requires that shipments of SNM be afforded additional physical protection in transit to prevent theft or sabotage. These protective measures include:

- NRC-approved preplanned carrier procedures
- Hand-to-hand receipts
- Tamper-indicating seals on all containers
- Minimum container weights of 227 kilograms for shipments in open vehicles
- No cargo transfers en route
- Periodic radio communication between transport vehicle and carrier dispatcher
- Armed escorts or specially designed transport vehicles with disabling features

E.3.1.7 Radiation Dose Rate Requirements. Radiation dose rates from shipments of radioactive materials are also limited by the regulations. Larger maximum radiation dose rates are allowed for vehicles used exclusively to transport radioactive materials than for vehicles not exclusively used.

For closed transport vehicles used exclusively for the transport of radioactive materials, the limits are as follows:

- 1,000 mrem/hr at 91 centimeters from the external surface of the package
- 200 mrem/hr at any point on the external surface of the vehicle
- 10 mrem/hr at 1.8 meters from the external surface of the vehicle
- 2 mrem/hr in any normally occupied position in the vehicle

For vehicles not used exclusively for the transport of radioactive materials, the limits are as follows:

- 200 mrem/hr at any point on the external surface of the package
- 10 mrem/hr at 91 centimeters from the package

E.3.1.8 Packaging. The type of packaging required for shipment of radioactive material depends on the type, quantity, form, and fissile classification of the material to be shipped. Department of Transportation (DOT) Regulation 49 CFR 178 allows radioactive material to be shipped in DOT specification containers, or in containers licensed by the NRC. The specification containers are described in the regulations. All radioactive material shipping containers must meet the general package requirements for radioactive material shipping containers, as specified in 49 CFR 173.393, and the general requirements for all packaging used in interstate commerce, as specified in 49 CFR 173.24.

Low-specific-activity material shipped in exclusive-use vehicles is exempted from most of the packaging requirements of 49 CFR 173. The only requirements for this material is that its packaging be strong and tight. The radiation dose limits discussed previously, however, are still applicable.

Type A quantities of radioactive material have to be shipped in DOT specification Type A packaging, approved Type B packaging, or packaging that meets the requirements of DOT Specification 7A (see 49 CFR 178.350).

In addition to meeting the general packaging requirements of 49 CFR 173.24 and 49 CFR 173.393, Specification 7A packaging must be capable of maintaining its shielding integrity and preventing the dispersal of its contents during tests that simulate conditions that normally exist during transportation. These test conditions are outlined below.

E.3.1.9 Test Conditions that Normally Exist During Transportation. These conditions are as follows:

- **Heat.** Direct sunlight at an ambient temperature of 54°C in still air
- **Cold.** An ambient temperature of -40°C in still air and shade
- **Pressure.** One half standard atmospheric pressure
- **Vibration.** That which normally occurs during transportation
- **Water Spray.** Sufficient to keep the exposed surfaces of the package, except the bottom, wet for 30 minutes
- **Free Drop.** 1.5 to 2.5 hours after the water spray test at a distance of 1.2 to 1.3 meters (depending on package weight) onto an unyielding surface
- **Corner Drop.** Onto each corner in succession (each quarter of each rim for cylinders) from a height of 30 centimeters onto an unyielding surface. (This applies only to wood or fiberboard containers of less than 45 kilograms in weight and to all Fissile Class II packages.)
- **Penetration.** Impact of the hemispherical end of a 5.9-kilogram steel cylinder, 3.2 centimeters in diameter, dropped from a height of 1 meter onto the most vulnerable surface of the package
- **Compression.** A compressive load against the top and bottom of the package in its normally transported condition. The load is either five times the weight of the package or 1.38 newtons/cm² (2 psi) times the minimum horizontal cross section of the package, whichever is greater. (This test applied only to packages weighing less than 5 tons.)

Type B quantities of radioactive materials may be shipped in DOT Specification 6M containers, in packaging authorized by the NRC, or in packaging meeting the 1973 International Atomic Energy Authority (IAEA) requirements (provided that the packaging has been approved by the competent foreign authority and has been validated by DOT).

E.3.1.10 Hypothetical Transportation Accident Test Conditions. Type B packaging submitted to the NRC for approval must be able to maintain its integrity under test conditions that simulate transport accident environments, as well as under test conditions that simulate conditions that normally exist during transportation. The following tests are performed sequentially to determine the cumulative effect on the package:

- **Free Drop.** Through a distance of 9.1 meters onto an unyielding surface and striking that surface in the position that would produce maximum damage
- **Puncture.** A free drop through a distance of 1 meter onto a cylindrical steel bar 15 centimeters in diameter with the edge rounded to a radius of 6 millimeters. The length of the bar is selected for maximum damage (minimum length is 20 centimeters). The bar is mounted on an unyielding horizontal surface, and the package is dropped in a position expected to produce maximum damage
- **Thermal.** Exposure to the equivalent of an 800°C fire for 30 minutes with no artificial cooling for 3 hours after the exposure
- **Water Immersion.** Immersed at a depth of 91 centimeters for 8 hours (for fissile material packages only)

Large quantities of radioactive materials are shipped in Type B packaging. The packaging used must provide adequate shielding and heat removal capabilities for the radioactive material to be transported.

Fissile material may be shipped in DOT Specification 6M or 6L (Type A quantities only) packaging or in packaging approved by the NRC.

Packaging approved by the NRC for fissile materials must meet some requirements not imposed on containers for other radioactive materials (see 10 CFR 71.38, 71.39, and 71.40). These conditions ensure a wide margin of safety on the number of packages that can be shipped at one time. They also ensure that a critical mass could not be formed in the event of a transportation accident.

E.3.1.11 Special Regulations Applying to Plutonium. Because of its hazardous nature, plutonium in excess of 20 curies per package can be shipped only in solid form (plutonium oxide). Present regulations for transportation specify that the transport system must meet the requirements of 10 CFR 71, 10 CFR 73, and 49 CFR 171 to 178. The plutonium must be double-packaged, and the total container system must maintain its containment integrity while undergoing the tests for Type B packaging.

Federal regulations prohibit shipments of plutonium in passenger aircraft in quantities exceeding 20 grams or 20 curies, whichever is less (10 CFR 71.42). However, a recently enacted federal statute has placed an embargo on shipments of plutonium by air until it is shown that the shipping containers will not release their contents in the event of an airplane crash.

E.3.2 State and Local Regulations

Although federal agencies dominate the regulatory process for the transportation of radioactive materials, state governments also exercise some control over these shipments. State highway departments regulate gross vehicle weights, vehicular dimensions, and other parameters for all shipments including radioactive shipments. About one half of the states have adopted the DOT Hazar-

dous Materials Regulations to cover intrastate shipments. Several states have adopted or proposed additional regulations concerning radioactive materials. These include:

- Special routing of radioactive shipments
- Advance notification for shipments of large quantities of radioactive materials
- State inspections of some types of radioactive shipments
- Prohibition of certain types of shipments within the state
- Prior approval for radioactive shipments
- Requirement of exclusive vehicle use for radioactive shipments
- Use of pilot vehicles
- Speed restrictions for radioactive shipments
- Accompaniment of all shipments by radiation monitoring personnel
- Specific hours of movement

The differences in regulations from one state to another can often require special considerations for interstate shipments. Specific attempts by state and local government to regulate nuclear shipments are outlined in Table E-5.

A potential conflict exists between some of the proposed state laws and the provisions of the National Transportation Act of 1974 (Public Law 93-633, signed in 1975), which prohibits the states from adopting laws or regulations more stringent than federal regulations unless the state regulations improve transportation safety. Even in this case, such rules can be adopted only if they do not unreasonably burden commerce.

Table E-5
STATE AND MUNICIPAL LEGISLATION AFFECTING THE
TRANSPORTATION OF RADIOACTIVE MATERIALS⁽¹⁶⁾

State	Statute	Provisions
I. BILLS ENACTED INTO LAW		
Arkansas	S-327	Arkansas State DOT authorized to promulgate additional rules governing transportation of hazardous materials.
California	A1593	Revises California State Code to require Department of Health to issue regulations and permits governing transport, handling, storage, and disposal of hazardous waste (9/22/77).
Colorado	HJ1045	Directs legislative council to study transportation of radioactive materials within the state by air, motor, or rail to determine and make recommendations to protect against hazards.
Connecticut	HB5708	Requires permit from Commissioner of Transportation to transport into or through the state any quantity of radioactive materials for waste disposal.
	H5358	Amends state statutes regarding permits required for radioactive materials to include any placarded shipment carried by commercial carrier (5/19/77).

Table E-5 (Cont'd)

Florida	DUM01	Bans all nuclear shipments through the Port of Miami (1/79).
	DUM02	Department of Health and Rehabilitative Services shall adopt reasonable regulations governing the transportation of radioactive materials. 1977 Florida Statute 381.512. Public Health Chapter 10 D-63.
	NUS00	Fort Pierce, Florida requires radioactive materials transporter to phone the fire marshall for escort before entering the city.
	NUS01	Fort Walton Beach, Florida requires radioactive materials transporter to phone city police for escort before entering the city.
	NUS02	Key West, Florida requires radioactive materials transporter to phone city police for escort before entering the city.
	NUS03	Tampa, Florida requires radioactive materials transporter to phone fire department outside of city before entering the city.
Illinois	HB1815	State DOT to develop legislative program regulating transportation of hazardous materials.
	S-245	Authorizes the Illinois Department of Transportation to regulate the transport of hazardous materials over state highways.
Kansas	H2559	Authorizes the Department of Health and Environment to adopt rules and regulations governing hazardous waste that is stored, collected, transported, or disposed of within state.
	NUS00	No radioactive shipments within the City of Wichita.
Kentucky	HR70	Appoints a special advisory committee to assume an oversight role of all matters pertaining to nuclear waste 78-80.
Louisiana	H420	Prohibits the transportation of high-level radioactive wastes into the state for the purpose of disposal or storage (signed 7/6/78).
	H1396	Created an Office of Environmental Affairs within the Department of Natural Resources which has jurisdiction over all matters affecting the regulation of solid and hazardous waste disposal, radiation control, and transportation.
Maine	NUS00	No radioactive loads permitted on the Maine Turnpike.
Maryland	S511	Alters certain bonding requirements as a condition to the issuance of a permit for those engaged in the transportation and disposal of hazardous materials in Maryland (4/29/77).
	HJR23	Maryland State Departments of Transportation, Health, and Natural Resources are to formulate recommendation for consolidating into one agency the authority to regulate and supervise the transportation of hazardous material through Maryland.

Table E-5 (Cont'd)

	NUS01	To cross Wm. Preston Lane, Jr. Memorial Bridge (formerly Chesapeake Bay Bridge) in Maryland, some types of radioactive materials must have prior permit from the Superintendent of Bridges (Toll Facilities Administration booklet, 7/76).
	NUS02	To cross Harry W. Nice Memorial Bridge (formerly Potomac Toll Bridge) in Maryland, some types of radioactive material must have prior permit by the Superintendent of Bridges (Toll Facilities Administration booklet, 7/76).
	NUS03	To cross Susquehannah River toll bridge, some types of radioactive material must have prior permit from the Superintendent of Bridges (Toll Facilities Administration booklet, 7/76).
	US00	No radioactive loads permitted in the Baltimore Harbor Tunnel: I-95.
Massachusetts	DUM01	Establishes the Department of Public Works as the agency governing transportation of hazardous materials on state highways.
	US00	No radioactive loads permitted on the Mass Turnpike.
Nebraska	H332	Requires transporters of hazardous materials to carry specified amounts of liability insurance (3/21/77).
New Jersey	A1953	Prohibits transportation of radioactive materials without certificate of handling from state.
	1978-141	Maple Heights, New Jersey city ordinance requires permits for transport of certain types of radioactive materials. Passed by council (12/6/78).
	NUS00	No radioactive loads permitted on New Jersey Turnpike.
New York	AB-7761B	Grants state DOT the authority to regulate transportation of radioactive materials (7/21/76).
	NYC-175	Article 175 of New York City health code extends control over and severely restricts shipments of certain categories of hazardous materials (including irradiated fuel, plutonium, and uranium) in NYC.
N. Dakota	NUS00	Any vehicle bearing radioactive placards must stop at all railroad crossings.
Ohio	NUS00	No radioactive loads allowed within city limits of Cleveland, Ohio, but no restriction on interstate around city.
Pennsylvania	NUSS0	No radioactive loads on Pennsylvania Turnpike.
Vermont	H549	Authorizes the State Department of Transportation to promulgate rule governing the transport of hazardous (including radioactive) materials within state (4/5/78).

Table E-5 (Cont'd)

W. Virginia	NUS00	No radioactive loads permitted on West Virginia Turnpike.
II. PENDING LEGISLATION		
Alabama	AL-H660	Would require advance notification to Departments of Public Health or Public Safety before spent fuel could be transported into or through Alabama.
Arkansas	DUM00	Would require prior notification of transportation of radioactive materials; time not specified.
Connecticut	DUM00	Would require a special permit for large quantities of radioactive materials and a police escort for spent fuel shipments.
Florida	H212	Would prohibit the transportation of certain radioactive materials unless the shipper received a certificate of emergency transportation 5 days prior to shipment.
Georgia	H2068	Would require shippers of radioactive materials to notify the Public Safety Commission not less than 10 hours prior to transportation in, to, or through state.
Illinois	H3350	Would authorize the Illinois Commerce Commission to regulate the transportation of hazardous materials by rail.
Maine	H892	Would regulate the transportation of radioactive materials within state.
	H1958	Would authorize the Commissioner of Public Safety to promulgate regulations for the possession, storage, handling, and transportation of hazardous materials.
Massachusetts	H3086	Would require registration of shipments of nuclear waste with state police, who would determine the route to be taken and the standards to be met.
	H5011	Would ban the transportation of radioactive material through any city/town unless a legislative body of municipality approves; and Public Health Department has been notified and certifies that safety standards have been met.
	H6075	Would establish a Nuclear Waste Advisory Committee to study the transportation of nuclear waste.
	DUM00	Newton City Council is considering an ordinance that would prohibit the transportation of radioactive wastes through the town.
Michigan	S523	Would license the transportation of radioactive waste material and nuclear medicine.
Minnesota	1619	Would prohibit transportation of radioactive materials without prior registration with the Commissioner of Public Safety.

Table E-5 (Cont'd)

New Jersey	S1122	Would amend state's radiation protection act by adding a new section to ban the transportation into the state of spent fuel rods or mixed fission products associated with the fuel elements (over 20 curies).
	A1566	Would authorize the Department of Environmental Protection to adopt regulations governing transportation of hazardous substances in state.
Oregon	DUM00	Would restrict shipments of nuclear material through Lane County.
	S238	Would authorize the Environmental Quality Commissioner to adopt rules which would require a permit to operate a motor vehicle that handles, transports, and disposes of environmental wastes and establishes trip reporting.
S. Carolina	H2234	Would require any motor vehicle transporting nuclear materials to be of such construction ...fastened to prevent escape of materials.
Tennessee	H2212	In the event of an accident involving hazardous materials, the party "who transported or caused to be transported or both" would be liable for damage and would reimburse state/locality for full cost of restoration of accident site.
Virginia	NUS00	Fairfax County asked Philadelphia Electric to avoid shipping irradiated fuel through the county.
W. Virginia	H1143	Would provide that Departments of Natural Resources and Health adopt rules governing disposal and transportation within the state.
	H1918	Would provide for defining of hazardous radioactive materials, require 5 days' prior notification (including proposed route), provide for approval of plans, and provide for notification of local official along route.
Wisconsin	H336	Would require that all vehicles transporting radioactive materials be equipped with dual controls and brakes, that there be a primary and secondary driver in the vehicle, and that the secondary driver be able to assume control of the vehicle at any time.
	H671	Would require a permit from the DOT before transportation begins.

E.3.2.1 Truck Size and Weight Limitation. Three basic gross vehicle weight (GVW) standards apply to the nation's interstate and primary highway system and must be considered in vehicle design. New interstate limits implemented in January 1975 permit a GVW up to 80,000 pounds; however, 10 states and the District of Columbia did not adopt this new limit and continue to allow a GVW of only 73,280 pounds, and Connecticut permits a GVW of only 73,000 pounds. For a complete listing, see Table E-6.

Table E-6
SINGLE AND TANDEM AXLE LIMITS AND MAXIMUM
GROSS VEHICLE WEIGHT, BY STATES⁽¹⁴⁾
(As of Jan. 1, 1978)

State	Single-Axle Load, 16 (a)		Tandem-Axle Load, 16 (a)		Gross Vehicle Weight, 16 (a)	
	Interstate System	Other Highways	Interstate System	Other Highways	Interstate System	Other Highways
Alabama	20,000	22,000	39,600	44,000	80,000	92,400
Alaska	—	20,000	—	34,000	—	109,000 ^(b)
Arizona	20,000	Same	34,000	Same	80,000	Same
Arkansas	18,000	Same	32,000	Same	73,280	Same
California	20,000	Same	34,000	Same	80,000	Same
Colorado	20,000	18,000	36,000	Same	80,000	85,000
Connecticut	22,848	Same	36,720	Same	73,000	Same
Delaware	20,000	Same	34,000	40,000	80,000	Same
District of Columbia	22,000	Same	38,000	Same	73,280	Same
Florida	22,000	Same	44,000	Same	80,000	Same
Georgia	20,340	Same	40,680	Same	80,000	Same
Hawaii	24,000	Same	34,000	Same	80,800	88,000
Idaho—Without permit	20,000	Same	34,000	Same	80,000	105,500
—By long-term permit					105,500	105,500
Illinois	18,000	Same	32,000	Same	73,280	Same
Indiana	18,000	Same	32,000	Same	73,280	Same
Iowa	18,540	Same	32,960	Same	73,280	Same
Kansas	20,000	Same	34,000	Same	80,000	85,500
Kentucky	20,000	21,000	34,000	35,700	80,000	82,000
Louisiana	20,000	Same	34,000	Same	80,000	Same
Maine	22,000	Same	34,000	38,000	80,000	Same
Maryland	22,400	Same	40,000	Same	73,280	Same
Massachusetts	22,400	Same	36,000	Same	80,000	Same
Michigan	20,000 ^{(c), (d)}	20,000	34,000 ^{(c), (d)}	34,000 ^(d)	80,000 ^{(c), (d)}	80,000 ^(d)
Minnesota	20,000	Same	34,000	Same	80,000 ^(c)	Same
Mississippi	18,000	Same	32,000	Same	73,280	Same
Missouri	18,000	Same	32,000	Same	73,280	Same
Montana—Without permit	18,000	Same	32,000	Same	76,800	Same
—By long-term permit	20,000	Same	34,000	Same	105,500	Same
Nebraska—Without permit	18,900	20,000	33,600	34,000	73,280	95,000
—By long-term permit	20,000	20,000	34,000	34,000	95,000	95,000
Nevada—Without permit	20,000	Same	34,000	Same	80,000	109,000
—By long-term permit					129,000	Same
New Hampshire	22,400	Same	36,000	Same	80,000	Same
New Jersey	23,520	Same	34,000	35,700	80,000	Same
New Mexico	21,600	Same	34,320	Same	86,400	Same
New York	22,400	Same	36,000	Same	80,000	Same
North Carolina	20,000	Same	38,000	Same	79,800	Same
North Dakota	20,000	Same	34,000	Same	80,000	105,500
Ohio	20,000	Same	34,000	Same	80,000	Same

Table E-6 (Cont'd)

State	Single-Axle Load, 16		Tandem-Axle Load, 16		Gross Vehicle Weight, 16	
	Interstate System	Other Highways	Interstate System	Other Highways	Interstate System	Other Highways
Oklahoma	20,000	Same	34,000	Same	80,000	90,000
Oregon—Without permit	20,000	Same	34,000	Same	80,000	Same
—By long-term permit					105,500	Same
Pennsylvania	23,072	Same	37,080	Same	73,280	Same
Rhode Island	22,400	Same	36,000	Same	80,000	Same
South Carolina	20,000	22,000	35,200	39,600	80,000	88,000
South Dakota	20,000	Same	34,000	Same	80,000	95,000
Tennessee	18,000	Same	32,000	Same	73,280	Same
Texas	20,000	Same	34,000	Same	80,000	Same
Utah—Without permit	20,000	Same	34,000	Same	80,000	Same
—By long-term permit			36,000	Same	105,500	Same
Vermont	22,400	23,520	36,000	37,800	80,000	Same
Virginia	20,000	21,000	34,000	35,700	79,800	Same
Washington—Without permit	20,000	Same	34,000	Same	80,000	Same
—By long-term permit					105,500	Same
West Virginia (c)	20,000	Same	34,000	Same	80,000	Same
Wisconsin	20,000	Same	34,000	Same	80,000	Same
Wyoming	20,000	Same	36,000	Same	80,000	101,000

(a) Includes tolerances.

(b) Interstate system. Gross weight determined by the number and spacings of axles within 70 feet (9 axles at 66 feet equals 109,000 pounds).

(c) On interstate and other designated highways.

(d) Higher gross weight allowed, determined by number and spacing of axles within 55 feet, with a maximum of 11 axles, but with lower axle and tandem weights.

(e) Gross weight determined by number and spacing of axles within 105 feet (9 axles at 101 feet equals 129,000 pounds).

In addition to weight limitations, size limitations also exist, as shown in Figures E-7 and E-8. Several states and the District of Columbia restrict the overall length to 55 feet, as shown in Figure E-8.

NEW INTERSTATE LIMITS (80,000 LB GVW)

$$W = 500 \left[\frac{LN}{N-1} + 12N + 36 \right]$$

WHERE: W = OVERALL GROSS WEIGHT ON ANY GROUP OF TWO OR MORE CONSECUTIVE AXLES TO THE NEAREST 500 POUNDS
 L = DISTANCE IN FEET BETWEEN THE EXTREME OF ANY GROUP OF TWO OR MORE CONSECUTIVE AXLES
 N = NUMBER OF AXLES IN THE GROUP UNDER CONSIDERATION, EXCEPT THAT TWO CONSECUTIVE SETS OF TANDEM AXLES MAY CARRY A GROSS LOAD OF 34,000 POUNDS EACH, PROVIDED THAT THE OVERALL DISTANCE BETWEEN THE FIRST AND LAST AXLES OF SUCH CONSECUTIVE SETS OF TANDEM AXLES IS 36 FEET OR MORE

Figure E-7.
80,000 lb GVW Limitations⁽¹⁴⁾

73,280 LB MAXIMUM GROSS
(73,000 LB IN CONNECTICUT)

LIMITS IN STATES STILL USING 73,280 GVW
(73,000 GVW IN CONNECTICUT)

STATE	DIM. A (MINIMUM)	DIM. B (MINIMUM)	DIM. C (MINIMUM)
ILL.	10'	36'	44'
D.C.	13'	35'	43'
IA.	15'	39'	47'
MD.	9'	30'	40'
MISS.	13'	36'	43'
MO.	14'	36'	43'
IND.	NONE	NONE	NONE
CONN.	NONE	NONE	NONE
PA.	NONE	NONE	NONE
TENN.	NONE	NONE	NONE
ARK.	NONE	NONE	NONE
NEB.	NONE	NONE	NONE

Figure E-8.
73,280 lb GVW Limitations⁽¹⁴⁾

E.3.2.2 Rail Car Size and Weight Limitations. The limits on size and weight for rail transportation are generally established by specifications from the AAR. Each railroad line has specific weight, height, and width limitations for every major route, and this information is published annually in the Railway Line Clearance Handbook. A review of the above reference indicates that most railroads limit unrestricted traffic to a minimum gross weight of rail car and load of 263,000 pounds.

Generalized boundaries for unrestricted interchange are:

● Maximum height above rails	15-1/2 ft
● Maximum width	10-2/3 ft
● Maximum weight per axle	65,750 lb
● Maximum total weight (4 axles)	263,000 lb
● Maximum weight per unit car length	6,000 lb/ft

The dimensional limitations for unlimited interchange are shown in Figures E-3 and E-4, which give the Association of American Railroads Plate B specifications.

E.3.2.3 Potential Problems Affecting Transport Regulations. The problems listed below are current concerns that may significantly affect transport regulations.

- The general reluctance of the railroad industry to carry nuclear materials
- Unnecessarily restrictive regulations imposed on nuclear waste shipments at state and local levels.
- Severely restrictive regulations on transportation resulting from the lack of public understanding of nuclear safety and accident probabilities and consequences

Because of their potential impact on the objectives of the Transport Logistics Study, the above concerns are identified as high-priority items in the National Waste Terminal Storage Program.

E.4 TRANSPORTATION LOGISTICS

A nuclear data base of 203 currently committed reactors is used to develop the logistics for transportation. The SFDS assumes that the distribution of reactors in the future will follow approximately the same pattern as the presently committed reactors in the United States. With this reactor distribution as background, resolution of transportation logistics considerations will properly locate the AFRs, P/E facilities, and repositories in order to minimize the total transportation cost.

The SFDS system considers: a group of reactors with a total spent fuel output of 5,000 MTHM/yr; 9 AFRs, each capable of receiving and shipping spent fuel at a rate of 555 MTHM/yr; 3 P/E facilities, each capable of about 1,500 MTHM/yr; and 1 repository which can receive wastes from processed and/or encapsulated spent fuel at a rate of 5,000 MTHM/yr. All of these individual installations are independently located.

E.4.1 Distance Between Facilities

Oak Ridge National Laboratory (ORNL) and Battelle Pacific Northwest Laboratory (BPNL) have developed two independent transportation logistics programs that can be used to minimize total transportation cost for the planning period, provided that the location of all the facilities involved is known. In this study, the United States is divided into nine regions, each containing 1 AFR and serving approximately equal numbers of reactors. A mean distance between reactor and AFR in each region is determined, and the average of these nine mean distances represents the average distance between the reactors and AFRs.

Similarly, the locations of the 3 P/E facilities and 1 repository are determined and the average distances from AFR to P/E facility, P/E facility to repository, and reactors to P/E facility are estimated. The average distances between the various facilities of interest of each case are summarized in Table E-7.

Table E-7
AVERAGE DISTANCES BETWEEN VARIOUS FACILITIES

Distance, mi	Ref. Case	Case 1	Case 2	Case 3
Reactor to an AFR	—	400	400	400
Reactor to a P/E facility	700	—	—	—
AFR to a P/E facility	—	700	700	700
AFR to the repository	—	1,100	1,100	1,100
AFR to shallow land burial	—	400	400	400
P/E facility to the repository	1,100	1,100	1,100	1,100
P/E facility to shallow land burial	700	700	700	700
P/E facility to the recycle facilities	400	—	—	—

E.4.2 Average Speed

The American Association of Railroads reports that the average daily rail car mileage for the United States is about 57 miles. At an average freight speed of 20 mph, including stops, it takes about 3 hours to cover 57 miles. Thus, the average effective speed of a rail car over an entire day is about 2.5 mph. Although some of the causes of this low average speed (such as terminal movements, classifying, and assembling of special train shipments) can be partially removed, the restrictions on speed, traffic priority, and rail car availability will probably more than offset any time saved in the above categories. However, since seasonal changes, loading time, and unloading time usually involved in computing the 2.5 mph figure are not factors in rail speed to move wastes, an average speed for rail car movements of 5 mph was used in the SFDS.

Experience with legal weight truck shipments of radioactive materials indicates that transit times are comparable to those of the trucking industry for all shipments. Average speeds of 30 to 40 mph are realized routinely. For trips of over 300 to 350 miles, it is necessary to provide two drivers to avoid stopoffs for driver sleep time. Table E-8 shows the approximate time it takes one or two drivers to drive a variety of distances. Apparently, longer distances dictate the use of two drivers.

Overweight truck shipments can also travel at average daytime speeds of approximately 35 mph, but they are limited by the conditions of the state permits, which may restrict or prohibit travel at night, on weekends, and on holidays. Accordingly, the transit times for overweight shipments will vary and are dependent upon the shipment conditions and routing.

On the basis of Table E-8, truck shipments are assumed to have an average speed of 33 mph.

Table E-8
TIME AND DISTANCE
ONE DRIVER VS. TWO

Mileage	Single Driver	Double Driver
100	3 hr	3 hr
200	6 hr	6 hr
300	9 hr	9 hr
400	20 hr	12 hr
500	23 hr	15 hr
600	1 day, 2 hr	18 hr
700	1 day, 13 hr	21 hr
800	1 day, 16 hr	24 hr
900	1 day, 19 hr	1 day, 3 hr
1,000	2 days, 6 hr	1 day, 6 hr
1,100	2 days, 9 hr	1 day, 9 hr
1,200	2 days, 12 hr	1 day, 12 hr
1,300	3 days	1 day, 15 hr
1,400	3 days, 2 hr	1 day, 18 hr
1,500	3 days, 5 hr	1 day, 21 hr
1,600	3 days, 8 hr	2 days
1,700	3 days, 19 hr	2 days, 3 hr
1,800	3 days, 22 hr	2 days, 6 hr
1,900	4 days, 1 hr	2 days, 9 hr
2,000	4 days, 12 hr	2 days, 12 hr
2,100	4 days, 15 hr	2 days, 15 hr
2,200	4 days, 18 hr	2 days, 18 hr

E.4.3 Shipments

A summary of annual material shipments from each facility (AFR, P/E facility, repository) is shown in Table E-9 for all cases.

Spent fuel from the reactors is shipped by truck and by rail. SFDS assumed that the number of truck shipments and the number of rail shipments are the same. This assumption results in about 90 percent (by weight) of the spent fuel being shipped by rail.

Table E-9 shows that spent fuel from AFRs and P/E facilities is shipped by rail only, since these facilities all have rail access. The use of rail minimizes the overall number of shipments. Since hulls, recovered Kr-Xe-CO₂, and HLW are contained in heavy shielded casks, these items are shipped only by rail. Uranium hexafluoride is routinely shipped by truck, and the small number of shipments shown presents no real transport problem. Plutonium oxide is assumed to be transported by truck. (A discussion of the transportation of plutonium oxide is given in Subsection E.5.2.) All TRU and non-TRU waste, LLW and ILW, is assumed to be transported by truck.

Table E-9
SUMMARY OF ANNUAL MATERIAL SHIPMENTS
FOR SFDS STUDY CASES

Discharged From	Material	Transport Mode	Cask Type and Number of Packages/Shipment	Annual Number of Shipments			
				Ref. Case	Case 1	Case 2	Case 3
Reactors	Spent fuel assemblies 1 year old 6,800 PWR, 10,200 BWR	Truck	NAC-1: 1 PWR or 2 BWR	1,240	1,240	1,240	1,240
		Rail	Conceptual: 8 PWR or 19 BWR	1,240	1,240	1,240	1,240
AFRs	Spent fuel assemblies 10 years old 6,800 PWR, 10,200 BWR	Rail	Conceptual: 14 PWR or 34 BWR	—	785	785	785
	Non-TRU, LLW 6,200 drums	Truck	Super Tiger: 42 drums	—	148	148	148
P/E facilities	Spent fuel canister 3,400 BWR and 6,800 PWR	Rail	Conceptual: 9 canisters	—	1,133	—	—
	Chopped spent fuel 8,070 canisters	Rail	Conceptual: 14 canisters	—	—	576	—
	Vitrified spent fuel 18,520 canisters	Rail	Conceptual: 9 canisters	—	—	—	2,057
	Vitrified HLW 2,380 canisters	Rail	Conceptual: 12 canisters	198	—	—	—
	Hulls and Kr-Xe-CO ₂ (Low-heat HLW)	Rail	Conceptual: 9 canisters	429	—	170	429
	Non-TRU, LLW	Truck	Super Tiger: 42 drums ^(a)	379	68	70	70
	TRU, ILW	Truck	Super Tiger: 42 drums ^(a)	40	13	20	29
	TRU and non-TRU, LLW	Truck	ATCOR AL-32-18D: 7 canisters	1,221	367	557	859
	PuO ₂ , 2,500 PPP-1M	Truck	PTV: PPP-1M	312	—	—	—
	UF ₆	Truck	2 cylinders	370	—	—	—
Total shipments			Truck	3,562	1,836	2,035	2,345
			Rail	1,867	3,158	2,771	4,511

^(a) Six drums are placed on a pallet, seven pallets per shipment

E.5 CASKS

Shipping casks ensure the safe transportation of highly radioactive materials during normal or accident conditions. Prior transport experience in the nuclear industry has been in the transportation of spent fuel and of low-activity and low-heat-generating radioactive materials such as fresh fuel, uranium, and LLW and ILW. The unique requirement for transporting radioactive materials in the SFDS is shipping the remaining fuel cycle materials (HLW, cladding hulls, and plutonium). Basically, the casks for HLW and hulls represent adaptations of currently available technology to meet the transportation requirements of a particular waste form. Special systems will have to be developed for plutonium oxide transport.

The following subsections describe available and conceptually designed casks, their current lease rates, turnaround times, and cask fleet requirements in the SFDS cases.

E.5.1 Available Casks

Existing casks are designed to transport spent light water reactor (LWR) fuel from current generation reactors after a minimum cooling time of 120 days. Casks are available for both truck and rail transport. Most casks can accept either PWR or BWR fuel by using different fuel baskets, although a few casks are designed only for a particular fuel type. A list of available casks is presented in Table E-10. Two of these casks, which represent the state of the art in cask design and construction, are described below. The IF-300 cask is a rail cask with a water-filled cavity and mechanical blowers for cooling. The NAC-1 cask is a truck cask with a water-filled cavity, and natural convection cooling.

The IF-300⁽⁷⁾ is a cylindrical cask designed to carry 7 PWR or 18 BWR fuel assemblies. Shielding is provided by water in the cask cavity, 10 centimeters of depleted uranium metal in the cask shell, and an exterior water-filled enclosure. Heat is transferred from the fuel to the cask cavity walls by the natural circulation of the contained water; from the cavity walls, the heat is conducted first through the uranium shielding layer, then through the water-filled outer jacket, and finally through the corrugated outer surface to the surrounding environment. The jacket is cooled by two diesel-driven blowers with a total air-moving capacity of 8.5 m³/sec, capable of removing 77 kilowatts of heat. This heat load corresponds to heat released from 7 PWR fuel elements having a burnup of 35,000 MWd/MTU at an average specific power of 40 MW/MTU after 120 days of cooling. The loaded cask weighs about 51 tonnes. The cask is normally transported by rail on a 100-tonne-capacity flatcar. It may, however, be transported for short distances on highways on a special nine-wheel truck. Such truck shipments require special overweight permits. This method has been used primarily to move the cask from reactors without rail facilities to the nearest rail siding.

The NAC-1⁽⁸⁾ is a cylindrical spent fuel cask used for legal weight truck shipments designed to hold 1 PWR or 2 BWR fuel assemblies. Shielding is provided by cask water in the cavity, 17 centimeters of lead (gamma shielding), and 11 centimeters of borated water solution (neutron shielding). Heat rejection is by convection through the water coolant in the cavity to the inner wall, conduction through the neutron shield, convection to the outer wall, and finally convection plus radiation to the atmosphere.

Other casks having the same capacity as the NAC-1 for legal weight truck shipments are the NL1-1/2 and the NFS-4.

Table E-10
CHARACTERISTICS OF LWR
SPENT FUEL CASKS

Cask Name	NFS-4	IF-200	IF-300	TN-8 TN-9	TN-12	NL-1/2	NL-10/24	NAC-1 ^(a)
PWR/BWR assemblies per cask	1/2	2/4	7/18	3/7	12/32	1/2	10/24	1/2
Loaded weight, MT	22.5	25.5	63.5	34.5	97	22	90	22.5
Cavity dimensions, dia. x length, in.	13.5x178	15x180	37.5x180	(9 in. 2x168) (5 in. 2x178)	48x180	13.375x178	43.5x180	13.5x178
Overall dimensions, dia. x length, in.	50x214	N/A	64x208	68x192 (68x202)	98x265	40x193	103x224	50x214
Gamma shield material	Lead	Uranium	Uranium	Lead	Steel	Lead	Lead	Lead
Neutron shield	Borated water	N/A	Water	Organic	Organic	Water	Water	Borated water
Cavity coolant	Water	N/A	Water	Air	Air	Helium	Helium	Water
Licensed design	Yes	Yes	Yes	Yes	Pending	Yes	Yes	Yes
Transport mode	LWT	OWT	OWT/rail	OWT/rail	Rail	LWT	Rail	LWT
No. of units (operational/under construction)	2/0	2/0	4/0	0/0 (0/1)	0/0	3/0	0/1	4/0
Supplier	Nuclear Fuel Services West Valley, N.Y.	General Electric Co. San Jose, CA		Transnuclear, Inc. White Plains, N.Y.		NL Industries, Inc. Nuclear Division Wilmington, Delaware		Nuclear Assurance Corp., Atlanta, Georgia

(a) Used as the basis for transportation cost calculation.

LWT—Legal weight truck

OWT—Overweight truck

Many forms of non-HLW have been shipped routinely in the United States. These shipments have been made primarily by truck, although rail transportation has also been used. This trend is expected to continue, with the choice of transport mode dependent on convenience and economic considerations. Available packaging is adequate for most anticipated forms of treated and untreated non-HLW waste. These casks are listed in Table E-11.

LLW includes both TRU and non-TRU wastes and is transported in a Super Tiger, which accommodates 42 drums (55-gallon capacity) or 33 drums (83-gallon capacity). ILW also includes TRU and non-TRU wastes and can also be transported in available shielded casks which accommodate 7 canisters.

Commercial packaging for uranium hexafluoride is currently available and is listed in Table E-12. For the purpose of cost estimates, the 48X "10-ton" container is being used.

Table E-11
TYPE B PACKAGING FOR POSSIBLE USE
IN THE TRANSPORTATION OF NON-HLW ⁽¹⁷⁾

Container Name	Cavity Dimensions Volume	Shielding	Heat Removal Capacity W	Approximate Weight MT	Usual Mode of Transport
ATMX-600 rail car ⁽¹⁰⁾	106"x106"x49'-11" (3,895 ft ³)	None	500	49.0(net)	Rail
(a)Super Tiger ⁽¹⁰⁾	75"x75"x14'-4" (560 ft ³)	None	235	12.6 (net)	Truck rail
ORNL Model III ⁽¹¹⁾	29"(D)x78"(H) (30 ft ³)	9" steel	440	19.1	Rail
SRP ⁶⁰ Co	15"x23"x3'-1" (7.4 ft ³)	8.5" lead	4,200	14.3	Truck
BMI-1 ⁽¹¹⁾	15"(D)x54"(H) (5.5 ft ³)	8" lead	1,500	10.7	Truck
Paducah demonstration cask	17"(D)x51S(H) (6.7 ft ³)	6" uranium	10,500	13.6	Truck
NCL 6502	12"x12"x10'-10"	10" lead	7,300	Unknown	Truck
GE IF200B140 ⁽¹¹⁾	15"(D)x140"(H) (14.3 ft ³)	8.3" lead	5,800	25.4	Truck
GE IF100X137 ⁽¹¹⁾	13"(D)x137"(H) (10.5 ft ³)	8.3" lead	5,800	20.4	Truck
GE IF100B129 ⁽¹¹⁾	13" (D)x129"(H) (9.9 ft ³)	8.3" lead	13,000	20.0	Truck
NECO B3-1	26"(D)x42"(H) (12.9 ft ³)	6" lead	Unknown	9.3	Truck
(a)ATCOR AL-32-180	81.5"x40"x12'-4" (278 ft ³)	1.5" lead	Unknown	14	Truck
ATCOR LL-50-100	63"(D)x75"(H) (135 ft ³)	4.5" lead	Unknown	24.0	Truck
ATCOR LL-56-65	36"(D)x114"(H) (67 ft ³)	7" lead	Unknown	25.8	Truck
ATCOR BC-48-220	73"(D)x88"(H) (213 ft ³)	7" concrete 25" steel	Unknown	21.8	Truck
Brookhaven waste shipment overpack	23"x63"x1'-2" (11.8 ft ³)	17" concrete	Unknown	13.6	Truck rail
(two sizes)	43"x24"x5'-11"	12" concrete	Unknown	13.6	Truck rail

(a)Used as basis for transportation cost calculation.

Table E-12
DATA FOR URANIUM
HEXAFLUORIDE PACKAGES⁽¹²⁾

Cylinder Model No.	Nominal Diameter in.	Material of Construction ^(a)	Minimum Volume ft ³	Approx. Tare Weight (without valve protector) lb	Maximum Enrichment wt % ²³⁵ U	Fill Limit	
						Maximum, lb UF ₆ (b)	Minimum, lb UF ₆
1S	1.5	Nickel	0.0053	1.75	100.00	1.0	0.01
2S	3.5	Nickel	0.0254	4.2	100.00	4.9	1
5A	5	Monel	0.284	55	100.00	55	11
8A	8	Monel	1.319	120	12.5	255	55
12A	12	Nickel	2.38	185	5.0	460	55
12B	12	Monel	2.38	185	5.0	460	55
30B	30	Steel	26.0	1,400	5.0 ^(c)	5,020	2,300 ^(d)
48X ^(e)	48	Steel	108.9	4,500	4.5 ^(c)	21,030	14,000
48Y	48	Steel	142.7	5,200	4.5 ^(c)	27,560	21,000

(a) For packaging normal depleted UF₆, cylinders of other materials, such as steel and Monel, may be substituted, provided that they have equivalent strength.

(b) Fill limits are based on 250°F maximum UF₆ temperature (203.3 lb UF₆/ft³), certified minimum internal volumes for all cylinders, and a minimum safety factor of 5%. The operating limits apply to UF₆ with a minimum purity of 99.5%. More restrictive measures are required if additional impurities are present. The maximum UF₆ temperature must not be exceeded.

(c) Maximum enrichments indicated require moderation control equivalent to a UF₆ purity of 99.5%. Without moderation control, the maximum permissible enrichment is 1.0 wt % ²³⁵U.

(d) Shipments from the Atomic Energy Commission and at the customer's request can be less than 2,300 lb in the 30B cylinders, providing the UF₆ is transferred from a larger cylinder.

(e) Used as basis for transportation cost calculation.

E.5.2 Conceptual Casks

To efficiently transport spent LWR fuel and a variety of radioactive materials, work is being done⁽⁹⁾ to develop the conceptual casks to serve complete fuel cycles like those expected in the SFDS. The shielding thickness and heat removal requirements for casks depend upon the type and characteristics of material to be transported. Characteristics such as fissile material content, activity, heat generation, and size of the canister are of concern. Another important factor to be taken into consideration is the cask/car packaging system weight limitation. Casks are optimized with respect to weight and payload to allow them to be used as unrestricted legal weight casks with regard to either railroad or road transport.

Longer term development in future cask design may include the following:

- Development of casks designed with optimum shielding thickness and cavity dimensions to give the best payload-to-cask weight ratios
- Development of casks with sufficient heat removal capacity for higher burnup
- Development of casks specially for intermodal transportation systems using water transport
- Improvements in safety and handling features
- Improvements in neutron shielding to provide adequate shielding at a lower weight

Conceptual casks for the various forms of radioactive materials studied in the SFDS are listed in Table E-13. They were developed by Oak Ridge National Laboratory using the PACRAT program⁽⁹⁾ and input data by Bechtel.

Table E-13
CONCEPTUAL CASKS

Cask	Spent Fuel PWR/BWR	Spent Fuel PWR/BWR	Spent Fuel PWR/BWR	Encapsulated Spent Fuel PWR/BWR	Chopped Spent Fuel	Vitrified Spent Fuel	Hulls	Vitrified HLW
Spent fuel age, yr	1	5	10	10	10	10	1	10
No of assemblies or canisters	8/19	11/27	14/34	9/9	14	9	9	12
Assembly or canister size, in.	8.4x8.4/ 5.5x5.5	8.4x8.4/ 5.5x5.5	8.4x8.4/ 5.5x5.5	14 (dia.)/ 16 (dia.)	14 (dia.) (a)	20 (dia.)	20 (dia.)	14 (dia.) (a)
Gamma shield	Uranium	Lead	Lead	Lead	Lead	Lead	Lead	Lead
Cavity diameter, in.	40	46	49	60	63	75	75	59
Overall diameter, in. x length, in.	61x219	69x221	71x219	75x216	104x166	92x143	88x140	99x166
Loaded weight, short tons	83	84	80	85	76	78	84	79
Transport	Rail	Rail	Rail	Rail	Rail	Rail	Rail	Rail

(a) Assumed to be overpacked 12-inch can.

A conceptual design has been carried out by AGNS-Sandia⁽¹³⁾ of a car/cask packaging system for safe and secure plutonium oxide transportation. The quantities of plutonium oxide in the packaging system are shown in Table E-14. This packaging system is used as the basis for transportation cost calculation. Figure E-9 shows four 8-kilogram canisters of plutonium product packaged in the modified overpack (PPP-1M); Figure E-10 shows one of the eight PPP-1Ms being removed from transport containment.

Table E-14
PLUTONIUM PACKAGE DESCRIPTION

Item	Quantity of PuO ₂ , kg	No of PuO ₂ Canisters	Remarks
Single PuO ₂ canister	8	1	Storage canister only, to be shipped in PPP-1M container
Plutonium product package (PPP-1M)	32	4	Self-contained shipping package
Plutonium transport vehicle (PTV)	256	32	Carries 8 PPP-1M packages

E.5.3 Turnaround Times and Lease Rates

The turnaround time of the cask at a facility is defined as the average time the cask remains at a facility for the purpose of loading/unloading and is the sum of the time for some or all of the following:

- Cask receipt
- Cask washdown
- Cask unloading from vehicles
- Cask cooldown
- Cask head removal
- Waste/fuel loading or unloading
- Cask head closure
- Cask exterior decontamination and maintenance
- Cask loading on vehicles
- Cask shipment

Figure E-9.
Overpack and Canisters for the Plutonium Oxide Product⁽¹³⁾

Figure E-10.
Handling of the Plutonium Oxide Overpack⁽¹³⁾

Turnaround times affect facility costs as well as transport costs. The four major contributing factors to cask turnaround are:

- General arrangement of the handling equipment in the facility and facility layout
- Experience and familiarity of the operating crew with the necessary operations and casks
- Reliability of the cask handling equipment
- Contamination of the cask exterior during unloading operations

Proper design in facility layout will provide smooth, continuous flow. Careful cask movement analysis for normal operations will eliminate excessive cask movements or retracing of cask movements. Operation experience and familiarity with the equipment and casks can be improved through operator specialization and minimizing the number of different kinds of casks to be handled if the latter can be controlled.

The importance of equipment reliability to the overall cask handling operation is obvious. Proper selection of cranes, lifts, power moving equipment, etc. is important.

Dry unloading may reduce the likelihood of cask contamination; however, wet unloading is currently the preferred approach. The ability to maintain low activity in the pool is an important factor: the lower the contamination level in the unloading pool water, the shorter the decontamination time. The use of a protective sleeve surrounding the cask will minimize contamination of irregular surfaces (fins, etc.) and other hard-to-reach surfaces in quick turnaround of casks.

Estimated turnaround times for both legal weight truck and rail shipping casks at the various facilities are shown in Table E-15.

The lease rate for all spent fuel rail casks is estimated to be \$3,600/day and for all road casks — \$900/day in January 1979 dollars. This is consistent with current commercial rates from operating companies. This lease rate includes such items as maintenance, insurance, expendables, and other related services.

Table E-15
ESTIMATED CASK TURNAROUND TIMES

Item	Cask Turnaround Time, hr	
	Road	Rail
Reactor	24	48
AFR	24	48
P/E facility	24	48
Repository	24	48

E.5.4 Cask Fleet Requirements

The number of casks required for the SFDS steady state transportation system is high in comparison with the number of currently available casks. Cask fleet requirements are shown in Table E-16. The estimate is based on 300 days/yr cask usage, the annual number of shipments shown in Table E-9, and the elapsed time required for the average round trip. The average round trip times are the sum of turnaround time and transportation time calculated from the average distances between facilities and the average speed. To meet these requirements, a sufficient lead time will be needed to get the desired cask fleet operational. Estimated lead times are 6 to 12 years,⁽¹⁴⁾ depending on the radioactive materials to be transported.

Table E-16
CASK FLEET REQUIREMENTS

From	Materials	Ref. Case Road/Rail		Case 1 Road/Rail		Case 2 Road/Rail		Case 3 Road/Rail	
Reactors	Spent fuel	17	66	13	46	13	46	13	46
AFRs	Spent fuel	—		0	42	0	42	0	42
	Non-TRU LLW	—		2	0	2	0	2	0
P/E facilities	Spent fuel	0	15	0	83	0	43	0	151
	UF ₆	8	0	—	—	—	—	—	—
	PuO ₂	4	0	—	—	—	—	—	—
	Hulls	0	32	0	0	0	13	0	32
	TRU ILW	21	0	6	0	10	0	15	0
	TRU LLW	1	0	1	0	1	0	1	0
	Non-TRU LLW	5	0	1	0	1	0	1	0
Total casks required		56	113	24	171	27	144	32	271

E.6 TRANSPORTATION COST ESTIMATES

Transportation costs are based on current commercial transport systems and tariffs. No attempt has been made to predict changes due to regulatory uncertainties. Transportation is assumed to be privately owned and operated, reflecting a system that meets the needs of all the SFDS cases.

E.6.1 Cost Summary

The transportation cost estimates for the various cases comprise the following transfer steps:

- Spent fuel from the reactors to the AFR
- Spent fuel from the reactors to the P/E facility
- Spent fuel from the AFR to the P/E facility
- Processed and/or encapsulated spent fuel from the P/E facility to the repository
- Uranium (UF_6) from the P/E facility to recycle
- Plutonium (PuO_2) from the P/E facility to recycle
- Cladding hulls from the P/E facility to the repository
- HLW from the P/E facility to the repository
- Non-TRU LLW from the AFR to shallow land burial
- Non-TRU LLW from the P/E facility to shallow land burial
- Non-TRU ILW from the P/E facility to the repository
- TRU LLW waste from the P/E facility to the repository
- TRU ILW waste from the P/E facility to the repository

The quantity of radioactive material to be transported annually is shown in Table E-1. The information in this table is translated into shipments by road and rail and presented in Table E-9. Average distances between facilities for transportation purposes are shown in Table E-7. Subsection E.5 outlines the transport packages to be utilized for the various radioactive materials.

In this study, truck freight rates are based on Tri-State Motor Company's rate.⁽¹⁵⁾ Railroad freight rates are based on the information provided in Items 80762-A and 89764-A of Supplement 3 to Uniform Freight Classification 12, ICC 8. Freight charges for loaded and empty rail casks in terms of \$/cwt versus distance, are shown in Figure E-11.

The overall transportation cost summary is given in Table E-17.

Table E-17
TRANSPORTATION COST SUMMARY
(in \$1,000s)

From	To	Materials	Ref. Case	Case 1	Case 2	Case 3
Reactors	P/E facilities	Spent fuel	92,536	—	—	—
	AFRs		—	64,680	64,680	64,680
AFRs	P/E facilities	Spent fuel	—	54,388	54,388	54,388
	Shallow land burial	Non-TRU(LLW)	—	244	244	244
P/E facilities	Repository	Spent fuel	—	107,832	53,597	192,171
		HLW	18,650	—	—	—
		Cladding hulls and misc.	40,469	—	16,036	40,469
		TRU(LLW)	135	44	68	98
		TRU and non-TRU(ILW)	4,132	1,242	1,185	2,903
	Shallow land burial	Non-TRU(LLW)	894	160	165	165
	Fuel recycle	UF ₆	278	—	—	—
	Fuel recycle	PuO ₂	2,082	—	—	—
Annual cost			159,176	228,590	191,063	355,118
Annual cost/MTHM			31.8	45.7	38.2	71.0

E.6.2 Transportation Cost Sensitivity

The transportation cost (the sum of freight and lease costs) is presented in Table E-17. The cost is based on specific assumptions regarding the following:

- Distance between various facilities
- Cask lease rate
- Average transportation speed

An analysis of the data presented in Table E-17 indicates that the cost of transporting spent fuel in various forms represents 99.3 percent, 90.4 percent, and 87.6 percent of the total transportation cost for Cases 1, 2, and 3, respectively. Therefore the sensitivity study is centered on the transportation of all forms of spent fuel in Case 1. The various transport steps considered are 1-year-old spent fuel from the reactors, 10-year-old spent fuel from an AFR, and encapsulated 10-year-old spent fuel from a P/E facility. The parameters to be varied are the distances between reactors and AFR, AFR and P/E facility, and P/E facility and repository; spent fuel rail cask lease rates; and average overall rail speeds.

Figure E-11.
Rail Freight Charges for Spent Fuel

E.6.2.1 Rail Transport Cost as a Function of Distance. The distances between facilities used to obtain the results in Table E-17 are shown in Table E-7.

These are:

- Reactor to AFR — 400 miles
- AFR to P/E facility — 700 miles
- P/E facility to repository — 1,100 miles

Figure E-12 shows rail transport cost for the various spent fuel forms with distances between facilities varied from 300 to 1,500 miles. The intercepts identified in Figure E-12 are the transportation costs as shown in Table E-17.

E.6.2.2 Rail Transport Cost as a Function of Rail Speed. The effect of varying average rail speed on rail transport cost for various spent fuel forms is presented in Figure E-13. The average speed was varied between 2 and 20 mph. The distances between reactor and AFR, AFR and P/E facility, and P/E facility and repository are 400, 700, and 1,100 one-way miles, respectively.

The overall average rail speed will probably vary between 3.5 and 7 mph, depending on geographical location. For short-haul situations of less than 500 miles, costs are high and truck transport is more economical.

E.6.2.3 Transportation Cost (by Rail) as a Function of Cask Lease Rate. Rail cask lease rates are varied between \$2,000/day and \$5,000/day using an average rail speed of 5 mph. The resultant costs for various spent fuel forms transported 400, 700, and 1,100 miles from reactor to AFR, AFR to P/E facility, and P/E facility to repository, respectively, are shown in Figure E-14.

The rail cask lease rate used in this study is based on information from current cask manufacturers. How these manufacturers established their rates is not clear, and a projection of these rates to future times is difficult. Transportation costs based on any lease rate between \$2,000 and \$5,000 can be obtained from Figure E-14.

E.6.3 Rail Cask Fleet as a Function of Distance

Figure E-15 shows the size of the cask fleet required, based on a 5 mph average rail speed and 300 day/yr operating time for transporting spent fuel from distances ranging from 400 to 1,100 miles. Other parameters are held constant as in Case 1.

Figure E-12.
Rail Cost Versus Distance for Spent Fuel (Case 1)
 (Case 1 cost data from Table E-17 are shown here for reference.)

Figure E-13.
Rail Transport Cost Versus Average Rail Speed (Case 1)
 (Case 1 cost data from Table E-17 are shown here for reference.)

Figure E-14.
Rail Transport Cost Versus Cask Lease Rate (Case 1)
 (Case 1 cost data from Table E-17 are shown here for reference.)

Figure E-15.
Rail Cask Fleet Versus Distance (Case 1)
(Case 1 cask fleet size from Table E-16 is shown here for reference.)

REFERENCES FOR APPENDIX E

- (1) "Spent Unreprocessed Fuel Facility," Rockwell International Atomic International Division, RHO-LD-2, February 1978.
- (2) "Capabilities of U.S. Domestic Transportation Systems for the Shipment of Radioactive Wastes," Nuclear Assurance Corporation with Union Carbide/OWI and USERDA, Y/OWI/SUB-77/2233, September 1977.
- (3) *Railroad Lines Clearance Handbook*, New York, N.Y., National Railway Publication Company, September 1977.
- (4) Association of American Railroads, "Specifications for Design, Fabrication, and Construction of Railroad Freight Cars," 1964.
- (5) *Nucleonics Week*, McGraw Hill, Inc., November 3, 1977.
- (6) S. Heaberlin, et al., "Evaluation of the Consequences of LWR Spent Fuel and Plutonium Loss at Sea," BPNL, August 1976.
- (7) "Preliminary Conceptual Design Description for National Waste Repository No. 1," Stearns-Roger Engineering Company, EY-77-C-05-5367, November 1977.
- (8) "Conceptual Design Criteria for Facilities for Geologic Disposal of Radioactive Wastes in Salt Formations," Union Carbide Corporation Nuclear Division, Y/OWI/TM-9, January 1977
- (9) "PACRAT — A Computer Program for Estimating Packaging and Cost of Radwaste Transport: Program Documentation and User's Manual," Union Carbide Corporation, Oak Ridge National Laboratory, Second Draft, June 1978.
- (10) *Directory of Packagings for Transportation of Radioactive Materials*, WASH-1279, USAEC Division of Waste Management and Transportation, October 1973.
- (11) *Directory of Shipping Containers for Radioactive Materials*, WASH-1145, USAEC Division of Construction, Washington, DC, October 1969.
- (12) D.A. Edling and J.F. Griffin, "Certification of ERDA Contractors' Packaging with Respect to Compliance with DOT Specification 7A Performance Requirements," Phase II, Summary Report, MLM-2228, June 1975.
- (13) "Plutonium Transportation Vehicle Concepts Descriptions and Analysis," Allied-General Nuclear Services, NMS-15, December 1977.
- (14) "Current Status and Future Consideration for a Transportation System for Spent Fuel and Radioactive Waste," Allied-General Nuclear Services, Y/OWI/SUB-77/42513, February 1978.
- (15) "Specific Truckload Commodity Rates," and "Mileage Truckload Commodity Rates," Tri-State Motor Transit Company, June 1978.
- (16) "Transportation Technology Legislative Data Base," Transportation Technology Pilot Plant Section, Chemical Technology Division, ORNL, March 1979.
- (17) "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," Vol. 3, ERDA-76-43, May 1976.

Appendix F

TECHNICAL CONSIDERATIONS FOR SAFEGUARDS DESIGN

TABLE OF CONTENTS

	Page
F.1 Safeguards Systems: General Considerations	F-1
F.1.1 Physical Protection System	F-2
F.1.2 Material Control and Accounting System	F-9
F.2 Safeguards Application to Fuel Cycle Facilities and Systems	F-13
F.2.1 Safeguards for a Transportation System	F-13
F.2.2 Safeguards for a P/E Facility	F-15
F.2.3 Safeguards for a Repository	F-19
F.2.4 Safeguards for an AFR	F-21
F.3 Sabotage and Theft	F-23
F.3.1 Adversaries	F-23
F.3.2 Threat	F-23
F.3.3 Fabrication of a Nuclear Explosive Device	F-24
F.3.4 Dispersal of Radioactivity	F-27
References for Appendix F	F-29

.

r

TECHNICAL CONSIDERATIONS FOR SAFEGUARDS DESIGN

F.1 SAFEGUARDS SYSTEMS: GENERAL CONSIDERATIONS

An effective safeguards system performs three basic functions:⁽¹⁾

- Detection of unauthorized activities and material balance discrepancies
- Delay of unauthorized activities until an appropriate response can be made
- Response to unauthorized activities and discrepancies in an adequate and timely manner

These three functions are provided by two major systems: a physical protection system and a materials control and accounting system. These systems are established and implemented in order to meet the following objectives:

- (1) To ensure that only authorized personnel and materials are admitted into material access areas and vital areas
- (2) To ensure that only authorized activities and conditions occur within protected areas, material access areas, and vital areas
- (3) To ensure that only authorized movement and placement of special nuclear material (SNM) occurs within material access areas
- (4) To ensure that only authorized and confirmed forms and amounts of SNM are removed from material access areas
- (5) To make sure that any unauthorized entry into the protected area of the facility is promptly detected
- (6) To ensure that the response to each unauthorized activity is timely and appropriate
- (7) To determine the quantity and to monitor the presence of all SNM in the plant

Objective (1) requires designs that provide barriers around material access areas, establish access authorization procedures and controls for each area, and specify devices and procedures to detect, assess, and communicate in a timely fashion any unauthorized access or penetration.

Objective (2) requires designs that establish specified authorized activities and conditions appropriate for each area of a facility and implement detection and surveillance measures for discovering and assessing in a timely fashion unauthorized activities and conditions in each such area.

Objective (3) requires designs that establish definite locations for such nuclear materials and procedures for their movement, set forth measures that maintain current knowledge of material location and movement, and establish measures that make possible the timely detection of unauthorized material placement or movement.

Objective (4) calls for designs that provide barriers that channel persons and materials, leaving a material access area to exit control points; specify controls and procedures that identify the properties and quantities of material being removed and the persons making the removal; and furnish detection and communication subsystems and procedures that will alert the security organization to any unauthorized attempt to remove material.

Objective (5) involves the same general design approach as the one used to control access into material access areas and vital areas, namely, barriers around the area, access controls, and devices to detect, assess, and communicate any unauthorized access in a timely fashion.

Objective (6) requires trained and qualified security personnel, an established plan for responding to emergencies and safeguards contingencies at each facility, appropriate equipment for the security organization, design features that facilitate rapid assessment of and reliable response to safeguards contingencies, and communications networks for rapid and accurate transmission of security information between onsite security personnel as well as to offsite assistance forces.

Objective (7) calls for monitoring processes and operations involving SNM to detect missing quantities of material; maintaining records of all SNM within the plant by location and quantity; measuring SNM as it is received into, moved within, and removed from the plant; and periodic inventories and audits to verify the location and quantities of SNM and the accuracy of the records.

These safeguards objectives must be coordinated with normal facility operational systems to obtain a safeguards design that is effective and has minimal cost and operational impact. The physical protection, materials measurement and accounting, and facility operations functions are coordinated using the authorization, information, and control channels shown in Figure F-1.⁽¹⁾ Safeguards responsibility is assigned to safeguards coordination just as operational responsibility is delegated to facility operations coordination. Safeguards coordination supervises access control, zone operations control, and materials control and accounting. It also coordinates safeguards information flow between these elements, management, and facility operations. The three primary functions of safeguards coordination are data collection and processing, assessment of safeguards alarms, and determination and initiation of appropriate response. Wherever appropriate, safeguards coordination relies on automatic decision and control; however, human assessment, decision, and response initiation are used when necessary.

Although primary responsibility for safeguards assessment lies with safeguards coordination, direct control from lower levels can be used to reduce response times. Direct control of area access and item handling is assigned to access control and zone operations control elements. Control of area access and of item handling involves a hierarchy of responses depending upon the discrepancy detected between the action authorized and the action performed. For minor discrepancies, control can be automated at the entrance to the operations control area. Serious discrepancies, calling for security force responses or facility shutdown, may require coordination between safeguards coordination, facility operations coordination, and facility management.

F.1.1 Physical Protection System

The discussion below of the physical protection system deals with the following three topics:

- Current NRC requirements
- Elements of physical protection systems
- Design approach

F.1.1.1 Current NRC Requirements. To ensure that licensees attain the desired level of safeguards, the NRC requires that the licensee assemble a configuration of technical hardware, procedures, and personnel that clearly meet specified requirements.⁽²⁾ Title 10 of the U.S. Code of Federal Regulations, Parts 70 and 73, contains the specific requirements for physical protection and accountability of SNM.

Implementation of these regulations by licensees is aided by a set of NRC Regulatory Guides that identify specific technical approaches that the NRC finds acceptable. These approaches serve as examples of acceptable methods of compliance. In addition, specific license conditions are imposed to ensure that site-specific aspects of safeguards are adequately addressed.

Each fuel cycle facility licensee authorized under NRC regulations to possess or use at any site certain specified quantities of enriched (in ²³⁵U) uranium, ²³³U, or plutonium (see Appendix E,

Figure F-1.
Safeguards Systems Overview

Subsection E.3) must comply with established NRC physical protection requirements. A physical protection plan must be submitted to the NRC for approval, and must demonstrate how the licensee will satisfy the regulatory requirements.

The licensee must maintain a physical security organization, including armed guards, to protect against industrial sabotage and theft. All guards or watchmen must be properly trained, equipped, and qualified, and must be requalified at least annually.

All vital equipment (defined as any equipment, system, device, or material whose failure, destruction, or release could directly or indirectly endanger public health and safety) must be located within a separate structure or barrier designated as a vital area. All vital areas and material access areas must be located within a larger protected area surrounded by a physical barrier. An isolation zone is required around the outer physical barrier. This zone must be kept clear of obstructions, must be illuminated and monitored to detect the presence of individuals or vehicles attempting to gain entry to the protected area, and must allow the armed members of the facility security organization to respond to suspicious activity or to the breaching of any physical barrier. SNM not in process must be stored in a vault or in a vault-type room equipped with an intrusion alarm.

Personnel and vehicle access into a protected area, material access area, or vital area must be controlled. A picture badge identification system must be used, visitors must be registered and escorted, and individuals and packages entering the protected area must be searched. Access to a vital area or material access area must be limited to those persons who require such access to perform their duties. To assure that special nuclear material is not being diverted, individuals within a material access area must be monitored continuously. All individuals, packages, or vehicles must be searched for concealed nuclear material before they may leave a material access area. Keys, locks, combinations, and related equipment must be controlled so they will not be compromised.

Intrusion alarms must be in operation on all emergency exits in the protected area, vital areas, and material access areas. Each unoccupied material access area must be locked and alarmed, and all alarms must annunciate in a continuously manned central alarm station located within the protected area and in at least one other manned station.

Each guard or watchman on duty must be capable of maintaining continuous communications with an individual in a continuously manned central alarm station within the protected area, who must be capable of calling for assistance from other guards and from local law enforcement agencies. To provide the capability of continuous communication with local law enforcement agencies, two-way radio voice communications must be available in addition to conventional telephone service.

Licensees must establish liaison with local law enforcement agencies and must be able to neutralize threats to the facility by taking appropriate direct action or by calling for assistance from local law enforcement agencies.

Security records must be maintained for all individuals authorized to have access to vital and material access areas, including visitors, vendors, and others not employed by the licensee. Routine security tours and all of the tests, inspections, and maintenance on security-related equipment and structures must be documented. A record that includes details of the response by facility guards must be maintained on each alarm, false alarm, alarm check, intrusion indication, or other security incident.

Immediate reports to the NRC must be made regarding all suspected thefts, unlawful diversions, and/or industrial sabotage. A detailed written report must follow within 15 days.

F.1.1.2 Elements of Physical Protection Systems. The elements of a physical protection system at fixed sites include physical barriers, intrusion alarm systems, access controls, contain-

ment mechanisms, internal surveillance systems, active delay devices, security forces, communication systems, and response forces.⁽²⁾

Physical barriers and alarm systems. The initial level of protection against overt sabotage and theft can be provided by multiple barriers around protected areas, devices at and between barriers to detect actual or attempted intrusion, and an alarm system to alert the guard force to the location and severity of an intrusion.

Automatic intrusion alarm systems, combined with remote assessment capability, can effectively monitor external areas under most environmental conditions. Redundant detection and alarm systems, functioning in conjunction with roving guards, improve effectiveness. A number of sophisticated intrusion alarm systems are available on an off-the-shelf basis.

Fixed barriers of steel and concrete can delay unauthorized intrusion into material access and vital areas through the exterior walls of buildings. Designs are also available for barriers at emergency portals. Various types of barrier systems are already widely used and have demonstrated a generally benign impact on the environment.

Access controls and admittance systems. Elimination of the need for access to any sensitive materials or records is the ideal access control. Where this is not feasible, a system of limited access based on need must be established.

Control of access to computer programs and computer files is not unique to the commercial nuclear industry, and systems have been developed to prevent the unauthorized use of data files and alteration of computer programs. Automatic admittance systems and card keys may be employed in conjunction with a system of passwords to identify and control access to sensitive data stored in computer systems.

Another technique for access control is the commonly used "buddy" system of surveillance, whereby two or more persons are present whenever access to SNM or vital equipment is granted. Surveillance of employees working with SNM can also be maintained by indirect means, such as closed-circuit television.

Vehicle access can be limited to essential traffic. If there is screening to detect obvious contraband and if continuous escort and surveillance are provided, time-limited access into a protected area can be granted to visitors and service personnel. Longer access to a specific area can be granted after a detailed search. Procedures for screening cargo entry and departure are significant, since attempts at material diversion may involve cargo-carrying vehicles.

Admittance systems work in concert with barrier and surveillance systems to limit facility access to authorized personnel only. A variety of such systems is available, and all require that identity be sufficiently established for an admittance decision to be made by site security personnel. The simplest and most common method is to display a picture identification badge to a guard. Variations of this approach include the use of remote viewing equipment to protect the guard, the use of stored images for comparison, and an exchange badge system that requires a special badge issued and worn only in a secure area and never removed from that area.

There are a number of automatic systems through which admittance is granted, based upon some unique aspect of personal identity, such as fingerprints, signature dynamics, hand measurements, or voice identification. With these systems, the basis for the admittance decision is automatic, and the potential for guard force collusion is substantially reduced.

Machine-readable cards, commonly called card keys, can be useful as an admittance system and can serve other purposes, such as time and attendance logging. However, card keys do not positively identify the bearer. A system that utilizes both card keys and ID picture badges provides a high level of effectiveness. Entry to secure areas is gained by means of a card key, which reduces the possibility of guard collusion, and the picture badge is then observed by a guard, who

verifies the authenticity of the badge and the identify of its bearer. Such combined ID photograph/card key admission systems are employed extensively in industrial and commercial establishments.

Containment mechanisms. Implicitly included in the containment control category are all the features of the industrial process that make the removal of material from normal process streams difficult. The primary function of these features is health and safety; a secondary function is safeguarding. Other features may be specifically designed for safeguards purposes, including vaults and vault-type rooms and methods for the surveillance of people working with the material. The removal of SNM by routine disposal paths is a special concern. Explicit containment controls can be provided for area-level entry and departure points, emergency exits, and the site perimeter entry and exit paths.

Relatively unsophisticated containment techniques include visual inspection of people's packages, combined with hands-on search, and the use of change rooms and clothing exchanges, possibly combined with surveillance augmented by SNM detection procedures. More sophisticated detection techniques are based on the use of X-rays or gamma or neutron detection.

Detection systems. Sensors capable of detecting concealed contraband items protect against both the introduction of weapons, explosives, or counterfeit SNM into a nuclear facility and the removal of concealed SNM from the facility.

Significant quantities of proscribed material can be detected by a physical (hands-on) search conducted by security personnel. But since such a search takes time, costs money, and tends to offend people, a variety of equipment has been developed that allows these searches to be conducted in a more expeditious, cost-effective, and socially acceptable manner.

Commercially available metal detectors are effective in detecting weapons. They also detect some forms of SNM shielding and, under certain circumstances, detect the metallic components of explosive devices.

X-ray inspection systems can be highly effective in detecting contraband concealed in packages or SNM concealed on or in the body. However, many factors discourage their use for personnel searches. X-ray package inspection equipment, which produces no personal radiation hazards, appears attractive.

Most devices for detecting explosives have significant limitations. Even the most promising, the trace vapor analysis system, cannot detect all types of explosives with good sensitivity. Dogs are very useful in detecting concealed explosives and can also perform other security functions; but they are expensive to maintain, may be objectionable to the work force, and, because of limited attention span, do not perform well in portal screening.

Currently available gamma detectors can detect unshielded SNM, but their effectiveness is reduced if the material is shielded. The use of neutron detectors in combination with gamma detectors eliminates this problem, since shielding needed to prevent both neutron and gamma detection is bulky and noticeable.

Detection system performance is highly procedure-dependent. For example, the use of change rooms and clothing exchange allows detection equipment to operate in a controlled environment free of substances that could interfere with its sensitivity. Thus, if company-provided coveralls are devoid of metal, the sensitivity of metal detectors is sufficient to detect very small amounts of metal being brought in.

Detection system performance can be enhanced by random selection of personnel for hands-on search or by operating detection equipment at its maximum sensitivity. Random selection adds an element of uncertainty that can deter attempts to defeat other detection methods. Operating at maximum sensitivity (with a resultant increase in false alarms) forces an adversary who is trying

to steal SNM to make more frequent sorties through the system with smaller amounts, thus increasing the likelihood of apprehension.

The most effective procedure for screening vehicles is to prohibit offsite vehicles from entering a protected area. If a vehicle must enter, the people associated with it can be required to pass through the site detection systems while their vehicle is being carefully screened for contraband. Search effectiveness can be enhanced by the use of hand-held detectors, effective in locating explosives and SNM, and by the use of dogs. Escorting the vehicle while it is within the protected area is a feasible added precaution.

Active delaying systems. Nonlethal active delaying materials and devices can substantially extend the time needed by an adversary to reach material access areas or vital areas. An unprepared adversary can be impeded by a markedly unfavorable condition, such as dense smoke, which reduces visibility; unbearably loud noises, which reduce or eliminate communications between adversaries; disorienting lights, which reduce efficiency; heat or skin irritants, which create discomfort and reduce effectiveness; and foams, extremely sticky substances, or slippery substances, which limit movement.

Additional active delaying materials are nonlethal gases or chemical agents, which tend either to immobilize the adversary or to reduce his effectiveness for a period of time. The effect of such materials is temporary, and they should be considered for use only in certain normally unoccupied places, such as vault storage areas, where SNM may be stored.

Active delay systems are at present rarely employed by industry, and very little information is available on their actual effectiveness, cost, or social acceptability.

Security forces. The element of a fixed-site safeguards program that is most effective in dealing with a malevolent act, whether by outsiders or insiders, is the guard force. The functions of the guard force include:

- Detection of unauthorized attempts to penetrate protective barriers and portals
- Detection of procedural violations that might be associated with the theft of SNM or with sabotage
- Assessment of detected events to determine the nature of the threat and the appropriate response
- Calling for assistance from offsite forces, as needed
- Immediate response to adversary actions in order to prevent theft or sabotage
- Prevention of successful attempts at diversion

In addition to performing these functions, fixed-site guards also carry out a variety of routine duties, including access control, escort, recordkeeping, plant activities surveillance, communications, incident investigation, and accident control.

In the event of an attack, the responsibility of on-duty guards is to alert response forces and to deploy themselves immediately in order to secure and defend SNM and predesignated vital areas of the facility. Other emergency measures, such as cordoning off the incident area, crowd or riot control, first aid, and pursuit and apprehension of adversaries, would be undertaken by secondary (offsite) response forces.

The number of guards required per shift varies with the circumstances at a specific facility, the availability of effective secondary response forces, and the response times needed for such forces to react.

Fixed-site communications. A number of systems are available to provide rapid and continuous communication between onsite security force personnel and between onsite and offsite response

forces. The needs of offsite communications, for example, can be met by using telephones for routine communications and a radio link for emergency communications.

Similarly, a radio communication system consisting of base stations, mobile radios, and hand-carried portable transceivers of good quality can meet onsite communications needs under most conditions.

Communications systems can be supplemented by a variety of signaling devices for notifying both onsite and offsite security personnel. These devices range from radio paging systems and two-way public address systems to klaxons, sirens, whistles, or pyrotechnic devices designed to alert all personnel to emergency conditions. Extensive precedents already exist for the use of such communications systems.

Response forces. In case of an attack on a nuclear facility, the response force must react quickly and effectively. The first response to such an attack would be the tactical deployment of onsite guards to fixed defensive positions in order to secure and defend SNM and predesigned vital areas of the plant.

Contingency plans must provide for any necessary emergency support to the onsite guards. Depending upon the site, off-duty guards may serve as reserves to reinforce the primary response, and local law enforcement agencies may undertake secondary response functions, such as cordoning off an incident area, blocking escape routes, crowd control, first aid, and pursuit and apprehension of adversaries. In some areas, the FBI, police, special weapons and tactics teams, military units, the National Guard, and state or local civil defense units can also play a role.

Local law enforcement agencies are likely to be familiar with the specific installation and the surrounding area and to be experienced in responding to local law enforcement problems. The NRC and the DOE are currently sponsoring programs to increase reliance on local law enforcement agencies as offsite response forces, while proceeding on the assumption that the safeguards system can delay attackers long enough for offsite response forces to be effective.

Detailed contingency planning is necessary to establish the primary and secondary response capability for each site. Training and frequent exercises are also needed to coordinate the operational procedures to be used by the various forces in the event of an overt attack or during other emergencies.

F.1.1.3 Design Approach. The following steps are used in the design of a physical protection system:

Facility characterization. The objective of this step is to assemble and process all of the technical information required to perform the physical protection system design and evaluation. The information needed for this step includes the overall physical protection system performance requirements, the range of threat attributes to be considered in the design, and the descriptions of facility buildings, processes, and systems. This information is used to perform theft, diversion, and sabotage target analyses and to identify safeguards concerns for the facility. Alternative process and plant layouts can be considered in this step.

Hardware-based physical protection systems configurations. The objective of this step is to develop and evaluate the portion of the physical protection system that detects and delays adversaries. This portion of the system is largely hardware-based and includes components such as barriers, detectors, and associated computers, as well as personnel involved in detection and delay (e.g., guards in fixed positions). Hardware configurations that satisfy all of the safeguards requirements identified in the first step are developed for each alternative facility design. If path-analysis

techniques are used, each configuration is evaluated to obtain an estimate of its relative effectiveness. Those configurations that meet minimal performance and cost restraints are considered further in the last step.

Hardware and response force tradeoff analysis. In this step, complete physical protection designs are developed by combining a range of guard response options with the hardware-based configurations selected in the previous step. In the development of guard response options, attributes such as numbers, armament, and protection are considered. Each design is evaluated to determine the relative level of protection provided by the particular combination of facility design, hardware design, and response force attributes. These designs are then compared, and the most effective and economical one is selected for detailed development and implementation.

F.1.2 Material Control and Accounting System.

The discussion below of the material control and accounting system deals with the following three topics:

- Current NRC requirements
- Elements of material control and accounting systems
- Design approach

F.1.2.1 Current NRC Requirements. Each person who is licensed or applies for a license to possess at any one time and location more than 1 effective kilogram of SNM (see Subsection E.3.2) in unsealed form must comply with detailed material control requirements. These requirements, which are stipulated in his fundamental nuclear material control plan and must have NRC approval, are discussed below:

Facility organization and operation. Responsibility for material control functions must be assigned to a single individual at an organizational level sufficiently high to allow independence of action. The SNM custodial, measurement, accounting, and audit functions must be separated in a manner which ensures that the activities of an organizational unit or individual performing one function serve as control over, and a check on, the activities of organizational units or individuals performing a different function.

A manual of approved material control procedures must be maintained and reflected in the facility process specifications, manufacturing instructions, and standard operating procedures. A formal program for the training and periodic requalification of personnel assigned to material control and accounting functions must be developed and documented.

Material balance areas or item control areas must be established for physical and administrative control of nuclear material. The custody of all nuclear material within any of these areas must be the responsibility of a single individual. Each material balance area must be an identifiable physical area so that (1) material assigned to a given area can be kept separate from material assigned to any other area and (2) the quantity of nuclear material moved into or out of a material balance area is represented by a measured value.

Item control areas may be established according to the same criteria as those used for material balance areas, except that material must be inventoried and moved into or out of item control areas by item identity and count. The validity of previously measured quantities of SNM must be assured by the application of tamper-indicating seals or devices to each item or container. The number of item control and material balance areas established at a plant must be sufficient to localize nuclear material inventory discrepancies.

Measurement and statistical controls. The licensee must determine by measurement the nuclear material content of all receipts, shipments, discards, and material in inventory. He must identify the various measurements that are used in nuclear material control, and must describe the measurement methods and procedures, stating the measurement uncertainties. Error models, including the basic statistical methodology techniques, must be provided to demonstrate the licensee's capability to meet adequate measurement criteria. A system of control must be established and maintained that ensures that measurement uncertainties during any material balance period do not exceed specified limits.

Inventories. Physical inventories of SNM must be conducted every 2 months, except in the case of material in the inaccessible portion of a plant. Inaccessible SNM and uranium enriched less than 20 percent in the isotope ^{235}U must be inventoried every 6 months. (Licensees authorized to possess less than 1 effective kilogram, but more than 350 grams of SNM, must conduct annual physical inventories.)

A principal measure of SNM control is the magnitude of inventory differences. This measure is a calculated value which represents the difference between the amount of material that is supposed to be present according to the accounting records, taking into account measured receipts, transfers, and discards (the book inventory), and the amount of material actually found to be present during a physical inventory.

The probability that no inventory difference will exist is very small, since the measurements required to establish the amount of material present are subject to error. A knowledge of the magnitude of these measurement errors is necessary for the proper interpretation of an inventory difference.

The NRC has proposed new guidelines to make sure that corrective action is taken when the amount of inventory difference reaches the NRC's allowable limits. Under the regulation published for public comment on July 17, 1975, absolute limits are specified for inventory differences. The new regulation requires that when such limits are exceeded, the licensee must take specific action, such as immediate reinventory, investigation of excessive differences, or adoption of new procedures to prevent recurrence.

Storage and internal control. A documented system of control over the SNM within a facility must be maintained, and all transfers of material between material balance areas and item control areas must be documented and validated. Storage and internal handling controls must be established, maintained, and followed to provide timely information on the identity, quantity, and location of all SNM within a facility in discrete items or containers. A unique item identification system must be established to ensure that no two items can have the same number. Records showing the identity, source, and disposition of all items must be kept.

A program must be developed and implemented for the control, processing, and disposition of scrap, since a large uncertainty in scrap measurements could be used to mask theft. No item of scrap with an SNM content whose measured uncertainty exceeds ± 10 percent is permitted to remain in inventory longer than 6 months when such scrap contains SNM.

Shipping and receiving. As a rule, shipments and receipts must be independently measured by the shipper and receiver. Shipper/receiver differences must be reviewed and evaluated, and appropriate investigation must be made of statistically significant differences to decide whether corrective action is necessary, or more important, whether diversion or theft has occurred. Missing, diverted, or stolen material must be detected promptly.

Audits. Annual audits are required under the material control program. The results of these audits must be documented, reported to appropriate plant management, and kept available at the facility for inspection. Losses of discrete items or containers must be investigated and the results of the investigation reported to licensee management and to the NRC.

F.1.2.2 Elements of Material Control and Accounting Systems.⁽³⁾ Material control and accounting systems can be divided into two primary activities: nuclear materials control (technical control), which is described below, and nuclear material accounting (administrative controls), which is covered in Subsection F.1.2.3. The technical elements of nuclear materials control are measurements, measurement control, statistical limits of error, and physical inventory. These elements are described below.

Measurements. Measurements comprise the equipment, techniques, and procedures used to determine SNM quantities for materials control and accounting purposes. Major elements include:

- Bulk measurements (mass and volume)
- Sampling
- Analytical measurements
- Nondestructive assay (NDA) measurements

Measurement control. Measurement control is required to monitor, evaluate, and report current SNM measurement performance, and comprises those actions, methods, tests, and procedures used to measure the random and systematic error variances of each component of SNM measurement. The measurement control program addresses the following areas:

- Volume (vessel) calibrations
- Mass (scale or balance) calibrations
- Mixing and sampling preoperational tests
- Nondestructive assay calibrations
- Preparing or acquiring, maintaining, storing, and using SNM measurement standards
- Controlling scale or balance measurement performance
- Controlling bulk vessel measurement performance
- Controlling analytical measurement performance
- Training, qualifying, and periodically requalifying all personnel that perform sampling and measurements for materials control and accounting purposes
- Maintaining sample integrity during transport and storage
- Frequency of SNM measurement system calibrations, of replicate measurements, and of standard calibrations of recertification

Statistical limits of error. Whenever an SNM measurement is made, the limit of error associated with the measurement must be determined. The measurement and measurement control programs are designed to supply the data required for the evaluation and calculation of limit of error for each item on inventory, for each material balance component, and for the material unaccounted for.

Physical inventories. Physical inventory is defined as the determination, on a measured basis, of the quantity of SNM on hand at a given time. The methods of physical inventory and associated measurements will vary, depending on the material to be inventoried and the process involved.

F.1.2.3 Design Approach. The following steps are involved in the design of the material control and accounting system.

Facility characterization. The objective of the first step is to assemble the details of the design, operating procedures, and layout of the facility. The facility is then subdivided into physical areas in which material balances can be maintained. These physical areas are called material balance areas, item control areas, and unit process areas. All material control and accounting functions are then accomplished by performing material balances in these areas.

Establishing material control procedures. Administrative procedures are established for controlling the SNM during receiving, handling, processing, and transfer from one material balance area to another. Within each material balance area, custodial responsibility of SNM is assigned to one individual or organization. The SNM custodial, measurement, and accounting functions are administratively separated in a manner that ensures that the activities of an organizational unit or individual performing one function serve as control over, and a check on, the activities of organizational units or individuals performing a different function.

Establishing data base for material control. The data base for material control is provided by the process control functions. The process control function requires a material measurement and quality control program, the analysis of data obtained in process monitoring, and control over the use of items and containers.

This information is made available from the process control computer to the material control and accounting computer. If some additional information or measurement is required for material control and accounting purposes (like the SNM content of a material), it should be coordinated through and included in the process control function.

Design of the accounting function. Material accounting consists of maintaining records of all receipts, transfers, locations, and shipments of SNM, and conducting periodic inventories. At present, the key measure used in accounting control is the comparison between book inventory and physical inventory. Material balance closures are performed at intervals specified by regulations around each material balance area identified in the first step. In the case of items and containers, the accounting function consists of confirming the identity, quantity, location, and integrity within each item control area identified in the first step.

F.2 SAFEGUARDS APPLICATION TO FUEL CYCLE FACILITIES AND SYSTEMS

F.2.1 Safeguards for a Transportation System

F.2.1.1 General. The following materials are transported in the various SFDS study cases:

- Spent fuel
- High level waste (HLW)
- Transuranic (TRU) waste
- Hulls
- Encapsulated spent fuel
- Uranium and plutonium

The transportation of all radioactive materials is governed by NRC Regulation 10 CFR 71 and DOT Regulations 49 CFR 170-179. Shipments of SNM are subject to NRC Regulation 10 CFR 73. The requirements specified in these regulations are discussed in detail in Subsection E.3. The shipment of spent fuel is specifically exempt from the requirements for SNM.

At present, NRC is studying the safeguard requirements for the transportation of various nuclear materials. The following features relating to safeguards during transportation are being studied:

- Provisions that delay the potential attacker's access to the material
- Loading and unloading features that require the use of specialized equipment and/or knowledge of the system
- A trailer hauled by a tractor or equivalent that has armor, weapons, guards, and/or communication systems
- The absence of built-in features that would automatically produce a lethal hazard to the public in its response to an attack
- Immobilization features (wheel locks) on the trailer that limit its movement during an attack (the tractor has remote-controlled trailer drop capability)

F.2.1.2 Measures for Providing Physical Security. Various measures are available for providing in-transit physical security for SNM shipments. They include the use of secure, penetration-resistant transport vehicles; special material containers; armed escorts and convoys; in-transit communications; and thorough planning and control. A brief discussion of each of these measures follows:

Secure transport vehicles.⁽⁴⁾ Specially constructed, penetration-resistant vehicles can reduce susceptibility to theft or sabotage and can often delay an attack until the arrival of response forces. Depending upon the specific design of the vehicle and the form of the SNM, a delay of minutes to hours against an adversary action can be achieved.

AGNS, under a subcontract with Sandia, has developed a conceptual design for a plutonium transport vehicle to withstand attempts at theft or sabotage. It combines the material containers and the transport vehicles into an integrated container vehicle which offers improved penetration and crash resistance and is applicable not only to road transport but also to air, rail, and water transport.

Special material containers. Both safety and safeguards needs must be considered when transporting SNM. Protection must be furnished against radiation, heat generation, accidental dispersion, and the accumulation of critical quantities; penetration and theft of the container must be

prevented. A variety of safeguards features can be build into SNM containers to enhance their security, including tamperproof features and features that aid in locating a stolen container.

Escorts and convoys. SNM shipments are currently protected by armed escorts and convoys. The specific functions of escorts and convoys are similar to those of fixed-site security forces: to deter, detect, or delay adversary attacks and, if necessary, to defeat them. The number of escort vehicles and armed personnel in convoys transporting SNM can vary, depending upon the specific form of material being transported; the terrain, mode, and distance of the shipment; and the security level or hardness of the transport vehicle.

To accomplish their function, the escorting personnel must be adequately deployed, equipped, and protected; drivers and escort personnel must be trained in defense procedures; and the escort vehicles must be appropriately armored and equipped with intraconvoy and external communications to alert each convoy element and local law enforcement units.

In-transit communications. Effective in-transit communications between the vehicles in a transit convoy are needed. Such communications must also be maintained between the convoy and shipment control center to track the convoy's progress, to report attempted attacks or thefts, and to provide a means for reporting other problems and for requesting and coordinating appropriate aid.

Intraconvoy communications needs can be met by the use of high-quality VHF mobile and portable radio transceivers, which are readily available from commercial sources. Communications between the convoy and the shipment control center are more difficult to establish, since they may require a system that can provide fast and reliable communications on a nationwide basis. One currently operating system meeting these criteria is the DOE SECOM 11 High-Frequency System.^{(5),(6)} An alternative solution could be the development of a dedicated communication system to communicate directly with the local law enforcement agencies along various routes.

Shipment planning and control. Procedures used in safeguarding shipments require as much attention as the actual equipment and hardware employed. Each of the following helps reduce the risk of in-transit material theft, diversion, or sabotage:

- Selection of the most advantageous transportation mode, route, and schedule
- Command procedures
- Coordination with local government and law enforcement authorities along the route
- Documentation, receipt, and reporting controls
- Loading and unloading procedures. As the volume and frequency of SNM transportation increase, new and improved planning and control methods will have to be developed and adopted

Response to attacks. In the event of an attack on an SNM shipment, a neutralizing force must be brought to bear quickly and effectively. Convoy escorts must be capable of rendering immediate assistance. If an SNM transporter is immobilized by adversary action, escorts need to establish defensive positions to protect the shipment and immediately alert the shipment control center and the nearest law enforcement agencies for assistance.

The law enforcement agencies should be given as much information as possible about the attackers (number, type of weapons, vehicles used, type of terrain, location, etc.). The escort force must try to delay the loss of control over the SNM as long as possible. After the response force (nearest law enforcement agency) arrives at the scene, the response force assumes the primary responsibility for the operation (i.e., protection of SNM, attack on the intruders for recovery of any stolen SNM, and search and recover of SNM). Throughout this operation, the escort force assists the law enforcement agency, as required.

F.2.2 Safeguards for a P/E Facility⁽⁷⁾

The safeguards system for a P/E facility is currently required to meet NRC requirements set forth in 10 CFR 70-75, appropriate NRC regulatory guides, and any future requirements developed by the NRC for a P/E facility. For the purpose of this study, the simplest P/E facility is a spent fuel encapsulation facility, while the most complex P/E facility is a spent fuel reprocessing plant containing uranium hexafluoride conversion, plutonium oxide conversion, and a waste vitrification step. The safeguards system discussed below addresses all aspects of the P/E facilities applicable to this study.

The responsibility for the safeguards system of the P/E facility is assigned to a single organization (Safeguards Coordination) headed by a responsible individual who is directly responsible to facility management. This individual coordinates all safeguards action with the process control function and is in charge of both the physical protection system (access control, zone control) and the material control and accounting system. To ensure that the activities of one organizational unit (or individual) control and check the activities of another organizational unit (or individual), the nuclear material custodian service is separated.

The system described below is a state-of-the-art system, some of whose features exceed current NRC requirements. Since there is a possibility that the system is not cost-effective, further studies are required to evaluate its cost-effectiveness.

F.2.2.1 Physical Protection System. The physical protection systems of P/E facilities are designed to meet the NRC requirements described in Subsection F.1.1.1. They consist of the elements described in Subsection F.1.1.2 and are designed in accordance with the approach described in Subsection F.1.1.3. The actual design of the physical protection systems of P/E facilities are classified, and further details are available in classified reports.⁽³⁾

F.2.2.2 Material Control and Accounting System. For the purpose of this study, the P/E facility consists of the main material control areas listed below. Each of these areas is further subdivided into material balance areas and item control areas for the material control and accounting system. The main material control areas are:

- Fuel receiving and storage area
- Separation process area (including the fuel chopping, voloxidation, dissolution, accountability tank, solvent extraction, product purification, and solvent recovery operations)
- Waste management areas (including the HLW handling area, hull handling area, TRU handling area, and offgas handling area)
- Plutonium nitrate storage area
- Hot and cold laboratory area
- Plutonium oxide conversion and storage area
- Uranium hexafluoride conversion and storage area
- Spent fuel encapsulation area

The complete P/E facility must be inventoried twice a year. This semiannual inventory includes:

- An item-count identification and verification of the integrity of all fuel-assembly canisters in the fuel receiving and storage station
- A shutdown, flushout, and physical inventory of the separations process area
- Volume measurements, sampling, and analysis of all solutions in the plutonium area, or confirmation that tamperproof seals are intact and that prior measurements are valid

- A cleanout of extraneous samples, and an item-count verification of remaining material in the hot and cold laboratory area

The following is a discussion of the material control and accounting system in each material control area.

Fuel receiving and storage area. Spent fuel assemblies are received in shielded casks. Each assembly is identified by visual sighting of its serial number and checked against the information included in the accompanying shipping papers. After appropriate identification by two individuals, the assembly is stored in an identified location in an identified fuel storage basket. Baskets are stored in racks in a fuel storage pool, and basket and fuel assembly locations are recorded. At the start of reprocessing (or encapsulation), each assembly is identified by two individuals before transfer to the fuel transfer conveyor. Material control is by item accounting of individual fuel assemblies.

The spent fuel containment and surveillance module permits surveillance and containment of the spent fuel assemblies from the time the transport vehicle enters the facility until the assemblies are transferred to the fuel shear (or encapsulation) area.

Physical safeguards at the vehicle admittance gate house include authorization checks of the driver and truck, tamper inspection of the cargo, and a contraband search of the vehicle. Onsite movement, washdown activities, and movement into the unloading bay are also subject to surveillance. A set of operations monitoring and operations control elements is furnished for the spent fuel residing in the receiving and storage zone.

Cask unloading operations are monitored to make sure that all assemblies are removed from the cask and that theft or diversion of the assemblies from the unloading pool has not occurred. Physical safeguards for the storage pool area ensure that theft or diversion of fuel assemblies does not occur.

Separations process area. The separations process area includes the fuel chopping and voloxidation, dissolution, accountability, solvent extraction, product purification, and solvent recovery operations. After shearing and dissolution, the dissolved material is transferred in batches to the accountability tank. The volume is measured, and the uranium and plutonium contents are determined using mass spectrometric methods.

Following uranium-plutonium partitioning, purification, and concentration, the SNM content of the uranium product is determined by measuring the uranyl nitrate concentration in the uranium product sample tank.

Measurements are also made in the various waste streams. These measurements, along with the storage tank measurements, constitute the material balance output of the separations process area.

A dynamic material accountability system is being developed to provide near-real-time accountability information. The cost-effectiveness of this system is currently being evaluated, and the system could be used in the future for parts of the separations process area.

The dynamic accountability system can be applied to the plutonium purification process (PPP). The PPP unit process boundary is determined by in-line or on-line flow and SNM concentration measurements. Waste and recycle streams from the columns and the concentrator could be monitored for plutonium concentration by a combination of flow meters and NDA alpha detectors.

Estimated thresholds for diversion detection sensitivity in a typical plutonium purification process range from 1.8 kilograms of plutonium for 1 day (1-hour balance periods) to 4 to 6 kilograms of plutonium for 1 week, depending on the measurement strategy.⁽⁸⁾

Waste management areas. The waste management areas include the HLW handling area, the hulls handling area, TRU handling areas, and offgas handling areas. The material control and accounting system for waste management areas is similar to that discussed for repositories in Subsection F.2.3.

Each of these areas is subdivided into discrete item control areas. Waste packages in each item control area are individually identified, worked, and recorded. Item control areas are kept separate from one another, and all transactions between them are accounted for by a double-entry accounting system.

Plutonium nitrate storage area. Plutonium nitrate is transferred to the plutonium conversion process area for conversion to plutonium oxide. The plutonium content of the solution leaving storage is determined by measurements, which provide the bases for closing the plutonium balance around the plutonium nitrate storage.

The plutonium nitrate storage area surveillance and containment module has monitoring elements that perform surveillance and containment functions, and operations control elements that perform detection, delay, and response functions. Each of these elements is based on closed-loop surveillance, monitoring, and control of personnel and material. The closed-loop system uses a variety of detectors and protective devices coupled with computer controls and checks throughout all phases of the transfer and sampling operations. The protective devices include personnel identification matched with the appropriate authorized activity at critical locations, glove-box access control, and material flow control.

The first safeguards concern relates to unauthorized entry into the material access areas — the pump and valve glove boxes and the sampling glove box of the plutonium nitrate storage area.

Access to these areas is normally gained through remotely operated portals containing personnel identification devices and equipment for detecting contraband. This access control is assumed and is not part of the closed-loop control system that controls operations within the various material access areas. Safeguards concerns related to the pump and valve glove boxes include authorized glove-box entry, operations of unauthorized valves, tampering with the valves, and tampering with the material lines. These actions could result in the transfer of material to an unauthorized location or, in conjunction with other actions, the diversion of material. Protective measures for these glove boxes include:

- Personnel identification at the glove boxes
- Access restriction to only those ports required for a specific operation
- Valves provided with status and tamper monitors

The sampling glove box is a material access point. The principal safeguards concern is the diversion of material through the needle block, with subsequent removal through the glove ports, bottle load-in port, bag-out port, or the pneumatic transfer tube. Protection measures at this station include:

- Personnel identification which, when correlated with the sampling authorization, allows admittance to the glove box
- The same kind of access restriction to the two glove ports as exists at the pump and valve glove boxes
- Securing and monitoring the bottle load-in port, allowing its use only during the specified operational sequence
- Needle block design, which precludes insertion of any object other than the sample bottle into the needles; this provides an electrical signal for the bottle count
- Air system control and monitoring, permitting air flow (and, hence, material flow) for limited time periods

The sample receiving station in the analytical services laboratory represents the other material access point directly associated with the sampling operation. Material control and accountability is maintained by the laboratory. Safeguards applied at this station ensure that counterfeit material is not transferred from the sampling glove box, since a substitution may not be detected until much later if an analysis is not performed immediately.

Hot and cold laboratory area. There are numerous remote and local sampling points throughout the plant. All samples containing significant amounts of SNM are taken via needle samplers located in shielded caves or glove boxes in the separations process area. Samples are transferred to the hot and cold laboratory area using a pneumatic transfer system. All SNM entering and leaving this material control area is controlled by actual measurement or by estimation of SNM content based on historical data.

Protective measures at the laboratory sample receiving glove box consist of:

- Personnel identification which, when correlated with the sampling authorization, allows ample removal operations to proceed
- Control and monitoring of the pneumatic tube access port within the laboratory sample receiving glove box
- A detector that verifies that the sample bottle delivered to the laboratory contains plutonium nitrate

A material accounting and operations control system handles materials measurement and accounting. This system consists of a modular network of computers, communications equipment, and data storage and retrieval devices, and furnishes a sophisticated, automated nuclear materials safeguards and management capability. The system consists of two major subsystems: the laboratory data system, and the material control and accounting system. These subsystems provide mutual redundancy; that is, one system can cover both functions (at a slower pace) in the event of an outage of the other system.

The material control function includes those measures that provide automatic surveillance of the material in the process. The materials accountability function includes those measures involving the physical inventory of the materials, their identification, and their location, both in process and in storage. The physical security function includes those measures involving access control (e.g., physical barriers, penetration alarms, portal monitors, guards).

Plutonium oxide conversion and storage area.⁽⁹⁾ The plutonium conversion process offers a unique safeguards challenge. Unlike the materials in other parts of the fuel cycle, input, in-process, and product materials are all attractive, high-purity, concentrated targets for diversion, unhampered by high-level radiation, heavy shielding barriers, or impractically low concentration levels. In addition, the conversion process tends to become the process buffer between the loosely coupled functions of plutonium separation (chemical reprocessing) and plutonium use (fuel fabrication). It has significant inventories of extremely attractive feed and product materials that offer the greatest potential for diversion.

Recently developed NDA technology, state-of-the-art conventional measurement methods, special in-plant sensors, process instrumentation, and data analysis techniques, supported by computer and data base management technology, combine to form an effective, timely safeguards materials management system.

Conventional materials accounting methods can be supplemented by dynamic materials accounting in which the facility is partitioned into discrete material balance areas. Dynamic material balances drawn around such unit material balance areas during the course of normal plant operation permit more timely materials accounting and facilitate detection of repeated small thefts of SNM.

The related concept of overlapping material balance areas is useful for cross-checking, and allows continued materials accounting, with less sensitivity and timeliness, in case of instrument malfunction.

The key measurement points in the conversion process are the receipt tank (volume and concentration) and the product cans (plutonium content). Absorption-edge densitometry is used to measure the plutonium concentration in the receipt tank solution; a neutron well counter or calorimeter is used to measure the plutonium content in a product can. Each measurement assumes that a companion measurement of the plutonium isotopic composition by gamma or mass spectroscopy is available. These on-line measurements are amenable to near-real-time accountability.

Uranium hexafluoride conversion and storage area. The uranium in this area is not SNM since it is well below the 20 percent ^{235}U enrichment level, above which the NRC requires special safeguards requirements. Thus, only normal material control and accounting procedures are necessary in this area.

Spent fuel encapsulation area. Each cell of the spent fuel encapsulation area is treated as an item control area. As the spent fuel assembly enters the encapsulation area (from the surge pool or the receiving area), its identity is verified and recorded. After encapsulation, the identification number of the fuel assembly is marked on the outer surface of the encapsulated package. The encapsulation area's item control uses a double-entry system in which all receipts and shipments are recorded. Any discrepancy is promptly investigated, and the results are reported to safeguards coordination.

F.2.3 Safeguards for a Repository⁽¹⁰⁾

A waste repository for disposal of waste generated by the commercial nuclear fuel cycle must meet the NRC requirements set forth in 10 CFR 70 and 10 CFR 75 and any future safeguards requirements developed by the NRC for waste repositories. Detailed safeguard requirements have been developed primarily for commercial nuclear power plants, and little attention has, as yet, been given to waste repositories. At present, the NRC is studying the safeguards requirements for all fuel-cycle facilities. General safeguards considerations, as they may apply to a waste repository, are discussed below.

The responsibility for the safeguards system of the repository is assigned to a single organization headed by a responsible individual with independence of action. This individual is in charge of both the physical protection (access control, zone control) and the material control and accounting systems. The nuclear waste custodian service is separated to ensure that the activities of one organizational unit (or individual) control and check the activities of another organizational unit (or individual).

F.2.3.1 Physical Protection System. The same physical protection requirements that apply to the confinement of radioactive materials are also required for the repository buildings because of the form of the waste, the radiation levels, and the size and weight of the canisters. These requirements also apply to the area around the repository.

For physical protection requirements, the site is divided into three areas:

- **Area A.** Area A consists of 200 acres and contains all the surface facilities
- **Area B.** Area B consists of 2,000 acres of the inner controlled area surrounding all surface facilities and encloses the maximum extent of underground development
- **Area C.** Area C consists of 18,000 acres of the outer controlled area

The actual level of protection required (i.e., the number of guards, the size of the isolation zone, etc.) is developed specifically for each detailed design of the repository. Detailed plans are submitted to the NRC before repository operation is approved. The detailed plan includes the items discussed below.

Area A requirements. Precautions for Area A are the most extensive and cover perimeter and portal safeguards. Perimeter safeguards include:

- **Isolation Zones.** These zones are used to provide the assessment system with accurate information for all contingencies
- **Physical Barriers.** The most common types of barriers are constructed of stone, brick, cinder block, concrete, steel, or comparable material
- **Illumination.** The illumination must be great enough to permit (1) the monitoring of outdoor areas adjacent to the physical barrier at the perimeter of the protected area and (2) the observation of potential penetration of the protected area in the isolation zone adjacent to the protected area
- **Surveillance.** At least one member of the physical security force must be able to observe activities in the isolation zone at all times and to transmit the required surveillance data to the central and secondary alarm stations
- **Security Locks.** These locks must be manipulation-resistant. In addition, an effective system is required for managing all locks, keys, combinations, and other related equipment employed in securing the perimeter of the protected area
- **Intrusion Detectors.** These detectors must be tamper-proof and self-checking to provide high-assurance detection of all penetrations into the protected areas, and must have sufficient emergency electrical power for operation during the attack
- **A Guard Force.** The guard force includes fixed guards positioned at security posts at the perimeter of the protected area and guards patrolling the barrier at the perimeter

Portal safeguards considerations covering personnel and vehicle portals include:

- Physical-structure requirements
- Security locks
- Guard force
- Detection capabilities
- Intrusion detectors of men and materials
- Access authorization
- Communications

Areas B and C requirements. Safeguards for Area B consist of fencing and daily patrols by a guard force around the periphery. A patrol road is built around the perimeter of Area C. In addition, signs are placed around the periphery stating that the government owns subterranean rights to the area and that permission must be obtained before drilling or mining can be started within the area. Weekly patrols are made around the periphery by a guard force.

F.2.3.2 Material Control and Accounting System. The material control and accounting system for the repository consists mostly of control and identification of discrete waste packages. Some of the waste packages received at the repository may not contain significant quantities of SNM. The material control and accounting requirements are assumed to be applicable to all waste packages received at the repository. A discussion of some of the features of such a material control and accounting system for a repository is found below.

Repository operation. Item control areas are established for physical and administrative control of nuclear waste. Enough of these areas are established at a repository to localize nuclear-material inventory discrepancies. The custody of all nuclear waste within any item control area is the responsibility of a single individual. Item control areas are kept separate from one another. The quantity of waste packages moved into or out of an item control area is represented by a number of discrete units.

Identification controls. At the repository, a check must be made to determine that all individually identifiable waste packages received at the repository either are still on hand or have been shipped to other facilities. This is done by identifying each waste package by a unique number. If one or more packages are missing and not accounted for, an investigation is called for. A system of periodic management audits and reviews is required to ensure the adequacy of these controls. A unique item-identification system is established to ensure that no two waste packages have the same number, and records showing the identity, source, and disposition of all packages are kept.

Storage and internal transfers. A documented system of control over the nuclear material within a facility is maintained. All transfers of nuclear waste between item control areas are documented and validated for individual packages by identity and count. A centralized accounting system using double-entry bookkeeping with subsidiary accounts for each item control area is established and maintained. This system can furnish information on a timely basis about the identity, quantity, and location of all waste within a facility.

Shipping and receiving. Shipments and receipts are independently accounted for by both the shipper and receiver. Shipper/receiver differences are reviewed and evaluated on an individual package or lot basis, on a shipment basis, and on a cumulative basis. Whenever there is a discrepancy, the matter should be investigated promptly. When the materials are received, they are piece-counted and item-identified for comparison with the shipment bill of lading. The integrity of the tamperproof devices on the shipping casks is verified, and records of shipper/receiver difference evaluations, investigations, and corrective actions are kept on file at the facility.

F.2.4 Safeguards for an AFR

The safeguards systems for an AFR must meet the requirements of 10 CFR 70 and 75 and any future safeguards requirements developed by the NRC for AFRs. The NRC is at present studying the requirements for safeguards systems for all fuel cycle facilities.

The responsibility for the safeguards system of the AFR is assigned to a single organization headed by a responsible individual with independence of action. Both the physical protection and the material control and accounting systems are the responsibility of this individual. The nuclear material custodian service is separated to ensure that the activities of one organizational unit (or individual) control and check the activities of another organizational unit (or individual).

F.2.4.1 Physical Protection System. The physical protection system of an AFR is designed to meet the NRC requirements described in Subsection F.1.1.1. The physical protection system is designed in accordance with the design approach outlined in Subsection F.1.1.3 and uses the elements described in Subsection F.1.1.2. The details of a physical protection system of an AFR are classified and subject to review and approval by the NRC.

F.2.4.2 Material Control and Accounting System. The material control and accounting system for the AFR consists of control and identification of discrete fuel assemblies. For the purpose of the material control and accounting system, the AFR can be divided into the following major areas:

- Spent fuel receiving area

- Spent fuel storage pool
- Spent fuel shipping area

The following is a discussion of some of the features of a material control and accounting system for an AFR.

AFR operation. Item control areas are established for physical and administrative control of spent fuel assemblies. Enough of these areas are established at an AFR to localize spent fuel inventory discrepancies. The custody of all spent fuel assemblies within any item control area is the responsibility of a single individual. These areas are kept separate from one another, and spent fuel is inventoried and moved in or out of them by unique item and identity count.

Identification controls. The AFR is required to determine that all individually identifiable fuel assemblies received in irradiated or nonirradiated form are still on hand or have been shipped to other facilities. This is done by identifying each fuel assembly by a unique number. If one or more assemblies are missing and not accounted for, an investigation is required.

A system of periodic management audits and reviews is needed to ensure the adequacy of these controls. A unique item-identification system is established to make sure that no two fuel assemblies have the same number. Records showing the identity, source, and disposition of all assemblies are maintained.

Storage and internal transfers. All transfer of fuel assemblies between item control areas are documented and validated for individual assembly by identity, count, and a previously measured valid SNM content (at the time of fabrication). A centralized accounting system using double-entry bookkeeping with subsidiary accounts for each item control area is established and maintained. This system can furnish information on a timely basis related to the identity, quantity, and location of all fuel assemblies within the facility.

Shipping and receiving. Shipments and receipts are independently accounted for by both the shipper and receiver. Shipper/receiver differences are reviewed and evaluated on an individual assembly or lot basis, on a shipment basis, and on a cumulative basis. Whenever there is a discrepancy, the matter should be investigated promptly. When the materials are received, they are piece-counted and item-identified for comparison with the shipment bill of lading. The integrity of the tamperproof devices on the shipping casks is verified. Records of shipper/receiver difference evaluations, investigations, and corrective actions are maintained on file at the facility.

F.3 SABOTAGE AND THEFT

There are several classes of social groups and individuals that could pose a potential threat to the nuclear industry. The fact that such groups exist leads some to perceive them as threats to the nuclear industry. The motivations to commit hostile acts include desire for revenge, financial gain, publicity, and intent to create mass destruction. There are two types of threats that can be made to satisfy these motives: high-consequence threats, such as sabotage of nuclear facilities that endanger local populations and theft of SNM for the purpose of constructing a clandestine nuclear device; and low-consequence threats, such as nuclear hoaxes, low-level sabotage of a nuclear facility, seizure of a nuclear facility, radioactive contamination of a target, and dispersal of plutonium.

F.3.1 Adversaries

For safeguard purposes, adversaries are defined as individuals who would attempt an act of sabotage or theft. These include groups or individuals outside the nuclear industry, individuals within the industry, and individuals inside and outside the industry acting in collusion.

F.3.1.1 Insiders. There are numerous credible, though not necessarily probable, threat scenarios involving employees within the nuclear industry. Those scenarios with the greatest potential significance, insofar as public health and safety are concerned, include sabotage of a facility and collusion for purposes of diverting SNM.

F.3.1.2 Organized Crime. Organized crime is regarded as a potential adversary because its members possess a wide variety of skills (in hijacking, burglary, extortion, black market operations, etc.) that could be used against the nuclear industry. However, a good argument can be made that organized crime — a relatively stable, profit-making enterprise — will not want to jeopardize its present position by involving itself in the kind of illicit and dangerous nuclear activities that could cause an aroused public to demand retaliatory action.

F.3.1.3 Terrorists. Although the popular press often dwells upon the possibility that terrorists will ultimately acquire a nuclear capability and that this capability will give them extraordinary powers of extortion, several recent serious studies indicate that such a scenario is not very likely.^{(11),(12),(13)}

F.3.2 Threat

Between May 1969 and the end of December 1975, 99 threats of violence against licensed nuclear facilities were recorded.⁽¹⁴⁾ Of these, 75 were bomb threats to reactors which were never carried out. A small number of the remaining threats actually involved small pipe bombs or incendiary devices that were found at research facilities, or at the offices of corporations in the nuclear industry. A smaller number yet of these 99 incidents involved break-ins, or apparent attempts to break in.

The intentionally set blaze at the Indian Point No. 2 reactor site, the bombing of the Fessenheim reactor construction site in France, and the bombing at the Mt. D'Arée reactor site in France are the best known incidents involving low-level sabotage of a nuclear facility. The Indian Point incident was caused by a maintenance worker before the reactor became operational. Damage was extensive, but confined to the reactor auxiliary building.

There have been no instances in the United States of seizure of a nuclear facility. The single publicized incident of this nature was at the Atucha reactor construction site in Argentina, where urban guerrillas briefly occupied a reactor building, painted slogans, stole some weapons, and then left.

Radioactive contamination of a symbolic target has never occurred in the United States. A mentally deranged individual did, however, contaminate two or more railroad coaches in Austria with radiopharmaceuticals stolen from a hospital.⁽¹⁴⁾

Procuring materials for a nuclear device, designing and manufacturing the device, and deploying it require very thorough and detailed advance planning and considerable technical expertise. There are much easier and more certain ways of bringing about mass destruction than stealing SNM and attempting to fabricate a nuclear device, which may or may not function. Chemical and biological weapons require less technological skill in their fabrication than nuclear weapons, can be handled with less risk of detection and personal harm, call for orders of magnitude less in terms of resources, and use raw materials that are more easily obtained than SNM. Moreover, the techniques of causing mass destruction with a chemical or biological weapon are fully described in the open literature.

There is the scenario of terrorists holding a city or government hostage for purposes of extortion by threatening the use of a weapon of mass destruction. However, effective nonnuclear means exist for carrying this out. Terrorists are generally aware that their demands would be unenforceable unless they could maintain a long-term enforcement capability. The more the city or government hostage scenario is examined, the more unlikely it seems to become. A credible scenario, however, is that in which city or government might be held hostage for the release of prisoners. In this case, the time for compliance could be relatively short, and once the demand was met, there would be no continuing requirement for the terrorist to maintain the threat.

Plutonium dispersal is potentially a low-consequence threat for reasons relating to its toxicity. The problems involved in creating and effectively dispersing an aerosol of plutonium serve to limit the actual threat represented by plutonium toxicity. The effects from plutonium dispersal are difficult to predict and the effects on individuals are not likely to be noticeable until many years after exposure; it is almost certain that there would be no immediate measurable radiobiological effects. This delay limits the utility of plutonium dispersal to a terrorist.

As yet, there has been no identifiable threat, of the type under discussion, to the nuclear power industry. Nor has any group with the motivation to establish such a threat been identified.

In summary, if an adversary should acquire significant quantities of SNM, or attain a position where he could commit an act of sabotage to a facility containing such materials, there would be a broad spectrum of possible threats. The most important of these would be:

- Fabrication of a nuclear explosive device
- Sabotage of a nuclear facility
- The dispersal of radioactivity
- Threats to use the nuclear materials for malevolent purposes unless certain demands are met (blackmail or extortion)

F.3.3 Fabrication of a Nuclear Explosive Device

Some of the plutonium-containing materials that would be found at various points in the reprocessing facility could be suitable for use in a nuclear explosive device with little or no processing; spent fuel, however, could not be so used without major facilities and major processing efforts.⁽²⁾ Accordingly, public concern and speculation have centered on the ease or difficulty of

making a crude nuclear explosive device from illicitly obtained reprocessed nuclear materials. Nuclear weapons design depends on many factors, such as the type, form, and quantity of nuclear material; the availability of essential accessory equipment; the capability of handling hazardous components, including radioactive materials and explosives; and knowledge of the technical features of a nuclear device. There is little likelihood that a terrorist group could fabricate an efficient bomb such as those in military inventories. There is a low, but credible, probability that such a group could assemble a crude device, and it is this possibility that safeguards systems are designed to minimize.

Although a small group of competent people might be able to carry out such a project, the successful design and construction of nuclear explosives is not easy, and the chances of success are highly dependent on the time spent on the project and the effort and abilities of the people involved.

A theoretical knowledge of nuclear physics and the unclassified literature on nuclear weapons, while helpful, is not sufficient for the design of a practical nuclear explosive. Most theoretical designs will fail. The knowledge required comes from practical experience in design, fabrication, and testing of actual nuclear devices, and this kind of knowledge is not widespread.

Knowledge of the design details of military nuclear weapons is not sufficient for the design of a reliable, simple nuclear explosive device. The designs for military nuclear weapons emphasize efficiency, low weight, small size, and reliability more than simplicity or ease of fabrication. Such designs are not readily adaptable to an illicit nuclear explosive device. Sophisticated knowledge and skills are required that depend on the design approach and materials to be used. They include knowledge of and skills in precision machining, chemical processing, foundry processes, use of electromechanical devices, electronics, and high-explosives handling. Experts with such skills and with the motivation to build and detonate a nuclear weapon for unlawful purposes would be difficult to gather together in a clandestine project. The equipment required for such a project would be related to the importance of the lives of the participants, the reliability and yield of the device, and the kind of nuclear materials available. Equipment needs could approach those of a large nuclear laboratory.

The simpler and less sophisticated the design of the explosive device, the larger the size and weight of the device and the greater the requirements for nuclear materials and high explosives. If large quantities of SNM are available, the design can be unsophisticated, but the resulting device is also likely to be heavy and a team of people and special equipment will be needed to assemble and transport it. If the available amount of nuclear material is small, the design must be sophisticated, and additional skills and more time will be required for fabrication.

There are many risks in fabricating a crude nuclear explosive device. The activities in such a project, the kinds and numbers of people required, and the materials involved increase the time and activity needed for completion. This increases the probability of detection.

The manufacture of nuclear explosive devices involves the use of hazardous materials and increases the chance that the project might suffer accidents involving criticality, plutonium poisoning, conventional chemical hazards, or the mishandling of high explosives. In the history of nuclear weapons fabrication, such accidents have occurred, even under highly controlled conditions, and their probability would be increased by the conditions prevailing in a clandestine project. Lethal radiation exposures, premature nuclear detonation, or accidental explosion of conventional high explosives could occur during assembly and delivery of the nuclear explosive devices. A crude device successfully assembled may fail to explode, may simply disperse radioactive material, or may give a wide range of possible nuclear yields.

Experts are divided on how formidable these problems are and on the minimum qualifications, equipment, and time that a determined group would need to create a crude nuclear explosive. The opinions quoted below represent a range of views:

- **Willrich and Taylor.**⁽¹⁵⁾ "As a result of extensive reviews of publications that are available to the general public and that relate to the technology of nuclear explosives, unclassified conversations with many experts in nuclear physics and engineering, and a considerable amount of thought on the subject we conclude:

"Under conceivable circumstances, a few persons, possibly even one person working alone, who possessed about ten kilograms of plutonium oxide and a substantial amount of chemical high explosive could, within several weeks, design and build a crude fission bomb. By a 'crude fission bomb' we mean one that would have an excellent chance of exploding, and would probably explode with the power of at least 100 tons of chemical high explosive. This could be done using materials and equipment that could be purchased at a hardware store and from commercial suppliers of scientific equipment for student laboratories.

"The key persons or persons would have to be reasonably inventive and adept at using laboratory equipment and tools of about the same complexity as those used by students in chemistry and physics laboratories and machine shops. They or he would have to be able to understand some of the essential concepts and procedures that are described in reactor technology and chemical explosives, and would have to know where to find these publications. Whoever was principally involved would also have to be willing to take moderate risks of serious injury or death."

- **J. Carson Mark** (former head of the Theoretical Division, Los Alamos Scientific Laboratory).⁽¹⁶⁾ "If one thinks of a small group wanting to build a bomb, and if one supposes that their primary requirement is that it give a nuclear yield (as to say, for example, 'the yield must be at least so much; but it is all right if it should turn out to be a few times larger'), then I think that such a device could be designed and built by a group of something like six well-educated people, having competence in as many different fields. As a possible listing of these, one could consider: a chemist or chemical engineer; a nuclear or theoretical physicist; someone able to formulate and carry out complicated calculations, probably requiring the use of a digital computer, on neutronic and hydrodynamic problems; a person familiar with explosives; similarly for electronics; and a mechanically-skilled individual. Among the above (possibly the chemist or the physicist) should be one able to attend to the practical problems of health physics which would arise. Clearly depending on the breadth of experience and competence of the particular individuals involved, the fields of specialization and even the number of persons could be varied, so long as areas such as those indicated were covered."
- **E. Zebroski and M. Levenson.**⁽¹⁷⁾ "Perhaps a more skeptical view of this possibility (of producing a crude nuclear weapon) would be by analogy to the ability of a reasonably well-informed technical person to sketch up a workable concept for a small jet propelled airplane, or medium sized computer. Given access to manufactured modules for most of the critical parts, construction of such a project by a small, dedicated group of artisans is conceivable. However, if the project must literally start from the raw materials in inconvenient chemical and physical form, and with very substantial hazards associated with handling and processing the materials, one obtains a rather different view of the probability of the 'garage operation weapon'."
- Secretary of Defense Harold Brown has said that "an atomic bomb could be fabricated within a year by perhaps 100 skilled workers with access to a machine shop and fissionable material."⁽¹⁸⁾ Brown has had wide experience in nuclear weapons design and fabrication at Lawrence Livermore Laboratory.

There is not as much difference among the foregoing views as there may initially appear to be. Willrich and Taylor assume that the adversary has approximately 10 kilograms of plutonium oxide and conclude that he could fabricate a crude nuclear device within several weeks. Levenson and Zebroski assume that the adversary must chemically process and refine his plutonium from some much less readily usable substance and point out the difficulties inherent in obtaining such readily usable materials as 10 kilograms of plutonium oxide. J. Carson Mark lists six different skills required, but does not deny that these skills could be gathered together in a group of fewer than six persons.

The following conclusions can be drawn from the preceding statements:

- To fabricate an illicit nuclear explosive requires a group of individuals with special knowledge, skills, and experience covering several technical fields
- To assemble persons with the qualifications and motivation to use a nuclear weapon would be difficult, but not impossible
- The difficulties involved in the job — the acquisition of significant quantities of a heavily guarded material and the hazards of working with the material — could deter such a group
- Safeguards increase the difficulty of acquiring material and hence reduce the likelihood that any attempt will be made

F.3.4 Dispersal of Radioactivity

The dispersal of radioactivity can be brought about by the detonation of a nuclear explosive device, the sabotage of a nuclear facility, an accident at a nuclear facility, or by a willful act of dispersion of radioactive materials. The consequences of dispersion of radioactivity due to accidents are studied during the design of a facility and reviewed by regulatory authorities during the licensing stage. The effects of accidents have also been studied in detail and discussed under "Radiological Impacts" (Section 7 and Appendix H).

The destruction caused by a nuclear explosion depends on the energy yield of the explosion and the time and place of its detonation. The yield of a particular device depends upon such factors as the amount of fissionable material, the design, and the quality of fabrication. It is difficult to predict the yield of an illicit nuclear explosive device.

The willful dispersion of radioactivity, especially the dispersal of plutonium, is generally mentioned as one of the potential threats by terrorist groups. Although there have been isolated, small-scale incidents, neither the technical nor psychological effectiveness of utilizing radioactivity dispersal has been demonstrated. In principle, there are many toxins available that would kill tens of thousands of people in the most vulnerable target areas in the United States. Some, such as botulism toxin and anthrax spores, are much more lethal than plutonium.^{(19),(20)}

Except in large doses, inhaled plutonium produces no discernible effect until the possible development of cancer 15 or more years later. Thus, instantaneous dramatic effects usually sought by terrorists are not possible.

Widespread effective dispersal of plutonium is difficult to achieve. In fact, dispersal efficiency was a major problem in animal experiments on plutonium toxicity. Only a narrow range of plutonium particle sizes is suitable — particles that exceed about 7 microns in diameter will not penetrate deeply into the lungs. Of the small particulates that do penetrate deeply, only about 30 percent will be deposited. The rest will be expelled by exhaling.^{(21),(22)}

Other studies suggest that dispersal of plutonium in the atmosphere would have relatively small consequences compared to other disasters that can be more easily induced in our complex society. For example, Cohen calculates⁽¹⁹⁾ that dispersal of 1 gram of reactor plutonium without

warning in a crowded football stadium could eventually cause two fatalities. If dispersal occurred on an average city street, the estimated fatality rate would drop to one per 15 grams of reactor plutonium if there were no warning and one per 150 grams with warning. It has been estimated⁽²³⁾ that plutonium dispersed in the ventilation system of a large building could be more effective, causing perhaps 70 eventual deaths per gram of reactor plutonium dispersed. Any of several toxins dispersed in this manner would be highly effective. But if the threat of such dispersal is to be employed for rational purposes, such as extortion, it must be accompanied by an announcement of some sort, and such an announcement would permit countermeasures to be taken that could reduce the hazards. Among countermeasures would be temporary shutdown of ventilation systems in the more likely target areas. Gofman⁽²⁴⁾ however, estimates that the cancer-causing dose of inhaled plutonium is 3,250 times smaller than that calculated by Cohen.

Experts disagree by orders of magnitude on some radiological effects. The results of prolonged observation of 25 Los Alamos employees who in 1944-45 inhaled plutonium aerosols (10 microcuries of plutonium were deposited in their lungs) support the lower hazard estimates. Cohen's calculations⁽¹⁹⁾ using the low range of effects found in the BEIR Report,⁽²⁵⁾ shows a 50 percent probability that one case should have developed to date within this group. There have been no cancers in this group. Studies of British uranium workers, uranium miners, metal miners, and fluorspar workers also support the lower estimate⁽¹⁴⁾ of carcinogenic effectiveness of plutonium inhalation.

REFERENCES FOR APPENDIX F

- (1) "A Systematic Approach to the Conceptual Design of Physical Protection Systems for Nuclear Facilities," Sandia Laboratories, HCP/DO789-01, May 1978.
- (2) "Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat," U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, NUREG-0414, May 1978.
- (3) "Analysis of Targets and Safeguards Systems at the BNFP (Unclassified Version)," Allied-General Nuclear Services, NMS-17, January 1978.
- (4) "Plutonium Transportation Vehicle Concepts, Descriptions, and Analysis," Allied-General Nuclear Services, NMS-15, December 1977.
- (5) Sandia Laboratories, "System Description for SECOM II — ERDA Digital Security Communications System," April 1975.
- (6) J.M. Montmollin, Sandia Laboratories, "Communications for Safeguarding Cargo," SLA-73-0822, September 1973.
- (7) "Barnwell Nuclear Fuels Plant Applicability Study — Vol. III Appendices," U.S. Department of Energy, DOE/ET-0040/3, March 1978.
- (8) ERDA Manual Appendix 7401-C.
- (9) "Coordinated Safeguards for Materials Management in a Nitrate-to-Oxide Conversion Facility," Los Alamos Scientific Laboratory, Los Alamos, New Mexico, LA-7011, April 1978.
- (10) "Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations — Volume 22 Nuclear Considerations for Repository Design," Science Applications, Inc., Y/OWI/TM-36/22, April 1978.
- (11) Brian Jenkins, "Will Terrorists Go Nuclear?" California Seminar on Arms Control and Foreign Policy, November 1975.
- (12) Roberta Wohlstetter, "Terror on a Grand Scale," *Survival* 18, No. 3, pp. 98-104, 1976.
- (13) Bernard L. Cohen, "The Potentialities of Terrorism," mimeographed, University of Pittsburgh, Pittsburgh, 1976.
- (14) "Terrorist Use of Gas Feared," *Washington Post*, p. 3, May 13, 1975.
- (15) *Op cit* (4). NB pp. 20-21.
- (16) Fred H. Schmidt and David Bodansky, "The Energy Controversy: The Role of Nuclear Power," University of Washington, Seattle, Washington, February 1975, (revised August 1975). NB pp. 59-60.
- (17) E. Zebroski and M. Levenson, "The Nuclear Fuel Cycle," *Annual Review of Energy* 1, pp. 101-130, 1976.
- (18) "Brown Says 100 Civilians Could Build A-Bomb," *The Washington Star*, p. 1, April 1, 1977.
- (19) Bernard L. Cohen, "The Hazards in Plutonium Dispersal," University of Pittsburgh, Pittsburgh, July 1975.
- (20) J. H. Rothchild, *Tomorrow's Weapons*, McGraw Hill, New York, 1964.
- (21) Task Group on Lung Dynamics, "Deposition and Retention Models for Internal Dosimetry of the Human Respiratory Tract," *Health Physics* 12, pp. 173-207, 1966.

- (22) R. F. Phalen and O.G. Raabe, "Aerosol Particle Size as a Factor in Pulmonary Toxicity," Proceedings of the Fifth Annual Conference on Environmental Toxicology, AMRL-TR-74-125, Paper No. 23, p. 353, Lovelace Foundation for Medical Education and Research, 1974.
- (23) *Op cit* (23), *passim*.
- (24) John W. Gofman, "The Cancer Hazard From Inhaled Plutonium," *Congressional Record — Senate*, S-14610-S-14620, July 31, 1975.
- (25) National Academy of Sciences and National Research Council, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Report of the Advisory Committee on the Biological Effects of Ionizing Radiations, Washington, D.C., 1972.

Appendix G

TECHNICAL CONSIDERATIONS FOR LONG-TERM CRITICALITY

TABLE OF CONTENTS

	Page
G.1 Characteristics of Fissile Material in the Waste	G-2
G.2 Critical Mass Data	G-9
G.3 Thermal Effects in a Repository	G-10
G.4 Geological Mechanisms and Factors Affecting Criticality Conditions in the Repository	G-11
G.4.1 Intrusion of Liquids	G-11
G.4.2 Canister Failure	G-13
G.4.3 Fuel Assembly Failure	G-13
G.4.4 Destruction of Waste Matrix	G-13
G.4.5 Fissile Material Separation and Transport	G-13
G.4.6 Fissile Material Reaccumulation	G-15
G.4.7 Time Factors	G-16
References for Appendix G	G-17

Appendix G

TECHNICAL CONSIDERATIONS FOR LONG-TERM CRITICALITY

Within a repository several natural events, each with its own small probability of occurrence, are required to provide the conditions necessary for a criticality. In addition, certain of these mechanisms are time dependent and/or must occur in sequence. The probability of these events occurring naturally in such a way as to yield the conditions necessary to cause a criticality is judged to be diminishingly small. This appendix, however, discusses the technical considerations that evolve from the fact that a critical mass is theoretically achievable with the fissile material in the waste. It does not attempt to address the probability of ever achieving a criticality, but rather analyzes:

1. The fissile characteristics of the waste as a function of time.
2. What is available to form a critical mass from the waste and the conditions under which a criticality can be achieved.
3. Effect of temperature in the repository.
4. Geological factors affecting criticality.

G.1 CHARACTERISTICS OF FISSILE MATERIAL IN THE WASTE

The mass data used to assess the potential for a criticality during long-term isolation is based on the following initial waste forms:

- Spent uranium oxide fuel in Cases 1 and 2
- Spent uranium oxide fuel complexed in glass for Case 3
- High level waste based on 2/3 uranium oxide and 1/3 mixed oxide spent fuel complexed in glass for the Reference Case

Fissile masses for each study case are derived from the waste canister data in Table G-1. The uranium and transuranic isotopic content of a uranium oxide fuel assembly at various times after it is removed from a reactor is shown in Table G-2. The fissile enrichment for 10-year decayed spent fuel after burnup to 33,000 MWD/MT is 1.43 percent and is based on the mass of ^{235}U , ^{239}Pu , ^{241}Pu , ^{242}Am , ^{243}Cm , and ^{245}Cm .⁽¹⁾ This enrichment is used for the spent fuel description in each case.

Table G-1
WASTE CANISTER DATA FOR THE SFDS STUDY CASES

Case	Fissile Form	Canister Description	Canister Contents
Reference	Vitrified HLW		2.1 MTHM equivalent ^(a)
1	Spent fuel assemblies ^(b)		
	PWR - 1	14" dia.	0.460 MTHM
	BWR - 3	16" dia.	0.550 MTHM
2	Voloxidized fuel ^(b)	12" dia. x 10' long	0.620 MTHM
3	Vitrified fuel ^(b)	20" dia. x 10' long	0.386 MTHM

(a) Actinide content based on fuel cycle comprised of 2/3 UO_2 and 1/3 MOX fuel elements. The vitrified actinides with 0.5 percent of original uranium and plutonium remaining.

(b) Based on spent uranium oxide fuel.

Table G-2

**URANIUM AND TRANSURANIUM CONTENT OF LWR SPENT FUEL ASSEMBLY
FOLLOWING REMOVAL FROM THE REACTOR (Kilograms)^(a)**

	Shutdown	10 Yr	10 ² Yr	10 ³ Yr	10 ⁴ Yr	10 ⁵ Yr
U	440.1	440.7	440.8	441.0	442.1	444.2
234	0.07	0.0799	0.1099	0.135	0.135	0.111
235	3.6	3.71	3.72	3.78	4.32	6.04
236	2.1	1.89	1.90	1.99	2.54	2.90
238	435.2	435.0	435.1	435.1	435.1	435.1
²³⁷ Np	0.22	0.215	0.283	0.678	0.798	
Pu	4.54	4.14	3.74	3.56	2.47	
238	0.071	0.06	0.03	0.00004	0.0	
239	2.8	2.49	2.48	2.42	1.90	
240	1.0	1.03	1.02	0.931	0.370	
241	0.47	0.358	0.005	0.0	0.0	
242	0.2	0.205	0.205	0.204	0.202	
Am	0.115	0.276	0.562	0.162	0.0167	
241	0.07	0.235	0.521	0.124	0.0	
242	0.0003	0.00033	0.002	0.0	0.0	
243	0.045	0.0411	0.0408	0.0375	0.0167	
Cm	0.0136	0.00784	0.000363	0.000118	0.00005	
242	0.0003	0.0	0.0	0.0	0.0	
243	0.00005	0.000024	0.0	0.0	0.0	
244	0.011	0.00770	0.000237	0.0	0.0	
245	0.0022	0.00011	0.000113	0.000106	0.00005	
246	0.0	0.00001	0.000013	0.000012	0.0	

(a) Source: Calculations using ORIGEN code as reported by Cohen⁽¹²⁾ and Allen⁽¹⁾.

When the fuel assembly is used in a reactor, the isotopic composition of the fuel changes significantly, as shown in Figure G-1. The inventory of ²³⁵U is reduced from 15 to about 4 kg. In addition, some 11 kg of the ²³⁸U is consumed to produce about 10 kg of ²³⁹Pu. By the end of core life, about 2.8 kg of fissionable ²³⁹Pu remain and 0.5 kg of ²⁴¹Pu (also fissile) will be present. Figure G-2 shows the effect of burnup on the neutron multiplication of uranium oxide in a water moderated system. The fuel rod data include the buildup of the fission product poisons. The cylinder curves indicate the effect of fissile material depletion only.⁽²⁾ Over the life of the fuel assembly, more than half as many fissions originate from plutonium as from ²³⁵U. Because of its larger fission cross-section, plutonium is fissioning at almost twice the rate of ²³⁵U at the end of core life. The net effect is to reduce the total fissile content of the fuel assembly and to add neutron poison materials as the amount of irradiation (expressed as burnup) is increased. The design basis burnup assumed for spent fuel in the SFDS is 33,000 MWD/MT. The discussions to follow include comparisons between the extremes of no burnup (3.5 percent enriched ²³⁵U) and high burnup at 33,000 MWD/MT.

Comparisons of the criticality potential of various storage circumstances are generally made on the basis of the effective multiplication constant, k_{eff} . A critical system will have a k_{eff} equal to 1. The effect of the changes caused by burnup on the potential for criticality can be estimated, neglecting the neutrons produced by fast fission and the neutrons absorbed in the process of thermalizing. (Both of these effects relate primarily to the ^{238}U content whose inventory changes little over the life of the core.) To this approximation, changes in k_{eff} can be represented by changes in the product of the number of neutrons produced by fission in the fuel and the fraction of all neutrons which are absorbed in the fuel. This product, when evaluated at the beginning of life, is about 1.67. When 3.3 kg of fissionable plutonium has been substituted for 11.5 kg of fissionable uranium at the end of life, the product is about 1.69. Therefore, if the poison effect of fission products and ^{240}Pu is neglected, the criticality potential of spent fuel is approximately equal to that of the fresh fuel. If the fission products could be selectively washed out of the fuel elements, the spent fuel could approach criticality in essentially the same circumstances as new fuel.

The prime ingredient needed in a critical reaction is the presence of a critical mass of fissile isotopes. The amount of total mass changes over geologic time with the radioactive decay of the actinide elements. Figure G-3 shows the decay chain of the actinide elements present in spent reactor fuel. (Elements heavier than mass number 244 have not been included, as they are not present in sufficient quantity in spent reactor fuel to affect the criticality potential.) The minimum isotopic mass for the actinides required for a critical condition are summarized in Table G-3.⁽³⁾ An inspection of the table and figure shows certain production/decay changes which are important in the long-term criticality assessment. Note, for example, that ^{239}Pu , with a significant thermal fission probability, decays to ^{235}U , another thermal system fissile isotope. This implies that the criticality potential of plutonium remains relatively constant throughout long geological time periods unless chemical separation mechanisms are introduced into the storage array.

A single unirradiated and unrodded PWR fuel assembly (4% enriched) has a k_{eff} of about 0.9 when moderated and reflected by water.⁽⁴⁾ However, the potential of forming a more critical geometry than the original fuel during geological time spans must be considered. If the form of the assembly were destroyed over a period of time, as could occur through prolonged oxidation at high temperatures, it is possible that the fuel material could be redistributed. The effect of changing the geometry can be estimated by comparing the criticality of a cylinder (33 cm in diameter by 381 cm long) with that of a sphere having a radius of 25 cm. The sphere has a volume of only about one-tenth that of the cylinder but would have an equivalent potential for criticality. This results from the fact that neutron leakage from a spherical configuration is significantly less than from a long, narrow cylinder. Hence, under conditions at or near optimum moderation, it is theoretically possible for the fissile material in a single canister of spent fuel to produce a k_{eff} of 1.05 if isolated, and slightly higher ($k_{eff} = 1.06$) if reflected with a few feet of water.

In the context of long-term storage, additional questions arise regarding the criticality effects associated with preferential leaching which could remove nuclear poisons or enhance concentration of fissile material. Such effects can occur either in the immediate vicinity of the emplaced canister or at a different location following migration of either fissile isotopes or their oxide forms. This is addressed in Section G-4 of this appendix.

Figure G-1.
Summary of Isotopic Changes in LWR
Fuel Assembly During Burnup

Figure G-2.
Neutron Multiplication vs Burnup on Rods and Assemblies
 Source = Heaberlin and Selby⁽²⁾

Figure G-3.
Radioactive Production and Decay Chain
of the Actinide Elements

Table G-3
COMPUTED OR MEASURED MINIMUM CRITICAL MASS
FOR FISSIONABLE NUCLIDES^(c)

Nuclide	Spherical, Thermal Neutron Optimum Moderation		Spherical, Fast-Neutron Unmoderated Systems		
	Unreflected, kg	Water Reflected, kg	Unreflected, kg	Water Reflected, kg	Steel Reflected, kg
U-233	1.2	0.59	16.5	7.3	—
U-235	1.5	0.82	49.0	22.8	—
U-236	(a)	(a)	(a)	(a)	(a)
Np-237	(a)	(a)	68.6	64.9	43.1
Pu-238	(a)	(a)	7.2	5.6	4.5
Pu-239	0.905	0.53	10.0	5.42	—
Pu-240	(a)	(a)	158.7	148.4	96.4
Pu-241	(b)	0.26	(b)	6.0	—
Am-241	(a)	(a)	113.5	105.3	71.4
Am-242	(b)	0.023	(b)	(b)	(b)
Am-243	(a)	(a)	(b)	(b)	(b)
Cm-243	(b)	0.213	(b)	(b)	(b)
Cm-244	(a)	(a)	23.2	22.0	14.2
Cm-245	(b)	0.042	(b)	(b)	(b)
Cm-247	(b)	0.159	(b)	(b)	(b)
Cf-249	(b)	0.032	(b)	(b)	(b)
Cf-250	(a)	(a)	(b)	(b)	(b)
Cf-251	(b)	0.010	(b)	(b)	(b)
Cf-252	(a)	(a)	(b)	(b)	(b)

(a) Neutron chain reaction not possible.

(b) No question concerning possible criticality, but no calculations are known to have been made.

(c) Source: E. D. Clayton⁽³⁾

G.2 CRITICAL MASS DATA

E. J. Allen of ORNL calculated the minimum quantity of each of the isotopes in spent fuel which could become critical.⁽¹⁾ Repeating the calculation as a function of time produced a minimum critical mass for each of several points in time. By dividing the quantity of each isotope present in spent fuel by the corresponding minimum critical mass of that isotope in water, Allen calculated the maximum number of critical masses of that isotope contained in a single fuel assembly. The study showed that, except for plutonium, the maximum number of critical masses of any element contained in a spent fuel assembly in water at any time is less than 0.01. Similar studies were repeated to determine the maximum number of critical masses of each element for each time period when dispersed with an optimum concentration in brine, salt, and sand. The results were similar in that, except for plutonium, less than 0.02 critical masses of any element was present at any time.

A study of Allen's work shows that the criticality potential of spent fuel increases gradually to about 10^4 years. The primary reason for this is that radioactive decay causes a small increase in the ratio of fissile isotopes present to those that act as a neutron poison. This is predominantly due to the near constant inventory of the fissile isotope ^{239}Pu (from the decay of ^{243}Pu and ^{243}Am) during this period, coupled with the decrease in isotopes like ^{240}Pu that act as large neutron absorbers. In the unlikely event that the plutonium in a spent fuel assembly could become separated from the fission products, uranium, and other transuranic elements, and then redistributed in a spherical configuration in water at an optimum concentration, only about half the plutonium content of one fuel assembly would be required to produce a criticality. Under these same conditions, except with the material distributed in salt instead of water, the plutonium contained in about five fuel assemblies would be required to produce a criticality.

Note that no reference to uranium has been made relative to criticality. This omission is a consequence of Allen's conclusion that no quantity of uranium having a residual enrichment of 0.8 percent or less can be made critical in a homogeneous assembly with any of the moderating/reflecting materials considered. Allen also calculated the criticality of a mixture of all the elements in Table G-2, including uranium, and found that arbitrarily large quantities of the mixtures appropriate to any time could not be made critical in any of the media studied. This conclusion appears to conflict with the possibility that several critical masses of the plutonium and uranium mixture might exist in a single fuel assembly. This apparent conflict can be resolved by noting that the fission reaction in actinides other than uranium and plutonium is a fast spectrum reaction, so that they only detract from the criticality of a moderated assembly.

G.3 THERMAL EFFECTS IN A REPOSITORY

Thermal output on the order of a kilowatt per fuel assembly heats up the surrounding storage medium. Since about one percent of this power still persists after thousands of years, thermally induced effects can be of long duration. Even if the ensuing temperatures are not high enough to threaten the structural integrity of the assembly or storage media, they have the potential for accelerating chemical degradation. The increase in temperature could cause water held in nearby salt to migrate toward the waste. Water volumes attracted to a waste canister over decades of exposure have been estimated by others to vary from negligible to as high as tens of liters. This suggests that enough water might be available to cause corrosion. Corrosion is enhanced in the presence of chlorine, particularly if brine can produce hydrochloric acid in a radiation environment. Water may also contribute to the eventual breaching of protective canisters which may have enclosed the waste and the fuel assembly cladding.

Higher temperature may also lead to a greater susceptibility of the salt to creep. Although the mean density of the fuel assembly is comparable to most media that would be considered for disposal (3 g/cm^3), each assembly weighs about 500 kg and rests on a base of less than 500 cm^2 . The higher temperatures, high thermal gradients, and the addition of the moisture could conceivably result in very gradual downward migration of the canister. Vertical movement, if it should occur, might introduce the remote possibility of the canister moving into a different geological stratum. If this new stratum contains liquid such as water or petroleum, the potential for criticality would be increased. Vertical movement also suggests the possibility of some lateral movement, which could also increase the potential for neutron interaction between canisters.

G.4 GEOLOGICAL MECHANISMS AND FACTORS AFFECTING CRITICALITY CONDITIONS IN THE REPOSITORY

As discussed previously, the design and operation of the repository include positive controls to prevent criticality. However, when the repository is sealed, these positive controls are no longer provided and the fissile material is subjected to long-term geological effects. These long-term effects may result in both physical and chemical changes to the fissile material. Mechanisms and factors that can affect the criticality potential of the repository over geological time periods include:

- Intrusion of liquids
- Canister failure
- Fuel assembly failure
- Destruction of waste matrix
- Fissile material separation and transport
- Fissile material reaccumulation
- Time factors

G.4.1 Intrusion of Liquids

In general, intrusion of liquids will tend to thermalize the neutrons which are released from the wastes. High energy neutrons originating from actinides in the waste through such mechanisms as spontaneous fission and alpha neutron reactions will decrease in energy through collision with the hydrogen in water. Certain of the fissile nuclides, such as ^{235}U , ^{239}Pu , and ^{241}Pu , fission with thermal neutrons. Other actinides, specifically ^{238}U and ^{242}Pu , will not sustain a neutron chain reaction under any set of conditions and their presence acts as nuclear poison in either a moderated or unmoderated system. Several actinide isotopes such as ^{237}Np , ^{238}Pu , ^{240}Pu , ^{241}Am , and ^{244}Cm will not sustain a chain reaction via thermal neutrons (i.e., in the presence of water).

Water intrusion could occur by means of:

- Failure of shaft seals or seal grout in exploratory drilling
- Subsequent human intrusion
- Natural erosion
- Faulting of capping overburden (i.e., seismic activity)

G.4.1.1 Seal Failure. The failure of a borehole seal could permit groundwater or water from an overlying aquifer to enter the repository. To evaluate the integrity of borehole seals already in common use, a study was made of failures that have been experienced by the oil and gas industry in the United States. It was found⁽⁵⁾ that the probability of failure of seals on a typical oil well was 10^{-4} per year. Considering the importance of long-term closure of drill holes penetrating the overburden above a repository, more engineering will have to be invested in sealing a waste repository than typically is invested in sealing an oil well. Proper application of additional engineering should reduce this failure rate. It is even more difficult to predict the incidence of failure with time. As seals age, physical and chemical changes may accelerate failure. Alternatively, the seals may improve with age if the pressure of the rock overburden causing differential loading between the seal and the surrounding formations decreases as a result of natural phenomena.

G.4.1.2 Human Intrusion. The possible occurrence of drilling or excavating in the vicinity of a repository after backfilling and decommissioning must always be considered. Such activity could offer a potential pathway for groundwater to reach the material emplaced in the repository. Human

intrusion of this type can be avoided over a reasonable period of time (e.g., 100 to 200 years) by administrative controls. It could also be offset to a considerable degree by selecting sites which do not offer attractive potential areas for mining or other geologic exploration.

G.4.1.3 Natural Erosion. Natural erosion does not have a credible potential for removing rock overlying the waste and exposing it to groundwater. Erosion rates determined⁽⁶⁾ in the vicinity of one potential burial site are a few centimeters per 1,000 years. It would require millions of years after burial to expose the waste to groundwater by natural erosion. One thousand feet of erosion would require approximately 100 million years to accomplish.

G.4.1.4 Seismic Activity. The possibility of a fault offset causing a fracture through the repository horizon which could provide a direct connection to waterbearing strata is an alternative scenario for introducing water to the region in which the wastes are stored. Although prediction of the location and magnitude of earthquakes (and other improbable phenomena) is difficult and uncertain, one would expect such occurrences to be rare in a region chosen to specifically avoid severe natural phenomena.

G.4.1.5 Water Intrusion. Pure water is not likely to be present after intrusion into the repository because soluble salt will dissolve to form a saturated brine solution. To be conservative in this study, however, the impact of pure water was considered as well as brine.

As long as a canister remains intact and fissile material cannot redistribute into more favorable geometry and moderation condition, a single canister of waste in any of the cases cannot form a critical mass, even when the canister is surrounded by water. Water can only contribute indirectly to enhancing the potential for criticality by:

- (1) increasing the rate of corrosion of the sleeve, canister, and fuel clad (Case 1),
- (2) acting as a chemical leachant to gradually dissolve materials from the waste matrix and potentially cause matrix failure, and
- (3) acting as a transport medium for leached fissile materials which could agglomerate elsewhere in a more concentrated mass.

Even though such naturally occurring events are not probable, water must always be present to moderate the neutrons if a criticality is to be considered remotely possible.

G.4.1.6 Brine Intrusion. Heat produced by radioactive decay elevates temperatures of the salt surrounding the high-level waste. Because the solubility of salt in water increases with temperature, brine inclusions in the salt might move up the temperature gradient, toward the waste canisters. This inward movement of brine tends to inhibit transport of actinides away from the canisters. Such movement might permit the reaccumulation of the actinides which would raise the potential for criticality. The presence of brine pockets, however, would not significantly affect the criticality assessment of a repository.

Brine is a less efficient moderator/reflector of neutrons than water. Thus, a larger quantity of fissile material would have to be present to reach criticality with brine as opposed to water. Brine intrusion would probably hasten barrier failure compared to water alone, but the overall potential for criticality would be limited by preferential leaching, transport, and concentration phenomena. The exception would be Cases 1 and 2, in which the fuel pellets could form a critical mass if they are free to move and a mechanism is available to move the materials into a near spherical shape free of neutron poison. Normally, there are sufficient nuclear poisons present in the spent fuel pellets to prevent them achieving criticality.

G.4.2 Canister Failure

After several decades of operation, the repository is backfilled and sealed. The decay heat continues to elevate temperatures in the vicinity of the waste and pressures begin to reestablish hydrostatic equilibrium with the local geology. The temperature and pressure environment, together with the anticipated radiation field, accelerate corrosive effects which could cause failure of the canister. The canister, as a confinement barrier, is presumed to have corroded away within a century or so. Hence, canister failure should be assumed to occur during long-term storage.

G.4.3 Fuel Assembly Failure

Fuel assembly cladding in Case 1 provides an additional confinement barrier to surrounding the fuel materials. In terms of geologic time, this barrier would only survive slightly longer than the canisters. Failure of cladding could permit the mechanical displacement of the fuel pellets.

G.4.4 Destruction of Waste Matrix

This mechanism considers changes in the initial fissile matrix, either sintered oxide or vetrified glass. A failure of the matrix holding the waste permits increased concentration of the fissile elements or loss of poisons such that a higher criticality potential is obtained. For Cases 1 and 2, destruction of the sintered oxide matrix would initially tend toward a lower criticality potential since the homogeneous powder or solution is less reactive than the heterogeneous pellets in the presence of water. However, the homogeneous form is more susceptible to physical and chemical changes associated with the removal of fission product poisons from the actinides or preferential separation of various fissile actinides. These factors provide a higher criticality potential although the mechanisms for causing such changes have a low probability of occurrence.

In the Reference Case, if water is present, the canister has failed, and the waste matrix has failed, other factors would still have to occur before criticality conditions exist. The fissile materials (all actinides) from 30-40 canisters of waste would have to be transported into a more favorable geometry. If curium were to be preferentially leached from the failed canister and transported as a pure substance to form a more favorable (i.e., near optimum) concentration and geometry, then a single canister could theoretically provide sufficient material for a critical mass.

In Cases 1, 2, and 3, if water is present, the canister has failed and the waste matrix has failed, criticality would also require element separation, transport, and reassembly into a near optimum concentration and geometry.

The American Physical Society study group⁽⁷⁾ concluded that leachability of the available glass *may* (emphasis theirs) provide protection against dissolution and release of radionuclides by immobilizing high-level wastes on time scales of a thousand years or more. They suggest that uncertainties preclude reliance on the durability of the glass itself as the principal barrier. Accordingly, studies in support of the Waste Isolation Pilot Plant (WIPP) safety analysis assumed that, upon intrusion of water, the wastes will dissolve at the same rate as the surrounding salt. If destruction of the waste matrix is related to the leaching of the elements from the matrix this change in either the fuel or the glass will occur between several thousand and several million years.

G.4.5 Fissile Material Separation and Transport

Fissile material separation and transport can lead to criticality via two distinct but improbable mechanisms:

- Leaching and transport of poisons and other materials away from the canister site so that criticality is possible within the fissile materials remaining behind
- Leaching and transport of fissile materials from one or more canisters and accumulating into a critical mass at another location

Depending upon the geological time frame in which the destruction of the waste matrix occurs, critical masses of plutonium are available in a single waste canister in Cases 1, 2 and 3 and with curium from a single canister of high-level waste in the Reference Case.⁽¹⁾ However, radiological decay will reduce the curium content of the Reference Case waste below the critical mass in approximately 10,000 years, and the plutonium content of Cases 1, 2, and 3 below the critical mass in approximately 100,000 years. Beyond these time periods, the only possible criticality scenario requires the accumulation of fissile materials from several waste canisters. For any of the above conditions to occur, preferential leaching and transport must occur.

G.4.5.1 Leaching. Tests performed at Hanford on both HLW glass and on irradiated fuel pellets in water reported selective leaching of various elements at different rates.^{(8),(9)} These rates are shown for comparison in Table G-4.

Table G-4
LEACH RATES FROM HLW

Element	Leach Rate (g/cm ² -d)	
	Glass	Irradiated Fuel
¹³⁷ Cs	2.5 x 10 ⁻⁵	9 x 10 ⁻⁶
⁹⁰ Sr	2 x 10 ⁻⁵	1 x 10 ⁻⁶
^{239,240} Pu	1.5 x 10 ⁻⁶	3 x 10 ⁻⁷
^{242,244} Cm	1 x 10 ⁻⁷	4 x 10 ⁻⁹

These data are for leach rates in demineralized water at room temperature. Both experiments indicate that the leach rate decreases with time. The glass experiments were reported up to 182 days and the fuel experiments up to 210 days. These times are diminishingly short compared with the geological times of interest. However, as longer term experimental data are obtained, steady-state leach rates for both systems may be lower than the values shown in Table G-4. How well the reported leach rates can be correlated with a general reduction of the waste matrix itself is not obvious. The leach rate of the waste form and corrosion of the canister may be affected by the radiation field, elevated temperatures and pressures, and the water present. Actual environments to be experienced in the salt repositories will have to be used to predict anticipated leach rates for the selected site. Those leach rates will vary with such properties as mineral content of water, temperature, radiation effects and homogeneity of waste.

G.4.5.2 Transport in Solution. In considering migration of the isotopes away from the burial site, the solubility of the selected elements in the transport medium is important. Opposing this transport is a sorption interaction with the geological medium which, because of a chemical affinity quite comparable to the dissolving process, inhibits motion of the migrating isotopes.

This process is comparable to the performance of an ion exchange column. As sorption occurs, the migration velocity of the elements is reduced to a fraction of the velocity of the transport medium. The empirical quantity that describes this sorption process is the distribution coefficient, K_d . This coefficient can be used only if the considered ion is present in trace quantities, the reaction is reversible, and near-equilibrium conditions exist. The coefficient is defined as:

$$K_d = \frac{\text{Mass of ion sorbed on solid/mass of solids}}{\text{Mass of ion dissolved in liquid/volume of liquid}} = \frac{m}{g}$$

The value for K_d is determined experimentally. A dilute solution of the ion of interest is permitted to reach equilibrium with the absorbing material and the ratio of the concentration measured. Efforts to generate geochemical data, including values for K_d , are continuing.⁽¹⁰⁾ The experiments show that there is a significant range of values for K_d depending on the nuclide being considered.

The effect of the sorption on the velocity of the ion through a medium is given by the retardation factor, K_f :

$$K_f = 1 + \frac{\rho}{\theta} K_d$$

Where:

ρ = the density of the medium

and θ = the porosity of the medium

This factor is the ratio of the groundwater velocity to the ion velocity and is seen to have a minimum value of 1.

G.4.6 Fissile Material Reaccumulation

The above discussion indicates that transport through geological media and selective element separation by sorption interaction with the geomedia is possible. However, the extent to which these naturally occurring mechanisms may accumulate selected elements into a critical configuration of proper geometry, concentration, composition, etc., and the time frame within which these mechanisms could occur is judged to be diminishingly small.

It is worth noting the studies of the Oklo phenomenon in Gabon, Africa, a natural reactor that "operated" approximately 2 billion years ago. It was estimated that the critical reaction continued with water flowing through the formation for about 10^5 to 10^6 years. Studies of the Oklo site indicate that the rare earth elements (lanthanides and actinides) were essentially retained within the reactor operating zones.⁽¹¹⁾ This indicates that selective migration, especially of the actinides, was negligible. Hence, the probability of migration and reaccumulation of these fissile materials seems even more remote.

Another type of transport can occur as whole waste forms or intact canisters are displaced to a common location. For example, it is theoretically possible for multiple canisters to form a critical mass in the Reference Case if those canisters were to move into an optimum close-packed orientation free of neutron poisons. Over 40 waste canisters would have to be brought together for this criticality mechanism to occur. Any other conditions for criticality in the Reference Case or other cases would require canister failure coupled with material transport and concentration.

G.4.7 Time Factors

Some events have a greater probability of happening over a geologic time scale than others. Certain critical factors can only occur in sequential order if the conditions for criticality are to be met. Some single event may have a reasonably high probability of occurrence, yet not follow a prerequisite sequence because other events will have a relatively low probability of occurrence.

In Table G-5, individual mechanisms are arranged in three groups relative to their probability of occurrence with geologic time. The three groups are those with some, little, or very little probability of occurring relative to time.

Table G-5

**SORTING OF INDIVIDUAL MECHANISMS CONTRIBUTING TO
CRITICALITY POTENTIAL OVER GEOLOGIC TIME**

Probability	Mechanism
Some (Following Water Intrusion)	Brine Intrusion
Little	Water Intrusion
	Assemble Critical Mass (Cases 1, 2, and 3)
Very Little	Assemble Critical Mass (Reference Case)
	Multiple Canister Displacement to Common Location
	Waste Matrix Failure and Specific Element Leaching
	Transport, Redeposition, and Concentration of Elements of Materials

The difference in rankings for the Reference Case and Cases 1, 2, and 3 under "Assemble Critical Mass" category result from the lack of plutonium consideration of the more rapid decay of fissile curium in the Reference Case compared to ^{239}Pu and ^{235}U in other cases.

The probabilities grouped as very little may not occur over geologic time because the contributing mechanisms are not apparent. For example, to bring multiple canisters into a close-packed array would require atypical forces which cannot be reasonably postulated.

REFERENCES FOR APPENDIX G

- (1) E. J. Allen, "Criticality Analysis of Aggregations of Actinides from Commercial Nuclear Waste in Geological Storage," ORNL-TM-6458, Oak Ridge National Laboratory, Oak Ridge, Tennessee, August 1978.
- (2) S. W. Heaberlin and G. P. Selby, "Effect of Fissile Isotope Burnup on Criticality Safety for Stored Disintegrated Fuel Rods," PNL 2589, Pacific Northwest Laboratory, Richland, Washington, September 1978.
- (3) E. D. Clayton, "Anomalies of Criticality," BNWL-SA-4868, Battelle Pacific Northwest Laboratory, Richland, Washington, 1974.
- (4) "Demonstration and Conditions for Document 778, SNM-778, Docket No. 78-824," Babcock and Wilcox, Lynchburg, Virginia, 1978.
- (5) K. J. Schneider and A. M. Platt, "Advanced Waste Management Studies, High Level Radioactive Waste Disposal Alternatives," BNWL-1900, Battelle Pacific Northwest Laboratory, Richland, Washington, May 1974.
- (6) F. Gera and D. G. Jacobs, "Considerations in the Long-Term Management of High-Level Radioactive Wastes," ORNL-4762, Oak Ridge National Laboratory, Oak Ridge, Tennessee, February 1972.
- (7) "Report of the Study Group on Fuel Cycles," *Review of Modern Physics*, 50, No. 1, Part II (1978).
- (8) J. E. Mendel, et. al., "Annual Report on the Characterization of High-Level Waste Glasses," BNWL-2252, Battelle Pacific Northwest Laboratory, Richland, Washington, June 1977.
- (9) Y. B. Katayama, "Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Groundwater," BNWL-2057, Battelle Pacific Northwest Laboratory, Richland, Washington, July 1976.
- (10) R. G. Dosch and A. W. Lynch, "Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico," SAND 78-0297, Sandia Laboratories, Albuquerque, New Mexico, June 1978.
- (11) G. A. Cowan, "A Natural Fission Reactor," *Scientific American*, 235, 36 (1976).
- (12) B. L. Cohen, "High Level Radioactive Waste from Light Water Reactors," *Review of Modern Physics*, 49, 1 (1977).

Appendix H

TECHNICAL CONSIDERATIONS FOR THE ASSESSMENT OF RADIOLOGICAL IMPACT

TABLE OF CONTENTS

	Page
H.1 Technical Issues	H-2
H.1.1 Operating Phase Exposure	H-2
H.1.2 Long-Term Risks	H-3
H.1.3 Regulatory Controls	H-4
H.1.4 Repository Integrity and Waste Retrievability	H-6
H.1.5 Radiological Impact Assessment Methodology	H-6
H.1.6 Population Considerations	H-7
H.1.7 Public Perception of Risks	H-7
H.2 Evaluation Criteria	H-8
H.2.1 Near-Term Public Health and Safety and Environmental Impact	H-8
H.2.2 Radiation Protection to Operators	H-8
H.2.3 Long-Term Public Health and Safety and Environmental Impact	H-11
H.2.4 Accident Potential	H-14
H.2.5 Criticality Potential	H-16
H.2.6 Retrievability Potential	H-16
H.3 Considerations Affecting Radiological Impact Assessments	H-17
H.3.1 Waste Types and Inventories	H-17
H.3.2 Radionuclides in Waste	H-17
H.3.3 Radionuclide Release Mechanisms	H-18
H.3.4 Biological Response	H-19
H.3.5 Measurement of Radiological Impact	H-19
H.4 Radiological Impact Modeling	H-21
H.4.1 Generalized Approach to Assessment of Radiological Impact	H-21
H.4.2 Specific Radiological Impact Models	H-21
References for Appendix H	H-27

Appendix H

TECHNICAL CONSIDERATIONS FOR THE ASSESSMENT OF RADIOLOGICAL IMPACT

The radiological and long-term effects of nuclear waste disposal have been researched in numerous government and industry studies, and are the subject of many existing technical documents. Several of these (1)-(10) were consulted frequently in developing the basis for, and in conducting, the radiological impact assessment for the Spent Fuel Disposal Study (SFDS). This Appendix summarizes some of the bases and other information developed in support of the SFDS radiological impact assessment, which appears in Section 7 of Volume 2.

H.1 TECHNICAL ISSUES

Several key technical issues must be considered to determine impacts related to facility design and operation and to establish ALARA exposure levels. In an attempt to assess the radiological impact in a systematic manner, the assessment study was divided into seven main areas of investigation as listed below:

- Operating phase exposure
- Long term risks
- Regulatory controls
- Repository integrity and waste retrievability
- Radiological impact assessment methodology
- Population considerations
- Public perception of risks

A detailed discussion of these issues follows.

H.1.1 Operating Phase Exposure

The radiological safety considerations during the operating phase of nuclear waste disposal facilities can be categorized as follows:

- The impact of radiation safety considerations on the design of the above ground processing and handling facilities and the underground repository facilities (the design aspects affected by radiation safety are briefly discussed in Subsection H.1.1.1)
- The impact of radiation exposure to operators at a reprocessing facility during transportation and upon receipt of nuclear waste at the repository, and the impact on the public during normal operation, maintenance and decommissioning of related facilities
- The impact of radiation exposure to operators and maintenance personnel, the public, and the external environment due to accidents or abnormal events which potentially could occur in the facilities

The consequences of radiological impacts from normal operation and abnormal events during the operating phase of the facilities are discussed below.

H.1.1.1 Normal Operation. Normal operation exposure mechanisms must be considered in two categories; (1) exposure to facilities personnel and (2) exposure to the public.

Exposure mechanisms under this category may include airborne radionuclide inhalation or direct radiation exposure emitted from waste storage, handling, transportation or processing systems at the various facilities.

Appropriately engineered safety features including ventilation, filtering, decontamination, and monitoring systems should prevent airborne radionuclides from exceeding acceptable concentration levels within the facilities. Similarly, it is assumed that appropriate shielding and radiation area accessibility controls have been developed and will be implemented to maintain normal operating exposure levels to personnel to within the regulatory requirements. Such features are well established in the nuclear power industry and, to a limited extent, the reprocessing industry.

To restrict exposure to the public during normal operation of the facilities, the regulations and licensing requirements have specified extensive safety systems and procedures to be implemented to prevent accidental releases of airborne or other radioactivity. These regulations are summarized in a later section.

As a result of the regulations and engineering features controlling exposure, the radiological impact during normal operation is expected to be within the acceptable range.

H.1.1.2 Abnormal Events. Perhaps the most significant issue of concern regarding radiological impacts is that of unplanned (or accidental) events which may cause releases to the external environment. Even though nuclear facilities are designed to achieve essentially zero release, the potential for an unplanned release due to an accident or natural causes, although of low probability, does exist in waste disposal. To minimize the potential, the aboveground facilities, including AFRs and P/E facilities, will be designed to withstand events such as tornadoes, earthquakes, floods and other natural disasters. Similarly, nuclear waste shipping casks have been tested to withstand a head-on collision while on a truck or train at a rate of speed much higher than that specified for rail transport of nuclear material.⁽¹¹⁾

Although an accident in a facility may never cause a significant release to the external environment, the potential for release is present. The pre-emplacement operations in the aboveground facilities probably have a greater potential for accident than the repository itself. Such operations include waste solidification, and fission gas and plutonium storage.⁽¹²⁾ Likewise, the radiological impact of accidents may also be greater for personnel working in aboveground facilities than for their counterparts working in the repository.

Above and below ground facilities are designed with several layers of engineered safety features to mitigate the impact of potential accidents. Taking these features into account, the actual risk may indeed be less than that hypothesized for nuclear waste disposal.

H.1.2 Long-Term Risks

Siting criteria will be established to minimize the impact of long-term uncertainties involved in disposal of nuclear wastes in deep geologic repositories.

H.1.2.1 Time Scales of Radiological Concern. In evaluating the relative impact of radiological effects from the waste disposal cycle it is important to consider the time dependent decay characteristics of radionuclides. Putting the relative impacts of disposal of encapsulated reprocessed high level wastes and unprocessed spent fuel in perspective, the following relevant time scales of radiological concern are recognized:

- **<500 years** — represents the period of maximum radiological concern after the repository is sealed when activity levels and radiological hazards are dominated by radioactive fission products which may escape into the biosphere due to natural processes, human mining or drilling, or catastrophic breach of the repository containment
- **500-1000 years** — represents the period when activity of reprocessed waste or spent fuel has decayed to negligible levels because of the decay of most of the fission products. This period is dominated by actinides and fissile materials including plutonium, americium, and curium
- **10^3 - 10^6 years** — represents the period when the residual activity of actinides, ^{226}Ra , and long-lived ^{129}I may be of concern
- **$>10^6$ years** — the levels of residual radioactivity are expected to be of minor consequence⁽¹³⁾

H.1.2.2 Uncertainties in Long-Term Impact Assessment. Many events which are not of significant concern in the radiological impact assessment of aboveground operating facilities have to be considered to be significant in evaluating the long-term radiological impact of the geologic

repository. For example, the consequence of such events as unintentional or intentional human intrusion of the repository containment at some future time (e.g., by drilling or mining) and the leaching of spent fuel or waste containers by undetected underground aquifers that breach the disposal site have to be considered. Even such a low probability natural occurrence as an earthquake, volcano, meteorite impact or flood must be considered in light of the potential consequence of such an event.

The uncertainties in long-term impact assessment derive from the fact that the technical knowledge defining such factors as the leaching rate of nuclides from fuel, and the retardation and groundwater migration rates, is still under development. Furthermore, the rates themselves are dependent on such unpredictable and unforeseeable events as earthquakes and the presence of surface water. The repository excavation itself upsets the long-term stability of the investigated geologic strata and could have an effect on measured hydrogeologic parameters. Thus, results of assessments based on parameters which range in value from conservative to optimistic may have wide-ranging degrees of credibility.

H.1.3 Regulatory Controls

In the establishment of various regulatory agencies and appropriate licensing guidelines, the federal government seeks not only to control the impact of facilities on operating personnel, the public, and environment under normal and accident conditions, but also to minimize the potential impact on future generations. This section reviews the existing relevant federal regulations and guidelines affecting nuclear waste management. State regulations are not considered here as they either are not well defined or their applicability with respect to federal regulations is not completely delineated. The guidelines and regulations affecting nuclear waste disposal are discussed in the following subsections.

H.1.3.1 Licensing Requirements. The following regulations apply for siting and completing construction of a facility based on theoretical assessments of radiological or environmental impact:

- a.) **10CFR51: Licensing and Regulatory Policy and Procedures for Environmental Protection** — This code sets forth the policy and procedures for the preparation and processing of environmental impact statements and related documents. 10CFR51 provides that fuel reprocessing plants and waste disposal operations (for nonexempt quantities) will require an environmental statement before they are licensed.
- b.) **10CFR20 — Sections 301-305: Standard for Protection Against Radiation — Waste Disposal** — These sections provide for the disposal, by burial or movement to a sanitary sewer, of small quantities of radioactive material. Section 302 requires burial of such material on land owned by either the state or federal government. It also requires that pertinent information on the nature of the environment be provided and that procedures be followed to minimize unexpected exposures.
- c.) **10CFR30 — Rules of General Applicability to Licensing of Byproduct Material** — These rules define quantities and concentrations of byproduct material for which licenses are required. Byproduct material means any "radioactive material (except special nuclear material) yielded in or made radioactive by exposure to radiation incident to the process of producing or using special nuclear material."
- d.) **10CFR40 — Licensing of Source Material** — This provides that no person without a license shall own, use, or handle source material after mining. Source material means (1) uranium or thorium, or any combination thereof, in any physical or chemical form, or (2)

ores that contain, by weight, 0.05% or more of uranium, thorium, or any combination thereof. Source material does not include special nuclear material.

- e.) **10CFR50, Appendix F: Licensing of Production and Utilization Facilities, Policy Related to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities** — This Appendix provides that high-level waste be solidified in 5 years and shipped in 10 years to a federal repository on federally-owned and controlled land.
- f.) **10CFR70 — Special Nuclear Material** — This establishes license requirements for nuclear materials. Special nuclear material means (1) plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the NRC determines to be special material, but not source material; or (2) any material, other than source material, artificially enriched by any of the foregoing.
- g.) **10CFR72 — Storage of Spent Fuel in an Independent Spent Fuel Storage Installation** — This establishes licensing requirements for ISFSIs (also called Away-From-Reactor (AFR) Storage Facilities) to accommodate extended spent fuel storage. The regulation sets an annual operation dose limit of 25 mrem whole body, 75 thyroid, and 25 other organs for individuals located outside the controlled area of the ISFSI facility. These limits are based on the maximum allowed for the total uranium fuel cycle in accordance with 40CFR190. It also sets a limit of 5 rem in 2 hours as the calculated dose to individuals located off the ISFSI site from a design basis accident. The regulation cites experience that shows ambient air levels around ISFSI spent fuel storage areas as about 5% of 10CFR20 limits for occupied areas.
- h.) **10CFR60 — Licensing of Geologic Repositories (Draft)** — This code sets forth preliminary guidelines for licensing excavated geologic formations for the purpose of storage and/or disposal of processed and unprocessed high-level nuclear wastes. It requires traceability and accountability of disposed materials, an analysis of margins of safety against release of radioactive materials to the environment, taking into account the effectiveness of both engineered and natural barriers. It also requires plans for decommissioning and alternate storage of radioactive wastes should the repository prove to be unsuitable for disposal.

H.1.3.2 Operating Requirements. The following regulations and guidelines apply for sustaining and controlling facility operations and waste handling based on monitored or measured concentrations or releases of radioactivity:

- a.) **10CFR20 — Standards for Protection Against Radiation** — This section details maximum permissible radiation dose, radionuclide air concentrations, and radionuclide water concentrations that apply to both occupational and public exposure.
- b.) **ERDA Manual, Chapter 0524** — This chapter applies to 10CFR20 radiation dose, and air and water concentration limits to contractors who operate ERDA-owned or controlled facilities.
- c.) **49CFR171-179: Hazardous Materials Regulation Board, Department of Transportation** — These parts cover preparation and shipping of hazardous materials, including radioactive materials.
- d.) **10CFR71 — Packaging of Radioactive Material for Transport and Transportation of Radioactive Materials under Certain Conditions** — This describes regulations for the handling and shipping of licensed material. The requirements of 10CFR71 are to be applied in addition to the requirements of 49CFR171-179.

- e.) **ERDA Manual, Chapter 0511: Radioactive Waste Management** — This chapter applies to contractors who operate ERDA-owned or controlled facilities. It includes the provision that solid waste containing significant uranium 233 or transuranic contamination greater than 10 nCi/g be segregated and placed in retrievable storage.
- f.) **40CFR141 — Interim Primary Drinking Water Regulations** — This establishes concentration and dose limits in drinking water for man-made and natural radionuclides. Although the burden of compliance with this regulation lies with the drinking-water supplier, these limits must be noted in evaluating discharges into water.
- g.) **40CFR190 — Environmental Radiation Protection Standards for Nuclear Power Operations** — This regulation specifies 25 mrem/yr as the maximum individual dose resulting from the uranium fuel cycle and sets release limits in terms of Ci/GW-yr for some long-lived nuclides. Although it is not applicable to waste disposal sites, the regulation is mentioned for its unique measurement units: (1) dose resulting from fuel cycle rather than from a specific facility, (2) release per unit energy, rather than per unit volume.

H.1.4 Repository Integrity and Waste Retrievability

As a prerequisite toward meeting the objective of the safe long-term waste disposal, two conditions must be met: (1) proper initial siting, and (2) proper design to allow safe emplacement of wastes and, if necessary, retrieval of disposed wastes during the repository life as a result of safety considerations. The first condition relies on appropriate siting criteria to assure the basic geologic and hydrologic integrity of the repository facility during both near-term mining and filling operations and during long-term geologic time. An assessment of the geologic repository siting criteria is discussed in Subsection H.2.3.2.

The second condition, addressing repository design, includes such considerations as appropriate radiation protection criteria, dose exposure limits, and engineered safety features, including monitoring, shielding, radiation zone isolation, access to any room from any corridor, fire protection equipment and alarms, ventilation systems, decontamination rooms, shielded waste handling equipment or manipulators, and appropriately designed structures in aboveground waste transfer and handling areas. In addition to the above waste emplacement safety features, considerations such as storage well sleeves, shield plugs, and over-boring are provided to assure retrievability during the first several years after emplacement, if unforeseen safety considerations should develop. The above design details on engineered safety features and retrievability considerations are the subjects of Section 8 and Appendices D and I of this report.

H.1.5 Radiological Impact Assessment Methodology

Several methods have been developed by the industry to analyze the radiological impact of normal facility operations or accidents. The methods involve several factors which can be useful for quantifying the consequence of an event. These factors can be summarized as follows:

- Radionuclide inventories in different types of nuclear waste
- Radionuclide release mechanisms
- Radioactivity concentration reduction mechanisms
- Radiological dose response
- Spatial distribution of exposed populations
- Probabilities of unplanned events or accidents
- Measurement of radiological risk

For each of the above factors there are inherent uncertainties which compromise the value of analytical results and these uncertainties require an analysis to weigh the consequence toward a conservative assessment. These factors are discussed in more detail in Section H.3 of this Appendix.

H.1.6 Population Considerations

This issue is concerned with the impact borne by populations near the source of release, relative to those far removed from the site. Clearly, during the placement of the nuclear waste in the repository and within the short-term future after the repository is closed, the nearby population is exposed to a higher degree and perhaps inequitable burden of impacts than the rest of the public.

The nuclear waste management criteria will ultimately be expressed in terms of indices for specific spatial extents of land areas. Some evaluations may be averaged over a 100-km distance around the repository, where other evaluations may be over the entire United States land area or even on a global basis (e.g., for krypton-85 and carbon-14). To protect the local, as well as national, population properly, an appropriate spatial parameter must be defined, and the proper weight must be given to that parameter.⁽¹³⁾

However, for long-term impact assessments, it is impractical to attempt an estimate of the population distribution for time periods greater than 100 years after a repository is sealed.

H.1.7 Public Perception of Risks

Acceptability of the potential impact is strongly dependent upon the public perception of the hazard resulting from nuclear radiation exposure.

H.1.7.1 Perception of Operating Phase Radiological Risks. For near-term operating phase consequences, such low probability, high consequence events as jet airplanes, meteorites, or nuclear warheads hitting facilities are excluded from licensed facilities' safety evaluations.

On the other hand, the evaluation of consequences from such low probability occurrences as tornadoes, earthquakes, floods, and other natural disasters is considered important because of the perceived seriousness of the consequences.

H.1.7.2 Perception of Long-Term Radiological Impact. Because of the difficulties and uncertainties involved in predicting and quantifying the long-term impacts from nuclear waste disposal, it is difficult to assess what portion of our generation's radiological burden may be placed on future generations. Because it is not certain that future generations will retain the knowledge to monitor or perceive nuclear residues from today's wastes in drinking wells, the ethical issue remains of how to weigh the potential future impact as compared to those real benefits received today from nuclear power.

It is interesting to note that, although the public associates long-term danger with radioactive wastes, larger quantities of more toxic chemical wastes are regularly transferred to the biosphere with little public concern. Such chemical wastes may migrate from surface disposal sites to underground water tables more quickly and with higher toxicity than radionuclides which must migrate to surface aquifers from a repository 2000 feet under ground. Migration of such nuclides may take thousands of years.

H.2 EVALUATION CRITERIA

Several evaluation criteria are used to assess the technical issues relating to the radiological impact of the spent fuel disposal and waste processing components of the nuclear fuel cycle.

H.2.1 Near-Term Public Health and Safety and Environmental Impact

The most important evaluation criterion is the assurance that the health and safety of the public is protected within the guidelines of the applicable federal and state regulations. Current federal regulations limit population dose exposure to 0.17 rem/yr.

The operating experience of the Nuclear Fuel Services reprocessing plant and the design of the Barnwell Nuclear Fuel Plant, as well as government-owned Purex reprocessing plants, have shown that spent fuel can be reprocessed without creating undue radiation hazards for the public or environment. Similarly, 20 years of experience in handling, transporting, and storage of spent fuel assemblies in surface facilities have demonstrated the safety of these components of the fuel cycle.

H.2.2 Radiation Protection to Operators

Radiation protection for operators and maintenance personnel must be considered a part of the essential evaluation criteria in the facility design process to maintain radiation exposure to operators within acceptable levels.

Radiation protection methods include shielding, restricted access zoning and isolation, protective clothing and breathing apparatus, occupancy control and other engineered safety features including ventilation, remote handling and decontamination. In all cases the presence of health physicists and appropriate monitoring equipment is required to assure radiation exposure is kept within acceptable levels. The ALARA (as-low-as-reasonably-achievable) radiation protection guidelines require that personnel in unrestricted areas not be subjected to dose levels above 0.5 mrem/hr. Similarly, for personnel in restricted access, the maximum whole body dose exposure is generally limited to 5 rem/yr.

The following sets of criteria define the radiation protection bases which will be used by the SFDS to meet the requirements of ALARA and the regulatory guidelines and assure the protection of the public and plant facilities personnel.

H.2.2.1 Radiation Dose Limits. Table H-1 lists the recommended radiation guidelines for external and internal exposure as defined in 10CFR20 and ERDAM-0524, and as summarized by B. L. Cohen.⁽¹⁰⁾

H.2.2.2 Radiation Zones. Several systems for designating radiation zones have been used over the years to facilitate nuclear power plant design and meet the requirements of plant operations and maintenance personnel. Table H-2 lists radiation exposure limits for various facility operating zones. Column A of this table lists the updated radiation dose rates for restricted areas which are consistent with NRC regulations (10CFR20) and which are recommended by the Stearns-Roger Engineering Company (SRENCO) NWTs Repository No. 1 Conceptual Design Report.⁽¹⁴⁾

These zones and dose limits are also recommended for use in the SFDS as radiation zoning criteria for most fuel cycle components, including AFRs and P/E facilities. The designated zone dose rates given in Column A are considered to be conservative upper limits.

Column B lists the repository zone classifications and dose limits recommended as a design basis for the SFDS repository, based on several somewhat atypical operating conditions which dis-

Table H-1
PERSONNEL RADIATION EXPOSURE LIMITS^(a)

	Annual Exposure	Quarterly Exposure
A. Occupational exposure limits		
(maximum permissible dose equivalent)		
(1) Combined whole body occupational exposure:		
Prospective annual limit	5 rems	
Retrospective annual limit	10-15 rems	
Maximum accumulation to age N yr	$(N-18) \times 5$ rems	
(2) Skin	15 rems	
(3) Hands	75 rems	25 rems
(4) Forearms	30 rems	10 rems
(5) Other organs, tissues and organ systems	15 rems	5 rems
(6) Fertile women (with respect to fetus)	0.5 rem in gestation period	
B. Dose limits for the public (or occasionally exposed individuals)		
(1) Individual or occasional	0.5 rem	
(2) Students	0.1 rem	
C. Population dose limits (averaged)		
(1) Genetic	0.17 rem	
(2) Somatic	0.17 rem	
D. Emergency dose limits^(b)		
(1) Individual (older than 45 years, if possible):		
Whole body	100 rems	
Hands and forearms	200 rems additional	
(2) Less urgent:		
Individual (whole body)	25 rems	
Hands and forearms	100 rems total	

(a) Source: B. L. Cohen⁽¹⁰⁾

(b) Exposure values for emergency dose limits are for each occurrence

Table H-2
RADIATION EXPOSURE LIMITS FOR VARIOUS
REPOSITORY OPERATING ZONES^(a)

Area Radiation Classification	Annual Whole Body Dose Limit (m-rem)	A Radiation Zoning For Above Ground Facilities		B Radiation Zoning for Repository		C Individual Access Limits	
		Zone	Dose Rate (m-rem/hr)	Zone	Dose Rate (m-rem/hr)	(Hrs/yr)	(Hrs/wk)
Unrestricted	500	offsite	< 0.06	offsite	< 0.06	No Limit	No Limit
Very Low Restricted	5000	1	0.06-0.6	1	0.06-0.1	No Limit	No Limit
Low Restricted	5000	2	0.6-2.5	2	0.1-0.5	2000	40
Intermediate Restricted	5000	3	2.5-15	3	0.5-15	333	6
High Restricted	5000	4	15-100	4	15-100	50	1
Very High Restricted	5000	5	> 100	5	> 100	< 50	< 1

(a) Adapted from Stearns-Roger NWTS No. 1 Conceptual Design Report⁽¹⁴⁾

tinguish the repository from nuclear plants and reprocessing facilities. The recommended zoning accepts ALARA dose limits for Zones 1 and 2 of 0.1 and 0.5 m-rem/hr, respectively, based on the fact that large numbers of non-nuclear mining personnel may be required to work on a continuing basis in areas of potentially high background radiation from radon 220 and 222.

A survey of operating salt mines revealed that while most mines had no radon present in the mine atmosphere, others had radon concentrations of 0.01 to 0.1 working levels. Based on a radon concentration of only 0.03 working level and a pool of 218 personnel in the mine, Kaiser estimated the annual integrated lung dose rate above background at 1220 man-rem or an average individual occupation dose of 5.6 rem/yr which exceeds the current quarterly occupational exposure limits.⁽¹⁵⁾ It is therefore reasonable to restrict the maximum dose rate limit above background to only 0.5 m-rem/hr to non-nuclear personnel in the mine, in order to allow unlimited access to Zone 1 and Zone 2 areas.

There are fewer radioactive systems in a repository than in aboveground reprocessing facilities which require repository maintenance and radioactive system operation. Therefore, because occupancy times for repair and maintenance are expected to be minimal, the higher dose rate limits defined in Column B for Zones 3 through 5 can be justified. In repositories, the occupancy times for such primary activities as handling of waste canisters are more predictable, thus facilitating control of dose exposure, as compared to nuclear power plants, for example, which have such unpredictable activities as valve, instrumentation and system maintenance (see Column C).

H.2.2.3 Nuclear Waste Handling Experience. Besides the demonstrated radiation safety experience of the nuclear power industry and the government-owned reprocessing plants, the feasibility and safety of transferring and handling highly radioactive materials in underground

mines has been demonstrated using Engineering Test Reactor fuel assemblies without the use of hot cells or any unusual form of radiation shielding. The equivalent yearly radiation exposure burden was only 40 mrem/canister per year, when each canister contained over 4 million curies.

Assuming the use of either shielded canister transporter trucks or remote control gantry crane transfer systems in current repository design concepts, this exposure burden should be reduced significantly.

H.2.3 Long-Term Public Health and Safety and Environmental Impact

In the case of the repository, public health and safety and environmental qualities are to be maintained, not only for the operational phase, but for the long-term as well.

While assessment of the environmental and radiological impact of spent fuel reprocessing is well documented, complete technical evaluation of the longer term environmental and radiological impacts of a repository facility is still in progress. Some of this work is being done, for example, as part of the Waste Isolation Safety Assessment Program (WISAP) by Battelle Pacific Northwest Laboratory and others. Those assessments are both unique and difficult because of the uncertainties involved in ascertaining the potential release or transport pathways for nuclides over a million year time scale.

For SFDS Cases 1, 2, and 3, the initial inventories of plutonium and uranium-235 will exceed those of the Reference Case by a factor of 200. Thus, the long-term impact from these isotopes and their daughters will be expected to exceed the impact of the Reference Case nuclides proportionately. Whether or not the long-term impact to some individual living during a specified future time period would exceed background depends upon the interaction of several significant factors affecting and inhibiting release of these nuclides to the environment.

The factors include:

- Decay of the initial inventory
- Biological hazard of the decay daughter products
- The probability of events occurring which will breach the essentially non-permeable media of the repository
- The geophysical and chemical barriers to nuclide release, which are functions of the path length from the repository site to the nearest groundwater supply and the leachability of the waste form itself

H.2.3.1 Radiological Impact Considerations. Several long-term radiological impact considerations related to repository waste material package design are independent of the study cases and are summarized below:

- (1) Because of the difference in migration rates of different nuclides, certain nuclide daughters (e.g., ^{226}Ra) can be released later and catch up to their parent nuclides (e.g., ^{230}Th) which have a much lower migration rate. This causes a phenomenon called reconcentration.
- (2) Although sorption in the geologic medium may tend to negate the benefits of low leach rate waste forms (because of the initial holdup of certain daughter products), reconcentration is not likely to be a significant influence on radiological impact.
- (3) The impact of gaseous fission product releases from a repository does not seem to be a problem because of the short half-lives involved and more particularly because of the dispersability of the gases once released. Even ^{129}I , which has an extremely long half-life (in

time periods greater than 10^6 years it becomes the dominant species), is not expected to be a problem because of its small contribution to the initial inventory.

- (4) If the path length to the nearest potable water supply is sufficiently long, low leachability waste forms, canister integrity, and site stability are of lesser concern.
- (5) Mechanisms which could affect the pathway for release of nuclides through the geosphere must be considered, including meteorites, volcanoes, and earthquakes, because of the potentially significant residual effects of such events causing a breach of the earthen confinement.
- (6) Although Burkholder indicates⁽¹⁶⁾ that the consequence from the unpartitioned waste in geologic isolation is no greater than the consequence of the low impact fraction of the partitioned waste isolated in the same manner, an analysis of Case 1 particularly indicates some incentive to partitioning plutonium and finding alternate means for its disposal because of the hazard of its decay daughters.
- (7) The highest potential impact on an individual is represented by the maximum potential hazard of ^{90}Sr for the Reference Case and plutonium for the spent fuel cases, during the first 500 years. The actual impact from the fission product inventory is not expected to be significant during this time period unless catastrophic breach of the repository confinement occurs.
- (8) After the first 500 to 1000 years, the impact on an individual is not expected to exceed background for the Reference Case waste inventories; however, the impact on individuals due to inventories from the spent fuel cases may exceed background by a slight moderate degree after this time period.
- (9) For time periods beyond one million years, the radiological hazard represented by uranium and thorium daughters is expected to be about equivalent to the hazard from current uranium mill tailings.

H.2.3.2 Site Selection Criteria. One of the major difficulties in assessing radiological impact of repositories is determining the probability that particular events (having potentially significant and radiological consequences) will occur. To minimize the impact of uncertainties regarding such event probabilities, the National Academy of Science (NAS) has identified certain geologic and geochemical criteria for selection of suitable sites for subsurface waste disposal.⁽¹⁷⁾ These criteria, summarized below in six categories, are also important for assessing the long-term radiological impact of geologic waste repositories.

(1) Geometrical and dimensional criteria. The repository should be at a depth sufficient to separate the repository from any surface process or event that might cause a breach of the repository. These include ice caps, stream erosion, meteorite impact, and man-made explosions. Knowledge of the size and shape of the specific body of rock in which a repository is to be placed is essential in advance of development of the site. In this regard, a potential conflict exists between core drilling to obtain data and the danger of subsequent fluid leakage into the repository, or nuclide release from the repository, resulting from such drill holes.

(2) Long-term geological stability. The choice of a repository site must assume the long-term geological and hydrological stability of the chosen site.

- The repository should lie within a structurally stable geological block and not near a tectonic boundary
- Faults along which rupture could occur must be avoided
- Areas with abnormally high geothermal gradients or with evidence of relatively recent volcanic activity should be avoided

For certain sites under consideration, a large domed salt deposit presumes the stability of the media and the unlikely consequence of tectonic or volcanic activity which could breach the earthen confinement and provide a release pathway for radionuclides. The mechanical and geophysical properties and the state of stress in the repository host rock should be such as to ensure the stability of the repository during its operation. The NAS study stresses the importance of proper design of vaults and passageways to meet these criteria.

(3) Hydrological criteria. Hydrological analysis of the perturbed geologic system involving a repository must determine that fluid transport will not move hazardous material to the biosphere in amounts and rates above prescribed limits. A geological system should be selected that can be satisfactorily sealed when the repository is closed. The geological record of previous hydrological conditions should be such that long-term hydrological isolation of the repository site in a perturbed geologic environment is favorable.

Because groundwater is assumed to be the most likely escape pathway for radionuclides, a favorable groundwater regime, as evidenced by the salt, provides assurance that radionuclides will be transported sufficiently slowly under normal conditions to prevent contact with the biosphere, until the potential impact of such nuclides has been reduced to acceptance levels.

(4) Repository containment integrity and geochemical criteria. A basic criterion to assure both short- and long-term waste isolation must presume confinement integrity including the following considerations:

- (a) The integrity and long-term stability of the geologic medium, as discussed above.
- (b) The ability of the medium, such as salt, to maintain its integrity under a continuous environment of high radiation and thermal loading. Small cracks or fissures opened in salt, for example, tend to be self-sealing in the presence of thermal and radiation fluxes anticipated for the repository. In this regard:
 - Radioactive heat and radiation should not reach levels high enough to produce physical and chemical reactions in the repository rock that would compromise the geological integrity. While the chemical and physical effects of radiation may be minor, radiation-induced distortion of the crystal lattice of minerals will generally increase their solubility and decrease their thermal stability.
 - Water in the repository, if present, should not react chemically or physically with the repository rock to increase its permeability, which would compromise geological integrity
 - The properties of the geochemical system of the radionuclides, the repository rock, and its associated water should be such as to restrict or prevent the mobility of the radionuclides and to delay their migration to the active biosphere.

With regard to the latter two criteria, the existence of the salt mine should itself be indicative of a minimal groundwater residual. However, some studies have shown the existence of brine located in minute cavities in salt (as much as 0.5% by volume) which when subjected to thermal gradients and elevated temperatures is found to move toward a heat source.⁽¹⁸⁾ The existence of brine next to waste packages could result in the generation of radiation and thermal reaction products with potentially adverse radiological and safety consequences, but more importantly, corrosion and degradation of the waste containment boundaries which may hasten the process of leaching and nuclide release. Additional results include the formation of explosive gas mixtures, enhanced transport of fission products, and existence of noxious gases in working areas.⁽¹⁹⁾

- (c) The inherent integrity of the waste fuel form itself, particularly for the required period of isolation and retrievability, should inhibit degradation, leachability and subsequent loss of radionuclides to the external environment. Any interaction of water, repository rock, and the waste material should be controlled in such a way as to minimize the rate of dissolution of the waste form.
- (d) Perhaps one of the single most important long-term impact criteria is that the overall integrity of the repository site should not depend upon continued human supervision or action.

(5) Retrievability. It is noteworthy that the NAS and others have recommended against retrievable storage of spent fuel elements or similar variations of the Case 1 spent fuel disposed of in a permanent repository, based on a desire to keep the design and operation of the repository simple and minimize safeguards and security requirements. It is assumed that disposal of unprocessed spent fuel would make the disposal site attractive even after the mine is sealed to a society which wants to mine the site area to recover fuel values. However, with retrievability as a criterion:

- The repository should be designed and waste received so as to limit retrievability to a period not to exceed the time period supported by testing in the specified host rock. In this regard, retrievable storage is important for the first repository sites to allow opportunity for developing engineering data and testing the principles of waste disposal in geologic media
- Backfilling and sealing of each segment of the mine cavity should be accomplished at the end of the retrievability period

(6) Geo-economic criteria, long-term waste isolation and security. Considerable importance is attached to this set of criteria which assumes that the potential repository site location should also be unattractive as a potential resource reserve. To maintain long-term waste isolation and security, it would appear desirable that the potential repository site not be located near potential resource sites, such as ore bearing or oil bearing shale strata, to prevent inadvertent breach of the repository by mining or drilling operations. In this regard, NAS supports the view of disposing radioactive wastes in media located at "unattractive" sites and further requires that:

- No area with a present or past record of resource extraction, other than by surface quarrying, should be considered as a geological site for radioactive wastes. From this perspective, the report also criticizes disposal of spent fuel in repositories since remaining fissionable materials would be "potentially highly valuable to future mankind, and therefore make the repository highly attractive to future explorationists..."
- No area should be considered as a potential site for a repository unless sufficient geological information is at hand to provide a basis for a reasonable analysis of resource potential
- No area adjacent to an actual or potential major dam site should be considered as a potential site of a repository

H.2.4 Accident Potential

Because the probability for an accident, as well as the type of accident, is unique for a particular facility (AFR, P/E facility, transportation system, or repository), the essential criterion is that any release of activity caused by such an accident does not exceed the allowable short-term dose limits for accidents as defined by 10CFR100 and applicable codes or regulations.

- The regulations limit whole body dose exposure to individuals living in the low population zone next to a nuclear facility to a 25 rem once-in-a-lifetime dose due to an accidental release

- The regulation may also allow credit for accidental dose reduction for engineered safety features which have demonstrated effectiveness

The relative impact of a waste handling accident, for example, is very dependent on the form of the waste, comparing unprocessed spent fuel assemblies to vitrified HLW for example, and/or the waste type, viz., HL, IL, or LL waste.

For the sake of definition, accidents are assumed to occur only during the operating phase of the facility and are man or machine-caused events. They would not include natural catastrophic events, which could also occur during both the operating or post-operational phase of the facility, or nuclear warfare or sabotage.

In this regard, some consideration in assessing the potential impact of an accident involved in either the transportation, handling, reprocessing, disposal, or retrieval of spent fuel are:

- (1) **Burnup** — the length of time of exposure in the reactor increases its potential radiological impact.
- (2) **Storage age of fuel** — the longer fuel decays after removal from the reactors, the less its potential impact.
- (3) **Source of fuel and presence of fission gases** — PWR fuel has a higher potential impact from an accident than BWR fuel because of the higher pressure of gas in the fuel element plenum. Obviously, the presence of fission gases which could escape from the fuel to the environment in unprocessed spent fuel assemblies provides a higher impact for accident potential than processed fuel or HLW.
- (4) **Extent of processing** — it might be reasoned that the greater the number of steps involved in processing, the higher would be the probability of an accident and, therefore, the higher the potential impact from the accident.
- (5) **Number of operating and maintenance personnel** — the greater number of personnel associated with operating and maintaining more complex processing systems or handling larger quantities of radioactive waste materials or shipping containers has a direct impact on the potential radiological impact to personnel working at the facility in terms of the facility's man-rem dose burden.
- (6) **Form of packaged waste** — voloxidized spent fuel has a greater potential impact from an accident than vitrified waste because the waste is more easily dispersed in case of an accident.
- (7) **Presence of plutonium** — the presence of plutonium and other toxic substances in spent fuel creates a greater hazard potential in the event of an accident than HLW from which most Pu has been extracted.
- (8) **Storage media and canister filler material** — the potential interaction with water, air, or other canister filler or storage media is considered because neutron irradiation and radiolytic interaction of these materials and spent fuel could occur in an accident.
- (9) **Remote handling** — the use of flangeless and valveless pneumatic fluid transfer systems as engineered for the BNFP is an engineering concept in process design which results in much less dose exposure to maintenance personnel than what might be expected in a conventional power reactor system. Also, the ability to perform repairs of equipment by remote control should prevent excessive exposure to operations and maintenance personnel even under unusual conditions.
- (10) **Decommissioning** — although it appears that decommissioning of AFRs and above ground repository facilities can be completed without major incident, decommissioning of P/E facilities could result in an above background release of nuclides in the event of a

severe earthquake coincident with final stages of decommissioning. NUREG 0278 defines a decommissioning incident during the equipment decontamination phase of the HLLW storage tanks. (There is a large penetration in the containment at this time due to the decommissioning activity.) Although the probability of such an earthquake-related event is low, and the radioactivity level in the tanks is orders of magnitude less than normal operation, the earthquake is postulated to destroy the greenhouse surrounding the tank and result in a release of about 3 curies (of material plated out on the sides of the tank) directly to the atmosphere. This results in a 50-year dose commitment only 1.5 x background. One of the most difficult tanks to decontaminate at a P/E facility would be the HLW storage tanks. Decommissioning of those storage tanks may consist of filling the tanks with concrete.

- (11) **Transportation system accidents** — potential transportation impacts are effectively proportional to the number of shipments of waste rather than the inventories of waste carried or transported.⁽¹¹⁾ An assessment of radiological impacts based on that consideration alone may tend to bias against shipment of nuclear waste by truck, even though this may be the most economical transport method, especially for ILW and LLW.

Nevertheless, the magnitude of the consequence is extremely low. The probability of someone receiving a dose exceeding background (.1 R/year) is less than 1 chance per 1 million people exposed per shipment in 1 year's time.

- (12) **Special safeguarding measures or devices** — in transporting wastes, certain methods proposed for safeguarding special nuclear materials (SNM) such as ⁶⁰Co spiking (i.e., mixing high activity waste or fission products with normal wastes) may also increase the radiological hazard if such wastes were involved in an accident while being transported. In the case of a severe accident, in which the train is derailed or the truck is turned over, the inability to handle or hoist the shipping casks by normal emergency vehicles would contribute more to the total handling time and expense than if the spike had not been used and normal emergency vehicles could quickly handle the accident.

H.2.5 Criticality Potential

Criteria and discussion associated with the radiological impact of criticality is discussed in Section 6 of Volume 2.

H.2.6 Retrievability Potential

Perhaps the greatest potential radiological hazard to personnel at the repository would be if circumstances required retrieval of spent fuel or high-level waste, or any other waste for any reason beyond the initial 5-year retrievability period, after which the rooms have been backfilled. The task of removing the backfilled rooms and overboring the canisters will be costly in terms of manpower and equipment and radiologically hazardous because of the high dose rates of direct gamma flux as well as the potential hazard represented by inadvertent penetration of the storage canister. Criteria and discussion associated with the radiological impact of retrievability is discussed in Section 8 of Volume 2.

H.3 CONSIDERATIONS AFFECTING RADIOLOGICAL IMPACT ASSESSMENTS

H.3.1 Waste Types and Inventories

Four major types of waste are considered as part of the Spent Fuel Disposal Study: Spent Fuel, High-Level Waste (HLW), Intermediate-Level Wastes (ILW), and Low-Level Wastes (LLW), including both TRU and non-TRU types of wastes. Characterization and quantification of these wastes are developed in Appendix C and tabulated in Appendix A of this volume.

H.3.2 Radionuclides in Waste

The source inventories (in millions of curies) for a 25-year 400 GWe nuclear power economy are given for the Reference Case HLW in time periods of concern up to 10^6 years in Table H-3.

Table H-3
TOTAL SOURCE STRENGTH — REFERENCE CASE^(a)
(Millions of Curies)
400 GWe Economy Through The Year 2000

Nuclide	Time After Initial Disposal (years)				
	10^2	10^3	10^4	10^5	10^6
¹³⁷ Cs	1840	0.0000017	0	0	0
⁹⁰ Sr	1100	0.0000252	0	0	0
⁹⁹ Tc	2.83	2.82	2.74	2.04	0.108
¹⁴ C	1.02	0.915	0.308	0.000006	0
¹²⁹ I	0.00804	0.00804	0.00804	0.00801	0.00772
²³⁸ Pu	186	0.232	0	0	0
²³⁹ Pu	0.986	1.08	1.55	0.193	0.0000124
²⁴⁰ Pu	3.13	2.88	1.15	0.000113	0.00172
²⁴¹ Pu	2.58	0.127	0.0598	0.000032	0
²⁴² Pu	0.00946	0.00982	0.00987	0.00844	0.00163
²²⁶ Ra	0.0000016	0.000247	0.0108	0.0844	0.0155
²²⁹ Th	0.000195	0.00173	0.0127	0.0472	0.0773
²³³ U	0.0188	0.0191	0.0221	0.0465	0.0773
²³⁴ U	0.101	0.17	0.165	0.129	0.0105
²³⁵ U	0.0000227	0.0000236	0.0000357	0.000115	0.000121
²³⁶ U	0.00559	0.00567	0.00616	0.00647	0.0063
²³⁸ U	0.000244	0.000244	0.000244	0.000244	0.000245
²³⁷ Np	0.0834	0.0959	0.0998	0.0971	0.0726
²⁴¹ Am	80.8	19.3	0.0598	0.000033	0
²⁴³ Am	5.07	4.67	2.07	0.000595	0

(a) Taken from Burkholder⁽¹⁶⁾

For the spent fuel case, the inventories of Pu, U, Np, Pa, Th, and americium-241 are expected to be a factor of 200 times as high as the Reference Case inventories for the same isotopes in this table. This is due to the recovery of about 99.5% of the U and Pu during spent fuel reprocessing. As time elapses, the greater inventories of U and Pu and their daughter products remaining in the spent fuel will increasingly bias the radiological impact of Cases 1, 2, and 3 compared to the HLW of the Reference Case.

H.3.3 Radionuclide Release Mechanisms

In the event of failure of safety systems which control release of radioactive materials, numerous potential release scenarios are possible. However, the significant pathways can be summarized under two mechanisms:

- (1) dispersal of gaseous or particulate releases to the atmosphere, and
- (2) migration of liquid releases and dissolved or suspended solids to an underground or surface aquifer.

Potential release scenarios which may result in an impact on the public are summarized for the different facilities as follows:

AFR. Radionuclides could be released by entrainment in liquids or venting of fission gases from failed fuel assemblies in the fuel storage pool. Failure of the offgas processing system at an AFR during a fire in the filters could release gaseous activity. Release of liquids could result from an earthquake-induced breach of the fuel pool containment.

P/E facility. In addition to the same type of release scenarios discussed for the AFRs, more severe events, such as earthquakes, could breach the temporary containment around HLLW storage tanks.

Because of the presence of untreated combustibles at the P/E facility, scenarios may involve drum failure as a result of a fire within the drum. Such scenarios, which could release gaseous and particulate matter and which might impact facility personnel involved in decontamination efforts, are not expected to result in significant releases to the public. The releases to the public are limited by the facility confinement barriers and filtering systems.

Transportation systems. The probability of release of materials from a shipping cask during an accident is very low. The casks have been observed to withstand the head-on collision of two trains (or trucks) moving at a combined speed of 70 m.p.h.

Repository. Although the repository is expected to receive only non-combustibles from P/E or other facilities, some combustibles such as low-level HEPA filters may be generated at the repository. A low-probability scenario reported for the repository by Science Applications⁽⁷⁾ postulates a spontaneous ignition and fire in one drum, which releases 5 percent of the radioactive inventory to the storage room. Assuming a 99.7 percent particulate filtering efficiency and a dilution factor of 10^6 , offsite dose levels from a one-drum fire are reduced to 2.5×10^{-6} μ Ci/m³ or 1/80 of the NRC guidelines for concentration of nuclides in unrestricted areas. The spread of the fire to 6 drums (an unlikely event) would result in offsite doses less than 1/10 of NCRP limits.

Scenarios for long-term releases. Releases from the repository are dependent upon behavior of specific nuclides and the release mechanisms which could lead to an impact on the public. Such considerations include:

- The leach rate of radionuclides from the spent fuel or HLW characterizes the radioactive source term for the repository. Once the metallic barriers fail, leachability depends on the temperature and chemistry of surrounding fluids and the rate of fluid movement

- The rate of groundwater movement or hydraulic conductivity depends upon the effective porosity of the host rock. Movement can be influenced by slow disruptive events and processes (e.g., groundwater dissolution of host rock), human-caused events (drilling), and faulting or fracturing in the host rock
- The degree of retardation of radionuclides due to chemical reactions with the host rock inhibits the net migration rate of radionuclides through geologic media to potential groundwater aquifers
- Direct release to ground or surface water supplies, or to the atmosphere as a result of tectonic activity would generate the most significant long-term impact on the public, particularly if such activities occur during the first 1000 years after the repository is sealed

From a probability analysis of catastrophic event scenarios on long-term impacts, Logan reports that 90% of the environmental and public dose impact is due to volcanism, 9% to earthquake-induced faulting and leaching and 1% due to meteorite impact.⁽⁸⁾

H.3.4 Biological Response

The dose effect of potential radionuclide releases due to waste disposal operations involve health risks to the population from chronic exposure to ionizing radiation near or below naturally occurring levels (less than 0.1 rem/yr).

Several researchers have used data based on extrapolations of high-level A-bomb survivor dose, therapeutic X-rays, and occupational radiation exposure to estimate the potential risks of various activities in units of cancer deaths/yr.

B. L. Cohen has derived biological health hazard data (risks of fatal cancers) as a function of low-level whole body radiation exposure, and as a result of radiation-induced genetic defects.⁽⁵⁾ Using his data and applying it to the radionuclide data from the SFDS cases, one can make rough comparative assessments between the cases from the standpoint of biological health hazards.

For example, using the data from Cohen's analysis, it is estimated that the Reference Case would cause only 0.4 cumulative cancer deaths in 1,000,000 years from waste disposal, assuming the effectiveness of engineered and natural barriers precluding the release of radionuclides for the first 200 years of isolation.

To put the dose response mechanism in perspective, it can also be shown that after a period of several thousand years, when the earliest nuclides would be expected to enter the biosphere, about 50 oz of spent fuel material or HLW would have to be directly ingested to result in one lethal cancer (50% probability).

H.3.5 Measurement of Radiological Impact

The method of assessment of radiological impact depends upon the ability to mathematically estimate the direct influence of radioactivity and the indirect impact of radionuclide release mechanisms.

H.3.5.1 Quantitative Measurement of Radiological Impact. Units of potential impact used in SFDS are represented by the following:

- (1) Man-rem dose per facility or transportation system accounts for the expected exposure of the population.

- (2) Curies/cc, curies/liter of radioactive materials released to the air or drinking water supply is a convenient unit of measurement of potential impact for both near-term and long-term events and can be compared with maximum permissible concentration, MPC limits of radioactive materials in the environment.
- (3) Cancer deaths per year can be estimated from dose estimates (in rems) using conversion factors based on statistically analyzed data from Hiroshima and Nagasaki A-bomb victims, operating personnel exposure, and X-ray and medical data.

H.3.5.2 Qualitative Measurement of Radiological Impacts. Potential impacts having high levels of uncertainty (e.g., long-term risks) can be compared qualitatively for each case if properly weighted for this uncertainty.

For example, significant factors which have a high degree of uncertainty would be the retardation properties of nuclides in the ground, the rate of leachability of nuclides from waste, the path length from the repository burial site to the groundwater supply, and the surrounding population density. The effects of the uncertainty of the above control variables on the incremental dose levels have been described quantitatively by H. C. Burkholder, et. al., for HLW⁽¹⁶⁾ and spent fuel disposal.⁽²⁰⁾

In SFDS, although the unit of measurement and formulation of radiological impact is intended to be qualitative rather than quantitative, the definition of certain factors is based on real-life characteristics and parameters. The qualitative representation of radiological impacts, for example, can be correlated to real-life measures of acceptable dose (e.g., a potential impact factor of 2 is related to background = 100 mrem/yr).

H.4 RADIOLOGICAL IMPACT MODELING

H.4.1 Generalized Approach to Assessment of Radiological Impact

The radiological impact of nuclear facilities can be determined for two principal categories:

- Impacts due to releases of radioactive nuclides which affect near-term and long-term radiological impacts to the public and near-term impacts to facility personnel. Both normal and abnormal operations are considered
- Impacts to facilities personnel (and to a lesser degree, the public) due to direct exposure to radiation

Impacts due to radioactivity releases are evaluated in the following steps:

- (1) The nuclide source term is evaluated.
- (2) A release fraction or release rate (as appropriate) is determined.
- (3) Decay of the individual radionuclides is considered.
- (4) The release (in curies or curies/cc of air or water) is corrected for the concentration reduction factor, for air dispersion, or for a groundwater dilution and migration factor to determine the concentration at some distance from the point of release. The results of those models can be modified to take into account nuclide pathways in food (i.e., milk, fish, etc.).
- (5) Evaluate the impact in units of dose rate using NCRP derived dose conversion factors (R/mCi) and convert to annual dose (mrem/yr, or man-rem). Man-rem is the product of the incremental dose at various distances from the site and the population within those distances. See Table H-4 for the dose conversion factors.⁽¹⁰⁾
- (6) Determine the impact of various nuclides as a ratio of the nuclide concentration to the maximum permissible concentration (MPCa for air and MPCw for water, as given in 10CFR20, 50).

The MPC values are given for unrestricted "non-occupational" exposure and for restricted "occupational" exposure.

- (7) Determine impacts on facilities personnel in the same way as impacts to the public are determined, except assume the only dose reduction factor is dilution in the surrounding air volume.

Near-term impacts to facilities personnel due to direct exposure to radiation is dependent upon the source strength, type of radiation (gamma, beta, neutron, or alpha), the energy of the radiation, and geometrical arrangement and material attenuation separating the source of radioactivity from the person exposed. Usually, direct radiation dose analysis requires sophisticated computations, although dose conversion relationships have been developed for certain materials and conditions.

H.4.2 Specific Radiological Impact Models

Groundwater Migration Model. S. E. Logan and M. C. Berbano of the University of New Mexico have developed formulas for the migration and transport of leached nuclides by groundwater through the geosphere.⁽⁸⁾ The model calculates the concentration of radionuclides in Ci/m³ of soil or μ Ci/ml in the groundwater of an underground aquifer at the point of entry at some path length from the repository.

Table H-4
NUCLIDE DOSE CONVERSION FACTORS^(a)

Nuclide	Dose Conversion Factors			Critical Org an	Max. Permissible C oncentration	
	Whole Body (R/mCi)	Ingestion (R/mCi)	Inhalation (R/mCi)		(MPC) _a	(MPC) _w
¹⁴ C	5.65(-4)	2.83(-3)	9.7(-3)	Lung-A		
¹³⁷ Cs	4.32(-2)	1.10(-1)	4.85(-2)	Liver-W	5.0(-9)	2.0(-4)
⁹⁰ Sr	1.66(-1)	8.31(0)	4.85(-2)	Bone-W	2.0(-9)	4.0(-6)
²³⁹ Pu	1.91(-2)	7.87(-1)	6.47(-2)	Bone-W	1.0(-11)	5.0(-5)
²⁴⁰ Pu	1.91(-2)	7.86(-1)	6.47(-2)	Bone-W	1.0(-11)	5.0(-5)
²²⁶ Ra	3.11(+1)	3.02(+2)	6.47(-2)	Bone-W	—	1.0(-7)l
²²⁹ Th	5.70(-2)	2.04 (0)	6.47(-2)	Bone-W	3.0(-12)	2.0(-5)
¹²⁹ I	1.29(-2)	1.03(+1)	9.70(-3)	Thyroid-W	2.0(-8)	4.0(-6)
⁹⁹ Tc	4.92(-5)	2.30(-3)	9.70(-3)	GI Tract-A	2.0(-8)	3.0(-3)
²³⁷ Np	5.54(-2)	1.37(0)	6.47(-2)	Bone-W	4.0(-11)	3.0(-5)
²⁴¹ Am	5.42(-2)	8.21(-1)	6.47(-2)	Bone-W	4.0(-11)	4.0(-5)
²⁴³ Am	5.30(-2)	8.19(-1)	6.47(-2)	Bone-W	4.0(-11)	4.0(-5)
²¹⁰ Pb	5.23(-1)	1.45(+1)	9.70(-3)	Bone-W	4.0(-11)	4.0(-5)
²¹⁰ Bi	2.62(-4)	4.52(-4)	4.85(-2)	GI Tract-A	—	—
²¹⁰ Po	8.64(-2)	2.47(0)	6.47(-2)	Kidneys-W	—	—

(a) Source: B. L. Cohen⁽¹⁰⁾

$$C = \frac{k_1}{v_p t} \exp \left\{ \frac{(x - k_2 v_p t)^2}{k_3 v_p t} + \frac{y^2}{k_4 v_p t} \right\} \exp(-\lambda_{Dt})$$

$$k_1 = \frac{M'}{4\pi(a_L a_T)^{1/2}/R_d}$$

$$k_2 = \frac{1}{R_d}$$

$$k_3 = \frac{4a_L}{R_d}$$

$$k_4 = \frac{4a_T}{R_d}$$

$$M' = \frac{M}{z_a}$$

$$R_d = 1 + \frac{\rho K_d}{\Sigma}$$

where C = radionuclide concentration in ground water at point of usage or discharge

M' = amount of particular radionuclide release by leaching during a release time interval (M in Ci) divided by the thickness of the aquifer in which the waste is released Ci/m

R_d = retardation factor, dimensionless

K_d = $\frac{f_m}{f_s}$ = distribution coefficient, defined as the ratio of the fraction of amount of species that is sorbed on the solid medium per unit mass of medium to amount of species remaining in the solution per unit volume of solution cm³/g

ρ = solid medium bulk density, g/cm³

Σ = porosity of solid medium, dimensionless

v_p = seepage velocity, m/d

a_L = longitudinal dispersivity

a_T = transverse longitudinal dispersivity

λ_{Dt} = the nuclide decay constant, year⁻¹

t = time after initial disposal, years

z_a = thickness of aquifer, meters

Typical values of the hydrogeologic parameters are shown in Table H-5. The K_d values are in Table H-6.⁽⁸⁾

Table H-5
INPUT DATA AND PARAMETERS FOR
GROUND WATER TRANSPORT MODEL

Symbol	Physical Parameter	Estimated Value
V_p	Seepage velocity [Xy75], m/d	4.00×10^{-3}
Σ	Porosity of porous media [Hi75], dimensionless	0.15
ρ	Solid bulk density, g/cm ³	2.3
z_a	Aquifer thickness, m	50
y_w	Effective plume width, m	
	Zone 2	2580
	Zone 8	3640
y_y	Transverse distance to average concentration, m	
	Zone 2	520
	Zone 8	730
a_L	Axial dispersivity, m	50
a_T	Transverse dispersivity, m	6
x	Source to point of usage distance, km	
	Zone 2	10
	Zone 8	20

Table H-6
ESTIMATED DISTRIBUTION
COEFFICIENTS FOR
SELECTED RADIONUCLIDES

Nuclide Number	Nuclide	Distribution Coefficient, cm ³ /g
1	C-14	1.4
2	Sr-90	14
3	Y-90	1400
4	Zr-93	1400
5	Nb-93m	1400
6	Tc-99	-0
7	I-129	-0
8	Cs-135	140
9	Cs-137	140
10	Pb-210	2800
11	Ra-225	70
12	Ra-226	70
13	Th-229	10500
14	Th-230	10500
15	Np-237	10.5
16	Np-239	10.5
17	Pu-238	1400
18	Pu-239	1400
19	Pu-240	1400
20	Pu-241	1400
21	Am-241	1400
22	Am-242m	1400
23	Am-243	1400
24	Cm-242	420
25	Cm-244	420

Gaseous Dispersion Model. A simplified model (ASME, Smith, 1968), used to calculate the radioactivity concentrations in curies/m³ of air directly downwind from an elevated or ground level release of radioactive gases or airborne particulates, has the form:

$$X(x, \sigma_y, \sigma_z) = \frac{Q}{\pi \sigma_y \sigma_z \bar{u}} \exp - \left(\frac{h^2}{2 \sigma_z^2} \right)$$

- where X = concentration (units/m³) (for example curies/m³)
 Q = pollutant release rate (units/sec) (for example, curies/sec)
 $\sigma_y \sigma_z$ = crosswind and vertical plume standard deviations (m)
 $\bar{u}$ = mean wind speed (m/sec) at h_s , the height of the stack
 h = effective stack height (m)
 x,y = downwind and crosswind distances (m)

Dose Models. Typical dose models use the dose conversion factors or MPC in curies/m³ of air or water, listed in Table H-4. The table is based on information provided by the ICRP Publications Numbers 2 and 6 (ICRP, 1959 and ICRP, 1962). A simplified dose model has the form:

$$D_{ij}(\text{Rem/yr}) = Q_i(\text{Ci/m}^3) \times d_{ij}(\text{Rem/Ci}) \times f \frac{(\text{m}^3)}{\text{yr}}$$

- where Q_i = The radioactivity concentration in curies/m³ of isotope, i
 d_{ij} = The dose conversion factor (Rem/Ci) or MPC_{ij}^{-1} (m³/Ci) for isotope, i, and for organ, j, from Table H-4. MPC values are used to determine the relative risk of a radioactive release. If MPC values are used, D_{ij} should be ≤ 1.0
 f = Biological uptake per year

For water ingestion, the constants ($f_w = 0.8 \text{ m}^3\text{w/yr}$) account for the assumption that waste is ingested at the rate of 2.2 liters/day ($w = \text{water}$).

For air inhalation, the constant ($f_a = 7300 \text{ m}^3 \text{ a/yr}$) is the volume of air inhaled/yr.

Fault Tree Analyses. A number of event trees have been developed by others, including those by Claiborne and Gera⁽²¹⁾ and S. E. Logan⁽⁸⁾, et. al., for geologic repositories; WASH-740 for nuclear power plants; the American Physical Society for HLW disposal⁽¹⁰⁾; and J. J. Cohen for the performance assessment of P/E facility, transportation systems and the repository components of the high-level solidified waste disposal cycle.⁽²²⁾ In all those evaluations, the uncertainties are significant, particularly in assessing the long-term impacts.

REFERENCES FOR APPENDIX H

- (1) "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," ERDA-76-43, May 1976.
- (2) "Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste," DOE/EIS-0046D, U.S. Department of Energy, April 1979.
- (3) "Report to the President by the Interagency Review Group on Nuclear Waste Management," TID-29442, March 1979.
- (4) J. J. Cohen, "Review of Issues Relevant to Acceptable Risk Criteria for Nuclear Waste Management," Lawrence Livermore Laboratory, February 22, 1978.
- (5) B. L. Cohen, "The Disposal of Radioactive Wastes from Fission Reactors," *Scientific American*, p. 21, June 1977.
- (6) "Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations; Volume 21, Groundwater Movement and Nuclide Transport," Y/OWI/TM-36/21, Dames and Moore, April 1978.
- (7) "Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations; Volume 23, Environmental Effluent Analyses," Y/OWI/TM-36/23, Science Applications, Inc., April 1978.
- (8) S. E. Logan and M. C. Berbano, "Development and Application of a Risk Assessment Method for Radioactive Waste Management," EPA520/6-78-005, U.S. Environmental Protection Agency, June 1978.
- (9) "Nuclear Waste Projections and Source-Term Data for 1977," Y/OWI/TM-34, Union Carbide Corporation-Nuclear Division, June 7, 1978.
- (10) B. L. Cohen, "High-Level Radioactive Waste from Light Water Reactors," *Reviews of Modern Physics*, Volume 49, No. 1, January 1977 and Volume 50, No. 1, January 1978.
- (11) "Capabilities of U.S. Domestic Transportation Systems for the Shipment of Radioactive Wastes," Y/OWI/SUB-77/2233, Nuclear Assurance Corporation, September 1977.
- (12) "Assessment of the Nuclear Fuel Cycle," Atomic Industrial Forum (Committee on Fuel), October 1977.
- (13) "Determination of Acceptable Risk Criteria for Nuclear Waste Management," UCRL-52404, Lawrence Livermore Laboratory, October 1977.
- (14) "NWTs Repository No. 1 Conceptual Design Report, Volume 3," EY-77-C-05-5367, Stearns-Roger Engineering Company, August 1978.
- (15) "Conceptual Design Report, NWTs Repository No. 2, Volume 1," NWTs-2990, Kaiser Engineers, August 1978.
- (16) H. C. Burkholder, et. al., "Incentives for Partitioning High Level Waste," *Nuclear Technology*, November 1976.
- (17) "The Geological Criteria for Suitable Sites of High-Level Radioactive Waste Repositories," National Academy of Sciences, August 3, 1978.
- (18) R. D. Cheverton and W. D. Turner, "Thermal Considerations and Analysis for a Radioactive Waste Repository," *Nuclear Technology*, July 1973.

- (19) "Report to the LMFBR Steering Committee on Resources, Fuel & Fuel Cycles, & Proliferation Aspects," ERDA 77-60, Energy Research and Development Administration, June 1977.
- (20) H. C. Burkholder, "The Reconcentration Phenomenon of Radionuclide Chain Migration," BNWL-2029, *Nuclear Waste Management and Transportation Quarterly Progress Report, January-March 1976*, June 1976.
- (21) H. C. Claiborne and F. Gera, "Potential Containment Failure Mechanisms and Their Consequences at a Radioactive Waste Repository in Bedded Salt in New Mexico," ORNL-TM-4639, Oak Ridge National Laboratory, October 1974.
- (22) J. J. Cohen, et. al., "Determination of Performance Criteria for High Level Solidified Nuclear Waste," NUREG-0279, Lawrence Livermore Laboratory, July 1977.

Appendix I

TECHNICAL CONSIDERATIONS FOR CANISTER RETRIEVABILITY

TABLE OF CONTENTS

	Page
I.1 Environmental Impact on Canisters	I-2
I.1.1 Effect of Corrosion on Canisters	I-2
I.1.2 Effect of Mechanical and Thermal Stresses on Canisters	I-4
I.2 Cost of Canister Materials	I-5
I.3 Facility Design Considerations	I-6
I.3.1 Facilities for Retrievability	I-6
I.3.2 Equipment for Retrievability	I-6
I.3.3 Instrumentation	I-8
I.3.4 Ventilation	I-8
I.3.5 Radiological Protection	I-8
I.3.6 Sleeve-Plug Design	I-8
I.4 Retrievability Operational Considerations	I-10
I.4.1 Subsurface Operations	I-10
I.4.2 Surface Operations	I-11
References for Appendix I	I-12

Appendix I

TECHNICAL CONSIDERATIONS FOR CANISTER RETRIEVABILITY

The retrievability period specified for the SFDS is 5 years. This means that spent fuel and HLW packages are to be stored in a repository in such a manner that they may be retrieved at any time during this period. The canister handling operations are assumed to be reversible to accommodate such retrieval. Storage rooms for canisters are therefore not back-filled during the 5-year period, and the storage wells in the room floors are provided with carbon steel sleeves and removable concrete shielding plugs or caps.

I.1 ENVIRONMENTAL IMPACT ON CANISTERS

The design criteria for canisters and overpacks include a requirement that they be capable of being retrieved (with relative ease) during the first 5 years of repository operation. However, the encapsulation design, canister overpack designs, and design of the sleeves all contribute to canister integrity for a period considerably longer than this 5-year retrievability period. The following subjects are considered below:

- Effect of corrosion on canisters
- Effect of mechanical and thermal stresses on canisters

I.1.1 Effect of Corrosion on Canisters

The main cause of potential corrosion in canisters is the presence of small inclusions of brine, which are typically found in the geologic salt medium. In the presence of a thermal gradient, these inclusions of adsorbed water may migrate toward the source of heat, and as a consequence, could cause the waste canisters to become surrounded by corrosive brine. This is undesirable. Another potential problem associated with the brine migration is the evolution of hydrogen, helium, and hydrochloric acid as a result of radiolytic decomposition and other reactions with brine.

Corrosion is seen, therefore, as the most severe problem for the waste canisters. To offset this, the geologic repository design (see Appendix D) includes sleeved wells for the canister plus sleeve plugs or caps. An assessment of the plug-to-sleeve interface design shows that the plug may not provide an adequate seal against the intrusion of brine into the space between the canister and the sleeve.

The effect of corrosion on simulated waste canisters has been studied by the Oak Ridge National Laboratory (ORNL) and the results presented in the Project Salt Vault Report.⁽¹⁾ Figure I-1 is based on the findings of that report and shows the effects of corrosion on various canister materials over a 25-year period. In the case of stainless steel, the effects of stress corrosion cracking are not included. While some information on this is discussed below, further study is required before it can be confidently selected.

Table I-1 summarizes the effects of stress corrosion cracking on 304L stainless steel for brine vapors over a range of temperatures. The surfaces of the stainless steel heater cans were maintained at temperatures well above the boiling point of the brine. Though superficially in good condition, the can components showed extensive stress corrosion cracking to depths of about one-half of the wall thickness, presumably as a consequence of the release of the included brine. (Although stress corrosion cracking occurs in an oxygenated aqueous solution at temperatures below the boiling point of brine, cracking has not been observed in such an environment at the temperatures the canisters are expected to reach in the repository.) Thus, if stainless steel is to be considered for canisters, the canisters must be protected from exposure to the brine or brine vapors. Since the SFDS repository design assumes that the plug cannot completely prevent the intrusion of brine or brine vapor into the gap between the canister and the sleeve, stainless steel may not be a suitable material for canisters.

The Project Salt Vault data reproduced in Figure I-1 show that the corrosion rate for carbon steel is high (for temperatures in excess of 300°C) in the salt environment. The Project Salt Vault experiments also showed that at temperatures of about 500°C, brine-induced corrosion of carbon steel is extremely rapid. therefore, if carbon steel is to be used for canisters at the expected canister temperatures of around 300°C in the repository, substantial corrosion allowances will be required. In the tests performed by ORNL, carbon steel did not exhibit stress corrosion cracking, however.

Figure I-1.
Corrosion in Canister Materials

Table I-1
STRESS CORROSION FOR 304L STAINLESS STEEL
HEATER CANS (6 mm thick) IN BRINE VAPORS

Can Location	Maximum Temperature (°C)	Duration (days)	Scaling	Intergranular Penetration (in.)	Carbide Precipitation
Wall	690	33	Severe	0.045	Yes
Floor	590	33	Moderate	0.002	Yes
Array	435	28	Slight	0.0015	No

According to the results of the Project Salt Vault experiments, the nickel-chrome alloys with a high nickel content (Inconel, etc.) have a relatively low corrosion rate in the salt environment even at higher (500°C) temperature levels. Hence, from the point of view of corrosion resistance, such alloys are attractive.

The migration of brine inclusions to the canisters and the chemical reaction of various materials with brine are associated with the evolution of hydrogen and other gases. The rate of generation of hydrogen is calculated to be sufficiently low so that the normal ventilation provided in the storage rooms (to keep heat levels tolerable during repository operation) is more than adequate to prevent the buildup of an explosive mixture of hydrogen in air. Research on this potential problem is needed to define and resolve it for an actual repository design.

Internal corrosion of the canisters is chiefly a result of radiolytic gases (nitrogen dioxide, oxygen, and halogens). For glass waste forms, if the wall temperature is maintained sufficiently below the glass melting temperature, no internal corrosive attack is expected. Since the maximum wall temperature in the repository is maintained below 375°C, internal corrosion for canistered glass should be negligible. For canistered voloxidized fuel material, corrosion may occur as a result of nitrate ion attack on the stainless steel canister. Corrosion is probably negligible for canister wall temperatures below 300°C.

I.1.2 Effect of Mechanical and Thermal Stresses on Canisters

There is an additional problem associated with maintaining the retrievability of HLW or spent fuel emplaced in salt. This problem is caused by the gradual mechanical deformation or creep of the salt in response to thermal expansion. The creep is likely to be accelerated by the high ambient temperatures associated with the storage of HLW or spent fuel. The engineering design must assure that access to the emplaced waste can be assumed during the retrievability period and that the canisters are not mechanically entrapped by the deformation of the surrounding salt. The proposed 3/8-inch-thick carbon steel sleeve is designed to provide minimum protection against canister entrapment and damage. The sleeve design calls for a gap around the sleeve which may be filled with crushed salt.

I.2 COST OF CANISTER MATERIALS

A variety of candidate steel alloys and nickel-chrome alloys with a high nickel content are being considered for the HLW and spent fuel canisters. Carbon steel, stainless steel, and Inconel alloy materials have been discussed briefly in Subsection I.1.1.

The relative costs of candidate materials are given in Table I-2 in order to show the preliminary impact of materials selection based upon available corrosion data. Use of the nickel-chrome alloys reduces the amount of material required for corrosion allowance, but the costs of the exotic materials, even with lower corrosion allowances, far exceed the cost of the basic carbon steel canister.

Table I-2
RELATIVE COST OF MATERIALS

Material	Cost Factor
Carbon steels	1
Low alloy steels	2
Nickel-chrome alloys	10-50
Titanium	10-20
Stainless steels	5-10
Glass-coated steel	1-3

1.3 FACILITY DESIGN CONSIDERATIONS

The repository design requires that spent fuel and HLW packages be stored in the mine in a manner that permits all of the canisters to be retrieved during the first 5 years of operation. Therefore, the repository design incorporates features that function only for retrievability. If a canister is breached while in storage, it must be removed to a special area by means of special equipment.

Retrievability studies have been made by Kaiser Engineering on the National Waste Terminal Storage (NWTS) Repository No. 2 design⁽²⁾ and by Boeing Engineering Company on HLW waste canister emplacement and retrieval.⁽³⁾ On the basis of information given in these reports, no major design problems are anticipated during a 5-year retrievability period.

1.3.1 Facilities for Retrievability

The facilities required for implementing retrieval within the repository are divided into two categories: underground facilities and surface facilities. These facilities are indicated in the block diagram of Figure 1-2.

The underground facility required for the retrieval function is the combined transfer/overpack cell, which is an integral part of the remote handling facilities. The cell is similar to the clean cell on the surface and has all the necessary equipment for transferring the canisters from the remote handling waste transporter to a transfer cask. A cell decontamination system is provided, as well as a system to include overpack equipment for servicing the retrieved damaged canisters.

The surface facilities required for the retrieval functions are the hot cell, the canister decontamination cell, and the overpack and repair room. In the hot cell, receipt, inspection, transfer, repair, and temporary storage of radioactive waste canisters take place. All operations use remote controls and manipulators, in addition to a shielded crane. All canisters are decontaminated in the canister decontamination cell. Space is required in this cell for the rail-mounted canister decontamination vehicles. The overpack and repair room has a supply of container overpacks and work space for receiving retrieved-contaminated waste containers.

1.3.2 Equipment for Retrievability

Equipment unique to retrieval and equipment requiring design modifications in order to support the retrieval operation are as follows:

- Cask to receive retrieved canisters
- Equipment to locate buried canisters
- An overcore drilling system
- A transporter that can contain a salt overcore
- A device to recover sleeves from disposal wells if the waste package has been damaged and high levels of radioactive contamination are present inside the sleeve

To retrieve an intact canister emplaced in a sleeved well requires devices identical to the equipment originally used for placing the canister in the well. This equipment is described in Appendix D. To retrieve waste canisters emplaced in an unsleeved well, an additional overcore drilling system similar to the waste isolation pilot plant (WIPP) design or slot drilling system is required. This overcore drilling system must include the equipment required for salt removal.

Figure I-2.
Retrieval Block Flow Diagram

1.3.3 Instrumentation

Devices must be installed to monitor the repository and collect experimental data during the retrievability period. These devices provide information that help determine whether canisters should be retrieved.

Conditions in the mine that have to be detected and monitored are:

- Repository instability and displacement
- Excessive radiation levels
- Unacceptable ventilation differential pressures
- Excessive temperatures
- Excessive water intrusion

The required radiological instrumentation includes monitors for radiation and samplers for particulates. Temperature is monitored at selected points by thermocouple scanning systems.

1.3.4 Ventilation

Ventilation of the storage rooms is required during the retrieval operation to keep the working temperature down to a reasonable level. If the ventilation system provided during the room filling is still operating, no modifications will be required. If the ventilation system to a room has been blocked off, it will have to be reopened and temperatures brought back to required levels for the retrieval operations.

1.3.5 Radiological Protection

Since the retrieval operation is similar to a step-by-step reverse process of the emplacement of canisters in the repository, the radiation safety problems encountered during the storage operation are also present during retrieval operations. However, contamination control considerations during retrieval could be more involved if canisters have failed. All facilities and equipment required to handle and store canisters are designed in compliance with applicable radiation safety criteria and regulatory guidance for the as-low-as-reasonably-achievable (ALARA) standards.

1.3.5.1 Dose Equivalent and Dose Commitment. The results of dose equivalent and dose commitment estimates for occupational radiation exposure are given in Section 7. The integrated occupational exposure from decommissioning the NWTS Repository No. 2 is estimated by Kaiser Engineers to be 20 man-rem.⁽²⁾ These data are assumed to apply for this study.

1.3.5.2 Area and Perimeter Radiation Monitoring System. Since the retrievability period ends before the repository becomes full, no monitoring system, other than the existing system provided for the repository operation, is required specifically for retrieval.

1.3.6 Sleeve-Plug Design

In the study, retrieval of the waste during the first 5 years is enhanced by providing double containment of the waste during storage. A protective liner or sleeve is used within the storage well in the mine floor. The radioactive waste, sealed in a separate container, is lowered into the sleeved well. This will make retrieval easier if it should be found necessary. The principal functions of the sleeve are to eliminate salt contact with the waste canister and to allow in-situ test on the canister and sleeve during the retrieval period.

The design described in Appendix D uses a carbon steel sleeve. In view of the potential problems with corrosion (especially if retrievability periods longer than 5 years are considered) nonmetallic materials such as concrete should perhaps be given consideration as alternatives. A study by ORNL shows that the low cost and ease of placement of concrete make it an attractive sleeve material. Different concrete materials are available with a wide range of properties, and several have potential application.⁽⁴⁾

High-temperature cements have been developed for use in the petroleum industry and can be used at temperatures up to 700°C and in salt formations.

I.4 RETRIEVABILITY OPERATIONAL CONSIDERATIONS

The operations required for implementing retrievability within the repository consist of both subsurface and surface activities. The sequence of operations for the retrieval of canistered waste is shown in Figure I-2.

The principal subsurface retrieval activities at the repository are as follows:

- Ventilating the storage room
- Trimming the storage rooms walls and floors to maintain access by retrieval equipment
- Locating and hoisting the waste canister from the emplacement well into the cask
- Transporting the waste cask to the transfer cell
- Transferring the waste canister to another cask and then via the mine shaft to the surface facilities

The principal surface retrieval activities at the repository are as follows:

- Transferring the waste canister from the cask to the hot cell
- Decontaminating and cleaning the waste canister
- Loading the waste canister into the offsite cask
- Transferring the offsite cask to the decontamination area
- Decontaminating the offsite cask for shipment
- Transferring the offsite cask to the transport vehicle

I.4.1 Subsurface Operations

Prior to waste canister retrieval, the storage room must be ventilated sufficiently to reduce air and surface temperatures to the required operating levels. The storage room is surveyed for clearance of retrieval hardware and remined, if necessary. Retrieval area radiation background is measured to establish personnel occupancy time limits and shielding requirements.

The empty transfer cask is brought into the storage room on the transporter. The shielding retrieval fixture is then positioned over the storage well and the plug adaptor moved onto the fixture. Next, the doors in the loading fixture are opened, the shield plug is hoisted from the storage well, the shield doors reclosed, and the plug adaptor lifted onto the transporter. At this point, the operator moves the transporter over the emplacement well, tilts the cask to an upright position, and removes the canister from the storage well. Finally, the shield plug is reinserted into the top of the storage well.

If the shielding plug cannot be removed by the hoist, the operator must resort to overdrilling or slot drilling to remove the shielding plug. This procedure may be required for the removal of the sleeve if the canister cannot be withdrawn; however, for the 5-year retrieval period, it is unlikely that the canister will be entrapped.

The waste canister is hoisted into the transfer cask, and the transporter moves the canister in the cask to the underground transfer cell. The necessary equipment for transfer of the canisters from the remote handling waste transporter to the waste cage-shaft conveyance is provided in the transfer cell. Decontamination operations can be performed in this cell, and the canister overpacked, if required. The cask is raised through the shaft to the waste handling facilities at the surface.

The operating crew requirements for retrieval in the subsurface waste facility are similar to the operating crew requirements given in Appendix D. The overall operating crew may require as many as 100 people, including supervisors. As many as 22,000 casks per year could be handled by the system, if required.

1.4.2 Surface Operations

The canister in its transfer cask is removed from the mine shaft to the transfer cell. The latter provides for the receipt, inspection, transfer, and temporary storage of radioactive waste canisters. The canister is removed from the transfer cask, processed through a smear test fixture, and inspected for corrosion. If the canister surface is not contaminated and does not have excessive surface corrosion, it is moved directly to a shipping cask or placed in a holding station. If tests and inspection indicate canister contamination or excessive corrosion, the canister is sorted for further processing. If the contamination cannot be removed or if there is evidence of inadequate structural integrity, the canister has to be repaired or overpacked. The canister is then placed in a shipping cask or holding station.

The operating crew requirements for retrieval at the surface facilities are similar to the operating crew requirements given in Appendix D. The overall surface operating crew may require as many as 160 people, including supervisors.

REFERENCES FOR APPENDIX I

- (1) "Project Salt Vault — A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines," Union Carbide Corporation, Oak Ridge National Laboratory, ORNL-4555, April 1971.
- (2) "Conceptual Design Report National Waste Terminal Storage Repository No. 2," Parts 1 and 2, Kaiser Engineers, Oakland, California, NWTS-2990, August 1978.
- (3) "High Level Waste Canister Emplacement and Retrieval Concepts Study," Boeing Engineering and Construction with Union Carbide Corporation, Office of Waste Isolation, Y/OWI/SUB-76/16509, September 1976.
- (4) "Sleeve Materials and Concepts for the Bedded Salt Pilot Plant Repository," Oak Ridge National Laboratory, Oak Ridge, Tennessee, ORNL-73-3-34, April 1973.

Appendix J

ABBREVIATIONS AND DEFINITIONS

J.1 ABBREVIATIONS

AAR	American Association of Railroads
AFR	Away-From-Reactor Storage Facility
AGNS	Allied General Nuclear Services
AIF	Atomic International Forum
ALAP	As Low As Practical
ALARA	As Low As Reasonably Achievable
BMI	Battelle Memorial Institute
BNFP	Barnwell Nuclear Fuel Plant
BPNL	Battelle Pacific Northwest Laboratory
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CG	Concentration Guide
CPI	Consumer Price Index
CWPP	Continuing Work Program Plan
DOE	U.S. Department of Energy
DOG	Dissolver Offgases
DOT	U.S. Department of Transportation
EPA	U.S. Environmental Protection Agency
ERDA	U.S. Energy Research and Development Administration
FP	Fission Products
FRC	Federal Radiation Council
FSAR	Final Safety Analysis Report
GEIS	Generic Environmental Impact Statement
GP	General Purpose
GVW	Gross Vehicle Weight
HAW	High Activity Waste
HEF	Hot Experimental Facility (Advanced Fuel Recycle Program)
HLLW	High Level Liquid Waste
HLW	High Level Waste
IAEA	International Atomic Energy Agency
ICA	Item Control Area
ICC	Interstate Commerce Commission
ILLW	Intermediate Level Liquid Waste
ILW	Intermediate Level Waste

IMT	Intermodal Transport
INFCE	International Nuclear Fuel Cycle Evaluation
ISFSI	Independent Spent Fuel Storage Installation
KO	Knock Out
LAW	Low Activity Waste
LLLW	Low Level Liquid Waste
LLW	Low Level Waste
LMFBR	Liquid Metal Fast Breeder Reactor
LP	Linear Programming
LSA	Low Specific Activity
LWR	Light Water Reactor
LWT	Legal Weight Truck
MFRP	Midwest Fuel Recovery Plant
MOX	Mixed Oxides (of Pu and U)
MPC	Maximum Permissible Concentration
MT	Metric Ton
MTHM	Metric Ton of Heavy Metal
MTU	Metric Ton of Uranium
NASAP	Nonproliferation Alternative Systems Assessment Program
NDA	Nondestructive Assay (Determination)
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
NWR	National Waste Repository
NWTS	National Waste Terminal Storage
ONWI	Office of Nuclear Waste Isolation (of Battelle Project Management Division)
ORNL	Oak Ridge National Laboratory
OWI	Office of Waste Isolation (of Union Carbide Corporation — Nuclear Division)
P/E	Processing/Encapsulation
PPP	Plutonium Purification Process
PPV	Primary Pressure Vessel
PSAR	Preliminary Safety Analysis Report
PTV	Plutonium Transport Vessel
PWR	Pressurized Water Reactor
RHO	Rockwell Hanford Operations
SAI	Science Applications, Incorporated
SF	Spent Fuel
SFDS	Spent Fuel Disposal Study
SNM	Special Nuclear Materials
SRENCO	Stearns-Roger Engineering Company

SRL	Savannah River Laboratory
SSFR	Scoping Study Final Report
STP	Standard Temperature and Pressure
SURF	Spent Unprocessed Fuel (Facility)
SWU	Separative Work Unit
TD	Theoretical Density
TOG	Tritium Offgas
TRU	Transuranic
UCC-ND	Union Carbide Corporation — Nuclear Division
VOG	Vessel Offgas
WIPP	Waste Isolation Pilot Plant
WISAP	Waste Isolation Safety Assessment Program
WL	Working Level

J.2 DEFINITIONS

Actinide	A chemical element belonging to the actinide series beginning with actinium (89) and continuing through all the known heavier elements (usually concerned with one of the long-lived radionuclides in that series).								
Activity	Radioactivity or the presence of radioactive materials. A measure of the rate at which a material is emitting radiation. Usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time. The standard unit of radioactivity is the curie (Ci).								
Alpha emitter	A nuclide which undergoes radioactive decay by emitting an alpha particle, a positively charged particle.								
CFR	Code of Federal Regulations, subdivided by titles and parts, available from U.S. Government Printing Office, Washington, D.C.								
Criticality	State of being critical. A self-sustaining neutron chain reaction in which there is an exact balance between the production and loss of neutrons.								
Curie (Ci)	A unit of activity. An activity of 1 curie is assigned to the quantity of any radioactive species decaying at a rate of 3.70×10^{10} disintegrations per second. <table><tr><td>mCi</td><td>= 10^{-3} curie</td></tr><tr><td>μCi</td><td>= 10^{-6} curie</td></tr><tr><td>Ci/m³ or Ci/liter</td><td>= activity measure per volume of material</td></tr><tr><td>Ci/MWe yr</td><td>= cumulative activity associated with the production of 1 MWe/yr</td></tr></table>	mCi	= 10^{-3} curie	μ Ci	= 10^{-6} curie	Ci/m ³ or Ci/liter	= activity measure per volume of material	Ci/MWe yr	= cumulative activity associated with the production of 1 MWe/yr
mCi	= 10^{-3} curie								
μ Ci	= 10^{-6} curie								
Ci/m ³ or Ci/liter	= activity measure per volume of material								
Ci/MWe yr	= cumulative activity associated with the production of 1 MWe/yr								
CWT	One hundred weight = 112 lb.								
Disposal	The emplacement of radioactive packages in a repository designed to permanently isolate radionuclides from the biosphere, with no expectation of retrieval after emplacement.								
Dose, Integral	The product of the radiation dose and the mass of the body receiving that dose.								
Dose, Radiation	The amount of absorbed energy imparted to matter, when ionizing radiation passes through that matter, per unit mass of the irradiated material. Generally measured in rads (for γ or X-radiation in air, see "Roentgen"; for biological effects, see "Rem").								
Dose, Rate	Amount of radiation dose absorbed in a given time. Measured in rad/hr; rem/hr (R/hr for γ radiation in air).								
Effective utilization factor	Represents the HLW waste (in MTHM) that can be stored in the repository. Given as: HLW (MTHM)/gross area (acre).								
Extraction ratio	In the repository, the quantity: (width of room)/(width of room + width of pillar).								
Fission Products	General term applied to the complex, highly radioactive mixture of nuclides, consisting of the fission fragments and their various decay products.								
Fuel age	For the purpose of this study, the amount of time that has expired since the removal of the fuel from the reactor vessel.								

HLW, HLLW	See "Waste"
Holdup measurement	A measurement whose execution requires the stopping of the processing operations.
Hulls	The metallic residue resulting from the processing of spent fuel assemblies. These generally include: zircaloy fuel cladding; Inconel spacer grids; and instrument tabs, springs, etc. These metallic pieces are assumed to carry some small quantity of residual uranium, plutonium, and fission products.
ILW, ILLW	See "Waste."
LLW, LLLW	See "Waste."
MWD/MTHM	Megawatt-day/metric ton heavy metal; unit measure of cumulative exposure (burnup) of a nuclear fuel in a reactor, $1 \text{ MWD/MTHM} = 8.64 \times 10^{10}$ joules of energy released from 1 MTHM.
Overpack	A secondary package enclosing an original, breached package. It is designed to meet all original package requirements.
Processing/encapsulation facility	A generic term that in this study represents any facility used for processing and/or encapsulation of spent fuel.
Rad	Radiation absorbed dose. Measures the radiation dose. Applies to any type of radiation and any exposed material. The quantity of radiation resulting in the absorption of 100 ergs of energy per gram of irradiated material.
Radwaste	A general term, meaning radioactive waste. Refers to LLW and ILW (see "Waste").
Rem	Roentgen equivalent man. A biological dose unit defined as: $\text{rem (dose)} = \text{rad (dose)} \times \text{RBE (factor)}$, where $\text{RBE} = \text{relative biological effectiveness of radiation (for X rays, } \gamma \text{ rays, } \beta \text{ particles, and electrons of all energies, RBE} = 1)$.
Repository	A geologic disposal area for radioactive waste packages.
Retrievability	In general terms, the ability to extract and return the fuel and waste canisters from geologic storage back to the entrance to the repository surface facility. This would occur only as the result of an unforeseen situation wherein it is indicated that geological disposal of fuel/waste may jeopardize public safety and/or adversely impact the environment or biosphere.
Roentgen (R)	Applies only to γ (or X) radiation in air. $1 \text{ R} = \text{Quantity of } \gamma \text{ (or X) radiation producing}$ $1 \text{ esu (electricity) in ionizing } 1 \text{ cm}^3 \text{ of air at STP.}$
Shipping cask	A heavily shielded enclosure in which the radioactive package is placed to provide protection and reduce surface dose rates to permissible levels during transportation.
SNM	Special nuclear material. Plutonium, ^{233}U , or uranium enriched in the ^{235}U isotope. Strategic SNM is a subset of SNM that includes only those materials that could be used in the fabrication of a nuclear explosive. This

	subset includes plutonium, ^{233}U , and uranium enriched to 20 percent or more in the 235 isotope.
"Spiked" Plutonium	Plutonium in which a lethal quantity of radioactive material is enclosed to deter its use by saboteurs.
Surge	The capacity to accommodate radioactive materials at the front and/or back ends of a facility. The requirements for surge capacity may occur because of the unforeseen receipt of shipments, facility shutdown, or restrictions on downstream facilities. The surge capacity is a means of normalizing material flow through the facility.
Throughput	A general term referring to the measurement of the rate of flow through a facility (e.g., 5,000 MTHM/yr).
TRU elements	Elements having atomic numbers greater than 92 (such as neptunium, plutonium, americium, curium, berkelium and californium).
TRU isotopes	Isotopes of transuranic (TRU) elements.
Unit Cell	A volume within the repository enclosing a waste canister. This volume is bounded by horizontal planes through the top and bottom of the canister and vertical planes through the centerline of the adjacent salt pillars.
Voloxidation	A process to promote the release of gaseous fission products from oxide fuel by oxidation of UO_2 to U_3O_8 at 400C to 500C in air.
Waste (radioactive)	<p>TRU waste — Contains more than 10 nCi of TRU elements per gram of waste.</p> <p>Non-TRU waste — Contains less than 10 nCi of TRU elements per gram of waste.</p> <p>Solid waste — Categorized by dose rate level (measured in mrem/hr or rem/hr).</p> <p>LLW: ≤ 10 mrem/hr ILW: > 10 mrem/hr to ≤ 15 rem/hr HLW: > 15 rem/hr</p> <p>Liquid waste — Rated by activity level (measured in Ci/liter).</p> <p>LLLW: ≤ 1 mCi/liter ILLW: > 1 mCi/liter to ≤ 1 Ci/liter HLLW: > 1 Ci/liter</p> <p>Comments:</p> <ol style="list-style-type: none"> 1. All HLW is TRU waste. 2. ILW and LLW can be TRU or non-TRU waste. 3. HLW can be high- or low-heat producing. 4. ILW and LLW are low-heat producing only.
Working level (WL)	Any combination of short-lived radon daughter products in 1 liter of air (at STP) that would result in the ultimate emission of 1.3×10^5 MeV of potential energy (10 CFR 12).

DISTRIBUTION LIST

ACRES AMERICAN INC
A S BURGESS

ALABAMA STATE GEOLOGIST
T J JOINER

ALASKA STATE GEOLOGIST
R G SCHAFF

ALLIED CHEMICAL CORPORATION
C M SLANSKY

ALLIED GENERAL NUCLEAR SERVICES
P F HIGHBERGER

AMERICAN NUCLEAR ENERGY COUNCIL
EDWARD M DAVIS

ANALYTIC SCIENCES CORPORATION
JOHN W BARTLETT
C KOPLIK

ANCO ENGINEERS, INC
ROBERT M WEBB

ANGELS STATE UNIVERSITY
CHESTER M ROWELL JR

ARGONNE NATIONAL LABORATORY
REX COUTURE
KEVIN FLINN
ARNOLD M FRIEDMAN
L J JARDINE
J H KITTEL
M STEINDLER

ARIZONA STATE GEOLOGIST
H WESLEY PIERCE

ARKANSAS STATE GEOLOGIST
NORMAN F WILLIAMS

ARMADILLO COALITION OF TEXAS
AMY MARTIN

ARTHUR D LITTLE INCORPORATED
CHARLES R HADLOCK

**ATOMIC ENERGY CONTROL BOARD
(CANADA)**
J L WALLACE

ATOMIC ENERGY OF CANADA LIMITED
P C FUNG
S A MAYMAN
EVA ROSINGER
TERRENCE RUMMERY
F P SARGENT
M TOMLINSON

**ATOMIC ENERGY RESEARCH
ESTABLISHMENT (UK)**
R L NELSON
H A C McKAY
JOHN RAE

ATOMIC INDUSTRIAL FORUM
P GARRETT

AUTOMATION INDUSTRIES INC
THOMAS T BRADSHAW
T M ROBERTSON

BATTELLE COLUMBUS LABORATORIES
BEVERLY AUSMUS
SANFORD BLOOM
DAVID HASTINGS
JEFF MEANS
V YOUNG PARK
KENNETH S WURM
W J ZIEFFENBACH

BATTELLE SEATTLE
A HENRY SCHILLING
LIBRARY SERVICES

BAYLOR UNIVERSITY
O T HAYWARD

BECHTEL NATIONAL INC
D S DUNCAN
J B KEMP
C W KUHLMAN
R A LANGLEY JR
N A NORMAN

**BHABHA ATOMIC ENERGY
ESTABLISHMENT (INDIA)**
K T THOMAS

**BOEING ENGINEERING AND
CONSTRUCTION**
C P BLACK
GEORGE DYMMEL
E S KEENE

BIRGER JARLSGATAN 41A (SWEDEN)
LARS B NILSSON

BOWLING GREEN STATE UNIVERSITY
C F KAHLE

BREAZEALE REACTOR
SAMUEL H LEVINE

BRIGHAM YOUNG UNIVERSITY
DEPARTMENT OF ARCHAEOLOGY
KIMBALL T HARPER
ROBERT L LAYTON
GLEN T NELSON
MORRIS S PETERSON
STANLEY L WELSH

**BROOKHAVEN NATIONAL
LABORATORY**
RICHARD E DAVIS
A J FRANCIS
P W LEVY
D MAJUMDAR
DONALD SCHWEITZER

BROWN UNIVERSITY
B GILETTI

BURNS AND ROE
JOHN PIRRO

**CALIFORNIA DEPARTMENT OF
CONSERVATION**
JAMES F DAVIS

**CAPITAL AREA (LA) GROUNDWATER
CONSERVATION COMMISSION**
A N TURCAN

CATHOLIC UNIVERSITY OF AMERICA
D D EBERT
PAUL E THIESS

**CAYUGA COUNTY (NY) PLANNING
BOARD**
ROBERT BROWER
JOHN SPOTO

**CENTER FOR ENVIRONMENTAL
INFORMATION**
ELIZABETH THORNDIKE

**CENTRAL NEW YORK REGIONAL
PLANNING AND DEVELOPMENT BOARD**
WALKER BENNING

CITIZENS AGAINST NUCLEAR TRASH
PETTUS WALLEY

CLARION STATE COLLEGE
J A LASWICK

**CLEVELAND ELECTRIC ILLUMINATING
COMPANY**
ALBERT OBERMEYER
FRANK STEAD

**CLEVELAND MUSEUM OF NATURAL
HISTORY**
DAVID R BUSH

**COLLEGE OF ENVIRONMENTAL
SCIENCES AND FORESTRY**
ROBERT HENNIGAN

COLORADO GEOLOGICAL SURVEY
JOHN W ROLD

COLORADO SCHOOL OF MINES
S S GOLDICH

COLUMBIA UNIVERSITY
WALLACE BROECKER

COMMONWEALTH ASSOCIATES INC
YOSSEF BALAS

**CONNECTICUT OFFICE OF
POLICY AND MANAGEMENT**
FAITH BRENNEMAN
TOM FITZPATRICK

CONNECTICUT STATE GEOLOGIST
HUGO F THOMAS

CONTROL DATA CORPORATION
R F KUECHENBERG

CORNELL UNIVERSITY
JOHN BIRD
ARTHUR L BLOOM
K B CADY
CYNTHIA DYBALLA
JACK OLIVER
ROBERT POHL

**CORPORATIONS AUTHORITIES AND
PUBLIC UTILITIES COMMITTEE (NY)**
JAMES McFARLAND

**CORTLAND COUNTY (NY) HEALTH
DEPARTMENT**
J V FEUSS

DALTON-DALTON-LITTLE-NEWPORT
STEVE NACHT

DAMES AND MOORE
A E AIKENS JR
ROBERT R BLICKWEDEHL
SUE FINGERMAN
TODD M GATES
LIBRARIAN
RICHAD M WINAR

**D'APPOLONIA CONSULTING
ENGINEERS INCORPORATED**
R D ELLISON

DARTMOUTH COLLEGE
J B LYONS

DATA-DESIGN LABORATORIES
PROGRAM DEVELOPMENT

DAWCON
DAVID A WEBSTER

**DEEP EAST TEXAS COUNCIL
OF GOVERNMENTS**
R E BLANKS
R L C WALKER

DELAWARE STATE GEOLOGIST
ROBERT R JORDAN

DEPARTMENT OF ENVIRONMENT (UK)
A G DUNCAN

EARTH SCIENCE LABORATORY
LIBRARIAN

**EASTGATE DEVELOPMENT AND
TRANSPORTATION AGENCY**
JOHN R GETCHEY

**EAST TEXAS COUNCIL
OF GOVERNMENTS**
TOM SMISER

**ECOLOGICAL CENTER OF
LOUISIANA INC**

EG&G IDAHO INCORPORATED
G B LEVIN
T H SMITH

ELSAM (DENMARK)
ARNE PEDERSEN

ENERGY INFORMATION CENTER
ROBERT B BURTCH JR

ENERGY RESEARCH GROUP INC
MARC W GOLDSMITH

ENERGY RESOURCES GROUP
JAMES CLINE

ENERGY SYSTEM GROUP
LIBRARY

ENVIRONMENTAL EVALUATION GROUP
ANN BANCROFT

ENVIRONMENTAL PLANNING LOBBY
MARILYN DU BOIS

ENVIROSPHERE COMPANY
J FRANCO

**ERIE (NY) COUNTY ENVIRONMENTAL
MANAGEMENT COUNCIL**
JOAN P SCHMIDT

EXXON CORPORATION
ARTHUR G RANDOL III

EXXON NUCLEAR COMPANY INC
GRAY WAYMIRE

FALCON RESEARCH AND DEVELOPMENT
WILLIAM J GALYEAN

**FIVE COUNTY (UT) ASSOCIATION
OF GOVERNMENTS**
RHEAD S BOWMAN

**FLORIDA DEPARTMENT OF NATURAL
RESOURCES**
CHARLES W HENDRY JR

FLORIDA POWER AND LIGHT COMPANY
JAMES R TOMONTO

FMC CORPORATION
MIKE WALLIS

**FORD BACON AND DAVIS ENGINEERS
CONSTRUCTORS**
DARRELL H CARD
ARTHUR SUTHERLAND
BURTON J THAMER

**FOSTER MILLER ASSOCIATES
INCORPORATED**
GREY L RILEY

**FOUNDATION SCIENCES
INCORPORATED**
LOU BATTAMS

**FREDERIC F MELLEN GEOLOGICAL
ASSOCIATES**
F F MELLEN

FRIENDS OF THE EARTH
JEFF NEWMAN
SALLY RODGERS
LORNA SALZMAN

FSU ENVIRONMENTAL TASK FORCE
SUC GENESSEO

FUGRO INCORPORATED
J CARL STEPP

GENERAL ATOMIC COMPANY
ROBERT M BURGOYNE
G W HANNAMAN
DOUGLAS D ORVIS
JAMES N SILTANEN
R F TURNER

**GEOLOGICAL SOCIETY OF AMERICA
INCORPORATED**
JOHN C FRYE

GEOLOGICAL SOCIETY OF CANADA
J GALE
B SANFORD
J S SCOTT

GEOLOGICAL SURVEY OF SWEDEN
OTTO BROTZEN

**GEORGIA DEPARTMENT OF NATURAL
RESOURCES**
JOHN R GEORGE

GEORGIA INSTITUTE OF TECHNOLOGY
GEOFFREY EICHHOLZ
LAWRENCE W FRY
J W POSTON
JOHN RUSSELL
CHARLES E WEAVER

GEOTECHNICAL ENGINEERS INC
RICHARD W TURNBULL

**GESELLSCHAFT FUR KERNFORSCHUNG
MBH**
HELMUT KRAUSE

GIBBS & HILL
JACK BADMAN

**HANFORD ENGINEERING DEVELOPMENT
LABORATORY**
R L KNECHT

HARVARD UNIVERSITY
R SIEVER

**HAWAII DEPARTMENT OF LAND AND
NATURAL RESOURCES**
ROBERT T CHUCK

HOWARD UNIVERSITY
GEORGE FERGUSON

IAEA (AUSTRIA)
ROBERT CATTLIN

**IDAHO BUREAU OF MINES AND
GEOLOGY**
MAYNARD M MILLER

IDAHO STATE UNIVERSITY
V HARRY CHARYULU

IIT RESEARCH INSTITUTE
E R BANGS

**ILLINOIS ATOMIC ENERGY
COMMISSION**
HAP DAY

ILLINOIS STATE GEOLOGICAL SURVEY
JACK A SIMON

INDIANA STATE GEOLOGIST
JOHN B PATTON

INDUSTRIAL RESEARCH/DEVELOPMENT

**INSTITUTE OF GEOPHYSICAL SCIENCES
(UK)**
DAVID A GRAY

INSTITUTE OF NATURAL RESOURCES
ANTHONY LIBERATORE

**INTERNATIONAL ENERGY
ASSOCIATION LTD**
JOHN B HENDERSON

**INTERNATIONAL ENGINEERING
COMPANY INCORPORATED**
JOHN COGAN
HEADQUARTERS OFFICE

**INTERNATIONAL RESEARCH AND
EVALUATION**
R DANFORD

INTERNATIONAL SALT COMPANY
R L FRIANT
D K PAUL

IOWA STATE GEOLOGIST
STANLEY C GRANT

IOWA STATE UNIVERSITY
RICHARD DANOFSKY

IRT CORPORATION
W E SELPH

**JAMES F MACLAREN LIMITED
(CANADA)**
VALERIE CASSADAY

**JAPAN ATOMIC ENERGY RESEARCH
INSTITUTE**
TARO ITO

JOHNS HOPKINS UNIVERSITY
JARED COHON

KAISER ENGINEERS INCORPORATED
W J DODSON
J S RITCHIE

KAMAN SCIENCES CORPORATION
PAUL A ELLIS

KANSAS STATE GEOLOGICAL SURVEY
WILLIAM W HAMBLETON

KBS (SWEDEN)
TONIS PAPP

KENT STATE UNIVERSITY
JIM COWDEN
RONALD W MANUS

KENTUCKY GEOLOGICAL SURVEY
DONALD C HANEY

**LAKE COUNTY (OH) PLANNING
COMMISSION**
DAVID F GILMER

LAW ENGINEERING TESTING COMPANY
J G LA BASTIE

LAWRENCE BERKELEY LABORATORY
JOHN A APPS
LEWIS COHEN
PAUL A WITHERSPOON

LAWRENCE LIVERMORE LABORATORY
L B BALLOU
R J BORG
DAVID G COLES
A DUBA
H C HEARD
ALFRED HOLZER JR
DANA ISHERWOOD
F R KOVAR
CAMILLE MINICHINO
L D RAMSPOTT
W G SUTCLIFFE
DONALD TOWSE

LEHIGH UNIVERSITY
D R SIMPSON

LOS ALAMOS SCIENTIFIC LABORATORY
K E APT
J R BRIDWELL
G A COWAN
BRUCE R ERDAL
D C HOFFMAN

W C LUTH
KURT WOLFSBERG

**LOS ALAMOS TECHNICAL ASSOCIATES
INCORPORATED**
S E LOGAN

**LOUISIANA AIR CONTROL
COMMISSION**
JAMES F COERVER

**LOUISIANA DEPARTMENT OF
CONSERVATION**
B JIM PORTER
R T SUTTON

**LOUISIANA DEPARTMENT OF
NATURAL RESOURCES**
WILLIAM C HULS

**LOUISIANA DEPARTMENT OF
PUBLIC SAFETY**
G W GARRISON

**LOUISIANA DEPARTMENT OF
TRANSPORTATION AND DEVELOPMENT**
GINO D CARLUCCI JR

LOUISIANA GEOLOGICAL SURVEY
CHARLES S GROAT
LEO W LOUGH

**LOUISIANA NUCLEAR ENERGY
DIVISION**
L H BOHLINGER

**LOUISIANA OFFICE OF
COMMERCE AND INDUSTRY**
ANDREW F FLORES
RESEARCH DIVISION

**LOUISIANA STATE PLANNING
OFFICE**
LIBRARY

LOUISIANA STATE UNIVERSITY
JEFFERY S HANOR
E B JERNIGAN
JOHN LOOS
J D MARTINEZ
ROBERT L THOMS
STEVEN WHIPPLE

LOZIER
WAYNE P ACKART

MAINE STATE GEOLOGIST
ROBERT G DOYLE

MARYLAND GEOLOGICAL SURVEY
KENNETH N WEAVER

**MASSACHUSETTS OFFICE OF
ENERGY RESOURCES**
PATRICK FORRESTER

MASSACHUSETTS STATE GEOLOGIST
JOSEPH A SINNOTT

MEMBERS OF THE GENERAL PUBLIC

ROHIT AHUJA
L Z BLANKENSHIP
LARRY BRADLEY
RANDY BREWER
HAL BRODIE
A L BROKAW
D W BYERLY
IRENE DICKENSON
JOHN ROGER DRAKE
BILL DUESING
JOHN GLOVER
EDWIN D GOEBEL
S GONZALES
CHARLES KILLGORE
K K LANDES
BRANDT MANNCHEN
HARVEY W MERRELL
J B MUCKERHEIDE
A M PIPER
M A SABET
WARREN SCHADT
MICHAEL L TABONY
L S TOBIAS
CHARLES TRAUTMANN
R F WALTERS
RICH WINDHOLZ

MICHIGAN STATE GEOLOGIST
ARTHUR E SLAUGHTER

**MICHIGAN TECHNOLOGICAL
UNIVERSITY**
J KALLIOKOSKI

MILLSAPS COLLEGE
CHARLES SALLIS
ERIK YENSON

MINNESOTA GEOLOGICAL SURVEY
MATT S WALTON

**MISSISSIPPI AIR AND WATER POLLUTION
CONTROL COMMISSION**
CHARLES T BRANCH
CHARLES CHISOLM

**MISSISSIPPI FUEL AND ENERGY
MANAGEMENT COMMISSION**
P T BANKSTON
PETER J WALLEY

**MISSISSIPPI GEOLOGICAL ECONOMIC
AND TOPOGRAPHICAL SURVEY**
ALVIN R BICKER JR
WILLIAM H MOORE

**MISSISSIPPI MUSEUM OF NATURAL
SCIENCE**
NATURAL HERITAGE PROGRAM

MISSISSIPPI POWER AND LIGHT
DONALD LUTKEN

MISSISSIPPI STATE UNIVERSITY
BOB T CHAPIN
RICHARD A MARSHALL
ROGER S REYNOLDS
VICTOR L ZITTA

**MISSISSIPPI WATER RESOURCES
DIVISION**
C A SPIERS

MISSISSIPPI WILDLIFE FEDERATION
CARLTON OWEN

MISSOURI GEOLOGICAL SURVEY
WALLACE B HOWE

MITSUBISHI METAL CORPORATION
JOAN C ABENA

MONSANTO RESEARCH CORPORATION
K V GILBERT

**MONTANA BUREAU OF MINES AND
GEOLOGY**
S L GROFF

**MORGANTOWN (WV) ENERGY
RESEARCH CENTER**
WILLIAM K OVERBEY

NASA JOHNSON SPACE CENTER
NORMAN HUBBARD

NASA LEWIS RESEARCH CENTER
JACK B ESGAR

NATIONAL ACADEMY OF SCIENCES
JOHN POMEROY

NATIONAL ATOMIC MUSEUM
GWEN SCHREINER

NATIONAL AUDUBON SOCIETY
DEDE ARMENTROUT

NATIONAL BUREAU OF STANDARDS
OFFICE OF MEASUREMENTS FOR
NUCLEAR TECHNOLOGY

NATIONAL SCIENCE FOUNDATION
ROYAL E ROSTENBACH

**NATURAL RESOURCES DEFENSE
COUNCIL**
T R LASH

**NEBRASKA CONSERVATION AND
SURVEY DIVISION**
VINCENT H DREESZEN

NEFCO
JAMES T KING

NETHERLAND SEWELL AND ASSOCIATES
CLARENCE M NETHERLAND

**NEVADA BUREAU OF MINES AND
GEOLOGY**
JOHN H SCHILLING

NEVADA DEPARTMENT OF ENERGY
NOEL CLARK

**NEVADA OFFICE OF PLANNING
COORDINATION**

**NEVADA STATE PLANNING
COORDINATOR**
BRUCE ARKELL

NEW HAMPSHIRE STATE GEOLOGIST
GLENN W STEWARD

NEW JERSEY STATE GEOLOGIST
KEMBLE WIDMER

NEW JERSEY INSTITUTE OF TECHNOLOGY
BEN STEVENSON

**NEW MEXICO BUREAU OF MINES AND
MINERAL RESOURCES**
FRANK E KOTTELOWSKI

**NEW YORK STATE DEPARTMENT OF
ENVIRONMENTAL CONSERVATION**
THOMAS J CASHMAN
TERRANCE P CURRAN
MELMER E WAGNER

**NEW YORK STATE ELECTRIC AND GAS
CORPORATION**
F S DOOLITTLE
PAUL ROMAR
LEWIS L STALEY

NEW YORK STATE ENERGY OFFICE
T K DE BOER

NEW YORK STATE GEOLOGICAL SURVEY
RICHARD DANA
JAMES F DAVIS
ROBERT H FAKUNDINY
ROBERT H FICKIES

**NORTH CAROLINA DEPARTMENT OF
NATURAL RESOURCES AND
COMMUNITY DEVELOPMENT**
STEPHEN G CONRAD
GEOLOGICAL SURVEY SECTION

NORTH DAKOTA STATE GEOLOGIST
LEE C GERHARD

NORTHEAST LOUISIANA UNIVERSITY
JAN FULLERTON
JOHN LEWIS
B E PRINCE

**NORTHEAST OHIO AREAWIDE
COORDINATING AGENCY**
FREDERICK E J PIZZEDAZ

**NORTHEAST OHIO FOUR COUNTY
REGIONAL PLANNING AND
DEVELOPMENT ORGANIZATION**
R C LARLHAM

NUCLEAR ASSURANCE CORPORATION
CLAUDIA DRUM
CAROL THORPU

NUCLEAR ENERGY AGENCY (FRANCE)
F GERA

**NUCLEAR FUEL CYCLE SECTION
(W GERMANY)**
M HAGEN

NUCLEAR SAFETY ASSOCIATES
J A LIEBERMAN

NUCLEAR SERVICES CORPORATION
FRANCIS J KENESHEA

**NUCLEAR SYSTEMS ASSOCIATES
INCORPORATED**

NUCLEAR WASTE WATCHERS
HELEN LETARTE

NUS CORPORATION
W G BELTER
STELLA BREGMAN
R J DAVIS
J DI NUNNO
M I GOLDMAN
BRUCE D GUILBEAULT
PAT MACHLEDT
SUSAN MURTHA
BARRY N NAFT
DAVID E RUTTER
L A STINNETT

OAK RIDGE ASSOCIATED UNIVERSITIES
H C MACPHERSON
A M WEINBERG

OAK RIDGE NATIONAL LABORATORY
G D BRUNTON
H C CLAIBORNE
A G GROFF
G H JENKS
T F LOMENICK
A L LOTT
G SCHWARZ
E G ST CLAIR

**OHIO DEPARTMENT OF ECONOMIC
AND COMMUNITY DEVELOPMENT**
ROGER B WILLIAMS

OHIO DEPARTMENT OF ENERGY
ROBERT RYAN

**OHIO DEPARTMENT OF NATURAL
RESOURCES**
HORACE R COLLINS

OHIO ENVIRONMENTAL COUNCIL
ERIC SMITH

OHIO EPA
JOHN NOYES

OHIO STATE UNIVERSITY
R N CHRISTENSEN
F A KULACKI
WAYNE PETTYJOHN
F UPHAM

OHIO UNIVERSITY RESEARCH INSTITUTE
GIRARD E KREBS
ADAM MARSH

OKLAHOMA GEOLOGICAL SURVEY
CHARLES J MANKIN

ONTARIO COMPANY
ROBERT BARROWS

ONTARIO HYDRO
R W BARNES
C F LEE

OREGON DEPARTMENT OF ENERGY
J TAPPAN

OREGON STATE GEOLOGIST
DONALD A HULL

OREGON STATE UNIVERSITY
BRIAN DODD

PACIFIC GAS AND ELECTRIC COMPANY
GENE BLANC

PACIFIC NORTHWEST LABORATORIES
DON J BRADLEY
A BRANDSTETTER
L L BURGER
J B BURNHAM
M KRIETER
A M PLATT
J R RAYMOND
R JEFF SERNE
R WALTER
E C WATSON
R D WIDRIG
L D WILLIAMS

**PANHANDLE-PLAINS
HISTORICAL MUSEUM**
BILLY R HARRISON

**PANHANDLE REGIONAL PLANNING
COMMISSION**
GEORGE LOUDDER

**PARSONS BRINCKERHOFF QUADE
AND DOUGLAS INC**
T R KUESEL

**PENNSYLVANIA DEPARTMENT OF
ENVIRONMENTAL RESOURCES**
ARTHUR A SOCOLOW

PENNSYLVANIA STATE UNIVERSITY
WILLIAM A JESTER
D M ROY
WILLIAM B WHITE
MICHAEL ZOLENSKY

**PERMIAN BASIN REGIONAL
PLANNING COMMISSION**
E W CRAWFORD

PERRY COMPANY
W F BOWEN

PIRGIM
RONALD W WILSON

**PLANNING AND ENVIRONMENTAL
ASSOCIATES**
THOMAS NIEDERKORN

**POLYTECHNIC INSTITUTE OF
NEW YORK**
RAPHAEL ARONSON

PORTLAND GENERAL ELECTRIC
J W LENTSCH

POTOMAC ALLIANCE
FRED MILLAR

PRINCETON UNIVERSITY
HARTMUT KRUGMANN
G E PINDER

PROJECT REACH INCORPORATED
ROBERT BATES

PUBLIC LAW UTILITIES GROUP

PURDUE UNIVERSITY
R H JOHNSON

RAILROAD COMMISSION OF TEXAS
J H MORROW

RALPH M PARSONS COMPANY
ALVIN E SMITH

REDDY COMMUNICATIONS INC
LIBRARY

RENSSELAER POLYTECHNIC INSTITUTE
BIMAL MALAVIYA

RE/SPEC INCORPORATED
PAUL F GNIRK
W C MCCLAIN

RIDIHALGH EGGERS AND ASSOCIATES
J RIDIHALGH

**ROCHESTER GAS AND ELECTRIC
CORPORATION**
L L PHILLIPS
RICHARD J WATTS

**ROCHESTER SAFE ENERGY
ALLIANCE**

ROCKWELL HANFORD OPERATIONS

H BABAD
G S BARNEY
R A DEJU
B DIETZ
GEORGE C EVANS
W J KURZEKA
C W MANRY
R A SEXTON
MICHAEL J SMITH
DAVE A TURNER
D D WODRICH
D E WOOD
P A YBARRA

ROCKWELL INTERNATIONAL

R H KARLSSON
R J MERLINI
LAWRENCE J SMITH
R E SMITH

SANDIA LABORATORIES

E H BECKNER
J F CUDERMAN
DEPARTMENT 4510
R G DOSCH
JERRY M FREEDMAN
LESLIE R HILL
THOMAS E HINKEBEIN
O E JONES
R D KLETT
R LINCOLN
R W LYNCH
G F RUDOLFO
A R SATTLER
L W SCULLY
A E STEPHENSON
DANIEL M TALBERT
L D TYLER
W D WEART
WIPP CENTRAL FILES

SAVANNAH RIVER LABORATORY

J L CRANDALL
R G GARVIN
H J GROH
C H ICE
F D KING
I W MARINE
W C REINIG

**SCIENCE APPLICATIONS
INCORPORATED**

SOPHIE CARMAN
C W CRAVEN
RONALD HOFFMANN
DAVID H LESTER
JOHN E MOSIER
WILLIAM R RHYNE
RICHARD W STAROSTECKI
M J SZULINSKI
ROBERT WILEMS

SCIENTIFIC RESEARCH EVALUATION
RONALD C CALLEN

SHAHER EXPLORATION COMPANY
WILLIAM E SHAHER

SIERRA CLUB MISSISSIPPI CHAPTER

**SIERRA CLUB NORTHEAST
OHIO GROUP**
T JENKINS

S M STOLLER CORPORATION
W KUPP

**SOLUTION MINING RESEARCH
INSTITUTE**
H DIAMOND

SOUTH CAROLINA STATE GEOLOGIST
NORMAN K OLSON

**SOUTH DAKOTA SCHOOL OF MINES
AND TECHNOLOGY**
W GRAMS

SOUTH DAKOTA STATE GEOLOGIST
DUNCAN J MCGREGOR

**SOUTHEASTERN UTAH ASSOCIATION
OF GOVERNMENTS**
WILLIAM K DINEHART

**SOUTHERN TIER CENTRAL REGIONAL
PLANNING AND DEVELOPMENT BOARD**
KEVIN BYRNE
ANN CLARKE
WILLIAM D HESS
LINDLEY V PRYOR

SOUTH PLAINS ASSOCIATION
TRUETT MAYES
COMMUNITY DEVELOPMENT
PLANNER

SRI INTERNATIONAL
R K WHITE

STANFORD UNIVERSITY
RONALD A HOWARD
KONRAD B KRAUSKOFF
P KRUGER
IRWIN REMSON

**STATE UNIVERSITY COLLEGE
(CORTLAND)**
DANIEL J BRENNAN

STATE UNIVERSITY COLLEGE (BUFFALO)
IRVING H TESMER

**STATE UNIVERSITY OF NEW YORK
(ALBANY)**
KEVIN BURKE

**STATE UNIVERSITY OF NEW YORK
(BINGHAMTON)**
ALBERT A DEKIN JR
T W DONNELLY

**STATE UNIVERSITY OF NEW YORK
(SYRACUSE)**
DONALD F BEHREND
PETER BLACK
ARTHUR R ESCHNER
NORMAN RICHARDS

**STEARNS-ROGER ENGINEERING
COMPANY**
J H JONES

STEPHEN F AUSTIN UNIVERSITY
JAMES E CORBIN
ELRAY NIXON

**STONE AND WEBSTER ENGINEERING
CORPORATION**
CRAIG F GROCHMAL
J PECK

SUL ROSS STATE UNIVERSITY
BARTON H WARNOCK

SYRACUSE UNIVERSITY
LEE HARRINGTON
ERNEST H MULLER

SYSTEMS SCIENCE AND SOFTWARE
PETER LAGUS

**TENNESSEE DEPARTMENT OF
PUBLIC HEALTH**
ROBERT H WOLLE

TENNESSEE ENERGY AUTHORITY
JACK THOMAS

TENNESSEE STATE GEOLOGIST
ROBERT E HERSHEY

TEXAS A&M UNIVERSITY
J HANDIN
ROY W HANN JR
D K PARRISH
STEPHEN RITER
J RUSSELL
JOSEPH L SCHUSTER
JIM SMATHERS
JAMES G TEER

TEXAS CHRISTIAN UNIVERSITY
JACK L WALPER

**TEXAS DEPARTMENT OF WATER
RESOURCES**
HARVEY D DAVIS

TEXAS ENERGY ADVISORY COUNCIL
ALVIN ASKEW
MILTON L HOLLOWAY

TEXAS ENVIRONMENTAL COALITION
STEVE FRISHMAN

TEXAS INDUSTRIAL COMMISSION
PHYLLIS PROCTER

TEXAS TECHNICAL UNIVERSITY
JOHN S MECHAM
BILL MEYER-OAKES

TIOGA COUNTY (NY) PLANNING BOARD
MARY S HARNON

TOMPKINS COUNTY (NY) DEPARTMENT OF PLANNING
FRANK R LIGUORI

TOUHILL, SHUCKROW AND ASSOCIATES
C J TOUHILL

TRW SYSTEMS AND ENERGY
E R CHRISTIE
LIBRARIAN
F MERTES
HERBERT N ROSENBERG

TULANE UNIVERSITY
CHARLES CARTER
WALTER MASON

TUSKEGEE INSTITUTE
IRA G DILLON

UNDE GMBH
DR OLINGER

UNIVERSITY OF AKRON
R G CORBETT

UNIVERSITY OF ARIZONA
JAAK DAEMEN
STANLEY N DAVIS
CHARLES LYON
JAMES C MCCRAY
R G POST
ROBERT L SEALE
DAVID L SOUTH

UNIVERSITY OF CALIFORNIA (BERKELEY)
NEVILLE G W COOK
KARI SAARI

UNIVERSITY OF CALIFORNIA (LOS ANGELES)
D OKRENT

UNIVERSITY OF FLORIDA
WILLIAM H ELLIS
M J OHANIAN

UNIVERSITY OF GEORGIA
J D SPAULDING
DAVID B WENNER

UNIVERSITY OF HAWAII AT MANOA
MURLI H MAUGHNANI

UNIVERSITY OF HOUSTON
JOHN R HOWELL

UNIVERSITY OF ILLINOIS
FRED A DONATH
GLENN GERDIN

UNIVERSITY OF INDIANA
J B DROSTE
HAYDN H MURRAY
CHARLES J VITALIANO

UNIVERSITY OF KANSAS
LOUIS F DELWIG

UNIVERSITY OF LOWELL
WILLIAM FILLIPONE
JOSE MARTIN
JAMES R SHEFF

UNIVERSITY OF MARYLAND
KAZGS ALMENAS
DICK DUFFEY
FRANK MUNNO

UNIVERSITY OF MICHIGAN
JOHN S KING

UNIVERSITY OF MINNESOTA
S L CROUCH
E N LINDNER
D H YARDLEY

UNIVERSITY OF MISSISSIPPI
CLYDE E COOK
FREDERICK MANLEY
ED MEEK
VELON MINSHEW
MARLIN PURDIE

UNIVERSITY OF MISSOURI (KANSAS CITY)
TRUMAN STAUFFER

UNIVERSITY OF MISSOURI (COLUMBIA)
W D KELLER
WALTER MEYER
DARROL H TIMMONS

UNIVERSITY OF MISSOURI (ROLLA)
NICK TSOLFANIDIS

UNIVERSITY OF NEW MEXICO
E LOGAN
D M WOODALL

UNIVERSITY OF NORTH CAROLINA (CHAPEL HILL)
JAMES H CRAWFORD

UNIVERSITY OF NOTRE DAME
JOHN LUCEY
K T YANG

UNIVERSITY OF OKLAHOMA
KENNETH S JOHNSON

UNIVERSITY OF PENNSYLVANIA
SATI SADHAL

UNIVERSITY OF PITTSBURGH
B L COHEN

UNIVERSITY OF RHODE ISLAND
EDWARD P LAINE
ROBERT L McMASTER

UNIVERSITY OF SOUTHERN MISSISSIPPI
CHARLES R BRENT
RALPH D CROSS
GEORGE PASSONEY
OSCAR L PAULSON JR
JAMES W PINSON
JOHN N SKATES

UNIVERSITY OF TENNESSEE
J B FUSSELL

UNIVERSITY OF TENNESSEE (CHATTANOOGA)
WILLIAM BLACK

UNIVERSITY OF TEXAS AT ARLINGTON
CHARLES I SMITH

UNIVERSITY OF TEXAS AT AUSTIN
RICHARD ASHLEY
L F BROWN
E LINN DRAPER JR
WILLIAM L FISHER
EARNEST F GLOYNA
JOHN B GORDON
CLARK HUBBS
E C JONAS
DALE KLEIN
JOE D LEDBETTER
G WERMUND

UNIVERSITY OF UTAH
JOHN S ALLEN
DUNCAN FOLEY
FRANK C HACKMAN
S K KAO
ROBERT C PENDLETON
HOWARD P ROSS
GARY M SANDQUIST
EDWARD B SISSON

UNIVERSITY OF VIRGINIA
T G WILLIAMSON

UNIVERSITY OF WISCONSIN
CHARLES W MAYNARD
JOHN B HEIL

U S COAST GUARD ACADEMY
MICHAEL HELFERT

US DEPARTMENT OF COMMERCE
L H GEVANTMAN

**US DEPARTMENT OF ENERGY
(ALBUQUERQUE)**

D DAVIS
DORNEST SCHUELER
WIPP PROJECT

US DEPARTMENT OF ENERGY (CHICAGO)
R NACK

US DEPARTMENT OF ENERGY (DENVER)
JACK O BRIEN
HARRY SMITH

**US DEPARTMENT OF ENERGY (GRAND
JUNCTION)**
FRANK ECKERSON
J ELLIS

**US DEPARTMENT OF ENERGY (IDAHO
FALLS)**
J P HAMRIC
J WHITSETT

**US DEPARTMENT OF ENERGY (LAS
VEGAS)**
R D BETTERIDGE
J B COTTER
D DUNCAN
M E GATES
J R GILPIN
K N JOY
M P KUNICH
H L MELANCON
R M NELSON JR
NEVADA TECHNICAL LIBRARY
R W NEWMAN
J ROBERTS
R W TAFT
T E WADE

**US DEPARTMENT OF ENERGY (NEW
YORK)**
CHARLES BAXTER

**US DEPARTMENT OF ENERGY (OAK
RIDGE)**
T ENSMINGER
DEWEY E LARGE
TECHNICAL INFORMATION CENTER

**US DEPARTMENT OF ENERGY
(RICHLAND)**
T A BAUMAN
R B GORANSON
D J SQUIRES
F R STANDERFER
KEN THOMAS

**US DEPARTMENT OF ENERGY
(RICHLAND-COLUMBUS)**
J O NEFF

**US DEPARTMENT OF ENERGY
(SAN FRANCISCO)**
C D JACKSON
JOHN MUHLESTEIN

**US DEPARTMENT OF ENERGY
(SAVANNAH RIVER)**
LAWRENCE W FRY
T B HINDMAN

**US DEPARTMENT OF ENERGY—PUBLIC
READING/DOCUMENT ROOMS**
ALBUQUERQUE OPERATIONS OFFICE
CHICAGO OPERATIONS OFFICE
DOE HEADQUARTERS
IDAHO OPERATIONS OFFICE
NEVADA OPERATIONS OFFICE
OAK RIDGE OPERATIONS OFFICE
RICHLAND OPERATIONS OFFICE
SAN FRANCISCO OPERATIONS OFFICE
SAVANNAH RIVER OPERATIONS
OFFICE
REGION IX OFFICE—SAN FRANCISCO
WASHINGTON DC

**US DEPARTMENT OF ENERGY REGION I—
BOSTON**
DUANE DAY

**US DEPARTMENT OF ENERGY REGION
II—NEW YORK**
LENORE LEDMAN

**US DEPARTMENT OF ENERGY REGION
III—PHILADELPHIA**
MARIA MARKS

**US DEPARTMENT OF ENERGY REGION
IV—ATLANTA**
DAVID ALANIZ

**US DEPARTMENT OF ENERGY REGION
V—CHICAGO**
PHILLIP T ZENI

**US DEPARTMENT OF ENERGY REGION
VI—DALLAS**
WILLIAM NIKOLIS

**US DEPARTMENT OF ENERGY REGION
VII—KANSAS CITY**
D FONTANE

**US DEPARTMENT OF ENERGY REGION
VIII—LAKEWOOD**
J L C GOSNEY
SIGRID HIGDON

**US DEPARTMENT OF ENERGY REGION
IX—SAN FRANCISCO**
JAMES RUSSELL

**US DEPARTMENT OF ENERGY REGION
X—SEATTLE**
LIBRARY
LEE JOHNSON

**US DEPARTMENT OF ENERGY
(WASHINGTON DC)**
W W BALLARD
EUGENE F BECKETT

C R COOLEY
J C DEMPSEY
JOHN M DEUTCH
WARREN EISTER
MARK W FREI
C H GEORGE
J L GILBERT
O P GORMLEY
DONALD H GROELSEMA
S L HACK
C A HEATH
KEITH KLEIN
CYRUS KLINGSBERG
JIM LANDERS
ROGER LE GASSIE
J L LIVERMAN
E F MASTAL
BEN McCARTY
S MEYERS
W E MOTT
J F MULLANEY
CARL NEWTON
RAY NG
A G PETTIT
MURIEL SCARBOROUGH
J E SEYMOUR
RALPH STEIN
OSCAR STRADINGER
JIM TURI
D L VEITH
SUSAN WELLS
J B WORK
H H YOUNG

US EPA
S GOLDBERG
A S GOLDIN
ANDREW J LETER
JAMES E MARTIN
PETER E McGRATH
JAMES NEIHEISEL
JOHN L RUSSELL
D S SMITH
W A WILLIAMS

USGS—BATON ROUGE
R L HOSMAN

USGS—DENVER
JOHN D BREDEHOEFT
JEFFE CLEVELAND
WILLIAM W DUDLEY JR
JULES FRIEDMAN
L M GARD
ROBERT J HITE
D L HOOVER
JOHN C REED JR
G W SHURR
ROBERT I TILLING

USGS—HOUSTON
JERRY CARR

USGS—ITHACA
O J COSNER

USGS—JACKSON
E H BOSWELL
CHARLES A SPIERS

USGS—LAKEWOOD
DON R MABEY
WILLIAM S TWENHOFEL
RICHARD WADDELL

USGS—MENLO PARK
J D BYERLEE
GERRY CZAMANSKE
EVERETT JENNE
ARTHUR H LACHENBRUCH
JACOB RUBIN

USGS—RESTON
G D DEBUCHANANNE
ROBERT M HAMILTON
EUGENE H ROSEBOOM JR
PETER R STEVENS
DAVID B STEWART
NEWELL J TRASK JR
WARREN W WOOD

**U S NUCLEAR REGULATORY
COMMISSION**
C BARTLETT
ORMEN E BASSETT
REGIS R BOYLE
ROBERT BUDNITZ
MICHAEL C CULLINGFORD
J J DAVIS
JAY B DURST
MARY A EADDEN
S FUCIGMA
EDWARD F HAWKINS
E E HELD
H E LEFOURE
J MALARO
MANAGER—WASTE ENVIRONMENTAL
STANDARDS PROGRAM

JOHN B MARTIN
T J NICKOLSON
EDWARD O'DONNELL
PROJECT OFFICER—SAFEGUARDS OF
HLW REPOSITORIES CONTRACTS
E REGNIER
I C ROBERTS
GARY ROBBINS
D M ROHRER
SHELDON SCHWARTZ
S H SMILEY

U S NUCLEAR REGULATORY RESEARCH
CLYDE JUPITER

UTAH ENERGY OFFICE
JAMES M BYRNE

**UTAH GEOLOGICAL AND MINERAL
SURVEY**
DONALD T MCMILLAN

UTAH STATE UNIVERSITY
JAY C ANDERSON
JACK T SPENCE

VANDERBILT UNIVERSITY
FRANK L PARKER

VAUGHN HANSEN ASSOCIATES
PAUL B BLACKER

VERMONT STATE GEOLOGIST
CHARLES A RATTE

VIRGINIA STATE GEOLOGIST
ROBERT C MILICI

WASHINGTON STATE GEOLOGIST
VAUGH E LIVINGSTON JR

WASTE ISOLATION PILOT PLANT
R K BROWN

WAYNE STATE UNIVERSITY
JAMES A WOODYARD

**WESTINGHOUSE ELECTRIC
CORPORATION**
P BRADBURY
A R HAKL
J HALEY
LIBRARY
GEORGE P SABOL

WESTINGHOUSE HANFORD COMPANY
ALBERT G BLASEWITZ

WESTINGHOUSE WIPP PROJECT
C L HOHMANN

WEST VIRGINIA STATE GEOLOGIST
ROBERT B ERWIN

**WISCONSIN GEOLOGICAL AND
NATURAL HISTORY SURVEY**
MEREDITH E OSTROM

WISCONSIN STATE ENERGY OFFICE
JOHN H THOMAN

WOODWARD-CLYDE CONSULTANTS
HANS M EWOLDSSEN
LIBRARY
CHARLES L TAYLOR

WYOMING STATE GEOLOGIST
DANIEL N MILLER JR

YALE UNIVERSITY
ROBERT M GARRELS

**YATES COUNTY (NY) PLANNING
BOARD**
ROBERT McNARY