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ONWI-39

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

**VOLUME 2
SYSTEM DEFINITION
AND ASSESSMENTS**

TECHNICAL REPORT

July 1979

ONWI
Office of Nuclear Waste Isolation
Battelle

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

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19

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:

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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.

TABLE OF CONTENTS

Section		Page
	FOREWORD	III
2	INTRODUCTION	2-1
2 1	Study Background	2-1
2 2	Study Purpose and Scope	2-4
2 3	Project Technical Approach	2-5
2 3 1	Systems Definition	2-5
2 3 2	Study Bases and Assumptions	2-5
2 3 3	Preconceptual Design Studies	2-5
2 3 4	Cost Estimates	2-7
2 3 5	Transportation Studies	2-7
2 3 6	Technical Assessments	2-7
2 3 7	Study Variations	2-7
2 3 8	Systems Evaluation	2-8
2 4	Study Products	2-9
3	STUDY DEFINITION AND ASSESSMENT APPROACH	3-1
3 1	Specific Considerations	3-4
3 1 1	Degree of Processing	3-4
3 1 2	Packaging Considerations	3-5
3 1 3	Colocation vs Independent Location	3-5
3 1 4	Recycling of Fissile Materials	3-5
3 1 5	Repository Geologic Media Options	3-5
3 1 6	Repository Retrievability Period	3-5
3 1 7	Fuel Age Leaving Reactor Storage Basins	3-6
3 1 8	Waste and Fuel Age at Time of Emplacement in Repository	3-6
3 1 9	Canister Emplacement Orientation in Repository	3-6
3 2	Description of Study Cases	3-7
3 2 1	Features Common to All Study Cases	3-7
3 2 2	Reference Case	3-9
3 2 3	Case 1	3-9
3 2 4	Case 2	3-10
3 2 5	Case 3	3-10

Section		Page
3.3	Description of Study Variations	3-11
3.3.1	Common Features for Study Variations	3-11
3.3.2	Specific Considerations within Study Variations	3-11
3.4	Assessment Approach	3-13
3.4.1	Description of a Detailed Ideal Methodology	3-13
3.4.2	Description of the Simplified SFDS Methodology	3-16
	References for Section 3	3-19
4	ASSESSMENT FOR TECHNICAL FEASIBILITY	4-1
4.1	Assessment Basis and Approach	4-2
4.1.1	Definitions	4-2
4.1.2	Measurement of Technical Feasibility	4-2
4.1.3	Assessment Methodology	4-5
4.2	Assessment Considerations	4-6
4.2.1	Aspects of Technical Feasibility	4-6
4.2.2	Basis for Current Approach	4-7
4.3	Assessment by Cases	4-8
4.3.1	Reference Case Technical Feasibility	4-12
4.3.2	Case 1 Technical Feasibility	4-15
4.3.3	Case 2 Technical Feasibility	4-17
4.3.4	Case 3 Technical Feasibility	4-19
4.3.5	Summary Comparison	4-21
	References for Section 4	4-23
5	SAFEGUARDS ASSESSMENT	5-1
5.1	Assessment Basis and Approach	5-2
5.1.1	Definition	5-2
5.1.2	Measurement of Safeguards Assessment	5-2
5.1.3	Methodology for Developing Assessment	5-4
5.2	Assessment Considerations	5-5
5.2.1	Four Factors that Affect the Potential Risk to the Public	5-5
5.2.2	Basis for Sabotage Assessment	5-7
5.2.3	Basis for Theft Assessment	5-12

Section		Page
5 3	Assessment by Cases	5-15
5 3 1	Assessment of the Reference Case	5-15
5 3 2	Assessment of Case 1	5-16
5 3 3	Assessment of Case 2	5-17
5 3 4	Assessment of Case 3	5-18
5 3 5	Summary Comparison	5-19
	References for Section 5	5-23
6	CRITICALITY ASSESSMENT FOR LONG-TERM DISPOSAL	6-1
6 1	Assessment Basis and Approach	6-2
6 1 1	Definition of Criticality	6-2
6 1 2	Approach	6-2
6 2	Assessment Considerations	6-3
6 2 1	Probability of a Criticality	6-3
6 2 2	Magnitude of a Criticality	6-10
6 2 3	Potential Radiological Impact	6-11
	References for Section 6	6-13
7	ASSESSMENT FOR RADIOLOGICAL IMPACT	7-1
7 1	Assessment Basis and Approach	7-5
7 1 1	Definition of Radiological Impact	7-5
7 1 2	Scope	7-5
7 1 3	Approach	7-5
7 2	Near-Term Assessment — Facilities	7-7
7 2 1	AFR	7-7
7 2 2	P/E Facility	7-7
7 2 3	Transportation System	7-12
7 2 4	Repository	7-15
7 3	Long-Term Assessment — Repository	7-19
7 3 1	General Discussion	7-19
7 3 2	Liquid Breach and Transport of Radionuclides	7-21
7 3 3	Direct Access Drilling	7-21
7 3 4	Catastrophic Breaching Events	7-22
	References for Section 7	7-24

Section		Page
8	ASSESSMENT FOR RETRIEVABILITY	8-1
8.1	Assessment Basis and Approach	8-2
8.1.1	Definition of Retrievability	8-2
8.1.2	Measurement of Retrievability	8-2
8.1.3	Methodology for Developing the Assessments	8-4
8.2	Assessment Considerations	8-5
8.2.1	Retrievability Subsystems	8-5
8.2.2	Basis for Current Approach	8-6
8.3	Assessment by Cases	8-7
8.3.1	Reference Case	8-8
8.3.2	Case 1	8-9
8.3.3	Case 2	8-9
8.3.4	Case 3	8-10
8.3.5	Summary of the Four Cases	8-10
9	ECONOMIC ASSESSMENT	9-1
9.1	Background	9-2
9.2	Approach to the Economic Assessment	9-3
9.3	Bases of the Economic Assessments	9-4
9.4	Assessment by Cases	9-8
9.4.1	The Reference Case	9-8
9.4.2	Case 1	9-8
9.4.3	Case 2	9-11
9.4.4	Case 3	9-11
9.4.5	Comparison of Cases	9-11
9.4.6	Sensitivity to Discount Rate	9-15
9.5	Approximate Charges in Commercially Operated Systems	9-17
	References for Section 9	9-19
10	STUDY VARIATIONS	10-1
10.1	Approach to the Study Variations	10-1
10.2	Time Factor Variations	10-3
10.2.1	Age of Spent Fuel Leaving Reactor	10-3
10.2.2	Age of Spent Fuel at Time of Emplacement in the Repository	10-4
10.2.3	Retrievability Time Period Options	10-8

Section	Page
10.3 Canister Design Variations	10-14
10.3.1 Canister Void Fillers	10-14
10.3.2 Overpacks and Thicker Wall Canisters	10-19
10.3.3 Number of Spent Fuel Assemblies per Canister	10-20
10.4 Repository Design Variations	10-23
10.4.1 Room Cooling During Extended Retrievability Period	10-23
10.4.2 Horizontal/Diagonal Emplacement of Canisters	10-24
10.4.3 Combination Spent Fuel/HLW Repository	10-25
10.4.4 Non-Salt Repository	10-26
10.5 Colocation Options	10-30
10.5.1 Colocation of 3 AFRs and 1 P/E Facility	10-30
10.5.2 Colocation of 1 P/E Facility and 1 Repository at 3 Sites	10-34
10.5.3 Colocation of 3 AFRs, 1 P/E Facility, and 1 Repository	10-37
10.6 Spent Fuel Disassembly Option	10-42
10.6.1 Design Impact	10-42
10.6.2 Assessment Area Impact	10-43
10.7 Application of Study Variations	10-46
References for Section 10	10-47

Volume 2

TABLES

Table	Page
3-1 Summary of Study Cases	3-8
3-2 Spent Fuel Disposal Study Ideal Assessment Methodology	3-14
3-3 Spent Fuel Disposal Study Simplified Assessment Methodology	3-17
4-1 Values of Technical Development Rating, A_m , for All Facilities and Transportation Systems	4-3
4-2 Values of Complexity Factor, C_m , for All Facilities	4-4
4-3 Values of Relative Importance Factor, R_m , for All Facilities	4-4
4-4 Values of Complexity Factor, C_m , for the Transportation System	4-4
4-5 Degree of Technical Development P/E Facility Process Units	4-8
4-6 Degree of Technical Development AFR Process Units	4-10
4-7 Degree of Technical Development Repository Process Units	4-11
4-8 Degree of Technical Development Transportation System Elements	4-12
4-9 P/E Facility Technical Feasibility Assessment Reference Case	4-13
4-10 Repository Technical Feasibility Assessment Reference Case	4-14
4-11 Transportation System Technical Feasibility Assessment Reference Case	4-15
4-12 AFR Technical Feasibility Assessment Case 1	4-16
4-13 P/E Facility Technical Feasibility Assessment Case 1	4-16
4-14 Transportation Technical Feasibility Assessment Case 1	4-17
4-15 P/E Facility Technical Feasibility Assessment Case 2	4-18
4-16 Transportation Technical Feasibility Assessment Case 2	4-19
4-17 P/E Facility Technical Feasibility Assessment Case 3	4-20
4-18 Transportation Technical Feasibility Assessment Case 3	4-21
4-19 Comparison of the Relative Technical Feasibility of the Four Study Cases	4-21
5-1 Rating Scale for Potential Risk to Public from Sabotage	5-3
5-2 Rating Scale for the Potential Risk to the Public from Theft	5-4
5-3 Risk to Public from Sabotage during Transportation	5-10
5-4 Risk to the Public from Sabotage at a P/E Facility	5-11
5-5 Risk to the Public from Sabotage at the Repository	5-12
5-6 Potential Risk to the Public from Sabotage	5-19
5-7 Potential Risk to the Public from Theft	5-22

Table	Page
6-1 Fissile Mass Contained in One Study Case Waste Canister	6-4
6-2 Minimum Critical Mass for Each Study Case Waste Form When Moderated and Reflected by Pure Water	6-5
6-3 Minimum Critical Mass for Each Study Case Waste Form When Moderated and Reflected by Saturated Brine	6-6
6-4 Fraction of the Minimum Critical Mass Contained in One Waste Canister When Moderated and Reflected by Pure Water	6-7
6-5 Fraction of the Minimum Critical Mass Contained in One Waste Canister When Moderated and Reflected by Saturated Brine	6-8
7-1 Summary of Near-Term Radiological Impact Data for a 5000 MTHM/Yr System	7-2
7-2 Long-Term Consequences of Repository Breaching	7-3
7-3 Annual Radiation Dose to Occupational and Non-Occupational Personnel During Normal Operations of AFRs	7-8
7-4 Radiation Dose to Occupational and Non-Occupational Personnel Due to Accidents and Abnormal Events at an AFR	7-8
7-5 Occupational Dose During Normal and Abnormal Operation of a Single P/E Facility	7-10
7-6 Non-Occupational Dose During Normal Operations at a Single P/E Facility	7-11
7-7 Non-Occupational Dose Due to Accidents at a P/E Facility	7-11
7-8 Summary of Potential Offsite Doses by Postulated Accidents at the Barnwell Nuclear Fuel Plant	7-12
7-9 Occupational Dose During Normal Transportation of Wastes Between Facilities	7-14
7-10 Non-Occupational Dose During Normal Transportation of Wastes Between Facilities	7-14
7-11 Maximum Dose to a Non-Occupational Individual Due to a Single Transportation Accident	7-16
7-12 Occupational Dose During Normal and Abnormal Operations at a Repository	7-17
7-13 Non-Occupational Dose During Normal Operations at a Repository	7-18
7-14 Non-Occupational Dose From Design Basis Accident at a Repository	7-18
7-15 Long-Term Radiological Consequences of a Non-Catastrophic Breach of the Repository	7-22
7-16 Radiological Impact Due to a Catastrophic Event Causing a Breach of the Repository with the Simultaneous Release of All Waste	7-23

Table	Page
8-1 Values of Technical Development Rating, A_m , for Retrievalability	8-3
8-2 Values of Complexity Factor, C_m , for Retrievalability	8-3
8-3 Values of Relative Importance Factor, R_m , for Retrievalability	8-3
8-4 Retrievalability Assessment for the Reference Case	8-8
8-5 Retrievalability Assessment for Case 1	8-9
8-6 Retrievalability Assessment for Case 2	8-9
8-7 Retrievalability Assessment for Case 3	8-10
8-8 Retrievalability Assessment Summary	8-10
9-1 Average Levelized Economic Costs Reference Case (Based on January 1979 Dollars)	9-9
9-2 Average Levelized Economic Costs Case 1 (Based on January 1979 Dollars)	9-10
9-3 Average Levelized Economic Costs Case 2 (Based on January 1979 Dollars)	9-12
9-4 Average Levelized Economic Costs Case 3 (Based on January 1979 Dollars)	9-13
9-5 Average Levelized Economic Costs for Four Cases of Spent Fuel Disposal	9-15
10-1 Design Impact Summary Age of Fuel Leaving the Reactor	10-3
10-2 Assessment Impact Summary Age of Fuel Leaving the Reactor	10-5
10-3 Design Impact Summary Age of Spent Fuel at Time of Emplacement in the Repository	10-6
10-4 Assessment Impact Summary Age of Spent Fuel at Time of Emplacement in the Repository	10-7
10-5 Design Impact Summary Retrievalability Time Period Options	10-9
10-6 Assessment Impact Summary Retrievalability Time Period Options	10-10
10-7 Degree of Technical Development and Implementation	10-12
10-8 Comparison of Canister Filler Materials	10-15
10-9 Design Impact Summary Canister Void Fillers	10-16
10-10 Assessment Impact Summary Canister Void Fillers	10-17
10-11 Design Impact Summary Overpacks and Thicker Wall Canisters	10-19
10-12 Assessment Impact Summary Overpacks and Thicker Wall Canisters	10-20
10-13 Design Impact Summary Number of Spent Fuel Assemblies per Canister	10-21
10-14 Assessment Impact Summary Number of Spent Fuel Assemblies per Canister	10-21
10-15 Typical Media Properties	10-27

Table	Page
10-16 Design Impact Summary Colocation of Three AFRs and One P/E Facility	10-31
10-17 Assessment Impact Summary Colocation of Three AFRs and One P/E Facility	10-33
10-18 Design Impact Summary Colocation of a P/E Facility and a Repository	10-34
10-19 Assessment Impact Summary Colocation of a P/E Facility and a Repository	10-36
10-20 Design Impact Summary Colocation of Three AFRs, a P/E Facility, and a Repository	10-38
10-21 Assessment Impact Summary Colocation of Three AFRs, a P/E Facility, and a Repository	10-40
10-22 Design Impact Summary Spent Fuel Disassembly Option	10-42
10-23 Assessment Impact Summary Spent Fuel Disassembly Option	10-44

Volume 2
ILLUSTRATIONS

Figure	Page
2-1 LWR Fuel Cycle as Originally Envisioned	2-2
2-2 A Modified Uranium Fuel Cycle	2-3
2-3 Spent Fuel Disposal Study Project Technical Approach	2-6
3-1 Fuel Cycle Diagram Case 1, Case 2, and Case 3	3-2
3-2 Fuel Cycle Diagram Reference Case	3-3
5-1 Scope of the Safeguards Assessment Study	5-6
5-2 Comparison of Frequency and Consequence of Occurrences	5-8
5-3 Potential Risk from Sabotage During Transportation	5-20
5-4 Potential Risk from Sabotage at an AFR	5-20
5-5 Potential Risk from Sabotage at a P/E Facility	5-21
5-6 Potential Risk from Sabotage at the Repository	5-21
7-1 Radioactive Decay of Spent Fuel	7-20
9-1 Economic Analysis Diagram	9-5
9-2 System Average Levelized Economic Cost \$/MTHM for Four Cases at Various Discount Rates	9-16
10-1 Option I — AFRs and P/E Facility — Colocated	10-32
10-2 Option II — P/E Facility and Repository — Colocated	10-35
10-3 Option III — AFRs, P/E Facility, and Repository — Colocated	10-39
10-4 Geological Repository Spent Fuel Waste Emplacement Schematic During Retrieval Period Fuel Pin Close Spacing	10-45

Section 2

INTRODUCTION

TABLE OF CONTENTS

Section		Page
2.1	Study Background	2-1
2.2	Study Purpose and Scope	2-4
2.3	Project Technical Approach	2-5
2.3.1	Systems Definition	2-5
2.3.2	Study Bases and Assumptions	2-5
2.3.3	Preconceptual Design Studies	2-5
2.3.4	Cost Estimates	2-7
2.3.5	Transportation Studies	2-7
2.3.6	Technical Assessments	2-7
2.3.7	Study Variations	2-7
2.3.8	Systems Evaluation	2-8
2.4	Study Products	2-9

Section 2

INTRODUCTION

2.1 STUDY BACKGROUND

For many years, research and development has been conducted by government and industry throughout the world to demonstrate acceptable methods for the management and permanent disposal of radioactive wastes produced by commercial nuclear power plants. In the United States, these studies were originally begun by the Atomic Energy Commission (AEC), continued by the Energy Research and Development Administration (ERDA), and are now sponsored by the Department of Energy (DOE). Until 1977, the U.S. program was focused primarily on the reactor fuel cycle involving reprocessing, recycling of usable materials, and disposal of wastes from light water reactors (LWRs).

The U.S. DOE is currently evaluating alternative fuel cycle systems in response to President Carter's mandate of April 7, 1977 to recommend alternatives that will minimize access to nuclear materials that are directly usable in nuclear weapons. Whichever fuel cycle is ultimately selected, the DOE will have to demonstrate that the safeguards systems are adequate to prevent clandestine diversion of nuclear fuel for weapons production.

In response to a proposal by the President, an International Nuclear Fuel Cycle Evaluation Program was established. The charter of this program calls for (1) recommendations of alternative systems that might produce resistance to increased proliferation and (2) the development of international solutions for nuclear waste disposal that will also ensure countries an adequate nuclear fuel supply. Another program, the Nonproliferation Alternative Systems Assessment Program, is a DOE effort designed to provide technical support for the international program as well as to help define the future direction of the domestic nuclear program.

The principal features of the original concept of the LWR recycle fuel cycle are shown in Figure 2-1. Spent fuel from LWRs and later from liquid metal fast breeder reactors (LMFBRs) is reprocessed, and the uranium and plutonium are recycled to reactors. The high-level liquid wastes (HLLW) are converted to a stable solid form having low leachability and inserted in sealed canisters for emplacement in a repository within 10 years after reprocessing. Other wastes are also converted to stable solid forms, inserted into containers, and placed in a repository. Nontransuranic low-level waste (non-TRU LLW) is placed in shallow land burial.

A modified (once-through) LWR fuel cycle with spent fuel disposal in a repository, shown in Figure 2-2, is being considered by the U.S. government as an alternative to the original concept. Various other intermediate scenarios involving partial processing of spent fuel and alternate forms of waste for disposal are also being considered. The advantages and disadvantages of some of the scenarios are being studied both in the United States and in other countries.

In 1976, the U.S. commercial nuclear waste program was consolidated by ERDA. The management of a major share of this program was assigned to the Office of Waste Isolation (OWI) of Union Carbide Corporation — Nuclear Division (UCC-ND). This Spent Fuel Disposal Study (SFDS) was conceived by OWI management to perform an overview analysis of the various proposals, alternatives, and considerations relevant to the geologic isolation of spent reactor fuel from commercial reactors. Bechtel was selected to conduct the SFDS under subcontract to UCC-ND. The responsibility for the management of this program was reassigned by the DOE from UCC-ND to the Office of Nuclear Waste Isolation (ONWI) of Battelle Project Management Division in June 1978.

Figure 2-1.
LWR Fuel Cycle as Originally Envisioned

Figure 2-2.
A Modified Once-Through Uranium Fuel Cycle

2.2 STUDY PURPOSE AND SCOPE

The purpose of this study is to provide the DOE with an evaluation of ultimate disposition methods for spent nuclear fuel assemblies. Various alternatives for the treatment of spent fuel elements for the purpose of geologic isolation are analyzed, and contrasted to the geologic isolation of high-level waste (HLW) from fuel reprocessing. This is not, however, a study comparing the merits of spent fuel disposal with reprocessing and high-level waste disposal since the steps at the front end of the fuel cycle were excluded from the study.

To accomplish this purpose, the study is conducted as follows:

- Alternative forms of spent fuel are identified, and the physical systems and facilities necessary to process, handle, and transport each of them (from the reactor to the repository) are defined at the preconceptual design level
- A methodology is developed for comparing various disposal alternatives for spent fuel
- The spent fuel disposal alternatives are analyzed and rated in terms of six assessment criteria: technical feasibility, safeguards, criticality, radiological impact, retrievability, and economics
- Gaps are identified where additional research and development are required to develop and demonstrate technologies

2.3 PROJECT TECHNICAL APPROACH

The project technical approach is illustrated in Figure 2-3. Eight distinct activities are followed: systems definition, establishment of study bases and assumptions, preconceptual design, transportation studies, technical assessments, cost estimates, study variations, and systems evaluation.

2.3.1 Systems Definition

The systems definition activity has three elements:

- **Case identification.** Define the cases and variations. Identify the transportation, storage, processing, encapsulation, and disposal systems to be studied for each of the study cases and study variations
- **Mechanical and chemical processing technology.** Identify applicable mechanical and chemical processing technology for the various systems to be studied. Emphasize those unique technology steps utilized for spent fuel disposal as contrasted with fuel reprocessing and HLW disposal
- **Systems evaluation criteria.** Identify and develop the criteria that the project team will use in conducting the studies and in making systems evaluations

The definition of alternative systems studied is presented in Section 3. Section 3 also contains a brief description of the mechanical and chemical processing technology. A more detailed description of the technology is incorporated in Appendices B, C, and D of Volume 3.

2.3.2 Study Bases and Assumptions

This project activity involves the development of study bases and assumptions in order to assure consistency of technical approach. The results of this activity are given in Appendix A of Volume 3.

2.3.3 Preconceptual Design Studies

Design studies are conducted at the preconceptual level for away-from-reactor storage facilities (AFRs), processing/encapsulation (P/E) facilities, and repositories for each study case. In doing this, extensive use is made of available information from related design studies and engineering projects dealing with similar fuel cycle facilities.*

The descriptions and definitions that are developed are in the form of descriptive text, illustrated by block diagrams, flow diagrams, and preconceptual design layouts of facilities. The material is in sufficient detail to meet the following objectives:

- To provide a basis, in terms of facilities configuration and process technology definition, for making assessments and comparing the cases
- To furnish data for developing order-of-magnitude capital and operating cost estimates
- To investigate the state-of-the-art of equipment and process technology in order to provide input to the Technical Feasibility Assessment

The results of these design studies are contained in Appendices B, C, and D of Volume 3.

* It is important to note that at the time the SFDS was conceived, consistent comparable information of this type did not generally exist. In the interim other studies have generated similar designs. Within the context of the overall DOE program on waste management these studies have been closely coordinated for consistency.

Figure 2-3.
Spent Fuel Disposal Study
Project Technical Approach

2.3.4 Cost Estimates

Capital, operating, and decommissioning facility cost estimates, at the preconceptual design level, are developed for AFRs, P/E facilities and repositories, and are given in Appendices B, C, and D of Volume 3, following the design descriptions of the respective facilities.

2.3.5 Transportation Studies

The transportation studies analyze the special aspects of transporting all nuclear materials in the back end of the fuel cycle, including radioactive wastes. The following information is developed: shipping quantities, cask preconceptual designs, shipping logistics, regulatory aspects of shipping, and transportation costs.

- Reactor to AFR
- AFR to P/E facility
- P/E facility to repository
- AFR to shallow land burial
- P/E facility to shallow land burial
- P/E facility to mixed oxide fabrication and enrichment facilities

The results of these studies are contained in Appendix E of Volume 3.

2.3.6 Technical Assessments

The four study cases are assessed in terms of six evaluation criteria:

- Technical feasibility
- Safeguards
- Criticality
- Radiological impact
- Retrievability
- Economics

These assessments are discussed in Sections 4 through 9, respectively. Quantitative and/or qualitative results are given, and the advantages and disadvantages of each of the four cases are presented. Additional technical material providing support for some of the assessments is given in Appendices F through I of Volume 3.

2.3.7 Study Variations

As the study progressed, many alternatives or variant definitions of the study cases were identified. Several of the major variations are incorporated in the study as study variations of Case 1. Each of these study variations is, in effect, a sensitivity study, and includes a cursory assessment under each evaluation criterion listed above.

The study variations are described and analyzed in Section 10.

2.3.8 Systems Evaluation

Each assessment section (4 through 9) contains a summary of findings or conclusions based on that single assessment area or criterion. Section 1 in Volume 1 contains an overall summary of the technical assessments of the case and study variations. Although no specific optimum case is recommended, a systematic basis and methodology is presented that provides the reader with an insight into the major factors which affect spent fuel disposal, and allows the reader to combine preferences with the technical data generated to select a preferred case.

2.4 STUDY PRODUCTS

This report, "An Assessment of LWR Spent Fuel Disposal Options," satisfies the ONWI contract requirement for a Scoping Study Final Report. It consists of three volumes:

- Volume I — Summary and Conclusions
- Volume II — System Definition and Assessments
- Volume III — Study Bases and System Design Considerations (Appendices)

Section 3

STUDY DEFINITION AND ASSESSMENT APPROACH

TABLE OF CONTENTS

Section		Page
3.1	Specific Considerations	3-4
3.1.1	Degree of Processing	3-4
3.1.2	Packaging Considerations	3-5
3.1.3	Colocation vs. Independent Location	3-5
3.1.4	Recycling of Fissile Materials	3-5
3.1.5	Repository Geologic Media Options	3-5
3.1.6	Repository Retrievability Period	3-5
3.1.7	Fuel Age Leaving Reactor Storage Basins	3-6
3.1.8	Waste and Fuel Age at Time of Emplacement in Repository	3-6
3.1.9	Canister Emplacement Orientation in Repository	3-6
3.2	Description of Study Cases	3-7
3.2.1	Features Common to All Study Cases	3-7
3.2.2	Reference Case	3-9
3.2.3	Case 1	3-9
3.2.4	Case 2	3-10
3.2.5	Case 3	3-10
3.3	Description of Study Variations	3-11
3.3.1	Common Features for Study Variations	3-11
3.3.2	Specific Considerations within Study Variations	3-11
3.4	Assessment Approach	3-13
3.4.1	Description of a Detailed Ideal Methodology	3-13
3.4.2	Description of the Simplified SFDS Methodology	3-16
	References for Section 3	3-19

Section 3

STUDY DEFINITION AND ASSESSMENT APPROACH

The purposes of this section are to:

- Identify and define the study cases and study variations
- Present the approach used in assessing the study cases and variations

Three fundamental spent fuel disposal cases are identified, defined, and assessed in this report. Each case is compared with a Reference Case that includes fuel reprocessing, recycling of uranium and plutonium in light water reactors (LWRs), and high-level waste (HLW) solidification and disposal. In addition, a number of study variations related to Case 1 are defined and assessed to a lesser degree.

The scope of spent fuel disposal systems in this study is shown in the two fuel cycle diagrams in Figures 3-1 and 3-2. The facilities and transportation steps within the dotted lines are those included in the study. The Spent Fuel Disposal Study (SFDS) begins with the shipment of spent fuel from the reactors. It includes away-from-reactor storage facilities (AFRs), processing and encapsulation (P/E) facilities, geologic repositories, and transportation systems for moving nuclear materials between facilities.

The factors considered in defining the cases, study variations, and assumptions for the study include:

- Degree of processing of spent fuel in the P/E facility before encapsulation and disposal
- Packaging considerations
- Colocation vs. independent location of facilities
- Recycling of fissile materials
- Repository geologic media
- Length of repository demonstration period
- Fuel age leaving reactor storage basin
- Waste and fuel age at time of emplacement in repository
- Canister emplacement orientation in repository

These factors are discussed in Subsection 3.1. A description of the study cases is provided in Subsection 3.2, a description of study variations in Subsection 3.3, and a description of the assessment approach used in the study is in Subsection 3.4.

There were numerous assumptions and bases, either provided or developed for this study, as well as a considerable amount of derived data. Because many of these items affected more than one area of the study, they were documented on a formal basis. This complete documentation is included as Appendix A, Bases and Assumptions, in Volume 3.

Figure 3-1.
Fuel Cycle Diagram
Case 1, Case 2, and Case 3

Figure 3-2.
Fuel Cycle Diagram Reference Case

3.1 SPECIFIC CONSIDERATIONS

The factors considered in defining the spent fuel study cases are discussed in this subsection to illustrate that many aspects of waste management were considered — and some rejected — in the course of case definition for the study

3.1.1 Degree of Processing

The various degrees of processing spent fuel prior to encapsulation are considered below

3.1.1.1 Simple Encapsulation. The simplest operation in a P/E facility is encapsulation of intact spent fuel assemblies. This option is adopted as the SFDS Case 1. Every other case involves additional processing steps which introduce some advantages or disadvantages beyond this simple step.

3.1.1.2 Removal of End Fittings. The spent fuel assembly could have the end fittings mechanically removed. Thus, a shorter canister can be used to contain the spent fuel. This modification has the following apparent advantages:

- Lighter and less expensive fuel casks due to shorter canisters
- Apparent cost savings at the repository due to handling shorter canisters

These advantages would be offset somewhat by the physical requirements for removing the end fittings and their subsequent separate disposal. This step, by itself, was rejected as a study case.

3.1.1.3 Encapsulation of Fuel Pins. The next degree of processing is to remove the end plates with the end fittings, and then remove and encapsulate the intact fuel pins in a compact manner. The balance of the fuel assemblies would be packaged separately from the fuel pins. The features of this approach, in addition to those listed in 3.1.1.2, include:

- Smaller canister diameter per unit of fuel contained, or
- Pins from more than one fuel element can be placed in the original diameter canister

The potential benefits of this option warrant its selection as a study variation.

3.1.1.4 Chopping Spent Fuel. The potential benefits of this additional step are:

- Increased packing density of waste
- More flexibility in sizing of canisters
- The ability to use one canister size efficiently for all types of spent fuel

A preliminary analysis showed that packing density is not increased in comparison to fuel pin encapsulation. Hence, using this step as a final step prior to encapsulation was dropped from consideration.

3.1.1.5 Voloxidation. Voloxidation following fuel chopping partially removes volatile fission products from the spent fuel but adds the requirement of disposing of the recovered volatile fission products. The voloxidation option has been selected for Case 2.

3.1.1.6 Dissolution, Calcination, and Vitrification. In this process increment, the fuel material, after chopping and voloxidation, is dissolved in nitric acid, followed by calcination, vitrification, and encapsulation. This provides significant immobilization of the spent fuel, and is the basis for Case 3.

3.1.1.7 Dissolution and Supercalcination. In this consideration, supercalcination is used as an alternative fixation process to vitrification. The claimed potential benefits of supercalcination include:

- Reduced leachability
- Better long-term stability in repository environment

This option was eliminated because of the uncertainty of the claims and the immaturity of the technology.

3.1.2 Packaging Considerations

Possible spent fuel packaging considerations are:

- Effect of different void filler materials on heat transfer and canister handling characteristics
- Impact of canister and over pack wall thickness on canisters
- Impact of number of fuel assemblies per canister on transportation and repository operations

All these packaging options have been considered in the SFDS as study variations.

3.1.3 Colocation vs. Independent Location

Colocation of waste management facilities reduces the costs of some common requirements, such as security, safety and control, and reduces or eliminates some transportation requirements. Any combination of AFR, P/E facility, or repository could conceivably be colocated.

All the study cases are defined as having independently located facilities; the advantages and disadvantages of certain colocation combinations are considered as variations of the study.

3.1.4 Recycling of Fissile Materials

The recycle options are:

- Recycling of uranium only
- Recycling of both uranium and plutonium

The Reference Case is the full recycle option. The uranium-only recycle option was investigated earlier in the study but was dropped from the scope of work by direction from the Office Of Nuclear Waste Isolation (ONWI).

3.1.5 Repository Geologic Media Options

The geologic media options currently being considered for repositories in the United States are: salt, basalt, granite, and shale. Of these, salt appears to be the most studied and is considered by many to be the most promising. Therefore, a salt repository is the basis for all study cases. Other geologic options are incorporated in a study variation to assess in a preliminary way the value of other media.

3.1.6 Repository Retrieval Period

During this period, material placed in the repository is maintained in a retrievable state until a repository validation program has verified that the actual repository behavior (from a safety standpoint) conforms to a predicted behavior.

A 5-year repository retrievability period has been chosen as the fundamental basis in the SFDS. As a measure of sensitivity, repository retrievability periods of 10 and 25 years are considered as study variations. The length of actual validation or demonstration will ultimately depend upon initial performance experience of the repository.

3.1.7 Fuel Age Leaving Reactor Storage Basins

Until recently, nuclear power plants were designed to allow onsite storage capacity for one annual discharge plus full core removal capability. The annual discharges were originally intended to be sent to a reprocessing plant for recovery of uranium and plutonium. With the delay of reprocessing, most utilities plan to increase, or have increased, their onsite spent fuel storage capacities. In addition, both federal and commercial initiatives are being considered to provide AFR storage of spent fuel assemblies. A preliminary analysis indicates that the mix of onsite and AFR storage could have a significant impact on the comparison of cases.

The SFDS assumes that utilities prefer to return, if possible, to their original plan of providing only 1 year onsite storage of spent fuel. Thus, all study cases assume spent fuel is shipped from the reactors at age 1 year. A study variation investigates the effect of storage of spent fuel at the reactor for 5 years and 10 years.

3.1.8 Waste and Fuel Age at Time of Emplacement in Repository

The age of waste at the time of emplacement affects the design of the P/E facility, the repository, the transportation system, and the spent fuel storage capacity of the AFRs. The SFDS assumes a 10-year-old waste at emplacement. This basis is consistent with assumptions made in repository conceptual designs by Stearns-Roger Engineering Co.,^{(1),(2),(3)} Kaiser Engineers,^{(4),(5),(6)} and others. As a measure of the sensitivity of this assumption, 5-year and 25-year ages at emplacement are considered as study variations.

3.1.9 Canister Emplacement Orientation in Repository

Several alternate canister emplacement orientations have been proposed. These include:

- Vertical emplacement in wells in the floor of repository rooms
- Emplacement in diagonal holes in the floor of repository rooms
- Horizontal emplacement in wells drilled in walls of repository rooms
- Emplacement in trenches in the floor of repository rooms
- Multiple emplacement of several canisters in a single hole

Of these alternatives, emplacement in vertical holes has been the most thoroughly analyzed by other investigators and is used as the basis for the SFDS. Vertical emplacement is considered to be the simplest design for retrievability.

Horizontal emplacement may have some advantage in reducing room height. Slanting emplacement and trench emplacement may offer some minor advantages over horizontal or vertical emplacement. Multiple emplacement of several canisters in a single hole does not appear to offer major advantages since the thermal load constraints dictate the number of canisters per acre.

A study variation considers the merits of horizontal and diagonal emplacement. Trench emplacement and multiple emplacement were eliminated from further considerations in this study.

3.2 DESCRIPTION OF STUDY CASES

This subsection contains brief definitions and descriptions of the four study cases: Reference Case, Case 1, Case 2, and Case 3. Complete preconceptual design descriptions and cost estimates for the facilities and transportation systems for the study cases are included in Appendices B through E of Volume 3.

3.2.1 Features Common to All Study Cases

The following features are common to each of the four cases:

- The spent fuel disposal system has an equilibrium throughput of 5,000 MTHM/yr. Thus, the system ignores industry growth-related effects
- All facilities are independently located
- Spent fuel is transported from the reactors at age 1 year by truck and rail to a regionally located AFR in all cases except for the Reference Case, where spent fuel is transported to the P/E facility
- Spent fuel is stored at the AFR until it is 10 years old (except for the Reference Case). Cases 1, 2, and 3 require 9 AFRs, each with 555 MTHM/yr shipping/receiving capacity and 5000 MTHM storage capacity
- Only HLW waste or a single form of encapsulated spent fuel is considered in any one case
- All HLW and encapsulated spent fuel canisters are filled with helium
- All canisters for HLW and spent fuel are carbon steel
- All HLW or encapsulated spent fuel is transported to and emplaced in the repository at age 10 years
- All repositories are in salt formations
- During the first 5-year period in the repository, high-level waste (HLW) (or the various forms of encapsulated spent fuel) are maintained in a retrievable mode. After the retrievability period, repository rooms are backfilled as soon as canister emplacement is complete
- All HLW, intermediate level waste (ILW), and encapsulated spent fuel canisters are emplaced vertically in holes. Sleeves are used during the 5-year retrievability period
- Repository rooms are cooled only while occupied
- ILW and LLW-TRU wastes are transported from P/E facility to repository immediately after processing
- Nontransuranic (non-TRU) LLW is transported from the P/E facility and AFR to shallow land burial immediately after processing
- The AFR and P/E facility are decommissioned at the end of 40 years
- Repository surface facilities are decommissioned after filling the repository

Table 3-1 provides a summary of the four study cases.

Table 3-1
SUMMARY OF STUDY CASES

Item	Ref. Case	Case 1	Case 2	Case 3
Number of AFRs	0	9	9	9
Processing/encapsulation facility				
Recycle products	U, Pu	None	None	None
Mechanical treatment of fuel				
None		X		
End removal and chop	X		X	X
Chemical treatment				
None		X		
Voloxidation	X		X	X
Dissolution	X			X
Solvent extraction	X			
Calcination and vitrification	X			X
Age of waste being transported (years) ^(a)				
Spent fuel from reactor to AFR	—	1	1	1
Spent fuel from reactor to P/E facility	1	—	—	—
Spent fuel from AFR to P/E facility	—	10	10	10
HLW from P/E facility to repository	10	—	—	—
Spent fuel from P/E to repository	—	10	10	10
LLW from P/E facility to surface burial	1	10	10	10
ILW, hulls and/or end fittings from P/E facility to repository	1	—	10	10
Recovered plutonium from P/E facility to fuel fabrication facility	1	—	—	—
Recovered uranium from P/E facility to enrichment plant	1	—	—	—
TRU waste from P/E facility to repository	1	10	10	10

(a) Time is measured from the removal of the fuel element from the reactor.

3.2.2 Reference Case

The Reference Case represents the concept that the U.S. nuclear fuel industry was developing prior to the decision made by the U.S. government not to reprocess the light water reactor (LWR) spent fuel. This case fully utilizes the fuel values in spent fuel, with the terminal disposal of fission products and other wastes generated during reprocessing.

The Reference Case, shown in Figure 3-2, includes transporting the spent fuel to the P/E facility and the wastes to the repository and the handling of the wastes at the repository, as mentioned in Subsection 3.1.3.2 and summarized in Table 3-1. The principal operational considerations are:

- Processing operations at the P/E facility, including calcination, vitrification, and encapsulation of HLW in canisters
- Storage of vitrified HLW at the P/E facility until age 10 years
- Transportation of cladding hulls, ILW and TRU LLW, properly packaged, from the P/E facility to the repository immediately after processing
- Transportation of the uranium hexafluoride and plutonium oxide from the P/E facility to a mixed oxide fuel fabrication plant immediately after processing

The design of the P/E facility is essentially that of the Barnwell Nuclear Fuel Plant (BNFP), augmented, as required, to meet the as-low-as-reasonably-achievable (ALARA) concept for limiting radioactive effluent, and to add the missing facilities to the plant. Main operational steps performed within the P/E facility include:

- Removal of fuel element ends
- Chopping of fuel bundle
- Voloxidation of chopped fuel
- Dissolution of fuel material
- Extraction of plutonium and uranium
- Conversion of uranium to uranium hexafluoride and plutonium to plutonium oxide
- Calcination and vitrification of HLW
- Encapsulation of vitrified HLW
- Preparation for shipping plutonium oxide and uranium hexafluoride
- Processing and packaging other wastes for shipment

The geologic repository for the Reference Case is based upon the conceptual design produced for the Department of Energy (DOE) by Stearns-Roger Engineering Co.^{(1),(2),(3)} The design is adapted, as necessary, to the requirements of the study.

3.2.3 Case 1

Case 1 involves simple encapsulation and disposal of spent fuel after storage at an AFR for nine years. No processing within the P/E facility beyond sealing the spent fuel assemblies in canisters and processing and packaging of minor amounts of secondary wastes is required. Other operations include the basic transportation steps, summarized in Table 3-1, and repository operations. The geologic repository for Case 1 is an adaptation of the Reference Case repository to take into account the differences in quantity and form of the nuclear waste.

3.2.4 Case 2

In this case, as in Case 1, spent fuel is initially stored at an AFR for nine years. In the P/E facility, the following operations are performed:

- Removal of fuel assembly end fittings and spacers
- Chopping of the fuel bundle
- Voloxidation of chopped fuel to partially remove volatile fission products
- Encapsulation of fuel material following voloxidation, end fittings, and secondary wastes

Other operations in Case 2 include the basic transportation steps, summarized in Table 3-1, and repository operations. The geologic repository for Case 2 is also an adaptation from the Reference Case to be responsive to the different form and quantity of material.

3.2.5 Case 3

This case is identical to Case 2, except that in the P/E facility the processing is carried beyond the voloxidation step. Major steps performed within the P/E facility include:

- Removal of fuel assembly end fittings and spacers
- Chopping of the fuel bundle
- Voloxidation of chopped fuel to break down the pellets and partially remove volatile fission products
- Dissolution of fuel material in nitric acid
- Calcination of fuel material
- Vitrification of fuel material
- Encapsulation of vitrified fuel material, end fittings, hulls, and secondary wastes

Other operations in Case 3 include the basic transportation steps, summarized in Table 3-1, and repository operations. The geologic repository for Case 3 is also an adaptation from the Reference Case to be responsive to the different form and quantity of material.

3.3 DESCRIPTION OF STUDY VARIATIONS

This subsection contains definitions and descriptions of the study variations, all of which are variations of Case 1. For each variation, one aspect of the Case 1 definition is changed. Thus, the analysis is essentially a sensitivity analysis. The study variations are fully described and assessed in Section 10.

3.3.1 Common Features for Study Variations

The following features are common to all study variations:

- The spent fuel disposal system has an equilibrium throughput of 5,000 MTHM/yr, as in the case definition
- All AFRs have a storage capacity of 5,000 MTHM
- All P/E facilities have a processing capacity of 1,500 MTHM/yr operating 300 days/yr. For the purpose of the economic assessments, individual facility costs are multiplied by 3-1/3 to satisfy the 5,000 MTHM/yr system capacity
- Repositories are designed to operate 300 days/yr and to receive all of the spent fuel and/or waste within the 5,000 MTHM/yr system
- Each repository has 2,000 acres of underground development at a depth of 2,000 feet
- All ILW and LLW, as well as spent fuel, are emplaced in rooms at the same depth

3.3.2 Specific Considerations within Study Variations

3.3.2.1 Time Factor Variations. In Case 1, the spent fuel leaves the reactor after 1 year. To determine the effect of this arbitrary time period on the overall fuel disposal study, age variations of both 5 and 10 years are considered. The impact of increased storage costs at the reactor is not considered.

In Case 1, the age of fuel at the time of emplacement in the repository is 10 years. In a study variation, corresponding fuel ages are 5 years and 25 years.

In Case 1, a 5-year retrievability period is assumed. In the variations, 10-year retrievability and 25-year retrievability are assumed.

3.3.2.2 Canister Design Variations. In Case 1, the canisters of encapsulated spent fuel are 3/8-inch thick single wall. In a variation, overpacking and increasing the thickness of the wall of the primary canister are studied.

The spent fuel canisters in Case 1 are backfilled with gaseous helium before sealing. A study variation analyzes the effect of using several other gases and solid materials as void fillers.

In Case 1, a single PWR or 3 BWR fuel assemblies are encapsulated in a single canister. In a study variation, encapsulation of 2 PWRs or 6 BWRs in a single canister is considered.

3.3.2.3 Repository Design Variations. In the basic Case 1 repository design, the canisters are emplaced in vertical wells in the floor of the rooms, the repository is dedicated to spent fuel elements as the principal waste form, the geologic medium is salt, and no cooling is provided during the retrievability period. In several study variations, the following are described and assessed:

- Room cooling during extended retrievability periods
- Horizontal/diagonal emplacement of canisters

- Combination spent fuel/HLW repository
- Nonsalt repository media

3.3.2.4 Colocation Options. Case 1 assumes separately located facilities; study variations evaluate the impact of colocating:

- Three AFRs and 1 P/E facility
- One P/E facility and 1 repository
- Three AFRs, 1 P/E facility, and 1 repository

3.3.2.5 Fuel Disassembly Option. Case 1 involves encapsulating the intact spent fuel assembly in a canister. In a study variation, the end fittings are removed, the fuel bundle is disassembled, and a close-packed array of individual fuel pins is encapsulated.

3.4 ASSESSMENT APPROACH

Each of the cases in the SFDS is analyzed in Sections 4 through 9 in terms of the following assessment criteria:

- Technical feasibility
- Safeguards
- Criticality
- Radiological impact
- Retrievability
- Economics

To conduct these assessments, a generalized methodology was developed as part of the study. The methodology enables the assessment to emphasize important considerations, to minimize duplication of considerations, and to focus attention on areas where additional study is required. Each assessment produces a qualitative or quantitative measure of the cases, and is supplemented by an analysis of technical factors to explain perceived differences between cases.

The methodology used in this study is a simplified version of a more detailed ideal methodology which could, if fully developed, produce a single numerical summation of the values for the six assessment criteria.

The detailed methodology requires generation of preference (weighting) factors to sum the assessments of the individual criteria. Development of such preference factors is beyond the scope of this particular study. A discussion of the detailed methodology is included here in brief form because it represents a logical next step if it is desired to extend the present approach, and it serves as a basis for a description of the simplified methodology actually applied in the study.

3.4.1 Description of a Detailed Ideal Methodology

The detailed assessment methodology is illustrated by the matrix of Table 3-2. In this matrix, three assessment factors can be identified for each combination of an assessment criterion and a major operation or facility of a given case.

A_{ef} A measure of penalty of facility or transport step, f , for given assessment criterion, e .

$W_{e/f}$ A relative preference factor for the assessment criterion, e , for a given facility or transport step, f .

$W_{f/e}$ A relative preference factor for the facility or transportation step, f , for a given assessment criterion, e .

The $W_{e/f}$ and $W_{f/e}$ factors are used as normalizing or weighting factors to sum the A_{ef} assessments to provide a total A_c (case assessment). A comparison of the A_c values for the various cases would be used to determine a preferred case.

Table 3-2
SPENT FUEL DISPOSAL STUDY
IDEAL ASSESSMENT METHODOLOGY

MAJOR OPERATIONS OR FACILITIES

EVALUATION CRITERIA	TRANSPORT – REACTORS TO AFRs	AFRs	TRANSPORT – AFRs TO P/E FACILITIES	P/E FACILITY	TRANSPORT – P/E FACILITY TO REPOSITORY	REPOSITORY	TRANSPORT – P/E FACILITY TO ENRICH- MENT PLANT	TRANSPORT – P/E FACILITY TO MOX FUEL FAB PLANT	ASSESSMENT BY EVALUATION CRITERION
TECHNICAL FEASIBILITY	A_{ef} $W_{f/e}$ $W_{e/f}$								$A_e = \sum_f A_{ef} W_{f/e}$
SAFEGUARDS									
CRITICALITY									
RADIOLOGICAL IMPACTS									
RETRIEV- ABILITY									
ECONOMICS									
ASSESSMENT BY MAJOR OPERATIONS	$A_f = \sum_e A_{ef} W_{e/f}$								TOTAL ASSESSMENT $A_c = \sum_e A_e W_e$ $A_c = \sum_f A_f W_f$

3.4.1.1 Development of Preference Factors. The preference factors are a function of the value system of the decision maker. Where more than one decision maker is involved, multiple, and possibly conflicting, value systems would come into consideration. Aspects of a value system which affect the preference factors include:

- Political considerations
- National security
- Trade-off of risk versus social benefits

Several methods are available for the determination of relative preference factors. They range from the educated assignment of a value to elaborate polling techniques like the Delphi method. The recommended technique is the constant sum method developed by Whitty.⁽⁷⁾

3.4.1.2 Approaches to Determining A_c . Two analytically equivalent approaches are possible to develop A_c , which provide a theoretical basis to verify the result obtained.

Approach A

- 1) Determine for each facility or transportation step a total assessment value, A_f , for a case. This value is the sum of the assessments of the criteria for that facility or transportation step. This is equivalent to adding a vertical column in Table 3-2.
- 2) Sum these to arrive at the assessment, A_c , for a case. This is equivalent to adding the bottom row of Table 3-2.

This approach is described by the equations:

$$A_f = \sum_e A_{ef} W_{e/f} \quad (3-1)$$

$$A_c = \sum_f A_f W_f \quad (3-2)$$

Where:

A_f = Measure of assessment for a given facility or transportation step

W_f = Relative preference factor for the facilities and transportation steps

Approach B

- 1) Determine for each assessment criterion a total assessment value, A_e , for a case. This is determined by adding the assessment values of the facilities and transportation steps for a given criterion (i.e., adding the horizontal rows on Table 3-2).
- 2) Sum these to determine the overall assessment, A_c , for a case. This is equivalent to adding the right hand column of Table 3-2.

This approach is described by the equations:

$$A_e = \sum_f A_{ef} W_{f/e} \quad (3-3)$$

$$A_c = \sum_e A_e W_e \quad (3-4)$$

Where:

A_e = Measure of assessment for a given assessment criterion

W_e = Relative preference factor for the assessment criterion

3.4.2 Description of the Simplified SFDS Methodology

In the methodology actually applied to the SFDS, two major simplifications are made:

- Only Approach B, described above, is used, and only step (1) of that approach (establishing the relative preference factor W_e is beyond the scope of the study)
- The relative preference factors $W_{f/e}$ are assessed in an approximate and individual way for each of the assessment criteria

This SFDS methodology is illustrated by the matrix of Table 3-3 and discussed below.

3.4.2.1 Determination of Assessment Value, A_{ef} . The procedure for developing the assessment value is generally the same regardless of whether the detailed or simplified methodology is applied (i.e., the values are determined from a technical analysis of various attributes of the facilities and transportation steps based on the availability of technical information within the scope of the study).

The determination of the assessment values, A_{ef} , for each of the facilities and transportation steps may be based on either of two approaches:

Approach 1

The value of A_{ef} for a facility or transportation step is a function of the A_s for individual component operations of the facility:

$$A_{ef} = \sum_m A_{mef} W_{mf/e} \quad (3-5)$$

Where:

A_{mef} = The measure of penalty of a process unit or component, m , of facility, f , for criterion, e

$W_{mf/e}$ = The relative preference factor for a process unit or component, m , of facility, f , for criterion, e

Approach 2

The value of A_{ef} for a facility or transportation step is a function of probability and consequences:

$$A_{ef} = P_{j/f} C_{j/f} \quad (3-6)$$

Where:

$P_{j/f}$ = Probability of event, j , occurring for facility, f

$C_{j/f}$ = Consequence of event, j , for facility, f

The determination of A_{ef} may, of course, be a combination of the two approaches.

3.4.2.2 Determination of Relative Preference Factors. The only relative preference factor, or weighting factor, developed for the simplified analysis is $W_{f/e}$. These are used to sum horizontally the values of A_{ef} (Table 3-3). These factors are determined in an approximate and individual way for each assessment criterion as discussed in Sections 4 to 9.

The ground rules used in developing these factors include:

- It is only desired to produce a relative measure of the comparison of the cases. Thus, for some of the criteria, the values of $W_{f/e}$ take values of 0 to 1, depending upon which values of A_{ef} differentiate between the cases. For some criteria (e.g., economics), the values of $W_{f/e}$ are all 1 because a sum for the values of A_{ef} has meaning in the absolute (total dollar) sense. For other criteria, non-integer values of the $W_{f/e}$ are developed to reflect the relative importance of the individual major operations or facilities in summing the A_{ef}

Table 3-3
SPENT FUEL DISPOSAL STUDY
SIMPLIFIED ASSESSMENT METHODOLOGY

MAJOR OPERATIONS OR FACILITIES

EVALUATION CRITERIA	TRANSPORT – REACTORS TO AFRs	AFRs	TRANSPORT – AFRs TO P/E FACILITY	P/E FACILITY	TRANSPORT – P/E FACILITY TO REPOSITORY	REPOSITORY	TRANSPORT – P/E FACILITY TO ENRICH- MENT PLANT	TRANSPORT – P/E FACILITY TO MOX FUEL FAB PLANT	ASSESSMENT BY EVALUATION CRITERION
TECHNICAL FEASIBILITY	A_{ef} $W_{f/e}$								$A_e = \sum_f A_{ef} W_{f/e}$
SAFEGUARDS									
CRITICALITY									
RADIOLOGICAL IMPACTS									
RETRIEV- ABILITY									
ECONOMICS									

- The development of preference factors depends, in part, upon value judgments, which must be incorporated within the engineering judgment considerations used. No polling procedure such as that discussed in Section 3.4.1.1 was applied in determining the $W_{f/e}$ values

3.4.2.3 Assessment Procedure. The assessment is conducted in the following steps:

- 1) Define the assessment criteria such that:
 - The sum of the penalties and benefits associated with each evaluation criterion reflects the comparative ranking of each case, for that criterion
 - Each assessment criterion is reasonably independent of the other
 - The assessment criteria show the difference between cases, if such differences exist
- 2) Identify significant attributes or logical subsystems of the facilities and transportation steps that contribute to the measure of penalty for a case for each assessment criterion.
- 3) Using only the attributes or subsystems that contribute to the differentiation between cases, derive the assessments for each of the facilities and transportation steps. These assessments may be either quantitative (e.g., numerical) or qualitative (e.g., judgmental).
- 4) The relative preference factor, $W_{f/e}$, which is used to combine, for a given criterion, the assessment criteria for each facility or process step, is developed individually for each criterion.

REFERENCES FOR SECTION 3

- (1) "NWTs Repository No. 1 Conceptual Design Report," Vol. I, Stearns-Roger Engineering Co., DOE EY-77-C-05-5367, August 1978.
- (2) "NWTs Repository No. 1 Conceptual Design Report," Vol. III, Stearns-Roger Engineering Co., DOE EY-77-C-05-5367, August 1978.
- (3) "NWTs Repository No. 1 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) W.A. Whitty, "An Evaluation Procedure for Radioactive Waste Treatment Processes," Los Alamos Scientific Laboratory, 1978.

Section 4

ASSESSMENT FOR TECHNICAL FEASIBILITY

TABLE OF CONTENTS

Section		Page
4.1	Assessment Basis and Approach	4-2
	4.1.1 Definitions	4-2
	4.1.2 Measurement of Technical Feasibility	4-2
	4.1.3 Assessment Methodology	4-5
4.2	Assessment Considerations	4-6
	4.2.1 Aspects of Technical Feasibility	4-6
	4.2.2 Basis for Current Approach	4-7
4.3	Assessment by Cases	4-8
	4.3.1 Reference Case Technical Feasibility	4-12
	4.3.2 Case 1 Technical Feasibility	4-15
	4.3.3 Case 2 Technical Feasibility	4-17
	4.3.4 Case 3 Technical Feasibility	4-19
	4.3.5 Summary Comparison	4-21
	References for Section 4	4-23

Section 4

ASSESSMENT FOR TECHNICAL FEASIBILITY

This section contains the assessment of the technical feasibility of constructing and operating the facilities and transportation systems for the cases covered within the scope of the Spent Fuel Disposal Study (SFDS). The section comprises three parts:

- Subsection 4.1 contains definitions, descriptions, and measurements of technical feasibility, and a methodology for developing the assessment
- Subsection 4.2 contains an analysis of assessment considerations for the facilities and transportation links from the standpoint of technical feasibility. The basis for the current assessment approach is developed
- Subsection 4.3 contains the results of the assessments of the study cases

The conclusions developed in this assessment can be summarized as follows:

- The technology for all cases exists or can be developed from existing technology on time schedules consistent with DOE's current waste management program
- AFRs are not a factor in differentiating between spent fuel cases since they involve well-known processes with long operating histories
- The repositories are not a factor in case differentiation since the technologies are essentially the same for all cases. A sleeve testing program to verify corrosion resistance is required. A review of thermal loading design criteria and calculation methodology is recommended
- The relative technical feasibility between the study cases derives essentially from the P/E facility and, to some extent, from the transportation system. The assessment shows the Reference Case and Case 2 to be equal with Case 1 slightly higher and Case 3 with a somewhat lower technical feasibility
- Transportation of all nuclear materials is feasible, and a cask fleet could be provided by the private sector if a clear-cut commitment to nuclear power is made by the government and suitable guarantees against policy changes are given

Technical information to support this feasibility assessment (particularly preconceptual designs, cost information and technical references) can be found in Appendices B, C, D and E of Volume 3.

4.1 ASSESSMENT BASIS AND APPROACH

4.1.1 Definitions

For the SFDS, technical feasibility can be defined as the ability to design, construct, and place in satisfactory operation in a reasonable time any given process, system, facility, or study case. The purpose of the technical feasibility assessment is to produce, for each facility and transportation system within each study case, a numerical measure of its technical feasibility. This number reflects the degree of confidence that the spent fuel disposal facilities can be designed, constructed, and operated satisfactorily.

4.1.2 Measurement of Technical Feasibility

The technical feasibility for a given study case reflects the present degree of technical development and implementation of the facilities and transportation system for that case. A discussion of technical feasibility of all the cases is given in Subsection 4.3.

The degree of technical development and implementation, A_f , for a facility or transportation system is a weighted average degree of development and implementation for the process units or elements constituting that facility or transportation system.

$$A_s = \sum_m (A_m W_m) \quad (4-1)$$

where:

f = A particular facility or transportation system within a study case

m = A process unit of a given facility or a single element of the transportation system

A_m = Degree of technical development and implementation for each m

W_m = Relative contribution of each m to the entire facility or transportation system

4.1.2.1 Technical Development, A_m . A_m depends on the degree of prior successful demonstration. For this study, the values of A_m and the meanings of these values are given in Table 4-1.

Table 4-1
VALUES OF TECHNICAL DEVELOPMENT RATING, A_m ,
FOR ALL FACILITIES AND TRANSPORTATION SYSTEMS

A_m	Meaning	
	Facilities	Transportation System
1	Well known process with long history of plant operation	Long operating history of transportation without major problems
2	Process with all technical problems resolved. Minor improvements are possible through research	Transportation presents no technical problems. Equipment and procedures could be adapted from existing technology
3	Process that is well known but requires full-scale implementation. No commercial-scale operating experience has been achieved	Transportation step requires the solution of some problems and the demonstration of new equipment and procedures
4	Process needs substantial research and development and pilot plant experience before reaching maturity	Transportation step needs substantial development, equipment design, and testing
5	Process that is new, in the theoretical stage, and has not been experimentally evaluated	Transportation step has never been developed nor the equipment designed

4.1.2.2 Relative Contribution, W_m . The development of the relative contribution factors, W_m , for the facilities and transportation system are described below.

Facilities. For facilities, the relative contribution factor reflects, as nearly as possible, the following considerations:

- Process unit complexity
- Relative importance of the process unit

The following approach defines the method of evaluating W_m .

For the process units of the AFRs, the P/E facilities, and the repositories, the relative contribution factor is based on the relationship:

$$W_m = \frac{C_m R_m}{\sum_m (C_m R_m)} \quad (4-2)$$

where:

C_m = A measure of the complexity of the process unit

R_m = A measure of the relative importance of the process unit (from the perspective of possible system or facility failure)

The values of C_m and R_m for facilities and the meanings of these values are given in Tables 4-2 and 4-3.

Table 4-2
VALUES OF COMPLEXITY FACTOR, C_m , FOR ALL FACILITIES

C_m	Meaning
1	A system consisting of several pieces of equipment, performing routine operations, with contact operations and maintenance
2	A system of one or several pieces of equipment, operating remotely, but designed for contact maintenance
3	A system consisting of several interconnected pieces of equipment, designed for remote operation and maintenance
4	A system consisting of several interconnected subsystems, designed for remote operation and maintenance

Table 4-3
VALUES OF RELATIVE IMPORTANCE FACTOR, R_m , FOR ALL FACILITIES

R_m	Meaning
1	A mechanical or chemical system or process which, if it fails, can be repaired with nominal effort and in reasonable time
2	An important mechanical or chemical system. Its failure may cause long periods of plant shutdown for decontamination and repair

Table 4-4
VALUES OF COMPLEXITY FACTOR, C_m , FOR THE TRANSPORTATION SYSTEM

C_m	Meaning
1	Transportation of low level waste (LLW) materials
2	Transportation of highly radioactive materials with a heavy cask required

Transportation system. For the transportation system, W_m is assessed more or less as it is assessed for the facilities. For the transportation elements, however, R_m is always equal to 1.

Thus, Equation 4-2 becomes:

The values of C_m for the transportation system and the meanings of these values are given in Table 4-4.

$$W_m = \frac{C_m}{\sum_m C_m} \quad (4-3)$$

4.1.3 Assessment Methodology

The following methodology is used in developing the technical feasibility assessment for a given case:

- (1) Identify all process units or elements within each facility and transportation system in such a way that:
 - The sum of the process units or elements constitutes the whole of the facility or transportation system
 - If a given element is a process unit in one facility, it is also a process unit in similar facilities in other cases
- (2) Using the 1 to 5 scale defined in Subsection 4.1.2.1, identify the degree of technical development and implementation, A_m , attained for each process unit for all facilities and transportation elements
- (3) Identify the relative contributions, W_m , of each process unit and transportation element involved. These contributions are based on facility design considerations and on transportation studies, using Equations 4-2 and 4-3.
- (4) Combine the results of Steps 1, 2, and 3, using Equation 4-1, to obtain the degree of technical development and implementation, A_f , for each facility and transportation system.

The results obtained in Step 4 provide a basis for comparing the degree of technical development and implementation of the cases and study variations. A summary comparison of the four cases is presented in Subsection 4.3.5.

4.2 ASSESSMENT CONSIDERATIONS

This subsection describes the considerations that affect technical feasibility. It also describes the basis for the current assessment approach.

4.2.1 Aspects of Technical Feasibility

The scope of the technical feasibility assessment includes an analysis of the components of the four study cases:

- AFR
- P/E facility
- Repository
- Transportation system

Each study case includes the transportation system and at least two of these facilities, each of which consists of several elements or process units, as described in Subsection 4.1. In this assessment, the technical feasibility of the elements or process units of a facility is determined by examining the numerical values of A_m , C_m , and R_m for each element.

The values thus derived reflect the contribution of the process increment to the facility as a whole. They also reflect the relative penalty to a system for nonperformance of the incremental part, or the relative complexity of the incremental part.

4.2.1.1 Degree of Technical Development and Implementation, A_m . One method of judging the technical feasibility of a process is to analyze the extent of technical development and utilization of that process in differing types of operations, such as bench-scale, pilot-plant, small-scale production, and full-scale production. Another consideration is the degree of technical attention that has been devoted to conceptual processes in exploring their content, potential problems, and implications.

Fully developed technologies and operating facilities (e.g., the spent fuel receiving and offloading facilities and the fuel storage pool) inspire a good deal of confidence. Other technologies (e.g., high-level waste (HLW) calcination and HLW vitrification), although proven, have been demonstrated only in full-scale pilot plants and therefore give a somewhat lesser degree of confidence. A_m measures the degree of technical development on a scale from 1 to 5; any value of A_m also gives an indication of the additional development required to reach maturity.

The degree of technical implementation is related to the amount of testing that the process has received — in the laboratory or pilot plant, or in commercial operations — and is also a measure of the process development. Some of the processes, such as special welding applications, need little implementation in a pilot plant because they are proven, standard processes in both the non-nuclear and nuclear worlds. The A_m of such processes is 1.

4.2.1.2 Degree of Technical Complexity, C_m . The process units of a facility, or the elements of a transportation system that are simple and uncomplicated are generally feasible systems or facilities. Conversely, a process unit that is interdependent with other processes and complicated in nature is more likely to present problems in design, construction, and operation, and hence is less feasible.

The complexity factor affects mainly P/E facilities, where the chemical processes are carried out with remote operation and control. Some complicated technical process units rate high on a scale of 1 to 4, as measured by C_m . Other simple processes, such as heat treatment of the welded joints on canisters, are rated low, even if the operation is conducted remotely.

4.2.1.3 Relative Importance, R_m . The process units of a P/E facility may be subject to occasional failures which have varied impacts on the operation and reliability of the total facility. In some cases, the failure of one process unit requires a shutdown of the entire facility and complicated decontamination procedures before repairs can be made. In contrast, other process units can be placed back in operation in a short time with nominal effort.

The importance factor, R_m , when combined with the complexity factor discussed above, gives the appropriate weight within the total feasibility assessment of the P/E facility.

4.2.2 Basis for Current Approach

A cursory examination of the four cases under consideration in the SFDS might suggest that it would be possible to design, construct, and operate all of the facilities and to provide transportation between them if an effort were made to overcome design problems and if the cost of construction were acceptable. However, such an approach is unsatisfactory because it fails to distinguish between cases. For this study, an attempt is made to analyze objectively all criteria that affect feasibility by applying numerical values for the design complexity and degree of development. Weighting factors are used to minimize biases.

The range of scales for A_m , C_m , and R_m is somewhat arbitrary and is based on subjective evaluations. For instance, contributors to the study felt that there could be five levels of technical development and four degrees of complexity in a P/E facility process unit.

The rating scales are not strictly linear, because of their dependence on individual judgments, although linearity was the nominal goal. As a result, a rating of 1 does not indicate that the state of technical feasibility of a certain process unit is twice that of a unit with a rating of 2. It may only represent a relative degree of greater feasibility.

4.3 ASSESSMENT BY CASES

In this subsection, the principles developed in Subsections 4.1 and 4.2 are applied to evaluate the technical feasibility of the four study cases. As a first step, the process units of the facilities and the transportation elements for all cases are identified and assessed in tabular form.

Table 4-5 lists the process units for the P/E facilities, describes their current state of development, and assigns a value of A_m . Tables 4-6, 4-7, and 4-8 provide similar information on process units for an AFR, the repository, and the transportation system elements, respectively.

Table 4-5
DEGREE OF TECHNICAL DEVELOPMENT
P/E FACILITY PROCESS UNITS

Process Unit	Degree of Technical Development	A_m
Fuel receiving and offloading	Long history of reliable operation at General Electric Morris Plant, Savannah River Plant, Nuclear Fuel Services, and others	1
Surge storage pool	Wide experience with spent fuel storage pools at U.S. government facilities and private utilities	1
Waste loadout	Waste handling and shipping operations are routine activities at all private utilities and government facilities	1
Canister surge storage (dry)	Same operation as that of fuel storage (dry). Technology and procedures are known	1
Cement fixation of waste	Full-scale plant operation experience at U.S. government facilities and private utilities. Commercial equipment is available	1
Canister loading and welding (remote)	Technology is well known. Equipment and procedures are being tested for the spent fuel packaging program by Rockwell Hanford Operations. QA requirements are stringent	3
Ultrasonic testing (remote)	Well known commercial process. Needs development for remote operation and control	2
Helium leak test (remote)	Widely used process for checking welds. Requires testing for remotely operated operation and control	2
Heat treatment of welds (remote)	Reliable process, used commercially in stress relieving of welded joints. Needs adaptation for remote operation	2
Canister decontamination and drying (remote)	Process used commonly in all nuclear facilities operating by contact methods. Remote operation needs development	2
Incineration of waste	Process has been tried in a pilot plant at Atomic International and other plants. Some problems to resolve ⁽¹⁾	3

Table 4-5 (Cont'd)

Process Unit	Degree of Technical Development	A_m
Clay fixation of waste	Process tried in laboratory only. Equipment not developed	4
Offgas treatment	Process used at U.S. govt. facilities (Hanford, Savannah River, and Idaho Falls)	2
Offgas recovery and fixation	Some fission gas recovery processes are only in the laboratory experimental stage. Others are more developed	3
Spent fuel dissolution	Process plant experience at Hanford, Savannah, and Idaho ⁽¹⁾	1
Pu and U separation and purification	Significant experience at U.S. govt. plants. Limited commercial experience	1
PuO ₂ production and packaging	Laboratory-scale production at Battelle Pacific Northwest Laboratories and other private facilities. Plant scale design available for Barnwell Nuclear Fuel Plant ⁽²⁾	2
UF ₆ production	Commercial-scale operation by Allied Chemical and Kerr-McGee	1
HLW or dissolved fuel concentration	Experience with U.S. govt. facilities. No large-scale operation available	2
Spent fuel dismantling and chopping	Pilot plant demonstrations at Oak Ridge National Laboratory. Early plant operation at Nuclear Fuel Services	2
Voloxidation	Pilot plant demonstration at Oak Ridge National Laboratory for the liquid metal fast breeder reactor program. Applicable to light water reactor fuel	4
HLW calcination	Small-scale operation demonstration at Idaho ⁽³⁾	3
HLW vitrification	Pilot plant hot demonstration at Battelle Pacific Northwest Laboratories ⁽¹⁾	3
Hulls compaction and packaging	Limited laboratory demonstration with cold material at Battelle Pacific Northwest Laboratories	4
Dissolved fuel calcination	This concept is new and has not been developed beyond the conceptual stage	4
Dissolved fuel vitrification	This is a new concept that has not been developed	5

Table 4-6
DEGREE OF TECHNICAL DEVELOPMENT
AFR PROCESS UNITS

Process Unit	Degree of Technical Development	A_m
Fuel receiving and offloading	Successful plant operation for many years at General Electric Morris Plant, Savannah River Plant, and others	1
Spent fuel storage pool	Successful plant operation for many years at General Electric Morris Plant, Savannah River Plant, and others	1
Water treatment	Successful plant operation for many years at General Electric Morris Plant, Savannah River Plant, and others	1
Waste treatment	Successful plant operation for many years at General Electric Morris Plant, Savannah River Plant, and others	1
Cement fixation of waste	Successful plant operation for many years at General Electric Morris Plant, Savannah River Plant, and others	1
Loadout and shipping	Successful plant operation for many years at General Electric Morris Plant, Savannah River Plant, and others	1

Table 4-7
DEGREE OF TECHNICAL DEVELOPMENT
REPOSITORY PROCESS UNITS

Process Unit	Degree of Technical Development	A _m
Cask receiving, unloading, and return shipment	Operations proven at nuclear plants with similar casks under similar conditions	1
Transfer cell operations	Operations have been demonstrated at nuclear facilities (canister grappling, etc.). Some difficulties may have to be resolved	2
Onsite cask handling	Canister is put in cask and cask is transferred to the mine. A well-developed operation, not likely to cause problems	1
Low level waste handling	Operation requires contact methods and radiological controls. Requires design development	2
Mining in salt	Conventional methods for mining salt are well developed	1
Drilling in salt	Drilling in salt is a simple process that needs no development	1
Sleeving in salt	Some problems may arise, such as backfilling around sleeve, mating of upper and lower sleeves, and correcting misalignments	2
Salt handling	A conventional operation that presents no problems	1
Backfilling rooms	A conventional operation that presents no problems	1
Waste emplacement operation	This operation has been performed in project Salt Vault. But, tilting of cask, mating cask with hole, and removing shield plug require further development	3

Table 4-8
DEGREE OF TECHNICAL DEVELOPMENT
TRANSPORTATION SYSTEM ELEMENTS

Transportation Element	Degree of Technical Development	A _m
Spent fuel from reactor to AFR or P/E facility	Long history of operation without problems	1
Spent fuel from AFR to P/E facility	Long history of operation without problems	1
LLW (non-TRU) from P/E facility or AFR to shallow land burial	Long history of operation without problems	1
Canistered spent fuel from P/E facility to repository	Transportation presents no problem. Existing casks and carriers may be adapted	2
Canistered chopped fuel from P/E facility to repository	Transportation presents no problem. Existing casks and carriers may be adapted	2
Vitrified and canistered HLW or spent fuel from P/E facility to repository	Transportation presents no problem. Existing casks and carriers may be adapted	2
Hulls or end pieces from P/E facility to repository	Transportation presents no problem. Existing casks and carriers may be adapted	2
ILW from P/E facility to repository	Same as transportation of current wastes. Presents no problem	1
LLW (TRU) from P/E facility to repository	Same as transportation of current wastes. Presents no problem	1
UF ₆ cylinders from P/E facility to enrichment plant	Existing transports and containers to be used. Will not require changes	1
PuO ₂ containers from P/E facility to recycle	Transportation of PuO ₂ was designed for Barnwell Nuclear Fuel Plant	2

4.3.1 Reference Case Technical Feasibility

The components of the Reference Case are:

- P/E facility
- Repository
- Transportation system

4.3.1.1 P/E Facility. Appendix C contains a description of P/E facilities for all cases. The facility for the Reference Case is a spent fuel reprocessing plant designed to separate uranium and plutonium for recycle. The wastes resulting from the reprocessing are then shipped to an appropriate waste storage facility. The technical feasibility of a P/E facility is summarized in Table 4-9. In this table, the values of C_m and R_m are shown for all the process units of the P/E facility.

Table 4-9

**P/E FACILITY TECHNICAL FEASIBILITY ASSESSMENT
REFERENCE CASE**

Process Unit	A_m	C_m	R_m	$C_m R_m$	W_m	$A_m W_m$
Fuel receiving and offloading	1	1	1	1	0.011	0.011
Surge storage pool	1	1	1	1	0.011	0.011
Waste loadout	1	1	1	1	0.011	0.011
Canister surge storage (dry)	1	2	1	2	0.022	0.022
Cement fixation of waste	1	2	1	2	0.022	0.022
Canister loading and welding	3	2	1	2	0.022	0.066
Ultrasonic testing (remote)	2	2	1	2	0.022	0.044
Helium leak testing	2	2	1	2	0.022	0.044
Heat treatment of welds	2	2	1	2	0.022	0.044
Canister decontamination and drying	2	2	1	2	0.022	0.044
Incineration of waste	3	2	1	2	0.022	0.066
Clay fixation of waste	4	2	1	2	0.022	0.088
Offgas treatment	2	3	1	3	0.033	0.066
Offgas recovery and fixation	3	3	1	3	0.033	0.099
Spent fuel dissolution	1	4	2	8	0.088	0.088
Pu and U separation and purification	1	4	2	8	0.088	0.088
PuO ₂ production and packaging	2	4	2	8	0.088	0.176
UF ₆ production	1	2	1	2	0.022	0.022
HLW or dissolved fuel concentration	2	4	2	8	0.088	0.176
Spent fuel dismantling and chopping	2	3	2	6	0.066	0.132
Voloxidation	4	3	2	6	0.066	0.264
HLW calcination	3	3	2	6	0.066	0.198
HLW vitrification	3	3	2	6	0.066	0.198
Hulls compaction and packaging	4	3	2	6	0.066	0.264
Totals				91	1.000	2.244

$$A_f = 2.2$$

These are the complexity and importance factors defined in Subsection 4.1: the weighting factors, W_m , calculated from Equation 4-2, and the A_f values from Equation 4-1.

The final value of A_f , reflecting the degree of technical development of a P/E facility, is in fact the technical feasibility of the P/E facility.

4.3.1.2 Repository. The repository, described in Appendix D, has several surface facilities to receive and process waste, in addition to the underground mine where the waste is disposed of.

The technical feasibility of the repository is shown in Table 4-10. As in the case of a P/E facility, the complexity and the importance factors, C_m and R_m , are shown and, from them, the weighting factors, W_m , are calculated using Equation 4-2. The final value of A_f is calculated, using Equation 4-1.

Table 4-10
REPOSITORY TECHNICAL FEASIBILITY ASSESSMENT
REFERENCE CASE^(a)

Process Unit	A_m	C_m	R_m	$C_m R_m$	W_m	$A_m W_m$
Cask receiving, offloading, and return shipment	1	1	1	1	0.067	0.067
Transfer cell operations	2	2	1	2	0.133	0.267
Onsite cask handling	1	1	1	1	0.067	0.067
Low level waste handling	2	1	1	1	0.067	0.133
Mining in salt	1	2	1	2	0.133	0.133
Drilling in salt	1	1	1	1	0.067	0.067
Sleeving in salt	2	1	1	1	0.067	0.133
Salt handling	1	1	1	1	0.067	0.067
Backfilling rooms	1	1	1	1	0.067	0.067
Waste emplacement operations	3	2	2	4	0.267	0.800
Totals				15	1.000	1.800

$$A_f = 1.8$$

(a) The values shown are also valid for the repository for Cases 1, 2, and 3.

4.3.1.3 Transportation. The transportation systems that link the facilities of the Reference Case are described in Appendix E. The technical feasibility of the transportation system depends on the state of development, A_m , of the transportation elements and on the weighting factors, W_m .

Table 4-11 shows a list of the various transportation elements that pertain to the Reference Case. The weighting factors, W_m , are calculated from Equation 4-3, and the final value of A_f from Equation 4-1. The value of A_f is a measure of the technical feasibility of all the transportation steps that satisfy the requirements of the Reference Case.

Table 4-11

**TRANSPORTATION SYSTEM TECHNICAL FEASIBILITY ASSESSMENT
REFERENCE CASE**

Transportation Element	A_m	C_m	R_m	$C_m R_m$	W_m	$A_m W_m$
Spent fuel from reactor to P/E facility	1	2	1	2	0.166	0.166
Vitrified HLW from P/E facility to repository	2	2	1	2	0.166	0.333
Compacted hulls from P/E facility to repository	2	2	1	2	0.166	0.333
ILW from P/E facility to repository	1	1	1	1	0.083	0.083
LLW (TRU) from P/E facility to repository	1	1	1	1	0.083	0.083
LLW (non-TRU) from P/E to shallow land burial	1	1	1	1	0.083	0.083
UF ₆ cylinders from P/E facility to recycle	1	1	1	1	0.083	0.083
PuO ₂ containers from P/E facility to recycle	2	2	1	2	0.166	0.333
Totals				12	1.000	1.500

$$A_f = 1.5$$

4.3.2 Case 1 Technical Feasibility

Case 1 has the following components:

- AFR
- P/E facility
- Repository
- Transportation

4.3.2.1 AFR. An AFR is a facility that provides interim storage for spent fuel assemblies discharged from reactors, until the assemblies are transferred to a P/E facility. A description of an AFR is given in Appendix B.

Table 4-12 shows the technical feasibility of an AFR. All the process units of the AFR have achieved a high degree of development ($A_m = 1$), and none of them is complicated ($C_m = 1$). If any of them fails, the plant does not have to be shut down for a long time ($R_m = 1$). Thus, the technologies required to build and operate an AFR are very well developed.

4.3.2.2 P/E Facility. A P/E facility for Case 1 is described in Appendix C. The facility is a spent fuel packaging plant designed to encapsulate in steel canisters the spent fuel assemblies delivered from an AFR. Table 4-13 lists all the process units and their assigned ratings for technical development. The values of A_m , and C_m , and R_m are the same as those shown for the Reference Case. The value of A_f reflects the technical feasibility of a P/E facility for Case 1.

Table 4-12
AFR TECHNICAL FEASIBILITY ASSESSMENT
CASE 1^(a)

Process Unit	A _m	C _m	R _m	C _m R _m	W _m	A _m W _m
Fuel receiving and offloading	1	1	1	1	0.200	0.200
Spent fuel storage pool	1	1	1	1	0.200	0.200
Water treatment	1	1	1	1	0.200	0.200
Cement fixation of waste	1	1	1	1	0.200	0.200
Loadout and shipping	1	1	1	1	0.200	0.200
Totals				5	1.000	1.000

$$A_f = 1.0$$

(a) The values shown are also valid for the AFR of Cases 2 and 3.

Table 4-13
P/E FACILITY TECHNICAL FEASIBILITY ASSESSMENT
CASE 1

Process Unit	A _m	C _m	R _m	C _m R _m	W _m	A _m W _m
Fuel receiving and offloading	1	1	1	1	0.038	0.038
Surge storage pool	1	1	1	1	0.038	0.038
Waste loadout	1	1	1	1	0.038	0.038
Canister surge storage (dry)	1	2	1	2	0.077	0.077
Cement fixation of waste	1	2	1	2	0.077	0.077
Canister loading and welding	3	2	2	4	0.154	0.462
Ultrasonic testing	2	2	1	2	0.077	0.154
Helium leak testing	2	2	1	2	0.077	0.154
Heat treatment of welds	2	2	1	2	0.077	0.154
Canister decontamination and drying	2	2	1	2	0.077	0.154
Incineration of waste	3	2	1	2	0.077	0.231
Clay fixation of waste	4	2	1	2	0.077	0.308
Offgas treatment	2	3	1	3	0.115	0.230
Totals				26	1.000	2.115

$$A_f = 2.1$$

4.3.2.3 Repository. The level of technical development of the repository in Case 1 is the same as that for the Reference Case. Therefore, the value of A_f shown in Table 4-10 for the Reference Case also reflects the technical feasibility for Case 1.

4.3.2.4 Transportation. The transportation technical feasibility assessment for Case 1 is shown in Table 4-14. This table lists the transportation elements and their development ratings and gives the value of A_f .

Table 4-14
TRANSPORTATION TECHNICAL FEASIBILITY ASSESSMENT
CASE 1

Transportation Element	A_m	C_m	R_m	$C_m R_m$	W_m	$A_m W_m$
Spent fuel from reactor to AFR	1	2	1	2	0.200	0.200
Spent fuel from AFR to P/E facility	1	2	1	2	0.200	0.200
LLW (non-TRU) from AFR to shallow land burial	1	1	1	1	0.100	0.100
Spent fuel from P/E facility to repository	2	2	1	2	0.200	0.400
ILW from P/E facility to repository	1	1	1	1	0.100	0.100
LLW (TRU) from P/E facility to repository	1	1	1	1	0.100	0.100
LLW (non-TRU) from P/E facility to shallow land burial	1	1	1	1	0.100	0.100
Totals				10	1.000	1.200

$$A_f = 1.2$$

4.3.3 Case 2 Technical Feasibility

The components of Case 2 are:

- AFR
- P/E facility
- Repository
- Transportation

4.3.3.1 AFR. An AFR for Case 2 is identical to that for Case 1; consequently, Table 4-12 (for Case 1) also gives the technical feasibility assessment for Case 2.

4.3.3.2 P/E Facility. A P/E facility for Case 2 is described in Appendix C. The facility partially processes the fuel by chopping, voloxidizing, and packaging it in steel canisters prior to shipping it to the repository. Table 4-15 lists all the process units and their assigned ratings for technical feasibility. The value of A_f reflects the technical feasibility of a P/E facility for Case 2.

Table 4-15

P/E FACILITY TECHNICAL FEASIBILITY ASSESSMENT
CASE 2

Process Unit	A _m	C _m	R _m	C _m R _m	W _m	A _m W _m
Fuel receiving and offloading	1	1	1	1	0.024	0.024
Surge storage pool	1	1	1	1	0.024	0.024
Waste loadout	1	1	1	1	0.024	0.024
Canister surge storage (dry)	1	2	1	2	0.049	0.049
Cement fixation of waste	1	2	1	2	0.049	0.049
Canister loading and welding	3	2	2	4	0.098	0.294
Ultrasonic testing	2	2	1	2	0.049	0.098
Helium leak testing	2	2	1	2	0.049	0.098
Heat treatment of welds	2	2	1	2	0.049	0.098
Canister decontamination and drying	2	2	1	2	0.049	0.098
Incineration of waste	3	2	1	2	0.049	0.147
Clay fixation of waste	4	2	1	2	0.049	0.196
Offgas treatment	2	3	1	3	0.073	0.146
Offgas recovery and fixation	3	3	1	3	0.073	0.219
Spent fuel dismantling and chopping	2	3	2	6	0.146	0.292
Voloxidation of chopped fuel	4	3	2	6	0.146	0.584
Totals				41	1.000	2.440

$$A_f = 2.4$$

4.3.3.3 Repository. The level of technical development of the repository in Case 2 is the same as that for the Reference Case. Therefore, the value of A_f shown in Table 4-10 for the Reference Case also reflects the technical feasibility for Case 2.

4.3.3.4 Transportation. The transportation technical feasibility assessment for Case 2 is shown in Table 4-16. This table lists the transportation elements and their development ratings and gives the value of A_f .

Table 4-16
TRANSPORTATION TECHNICAL FEASIBILITY ASSESSMENT
CASE 2

Transportation Element	A_m	C_m	R_m	$C_m R_m$	W_m	$A_m W_m$
Spent fuel from reactor to AFR	1	2	1	2	0.166	0.166
Spent fuel from AFR to P/E facility	1	2	1	2	0.166	0.166
LLW (non-TRU) from AFR to shallow land burial	1	1	1	1	0.083	0.083
Chopped fuel from P/E facility to repository	2	2	1	2	0.166	0.333
Fuel element end pieces from P/E facility to repository	2	2	1	2	0.166	0.333
ILW from P/E facility to repository	1	1	1	1	0.083	0.083
LLW (TRU) from P/E facility to repository	1	1	1	1	0.083	0.083
LLW (non-TRU) from P/E facility to shallow land burial	1	1	1	1	0.083	0.083
Totals				<u>12</u>	<u>1.000</u>	<u>1.330</u>

$$A_f = 1.3$$

4.3.4 Case 3 Technical Feasibility

The components of Case 3 are:

- AFR
- P/E facility
- Repository
- Transportation

4.3.4.1 AFR. An AFR for Case 3 is identical to that for Case 1; consequently, Table 4-12 (for Case 1) gives the technical feasibility assessment for Case 3.

4.3.4.2 P/E Facility. A P/E facility for Case 3 is described in Appendix C. This facility is designed to process spent fuel beyond the chopping and voloxidation steps of Case 2 through a series of additional steps that result in packaging vitrified spent fuel for storage at the repository. Table 4-17 lists all the process units and their assigned ratings for technical development. The value of A_f reflects the technical feasibility of a P/E facility for Case 3.

Table 4-17
P/E FACILITY TECHNICAL FEASIBILITY ASSESSMENT
CASE 3

Process Unit	A _m	C _m	R _m	C _m R _m	W _m	A _m W _m
Fuel receiving and offloading	1	1	1	1	0.014	0.014
Surge storage pool	1	1	1	1	0.014	0.014
Waste loadout	1	1	1	1	0.014	0.014
Canister surge storage (dry)	1	2	1	2	0.027	0.027
Cement fixation of waste	1	2	1	2	0.027	0.027
Canister loading and welding	3	2	1	2	0.027	0.082
Ultrasonic testing (remote)	2	2	1	2	0.027	0.055
Helium leak testing	2	2	1	2	0.027	0.055
Heat treatment of welds	2	2	1	2	0.027	0.055
Canister decontamination and drying	2	2	1	2	0.027	0.055
Incineration of waste	3	2	1	2	0.027	0.082
Clay fixation of waste	4	2	1	2	0.027	0.110
Offgas treatment	2	3	1	3	0.041	0.082
Offgas recovery and fixation	3	3	1	3	0.041	0.123
Spent fuel dissolution	1	4	2	8	0.110	0.110
HLW or dissolved fuel concentration	2	4	2	8	0.110	0.220
Spent fuel dismantling and chopping	2	3	2	6	0.082	0.164
Voloxidation	4	3	2	6	0.082	0.329
Hulls compaction and packaging	4	3	2	6	0.082	0.329
Dissolved fuel calcination	5	3	2	6	0.082	0.411
Dissolved fuel vitrification	5	3	2	6	0.082	0.411
Totals				73	1.000	2.769

$$A_f = 2.8$$

4.3.4.3 Repository. The level of technical development of the repository in Case 3 is the same as that for the Reference Case. Therefore, the value of A_f shown in Table 4-10 for the Reference Case also reflects the technical feasibility for Case 3.

4.3.4.4 Transportation. The transportation technical feasibility assessment for Case 3 is shown in Table 4-18. This table lists the transportation elements and their development ratings and gives the value of A_f .

Table 4-18
TRANSPORTATION TECHNICAL FEASIBILITY ASSESSMENT
CASE 3

Transportation Element	A_m	C_m	R_m	$C_m R_m$	W_m	$A_m W_m$
Spent fuel from reactor to AFR	1	2	1	2	0.166	0.166
Spent fuel from AFR to P/E facility	1	2	1	2	0.166	0.166
LLW (non-TRU) from P/E facility to shallow land burial	1	1	1	1	0.083	0.083
Vitrified fuel from P/E facility to repository	2	2	1	2	0.166	0.333
Compacted hulls from P/E facility to repository	2	2	1	2	0.166	0.333
ILW from P/E facility to repository	1	1	1	1	0.083	0.083
LLW (TRU) from P/E facility to repository	1	1	1	1	0.083	0.083
LLW (non-TRU) from P/E facility to shallow land burial	1	1	1	1	0.083	0.083
Totals				12	1.000	1.330

$$A_f = 1.3$$

Table 4-19
COMPARISON OF THE RELATIVE TECHNICAL FEASIBILITY
OF THE FOUR STUDY CASES

Component Part	Ref. Case	Case 1	Case 2	Case 3
AFR	—	1.0	1.0	1.0
P/E facility	2.2	2.1	2.4	2.8
Repository	1.8	1.8	1.8	1.8
Transportation	1.5	1.2	1.3	1.3

4.3.5 Summary Comparison

The numbers that characterize the technical feasibility assessments of the facilities and transportation of each case are shown in Table 4-19.

An inspection of this data shows, first of all, that neither the AFRs nor the repositories contribute to a differentiation between the cases. This is because the AFR application is identical for all the spent fuel cases and the slight differences in repository design between the four study cases are not sensitive in the technical feasibility area.

As a result, the values for the P/E facility and the transportation system provide the case differentiation, although it is obvious that the differences are relatively small. One can, however, combine the P/E facility and transportation numbers either by direct averaging or, more appropriately, by

weighting the P/E facility over the transportation system, somewhat. When this is done, the Reference Case and Case 2 are seen to be almost equal, Case 1 shows a slight advantage and Case 3, a disadvantage. There is a clear difference indicated between Case 1 and Case 3.

The differences in A_f values for the transportation system, are due to the application of higher complexity factors, C_m , to shipments involving significant shielding and to the assignment of slightly larger (i.e., poorer) technical development values, A_m , to canistered fuel (of all types), high-level waste and plutonium oxide shipments.

The analysis regarding P/E facilities is rather straightforward. The Case 1 P/E facility is the least complex and requires the least additional development. Furthermore, all process units contained within that facility are also contained within all other P/E facilities for other cases.

The Case 2 P/E facility is the next least complex. All process units within that facility are also within the P/E facilities for Case 3 and the Reference Case. Those additional process units which are in the Case 2 P/E facility, but not in the Case 1 P/E facility, have the following measures of technical development and implementation, A_m :

● Dismantle and chop	2
● Voloxidation	4
● Offgas recovery & fixation	3

All of these processes are well known. Voloxidation and offgas recovery still require full-scale implementation to gather operating experience.

In the Reference Case P/E facility, the following additional process units, not present in Case 2, are required:

● Fuel dissolution	1
● PuO_2 production	2
● UF_6 production	1
● HLLW concentration	2
● HLW calcination and vitrification	3
● Hulls compaction	4

The hulls compaction process has insufficient experimental development for full-scale application. It is important to note that the hulls compaction step is not necessary for the Reference Case. Without this step, the facilities would function properly with a possible increase in operating cost.

The Case 3 P/E facility has the poorest technical feasibility assessment. In this facility, the calcination and vitrification processes are given A_m assessments of 4 and 5, respectively. Additional R&D would be required to identify design criteria associated with these processes when performed with material having high plutonium and uranium content.

REFERENCES FOR SECTION 4

- (1) "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," Vol. 2, ERDA-76-43, May 1976.
- (2) "Plutonium Conversion Facilities, Baseline Description," Allied General Nuclear Services, NMS-5, July 1977.
- (3) "Status of Commercial Nuclear High-Level Waste Disposal," Electric Power Research Institute, Palo Alto, California, NTIS, PB-260 559, September 1976.

Section 5

SAFEGUARDS ASSESSMENT

TABLE OF CONTENTS

Section		Page
5.1	Assessment Basis and Approach	5-2
5.1.1	Definition	5-2
5.1.2	Measurement of Safeguards Assessment	5-2
5.1.3	Methodology for Developing Assessment	5-4
5.2	Assessment Considerations	5-5
5.2.1	Four Factors that Affect the Potential Risk to the Public	5-5
5.2.2	Basis for Sabotage Assessment	5-7
5.2.3	Basis for Theft Assessment	5-12
5.3	Assessment by Cases	5-15
5.3.1	Assessment of the Reference Case	5-15
5.3.2	Assessment of Case 1	5-16
5.3.3	Assessment of Case 2	5-17
5.3.4	Assessment of Case 3	5-18
5.3.5	Summary Comparison	5-19
	References for Section 5	5-23

Section 5

SAFEGUARDS ASSESSMENT

The ability of a safeguards system to provide and maintain a secure environment for the nuclear materials for each of the four study cases has been investigated. The potential risk to the public from sabotage and theft of nuclear materials has been assessed, and those aspects relative to facility design and operation have been identified. The conclusions of the safeguards assessment can be summarized as follows:

- From the standpoint of sabotage there is little difference between cases, since the greatest sabotage risk (the dispersion of radioactive material from spent fuel) is common to all the cases
- The spent fuel disposal cycle (i.e., Cases 1, 2, and 3) does not present an attractive target to terrorists who seek to fabricate a nuclear explosive device, because materials which might be stolen require considerable additional processing to be of value. Hence Cases 1, 2, and 3 are essentially identical from the theft perspective
- The Reference Case is more attractive for diversion because of the existence of separated plutonium. However, the exposure to theft can be reduced considerably by coprocessing, spiking or the colocation of a P/E facility with a MOX fuel fabrication facility
- Safeguards considerations should not play a major role in the selection between spent fuel disposal alternatives

The analysis and assessment of safeguards risks, which is developed in this section, consists of the following:

- A definition of safeguards for assessment purposes and a description of the methodology used in assessing risks to the public
- Some of the basic safeguards assessment considerations that affect the potential risk to the public
- An assessment of the study cases

A discussion of safeguards systems considerations and criteria as they affect facility design and operation and a number of technical references are given in Appendix F of Volume 3 as support to this assessment.

The safeguards systems considered in this study are expected to meet the regulations and applicable guidelines of the Nuclear Regulatory Commission (NRC). Where NRC regulations are not established (e.g., the repository), regulatory requirements for similar facilities have been assumed to apply.

5.1 ASSESSMENT BASIS AND APPROACH

5.1.1 Definition

For the purpose of this assessment, safeguards is defined as protection of the nuclear material identified in the study cases from sabotage and theft. Sabotage is limited to the dispersion of radioactivity into the environment. Theft of nuclear material is considered to be for the sole purpose of fabricating a nuclear explosive device. This study does not address industrial sabotage (damage within the facility) or the theft of nuclear material for use as poison or in activities such as blackmail. The study addresses the question of safeguards in the back end of the fuel cycle, and in the domestic threat context (i.e., within the U.S.) only.

The purpose of the safeguards assessment is to compare, case by case, the risk to the public of sabotage and theft of nuclear material. No absolute measure of safeguards risk is intended.

5.1.2 Measurement of Safeguards Assessment

For the purpose of comparing risk to the public in the four spent fuel disposal cases, the safeguards assessment is analyzed in two parts: sabotage and theft. No attempt is made to derive weighting factors that would allow these two factors to be combined into one.

5.1.2.1 Sabotage Assessment Basis. The potential risk to the public from sabotage, A_s , has been defined as:

$$A_s = P_s C_s$$

where

P_s = probability of sabotage

C_s = consequences of sabotage to the public

Each of the spent fuel disposal cases consists of the following components:

- Transportation system
- Processing/encapsulation (P/E) facility
- Geologic repository

Cases 1, 2 and 3 also include an away-from-reactor storage facility (AFR).

The probability and consequences (i.e., the potential risk) of sabotage for each component depend on the four factors identified and discussed in Subsection 5.2.1. The absolute values of the probability and consequences cannot be determined within the scope of the Spent Fuel Disposal Study (SFDS). A scale of 1 to 3 is used to rate the relative risk of each type of component in a case. This scale, which ranges from lesser risk to greater risk, respectively, is defined in Table 5-1.

Table 5-1
RATING SCALE FOR POTENTIAL RISK
TO PUBLIC FROM SABOTAGE

A_s	Definition
1	Extremely low risk to the public. Comparable to the dispersal of radioactivity due to the sabotage of a low level waste (LLW) or intermediate level waste (ILW) transuranic (TRU) shipment
2	Very low risk to the public. Comparable to the dispersal of radioactivity due to the sabotage of a shipment of hulls
3	Low risk to the public. Comparable to the dispersal of radioactivity due to the sabotage of a shipment of high level waste (HLW) or spent fuel

Note: The scale in Table 5-1 represents a qualitative assessment rather than a quantitative assessment of each facility for each case, and is used to define a qualitative comparison of relative impact between the cases. The rating scale is not strictly linear.

5.1.2.2 Theft Assessment Basis. The potential risk to the public from theft, A_t , has been defined as:

$$A_t = P_t P_d C_t$$

where

P_t = Probability of theft of nuclear material

P_d = Probability of fabricating a nuclear explosive device from stolen material

C_t = Consequences of a nuclear explosive device to the public

The total risk from theft in each of the spent fuel disposal cases depends on the risk of theft from each of the components (i.e., transportation steps, AFR, P/E facility, or the repository). It also depends on the four factors identified and discussed in Subsection 5.2.1.

The probability of theft of a material depends on the attractiveness of that material for fabricating a nuclear explosive device. The attractiveness of a particular material like spent fuel, for fabricating a nuclear explosive device, is the same for all four cases. The consequences of the explosive device are constant. The absolute values of probability of theft and probability of fabricating a nuclear explosive device and the consequence to the public of fabricating the device have not been determined within the scope of the SFDS. A scale of 1 to 3 is used for rating the relative risk from each of the components (in all four cases) on the basis of the relative difficulty of fabricating a nuclear explosive device from material stolen in that component. This scale is defined in Table 5-2.

Table 5-2
RATING SCALE FOR THE POTENTIAL RISK
TO THE PUBLIC FROM THEFT

A_t	Definition
1	Extremely low. Stolen nuclear fuel material would have to undergo an extremely difficult step (such as separating low abundant plutonium from waste) before a nuclear explosive device could be fabricated
2	Very low. Stolen nuclear material would require reprocessing and conversion to fabricate a nuclear explosive device
3	Low. Stolen nuclear material would require conversion to fabricate a nuclear explosive

Note: The scale in Table 5-2 represents the relative attractiveness of the material stolen. It yields a qualitative assessment rather than a quantitative assessment of each facility for each case, and is used to define a qualitative comparison of relative impact between the cases. The rating scale is not strictly linear.

5.1.3 Methodology for Developing Assessment

The methodology used to assess safeguards risk for a given case is as follows:

- Review the currently available state-of-the-art information on safeguards, including the following sources:
 - Literature on safeguarding nuclear reactors, reprocessing plants, and repositories
 - Personnel at Sandia Laboratories, Los Alamos Laboratory, Savannah River Laboratory (SRL), Allied General Nuclear Services (AGNS), and Lawrence Livermore Laboratories
- Identify considerations for facilities and transportation links which influence either the probability or consequence of sabotage and theft
- Reduce the list of safeguards assessment considerations to those that may contribute to the comparison between cases
- Apply the list of safeguards assessment considerations to each facility or transportation link. The potential risk to the public from sabotage and theft are assessed separately
- The comparison of the cases is based on the comparison of risk to the public from similar components of the cases. For example, risk to the public from sabotage (or theft) of a P/E facility of one case is compared with that of the P/E facility of another case
- The overall assessment of a case is based on the weakest link principle (i.e., the risk from sabotage or theft of a case is represented by the highest risk from sabotage or theft of a component in that case)

5.2 ASSESSMENT CONSIDERATIONS

In this subsection, some of the basic considerations of sabotage and theft as they affect the safeguards assessment are described, and the basis for the current assessment approach is developed. Figure 5-1 shows the scope of the safeguards assessment study.

In each of the four cases, sabotage and theft of nuclear material can take place during transportation, at an AFR, at a P/E facility, or at the repository. Materials that are attractive from the standpoint of sabotage and theft are identified in Figure 5-1. Spent fuel, plutonium oxide, and plutonium nitrate are the only materials of interest from the standpoint of theft, whereas all nuclear materials, spent fuel, hulls, HLW, plutonium, uranium, TRU waste, and non-TRU waste are attractive from the standpoint of sabotage. The terms "spent fuel," "HLW," etc., have been generically used in this diagram and represent various forms of the materials present in the components.

5.2.1 Four Factors that Affect the Potential Risk to the Public

Potential risk to the public from sabotage and theft depends on:

- Adversary action characteristics
- Environmental conditions
- Safeguards systems design
- Properties of materials

A discussion of the above factors and how they affect the comparison of cases is given below.

5.2.1.1 Adversary Action Characteristics. The potential risk to the public depends on the probability of adversary action, the motivation of the adversaries, the number of adversaries, the tactics employed (i.e., use of explosives, type and quantity of explosives, etc.) and the adversary's relationship to the component (insider or outsider). The adversary action characteristics are of the same order of magnitude for a particular type of component (transportation, AFR, P/E facility, or repository) in each of the four cases. For example, the adversary action characteristics for sabotage during transportation in one case are similar to those for sabotage during transportation in another case. (However, adversary action characteristics for sabotage during transportation are not similar to those for sabotage at an AFR, or a P/E facility, or the repository.) Thus, the adversary action characteristics do not enter into a comparative assessment of the cases.

Some of the characteristics of potential adversaries and the threats posed by them in the past are discussed in Appendix F of Volume 3. In general, the probability of a sabotage or theft at a nuclear facility is estimated to be very low.⁽¹⁾

5.2.1.2 Environmental Conditions. The environmental conditions for a particular type of component (transportation, AFR, P/E facility, or repository) affect both the probability and consequences of sabotage and theft. Since these conditions are assumed to be similar for a given component in each of the four cases, they do not enter into a comparative assessment of the cases.

5.2.1.3 Safeguards System Design. The probability of sabotage or theft and the probability of success all depend on the effectiveness of the safeguards system design, which, in turn, depends on the type of material being safeguarded. The more desirable a material is from the point of view of sabotage or theft, the higher the level of safeguards system generally required. Although an acceptable level of potential risk has not yet been defined by NRC regulations, the design must conform to such a level of risk for licensing purposes. The SFDS assumes that the probability of success of sabotage (or theft) for each type of component (e.g., the P/E facility of Case 1 and the P/E facility of Case 2) is of the same order of magnitude in all four cases. Hence, the safeguards

Figure 5-1.
Scope of the Safeguards Assessment Study

system design does not enter into a comparative assessment of the cases. The probability of success, however, of sabotage (or theft) for different types of components (e.g., an AFR and a P/E facility) may not be of the same order of magnitude in each case.

The quantitative risk for facilities is being studied by various laboratories. Heinrich at SRL has done a safeguards assessment study of a proposed complex consisting of reprocessing plant and fuel fabrication facility,⁽²⁾ and has calculated frequency of occurrence and consequence comparisons for sabotage and theft. The frequency values for theft and sabotage are upper-limit boundaries and are shown on a preliminary basis in Figure 5-2 as a function of fatalities per event. This shows that the risk of sabotage and theft are lower than that of many natural phenomena such as earthquakes and dam failures. These risks are in the range calculated in the Rasmussen study for reactors.

5.2.1.4 Properties of Materials. The nature and properties of the materials affect both the probability of an act of sabotage or theft and the consequences of such an act. Certain nuclear materials attractive from the standpoint of sabotage and theft are identified in Figure 5-1 for the various components in each case. The physical and chemical properties, and the containment, of these materials can be different even in similar components of each case. These intrinsic properties of the materials have to be considered in comparing the risks from sabotage or theft.

5.2.2 Basis for Sabotage Assessment

As discussed in Subsection 5.2.1.1, the adversary action characteristics, environmental conditions, and the design of the safeguards system are similar for a particular component from case to case, and, therefore, do not enter into the comparative assessment of the cases. Thus, the risk to the public from sabotage of a component primarily depends on the properties of the nuclear materials handled in that particular component.

The sabotage of a nuclear material can result in two types of potential impact on the public:

- Risk due to direct radiation
- Risk due to ingestion of nuclear material

Since the risk due to ingestion of nuclear materials is essentially a long-term risk (e.g., increased incidence of cancer or genetic damage), it would not likely be of interest to potential saboteurs. Consequently, the relative risk to the public due to direct radiation from sabotage of various nuclear materials in each type of component is used for comparing the cases.

The total potential impact on the public is the sum of the risks from various materials in the component. The relative ratings, based on a scale of 1 to 3 for potential impact from a material, can be combined by using suitable normalization and weighting factors for each material in a component to some common units, so that the risks from each material can be added. Since the determination of the normalization and weighting factors for the materials in a component is not within the scope of the SFDS, the highest rating for a material in a component is used as a rating for that component, as a first approximation.

5.2.2.1 Sabotage During Transportation. If radioactive materials are dispersed during transportation as a consequence of sabotage, the severity of the resulting problem is a function of the nature, form, specific activity, and quantity per shipment. Due to the manner in which radioactive materials are packaged during transportation, it is very difficult to breach the containment (e.g., casks) and disperse the material. The difficulty in breaching a cask is dependent on the amount of radioactivity carried by the cask (and increases as the amount of radioactivity contained in each cask increases). The probability of sabotage is, however, dependent on the nature (or attractiveness from a sabotage point of view) of the material and the time required for its transportation.

Figure 5-2.
Comparison of Frequency and Consequence of Occurrences

The materials to be transported in the various cases are:

- Spent fuel
- HLW
- Hulls
- TRU waste
- Non-TRU waste
- Uranium hexafluoride
- Plutonium oxide

The discussion below presents the basis for assessing the potential risk to the public from sabotage of the above materials during transportation. These assessments are based on the rating scale of 1 to 3 (see Table 5-1).

Transportation of TRU and non-TRU waste. Low and intermediate level wastes may be classified as either TRU or non-TRU wastes in this study. The LLW has a surface dose of less than 10 mrem/hr; the ILW has a surface dose of less than 10 rem/hr. These wastes are usually transported by truck as Type A or Type B wastes. Since they are quite low in radioactivity, compared to HLW or spent fuel, their sabotage will result in much less exposure to the public. The risk to the public from sabotage of these materials is classified as "extremely low" and can be rated as 1 on the rating scale.

Transportation of hulls. The hulls have a surface dose rate of the order of 10^3 rem/hr and are transported in casks similar to those used for the transportation of spent fuel and HLW. The risk to public from sabotage of the hulls shipment is higher than that from a TRU or non-TRU waste shipment, but lower than that from a spent fuel or HLW shipment. The risk is classified as "very low" and rated as 2.

Transportation of uranium hexafluoride. Successful sabotage of uranium hexafluoride canisters will result in the release of solid material that is toxic chemically but not dangerous from a radiological standpoint.

Transportation of plutonium. Successful sabotage of plutonium canisters, although difficult to achieve, may result in the dispersal of radiotoxic particles which are extremely dangerous if inhaled. (See the discussion related to this in Appendix F-3.4 of Volume 3). As a consequence, plutonium is assigned a value of A_s of 3.

Transportation of spent fuel and HLW. The surface dose from a 1-year-old PWR spent fuel assembly is of the order of 10^5 rem/hr; the surface dose from a 10-year-old assembly is of the order of 10^4 rem/hr. The HLW has an activity level and dose rate equivalent to that of spent fuel assemblies. HLW and spent fuel are shipped in casks with a heavy lead or uranium shield. These casks are quite difficult to sabotage. In a test by Sandia Laboratories, a typical spent fuel cask survived a collision in which a 60-mph train hit the cask broadside. Even if the spent fuel cask could be damaged by a strategically placed explosive on the cask, it is highly unlikely that any appreciable amount of radioactivity will be dispersed. From the point of view of saboteurs, the spent fuel and HLW are very attractive, as large amounts of radioactivity can be potentially released from a successful sabotage. Since the probability of sabotage and the probability of success of a sabotage attempt are very small, the risk to the public from sabotage of a spent fuel or HLW shipment is classified as "low" and rated as 3 on the 1 to 3 rating scale.

Table 5-3 summarizes the rating of the risk from sabotage of various materials during transportation.

Table 5-3
RISK TO PUBLIC FROM SABOTAGE
DURING TRANSPORTATION

Item	Risk on a 1 to 3 scale
Spent fuel (any form/age)	3
HLW	3
Hulls	2
ILW TRU and non-TRU	1
LLW TRU and non-TRU	1
PuO ₂	3
UF ₆	1

5.2.2.2 Sabotage at an AFR. At AFRs, spent fuel is the primary material received, stored, and shipped out. Non-TRU LLW generated at an AFR is also handled at the AFR.

The AFRs are designed to process all radioactive effluents prior to release to the environment during normal operation and under accident conditions due to natural phenomena (such as earthquake, tornadoes, floods) and still maintain the release to the environment in accordance with the as-low-as-reasonably-achievable (ALARA) concept. It can, therefore, be assumed that the consequences of sabotage of nuclear materials at an AFR will be smaller than those from sabotage of the same material during transportation. Hence, the risk to the public from sabotage of spent fuel and TRU LLW at an AFR can be classified as 2 (very low) and negligible, respectively.

The overall rating of any case for sabotage at an AFR can, then, be expressed as 2, very low.

5.2.2.3 Sabotage at a P/E Facility. Although specific designs for P/E facilities are incomplete, health and safety standards require that they conform to applicable standards. Portions of plants used for storing or processing plutonium must currently be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

Further protection against sabotage is provided by the massive biological shielding including P/E facilities to protect plant personnel and the public from radiation hazards. This shielding may include underground construction and/or reinforced concrete walls up to 6 feet thick. In addition, the safety design basis of the P/E facility emphasizes system reliability, flexibility, redundancy, and protection against common mode failures which will also lessen the consequences of sabotage at the P/E facility.

Operations at P/E facilities having a potential for accidentally dispersing significant amounts of radioactive contaminants are designed to confine contaminants in the event of accidents or natural phenomena. During the life of P/E facilities, some process or equipment failures will occur. Accordingly, monitors are provided to detect these failures and to initiate, or signal the need for, corrective action. These facilities are also designed to cope with unplanned and sabotage-induced accidents; their cells, buildings, and equipment are designed to be decontaminated and to facilitate the repair or replacement of equipment; and their ventilation systems are designed so that the

contaminated air from any inadvertent releases within the facility are routed through high-efficiency filters to remove airborne radioactive particulates before the ventilation air is discharged.

These plants are analyzed over a wide spectrum of credible accidents and resulting consequences. Although major equipment failures or spills of radioactive materials within the facility might disrupt operations and cause the facility to shut down for cleanup and repair, such occurrences are not expected to result in the release of significant amounts of radioactive materials off site. Only a few accidents involving radioactive materials have occurred in existing fuel reprocessing facilities, and none has resulted in significant contamination beyond the immediate vicinity of the plant. The experience gained from these few accidents has resulted in the adoption of improved safety procedures and features.

It can, therefore, be assumed that the consequences of sabotage of nuclear materials at the P/E facility are smaller than the consequences of sabotage of the same material during transportation. The potential risk to the public from sabotage of various nuclear materials at a P/E facility is shown in Table 5-4.

Table 5-4
RISK TO THE PUBLIC FROM SABOTAGE
AT A P/E FACILITY

Nuclear Material	Risk 1 to 3 scale
Spent fuel (any form/age)	2
HLW	2
Hulls	1
Plutonium compounds	2
TRU and non-TRU ILW	Negligible
TRU and non-TRU LLW	Negligible
Uranium	Negligible

5.2.2.4 Sabotage at a Repository. The design of the repository probably will be required to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, etc., depending on the site.

The repository is designed to prevent radioactive effluents from being released to the environment during normal operation, as well as during postulated accident and sabotage conditions, and still maintain the releases in accordance with the ALARA concept. The radioactive materials are stored at about a depth of 2,000 feet, and air from the repository is filtered prior to release to the environment.

It can, therefore, be assumed that the consequences of sabotage of nuclear materials at a repository are smaller than the consequences of sabotage of the same material during transportation. The risk to the public from sabotage of various nuclear materials at the repository is shown in Table 5-5.

Table 5-5
RISK TO THE PUBLIC FROM SABOTAGE
AT THE REPOSITORY

Nuclear Material	Risk 1 to 3 scale
Spent fuel (any form/age)	2
HLW	2
Hulls	1
TRU and non-TRU ILW	Negligible
TRU LLW	Negligible

5.2.3 Basis for Theft Assessment

5.2.3.1 Assumptions. The following simplifying assumptions are made in order to develop the theft assessment:

- Theft of nuclear material is for the sole purpose of fabricating a nuclear explosive device
- The most significant attribute in the assessment of theft is the utility of the nuclear material for fabricating a nuclear explosive device. If the material cannot be used for a nuclear explosive device, the adversaries will not be interested in stealing it. Thus, the differentiation between cases depends mainly on the consequences of diversion and is represented by a qualitative judgment of the ability to use the nuclear material for a nuclear explosive device
- For each material a relative attractiveness value is assigned, as shown in Table 5-2. The relative attractiveness value represents the difficulty of fabricating a nuclear explosive device from stolen material
- For each component, the assessment of theft is based on the highest rating of the nuclear material transported, stored, or processed in that component
- The design level of protection provided by a safeguards system is related to the attractiveness of the material protected. The nuclear material of highest utility for producing nuclear explosives is protected to a much higher level than a material of lower utility
- The quantity of nuclear material to be diverted in each case is constant and is sufficient to fabricate a nuclear explosive device

Attributes that determine the probability of the occurrence of a theft, the probability of a successful theft, the probability of producing a nuclear explosive, and the consequences of a nuclear explosion should be considered in more detail in future work to achieve a more complete evaluation.

5.2.3.2 Theft Rating Factors. The nuclear materials considered in this study are HLW, hulls, ILW (TRU and non-TRU), LLW (TRU and non-TRU), uranium oxide, uranium hexafluoride, spent fuel, plutonium nitrate, and plutonium oxide. Of these, only spent fuel and plutonium compounds need be considered as potentially attractive materials for fabricating a nuclear explosive device.

HLW, hulls, ILW (TRU and non-TRU), and LLW (TRU and non-TRU). Since most of the uranium and plutonium have been removed from these substances, the amount of fissile material in them is at too low a concentration to be attractive for fabricating a nuclear explosive device. For example, there is about 0.4 gram of plutonium per kilogram of vitrified HLW, and the susceptibility of this

waste is not considered comparable to that of other materials.⁽³⁾ Thus, in this study, HLW, hulls, ILW and LLW are assigned a rating factor of 1 for all four cases.

Uranium oxide and uranium hexafluoride. Uranium of less than 20 percent enrichment in ^{235}U cannot be employed to make a practical nuclear explosive device, even though it can be employed in a reactor to sustain a fission chain. At 20 percent enrichment, the critical mass of metallic uranium with a good neutron reflector is about 250 kilograms⁽⁴⁾ and requires a high-explosive system and other equipment to be useful for fabricating a nuclear explosive device. This is considered too heavy and bulky to be practical. Since all the uranium in the LWR fuel cycle is below 5 percent enrichment, the uranium considered in this study is not useful as a nuclear explosive material.

Spent fuel. Spent fuel from the reactor to the reprocessing plant is shipped in 25- to 90-ton casks (see Subsection E.5). The 25-ton truck cask is designed to contain either two boiling water reactor (BWR) fuel elements containing about 3 kilograms of plutonium (approximately 60 percent ^{239}Pu) each, or one pressurized water reactor (PWR) fuel element containing about 4 kilograms of plutonium (approximately 60 percent ^{239}Pu). Each truck carries only one cask, which holds well below the 8 kilograms of plutonium defined by the International Atomic Energy Agency (IAEA)⁽⁵⁾ as the quantity of safeguards significance. It should be noted that the spent fuel emits so much radiation that any unauthorized removal of the fuel assembly from the cask would release a lethal dose of radiation to the diverter, unless handled in proper facilities.

The 90-ton cask planned for use in rail shipments is designed to contain 19 to 34 BWR fuel elements or 8 to 14 PWR fuel elements, depending on the age of spent fuel. The weight of this cask makes it extremely difficult to steal or transship on a truck. If it remains on a rail, it should be easy to locate. In Appendix F the various methods available for providing in-transit safeguards during the transportation of spent fuel are discussed.

Stealing spent fuel from a P/E facility or an AFR is even more difficult than stealing fuel during transportation. A thief would have to penetrate the safeguards system at the facility to gain access to the spent fuel and then remove the multi-ton spent fuel assemblies through the facility safeguards system. The probability of accomplishing this is almost zero.

Even if the spent fuel were stolen, it would have to be processed to obtain separated plutonium before a nuclear explosive device could be fabricated. Estimates made by experts indicate that the preparation time (from the decision to steal, and including design, construction, and testing) would be at least 18 to 24 months for reprocessing plants designed for simplicity of construction and operation. Construction costs of the plant would be in the range of several million dollars.⁽⁶⁾ Thus, to process the spent fuel to produce plutonium, a well organized, well financed, and skilled group of individuals would be necessary, and this group would have to overcome the many obstacles described in Appendix F to fabricate a nuclear explosive device.

A rating factor of 2 is assigned to spent fuel for all locations.

Plutonium. The remaining alternative is to steal the plutonium. If the plutonium were stolen, it would still need to be transported, converted to an appropriate form, and then used to fabricate a nuclear explosive device, which is no easy task.

To discourage the theft of plutonium, the safeguards system should reduce the probability of a success of theft to almost the level of probability of successfully stealing a nuclear weapon. Since the consequences of a thief's obtaining a nuclear explosive device could be very serious, the probability of obtaining a nuclear explosive by stealing the plutonium (or the explosive itself) must be made extremely small. The plutonium could be kept in a form so that it is never completely separated from other radioactive material, or it could be sufficiently spiked with radioactive isotopes so that fatal radiation levels would be present if a thief attempted to work with it. In addition, the reprocessing plant and fuel fabrication plant could be located at the same site so that a thief

would have to penetrate a very elaborate fixed-site safeguards system. This would eliminate the transportation of plutonium oxide, the in-transit physical security for special nuclear materials (SNM) shipments, and the integrated container vehicle described in Appendix F.

The safeguards system can be developed to a point where the probability of successfully stealing plutonium is almost zero. Reasonable estimates suggest that such a system can be built at a cost of about 5 to 7 percent of the facility cost. Thus, a safeguards system that reduces the probability of successfully stealing plutonium to an acceptable level can be constructed for a reasonable price.

A rating factor of 3 is assigned to the plutonium considered in this study.

5.3 ASSESSMENT BY CASES

Each of the spent fuel disposal cases has been divided into the following components:

- Transportation
- AFR
- P/E facility
- Repository

The potential risk to the public from theft of nuclear materials or sabotage within these components depends upon the adversary action characteristics, environmental conditions, safeguards system design, and properties of the nuclear materials. Also, the potential risk to the public from sabotage or theft depends on the probability of its occurrence and the consequences of such an occurrence as defined in Subsection 5.1. Since the probability of occurrence and its consequences cannot be determined quantitatively within the scope of the SFDS, the risk to the public from sabotage and theft (A_s and A_t , respectively), in each component of each case, have been rated from extremely low to low on a scale of 1 to 3, as defined in Subsection 5.1.

To derive a single value of potential risk of safeguards for each case, weighting factors must be developed that represent the relative importance of the components (transportation, AFR, P/E facility, and repository). These weighting factors can be defined in many ways, depending on the definition of the relative importance of the components.

The weighting factors cannot be defined within the scope of the SFDS. The comparison of the cases is, however, based on a comparison of one component of a case (e.g., a P/E facility) with the same component (P/E facility) of another case. The overall assessment of a case is based on the weakest link principle; i.e., the risk from sabotage or theft of a case is represented by the highest risk from sabotage or theft for a component in that case.

5.3.1 Assessment of the Reference Case

The Reference Case consists of the following components:

- Transportation
 - 1-year-old spent fuel from the reactors to a P/E facility
 - 10-year-old vitrified HLW, hulls, TRU LLW, TRU ILW, and non-TRU ILW from a P/E facility to the repository
 - Non-TRU LLW from a P/E facility to shallow land burial
 - Uranium hexafluoride and plutonium oxide from a P/E facility to recycle facilities
- P/E facility
- Repository

5.3.1.1 Potential Risk from Sabotage in the Reference Case. As discussed previously, the potential risk to the public from a component of a case is rated on a scale of 1 to 3 and is defined as the highest rating for the material handled in that component of the case. Thus, for the Reference Case:

Step	A _s	Description
Transportation	3	Transported spent fuel, HLW and PuO ₂
AFRs	N A	(Not applicable) No AFRs in this case
P/E facility	2	Spent fuel and HLW handled at the P/E facility
Repository	2	HLW at the repository

5.3.1.2 Potential Risk from Theft in the Reference Case. As discussed previously, the potential risk to the public from each component of a case is rated on a scale of 1 to 3 and is defined as the highest rating for the material handled in that component of the case. Thus, for the Reference Case:

Step	A _t	Description
Transportation	3	Plutonium transported from a P/E facility to the recycle facility
AFR	N A	No AFRs in this case
P/E facility	3	Plutonium handled in a P/E facility
Repository	1	Waste materials from which most uranium and plutonium have been removed in the repository

5.3.2 Assessment of Case 1

Case 1 consists of the following components:

- Transportation
 - 1-year-old spent fuel from the reactors to an AFR
 - 10-year-old spent fuel from an AFR to a P/E facility
 - Non-TRU LLW from an AFR to shallow land burial
 - 10-year-old encapsulated spent fuel and TRU LLW from a P/E facility to the repository
 - Non-TRU LLW from a P/E facility to shallow land burial
- AFR
- P/E facility
- Repository

5.3.2.1 Potential Risk from Sabotage in Case 1. As discussed previously, the potential impact on the public from sabotage from a component of a case is rated on an arbitrary scale of 1 to 3 and is defined as the highest rating for the material handled in the component of the case. Thus, for Case 1:

Step	A _s	Description
Transportation	3	Transported spent fuel
AFR	2	Spent fuel handled at AFR
P/E facility	2	Spent fuel handled at a P/E facility
Repository	2	Spent fuel stored in the repository

5.3.2.2 Potential Risk from Theft in Case 1. As discussed previously, the potential risk to the public from theft in each component of a case is rated on an arbitrary scale of 1 to 3 and is defined as the highest rating for the material handled in that component of the case. Thus, for Case 1:

Step	A _t	Description
Transportation	2	Transported spent fuel
AFR	2	Spent fuel stored at an AFR
P/E facility	2	Spent fuel handled at a P/E facility
Repository	2	Spent fuel stored in the repository

5.3.3 Assessment of Case 2

Case 2 consists of the following components:

- Transportation
 - 1-year-old spent fuel from the reactors to an AFR
 - 10-year-old spent fuel from an AFR to a P/E facility
 - Non-TRU LLW from an AFR to shallow land burial
 - 10-year-old voloxidized spent fuel, hulls, TRU LLW, and non-TRU ILW from a P/E facility to the repository
 - Non-TRU LLW from a P/E facility to shallow land burial
- AFR
- P/E facility
- Repository

5.3.3.1 Potential Risk from Sabotage in Case 2. As discussed previously, the potential impact on the public from sabotage of a component in a case is rated on an arbitrary scale of 1 to 3 and is defined as the highest rating for the material handled in that component of the case. Thus, for Case 2:

Step	A _s	Description
Transportation	3	Transported spent fuel
AFR	2	Spent fuel stored at an AFR
P/E facility	2	Spent fuel handled at a P/E facility
Repository	2	Spent fuel stored in the repository

5.3.3.2 Potential Risk from Theft in Case 2. As discussed previously, the potential risk to the public from theft in each component of a case has been rated on an arbitrary scale of 1 to 3 and is defined as the highest rating for the material handled in that component of the case. Thus, for Case 2:

Step	A _t	Description
Transportation	2	Transported spent fuel
AFR	2	Spent fuel stored at an AFR
P/E facility	2	Spent fuel handled in a P/E facility
Repository	2	Spent fuel stored in the repository

5.3.4 Assessment of Case 3

Case 3 consists of the following components:

- Transportation
 - 1-year-old spent fuel from the reactors to an AFR
 - 10-year-old spent fuel from an AFR to a P/E facility
 - Non-TRU LLW from an AFR to shallow land burial
 - Vitrified fuel, hulls, TRU LLW, non-TRU ILW from a P/E facility to the repository
 - Non-TRU LLW from a P/E facility to shallow land burial
- AFR
- P/E facility
- Repository

5.3.4.1 Potential Risk from Sabotage in Case 3. As discussed previously, the potential impact on the public from sabotage of a component in a case is rated on a 1 to 3 arbitrary scale and is defined as the highest rating for the material handled in that component of a case. Thus, for Case 3:

Step	A _s	Description
Transportation	3	Transported spent fuel
AFR	2	Spent fuel stored at an AFR
P/E facility	2	Spent fuel handled at a P/E facility
Repository	2	Spent fuel stored in the repository

5.3.4.2 Potential Risk from Theft in Case 3. As discussed previously, the potential risk to the public from theft in each component of a case has been rated on an arbitrary 1 to 3 scale and is defined as the highest rating for the material handled in that component of the case. Thus, for Case 3:

Step	A _t	Description
Transportation	2	Transported spent fuel
AFR	2	Spent fuel stored at an AFR
P/E facility	2	Spent fuel handled at a P/E facility
Repository	2	Spent fuel stored at the repository

5.3.5 Summary Comparison

The four spent fuel disposal cases are assessed above from the standpoint of risk to the public from sabotage and theft during transportation, at an AFR, at a P/E facility, and at the repository. These assessments are the basis of first-approximation comparisons. From these comparisons, it appears that safeguard considerations may not play any major role in the selection of a disposal alternate. The potential risk to the public of sabotage and theft is of very low level and similar for each of the four cases.

5.3.5.1 Sabotage. As discussed previously, the potential risk to the public due to sabotage in various components of the four cases is expressed on a 1 to 3 rating scale and is summarized in Table 5-6. As can be seen, the potential risk to the public of sabotage in Cases 1, 2, and 3 is similar and differences, if they exist, cannot be distinguished by an evaluation using a 1 to 3 scale. The risk due to sabotage during transportation, at a P/E facility, and at the repository are similar for all four cases.

Table 5-6
POTENTIAL RISK TO THE PUBLIC FROM SABOTAGE

Item	Ref. Case	Rating on 1 to 3 scale		
		Case 1	Case 2	Case 3
Sabotage during transportation	3	3	3	3
Sabotage at an AFR	NA	2	2	2
Sabotage at a P/E facility	2	2	2	2
Sabotage at the repository	2	2	2	2
Sabotage assessment based on the weakest link	3	3	3	3

Figures 5-3, 5-4, 5-5, and 5-6 graphically compare the potential risk to the public from sabotage during transportation, at an AFR, at a P/E facility, and at the repository, respectively, for all four cases. In these figures, each component of a case has been divided into various steps or process units and plotted against the rating scale. Although the rating scale defined in Subsection 5.1.2 is from 1 to 3, note that no assigned (or resultant) assessment values exceed a rating of 3. These diagrams show more differences between the various cases than are apparent from Table 5-6. Similarly, if the risk from sabotage is studied in detail by comparing and evaluating the various attributes discussed in Subsection 5.2, the actual impact from sabotage in each case may be different. These differences are not, however, expected to be significant. Therefore, it can be reasonably expected that the potential risk from sabotage is similar in magnitude in all four cases, which is true if one assesses risk from sabotage for each case on the basis of the weakest link.

5.3.5.2 Theft. As discussed previously, the potential risk to the public due to theft of nuclear materials in various components of the four cases is expressed on a 1 to 3 rating scale and is summarized in Table 5-7. Note that the potential risk to the public from theft in Cases 1, 2, and 3 are similar and that the differences, if they exist, cannot be distinguished by an evaluation using a 1 to 3 scale. Compared with the Reference Case, the overall risk to the public due to theft in Cases 1, 2, and 3 is lower during transportation and at a P/E facility but is higher at the AFR and at the repository. The main reasons for these differences are:

- Plutonium oxide, which is more attractive than spent fuel for fabricating a nuclear explosive device, is handled during transportation and at a P/E facility in the Reference Case

Figure 5-3.
Potential Risk from Sabotage During Transportation

Figure 5-4.
Potential Risk from Sabotage at an AFR

Figure 5-5.
Potential Risk from Sabotage at a P/E Facility

Figure 5-6.
Potential Risk from Sabotage at the Repository

- There is no AFR in the Reference Case
- The repository in the Reference Case contains HLW, hulls, and TRU waste only, which are less attractive than spent fuel for fabricating a nuclear explosive device

Table 5-7
POTENTIAL RISK TO THE PUBLIC FROM THEFT

Item	Rating on 1 to 3 scale			
	Ref. Case	Case 1	Case 2	Case 3
Theft during transportation	3	2	2	2
Theft at an AFR	N A	2	2	2
Theft at a P/E facility	3	2	2	2
Theft at the repository	1	2	2	2
Theft assessment based on the weakest link	3*	2	2	2

*Assumes separated PuO₂ being transported.

From Table 5-7, it appears that the risk to the public from theft in Cases 1, 2, and 3 is similar in magnitude. In the Reference Case, better safeguards protection is required for plutonium because of the latter's attractiveness as a weapons material.

Some of the current antiproliferation proposals, if incorporated into the Reference Case (e.g., CIVEX, coprocessing, colocated energy parks, etc.) would reduce the safeguards risk, and hence improve the assessment of that case.

REFERENCES FOR SECTION 5

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- (4) Mason Willrich and Theodore B. Taylor, *Nuclear Theft: Risks and Safeguards*, Ballinger Publishing Company, Cambridge, 1974. NB p.58.
- (5) G. Hough, T. Shea, and D. Tolchenkov, "Technical Criteria for the Application of IAEA Safeguards," IAEA-SM-231/112. NB p. 209.
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Section 6

CRITICALITY ASSESSMENT FOR LONG-TERM DISPOSAL

TABLE OF CONTENTS

Section		Page
6.1	Assessment Basis and Approach	6-2
6.1.1	Definition of Criticality	6-2
6.1.2	Approach	6-2
6.2	Assessment Considerations	6-3
6.2.1	Probability of a Criticality	6-3
6.2.2	Magnitude of a Criticality	6-10
6.2.3	Potential Radiological Impact	6-11
	References for Section 6	6-13

Section 6

CRITICALITY ASSESSMENT FOR LONG-TERM DISPOSAL

Criticality analyses on the design and operation of any waste handling facility (e.g., AFR, P/E facility, transportation system, repository) are always based on conservative assumptions regarding the fissile material content and physical/chemical composition of the nuclear waste. In considering the potential for criticality during such handling and storage, these assumptions are made sufficiently conservative to require the inclusion of positive design features such as geometry control, limits on the physical/chemical waste form, and operating or administrative controls. The rigid application of these features and controls are expected to prevent a criticality from occurring during the operating life of a licensed facility. For the Spent Fuel Disposal Study, then, the paramount question is whether the long-term geologic effects on the fissile material in a repository are sufficient to negate the established controls and create the conditions needed for a criticality to occur. It is this aspect of criticality that drew the principal focus in the SFDS.

Since fissile nuclides typically have long half lives, the probability and consequence of an agglomeration of fissile material causing a critical chain reaction are a concern in the long-term geologic disposal of nuclear waste. The importance of a criticality assessment in long-term isolation lies in the possible radiological risk to future populations. Such an event would not in itself be harmful unless the repository were coincidentally breached and the resulting radionuclides moved to the biosphere. The subject of criticality is addressed because there is sufficient fissile material in a single waste canister, in the cases considered in the SFDS, to form a critical mass under optimum or near optimum conditions. The investigation of criticality in the SFDS comprises a study of the mechanisms affecting long-term criticality and an assessment of ensuing risk. The results of this analysis provide the following conclusions:

- The probability of producing the conditions needed for a criticality in a repository by naturally occurring events is extremely low
- A criticality event, if it did occur, would be localized, self-limiting, and would not involve the violent reaction associated with the rapid accumulation of a critical mass. One analysis shows that the power level of such a critical system might be on the order of 4 kW
- The risks and radiological impacts to future populations due to a criticality within a deep geologic repository are diminishingly small

6.1 ASSESSMENT BASIS AND APPROACH

The approach used in making the criticality assessment is given in this subsection. The assessment itself draws upon a number of technical considerations and references which are identified and discussed in Appendix G of Volume 3.

6.1.1 Definition of Criticality

A criticality event is an uncontrolled nuclear chain reaction that is self-sustained for some period of time. This reaction occurs when the amount of fissile material exceeds the critical mass and when the necessary conditions of neutron moderation and reflection are achieved. The formation of a critical system may be the result of an internal change, such as a chemical reaction, or it may result from external changes which cause the required conditions for a criticality to be attained.

6.1.2 Approach

When criticality considerations are extended over geologic time spans, the differences between the spent fuel cases studied diminish and the criticality potential of the waste depends on its composition rather than the nature of treatment and encapsulation prior to disposal. This criticality assessment does not, therefore, attempt to differentiate between the cases. Instead it analyzes long-term criticality in the repository by: (1) reviewing the potential for critical mass formation in the canistered waste material for each case, (2) defining the conditions which must be present to achieve and sustain a critical reaction (including the unique combinations of these conditions), (3) postulating the magnitude of a potential critical event, and (4) estimating the radiological impact and risk from such an event.

6.2 ASSESSMENT CONSIDERATIONS

6.2.1 Probability of a Criticality

6.2.1.1 Potential for Criticality in the Waste. For each case in the study, the potential for achieving criticality can be determined by comparing the quantity of fissile material in spent fuel and HLW with available minimum critical mass data. These data are presented in Tables 6-1 through 6-5⁽¹⁾ and are summarized below:

- (1) Table 6-1 lists the amounts of fissile material contained in a single waste canister for each case considered in the study. The table presents this data as a function of time over one million years.
- (2) For every group of fissile materials there is a minimum amount required to achieve a critical mass. This amount varies as a function of the neutron moderating and reflecting conditions. Table 6-2, which has the same format as Table 6-1, lists the minimum critical mass for each waste type studied when redistributed in optimum configuration and moderated and reflected by pure water.
- (3) Table 6-3 is essentially the same as Table 6-2, only the minimum critical mass is moderated and reflected by a saturated solution of brine.
- (4) The fraction of the minimum required critical mass that is contained by a single waste container in a water moderated environment is presented in Table 6-4. These fractions were obtained by ratioing the data in Table 6-2 to that in Table 6-1. Some of these fractions exceed one. Thus, for all cases in the study, the table indicates where it is theoretically possible to achieve a criticality in situ with the fissile isotopes from one waste container if they are redistributed in pure water under near optimum conditions.
- (5) Table 6-5 lists the fraction of the minimum required critical mass contained in a single waste container in a brine environment. These values were obtained by ratioing the data in Table 6-3 to that in 6-1. The fractions obtained suggest that fissile isotopes from a minimum of about 3.5 waste canisters would have to be combined under near optimum conditions in brine for a criticality to occur.

The criticality potential of new fuel is higher than spent fuel only because of the lack of fission products. (Hence, if for some reason unirradiated fuel assemblies were to be stored in the repository, they should not be stored in the same location but spread throughout the storage area.) If the poison effects of the fission products and ^{240}Pu are ignored, the criticality potential of spent fuel is approximately equal to that of fresh fuel. This is due to the buildup of fissile isotopes of plutonium as uranium is irradiated. To the extent that fissile nuclides could be selectively leached out of the fuel elements, spent fuel could approach criticality in essentially the same circumstances as new fuel. Furthermore, due to the decay of ^{243}Pu and ^{243}Am to the fissile isotope ^{239}Pu , the total amount of fissile material in a spent fuel waste canister decreases only gradually over thousands of years. Hence, depending upon the geological time frame involved, 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. However, radiological decay, for a single container, 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.

Table 6-1

FISSILE MASS CONTAINED IN ONE STUDY CASE WASTE CANISTER

Case	Fissile Material	Fissile Mass per Canister (kg-oxide) After Indicated Years After Reactor Discharge				
		10 ² yr	10 ³ yr	10 ⁴ yr	10 ⁵ yr	10 ⁶ yr
1	Fuel rods ^(a) – BWR	147 rods	147	147	147	147
	– PWR	225 rods	225	225	225	225
	Mixture of actinides	603.6	603.6	603.4	601.9	NA ^(b)
	Plutonium	5.1	4.8	3.35	0.43 ^(c)	NA
2	Mixture of actinides	680.4	680.4	680.2	678.5	NA
	Plutonium	5.7	5.4	3.77	0.49	NA
3	Mixture of actinides	423.6	423.6	423.5	422.4	NA
	Plutonium	3.6	3.4	2.35	0.30	NA
Reference	Mixture of actinides	13.24	13.24	13.21	13.2	13.2
	Curium	0.080	0.063	0.029	NA	NA
	Uranium	NA	NA	NA	13.55 ^(d)	13.55 ^(d)

(a) Based upon 3-BWR assemblies 7×7, and 1-PWR assembly 15×15.

(b) NA – Data not available.

(c) Determined by delay calculation from fissile composition at 10,000 years.

(d) Based upon uranium content only (applicable if uranium-neptunium separation occurs).

Table 6-2

**MINIMUM CRITICAL MASS FOR EACH STUDY CASE WASTE FORM
WHEN MODERATED AND REFLECTED BY PURE WATER**

Case	Fissile Material	Minimum Critical Mass (kg-oxide) After Indicated Years After Reactor Discharge				
		10 ² yr	10 ³ yr	10 ⁴ yr	10 ⁵ yr	10 ⁶ yr
1	Fuel rods in water – BWR(a)	430 rods	430	430	430	430
	– PWR(a)	710 rods	710	710	710	710
	Heterogeneous actinides + H ₂ O(b)	225	225	225	225	225
	Homogeneous actinides + H ₂ O(c)	∞	∞	∞	NA(d)	NA
	Plutonium + H ₂ O solution	2.37	2.12	1.27	NA	NA
2	Heterogeneous actinides + H ₂ O(e)	455	455	455	455	455
	Heterogeneous actinides + H ₂ O(e)	225	225	225	225	225
	Homogeneous actinides + H ₂ O(c)	∞	∞	∞	NA	NA
	Plutonium + H ₂ O solution	2.37	2.12	1.27	NA	NA
3	Homogeneous actinides + H ₂ O(c)	∞	∞	∞	NA	NA
	Plutonium + H ₂ O solution	2.37	2.12	1.27	NA	NA
Reference	Actinides in glass	457.4	580.1	432.4	NA	NA
	Homogeneous actinides + H ₂ O	950.1	990.3	680.8	NA	NA
	Curium + H ₂ O solution	0.0593	0.0512	0.0492	NA	NA
	Uranium + H ₂ O solution	NA	NA	NA	17.9(f)	17.9(f)

(a) Based upon UO₂ + H₂O lattice calculations @ enrichments = 1.5% and no clad

BWR – W/F = 3.0; pellet dia = 0.50 inch

PWR – W/F = 2.8; pellet dia = 0.40 inch

(b) Based upon U(nat) O₂ + 1.5 PuO₂ lattice calculations, ²⁴⁰Pu = 5%

(c) No critical mass is possible in 1-meter diameter sphere for delay times up to 10,000 years

(d) NA – Data not available

(e) The higher number is for density $\rho = 3.79$ gm oxide/cm³, lower number for optimum moderation

(f) Based upon uranium-water solution, enrichment = 13.6% ²³⁵U

Table 6-3

**MINIMUM CRITICAL MASS FOR EACH STUDY CASE WASTE FORM
WHEN MODERATED AND REFLECTED BY SATURATED BRINE**

Case	Fissile Material	Minimum Critical Mass (kg-oxide) After Indicated Years After Reactor Discharge				
		10 ² yr	10 ³ yr	10 ⁴ yr	10 ⁵ yr	10 ⁶ yr
1	Fuel rods – BWR	NA(a)	NA	NA	NA	NA
	– PWR	NA	NA	NA	NA	NA
	Heterogeneous actinides	NA	NA	NA	NA	NA
	Homogeneous actinides	∞	∞	∞	NA	NA
	Plutonium solution	19.3	19.1	12.3	NA	NA
2	Heterogeneous actinides: $\rho = \frac{\text{gm}}{\text{cm}^3}$	NA	NA	NA	NA	NA
	Heterogeneous actinides: Optimum moderation	NA	NA	NA	NA	NA
	Homogeneous actinides	∞	∞	∞	NA	NA
	Plutonium solution	19.3	19.1	12.3	NA	NA
3	Homogeneous actinides	∞	∞	∞	NA	NA
	Plutonium solution	19.3	19.1	12.3	NA	NA
Reference	Homogeneous actinides	1084	1139.6	818.6	NA	NA
	Curium solution	0.556	0.467	0.450	NA	NA
	Uranium solution	NA	NA	NA	403.1(b)	403.1(b)

(a) NA – No data available

(b) Based upon uranium-brine solution, enrichment = 13.6% ²³⁵U

Table 6-4

**FRACTION OF THE MINIMUM CRITICAL MASS CONTAINED IN ONE WASTE CANISTER
WHEN MODERATED AND REFLECTED BY PURE WATER**

Case	Fissile Material	Critical Mass Fraction After Indicated Years After Reactor Discharge				
		10 ² yr	10 ³ yr	10 ⁴ yr	10 ⁵ yr	10 ⁶ yr
1	Fuel rods in water — BWR ^(a)	0.342	0.342	0.342	0.342	0.342
	— PWR ^(a)	0.317	0.317	0.317	0.317	0.317
	Actinide pellets in water ^(a)	2.68	2.68	2.68	2.68	2.68
	Actinide solution	0.0	0.0	0.0	NA	NA
	Plutonium solution	2.15	2.26	2.64	NA	NA
2	Actinide pellets: $\rho = 3.79 \text{ gm/cm}^3$ ^(a)	1.50	1.50	1.50	1.50	1.50
	Actinide pellets: Optimum moderation ^(a)	3.02	3.02	3.02	3.02	3.02
	Actinide solution	0.02	0.02	0.0	NA	NA
	Plutonium solution	2.41	2.55	2.97	NA	NA
3	Actinide solution	0.0	0.0	0.0	NA	NA
	Plutonium solution	1.52	1.60	1.85	NA	NA
Reference	Actinides in glass	0.029 ^(c)	0.023 ^(c)	0.031 ^(c)	NA	NA
	Actinide solution	0.014	0.013	0.019	NA	NA
	Curium solution	1.35	1.23	0.589	~ 0.0	~ 0.0
	Uranium solution	~ 0.0	~ 0.0	~ 0.0	0.757 ^(b)	0.757 ^(b)

(a) Criticality estimates based upon data at constant isotopic compositions — no radiological decay considered for this evaluation.

(b) In this time frame (10,000 — 100,000 years), ²³⁹Pu decays to ²³⁵U increasing the uranium enrichment to ~ 13.6%. This results in the increase in critical fraction noted.

(c) Based upon data for actinide elements moderated and reflected by granite to simulate glass.

Table 6-5

**FRACTION OF THE MINIMUM CRITICAL MASS CONTAINED IN ONE WASTE CANISTER
WHEN MODERATED AND REFLECTED BY SATURATED BRINE**

Case	Fissile Material	Critical Mass Fraction After Indicated Years After Reactor Discharge				
		10 ² yr	10 ³ yr	10 ⁴ yr	10 ⁵ yr	10 ⁶ yr
1	Fuel rods – BWR	NA ^(a)	NA	NA	NA	NA
	– PWR	NA	NA	NA	NA	NA
	Actinide pellets	NA	NA	NA	NA	NA
	Actinide solution	0.0	0.0	0.0	NA	NA
	Plutonium solution	0.264	0.251	0.272	NA	NA
2	Actinide pellets: $\rho = 3.79 \text{ gm/cm}^3$	NA	NA	NA	NA	NA
	Actinide pellets: Optimum moderation	NA	NA	NA	NA	NA
	Actinide solution	0.0	0.0	0.0	NA	NA
	Plutonium solution	0.295	0.283	0.307	NA	NA
3	Actinide solution	0.0	0.0	0.0	NA	NA
	Plutonium solution	0.187	0.178	0.191	NA	NA
Reference	Actinide solution	0.012	0.012	0.016	NA	NA
	Curium solution	0.144	0.135	0.064	~0.0	~0.0
	Uranium solution	~0.0	~0.0	~0.0	0.034	0.034

(a) Data not available.

It should be noted that, relative to the other cases, the Reference Case has less potential for causing a criticality during long-term isolation. This is because:

- (1) It contains less total fissile material capable of forming a critical mass (and only trace amounts of plutonium), and
- (2) The time needed for the fissile material to deplete to a mass lower than that which could go critical is shorter in the Reference Case.

No great significance has been attached to this, however, for two reasons: first, the high degree of the practical difficulty in achieving a criticality with any of the waste forms, and second because it is essentially impossible to define the degree of Reference Case benefit achieved over the long time periods involved in comparison to the extreme improbability of naturally occurring events establishing the conditions needed for criticality.

For the waste to form a critical mass, preferential leaching, transport, and deposition and re-concentration in an appropriate volume must occur. Note that in all cases a neutron moderator is required to achieve a critical system. In the Reference Case, nuclides that fission in a fast neutron spectrum (if aggregated in a nearly pure configuration) are present in more than trace amounts. Forming a critical system with such nuclides would require so large an agglomeration of waste and the subsequent aggregation and isolation of these exclusive isotopes that the likelihood is extremely improbable.

6.2.1.2 Conditions Needed to Achieve Criticality in a Repository. The ability to make a mass of fissile material achieve and sustain a critical chain reaction depends upon a number of complex factors, including (1) the particular fissile isotope involved, (2) its geometric configuration, (3) the presence of neutron reflector, and (4) the absence of neutron poisons. The reaction is also highly spectrum dependent. Those isotopes that require a fast spectrum to fission act as a neutron poison in a system that fissions with a thermal spectrum. All of the parameters affecting criticality are highly interdependent (i.e., the range of values that can be assigned to each parameter is extremely restricted once any other parameter is set).

Redistributing the fissile material into a more optimum configuration is the principal requisite in achieving a criticality. The only mechanisms that could significantly redistribute the waste are the intrusion of groundwater and severe natural phenomena such as earthquakes. (In all cases, while these events *could* cause the required optimum redistribution of material, they might also occur without producing such conditions.) Water could enter the repository through unsealed boreholes, mine shafts, or fissures in the geological strata. Unless the water entered very near a waste container, or was flowing at a high rate, it would rapidly become a saturated brine solution because of the solubility of salt.

To achieve criticality in the repository, several uncontrolled natural events must occur in a proper sequence and within a specific period of time. These events, which establish the exact set of conditions needed for a criticality to occur, are:

- (1) Water must enter the repository in the required volume.
- (2) The waste canister and/or the cladding must fail.
- (3) The matrix in which the fissile isotopes are held must be physically moved or destroyed, or selected fissile isotopes leached out.
- (4) The fissile isotopes must be transported into a more optimum geometry and reconfigured into a critical mass. This also requires that the transport mechanism must effectively remove large amounts of material that would otherwise act as a neutron poison in the fissile system.
- (5) Sufficient conditions of moderation and reflection must exist to yield the neutron spectrum required for the fissile mass to achieve criticality.

It is impossible to quantitatively calculate the probability of these events occurring over millions of years in the future. It is reasonable to assume that during so great a period of time it may be possible for some of them to occur. However, it must be remembered that the same forces that may be acting to create the conditions needed for criticality have an even greater potential for making such an event less plausible. Therefore, the probability of achieving all of the conditions needed to reach criticality with uncontrolled events is judged to be diminishingly small. Similar conclusions were reached by Brookins in a recent study of geochemical constraints to actinide accumulation.⁽²⁾

6.2.2 Magnitude of a Criticality

This subsection presents a reasoned argument to show that a criticality incident in a repository would not produce significant results even if the improbable chain of events previously described caused such an occurrence. The results appear insignificant because of the extremely slow rate at which criticality would be approached, and because of the self-limiting nature of the event.

6.2.2.1 Limitations on Criticality. To achieve a criticality a flow of water must be introduced into the repository. This water would be required to move the fissile nuclides, and would be needed to moderate and reflect the neutrons. However, for water to flow, a pressure gradient is required. If the gradient is eliminated by increases in local pressure, the flow will stop or even reverse direction. A system approaching criticality will produce energy at an increasing rate. This causes both the temperature and related pressure of the critical system to increase. When the ensuing pressure exceeds the hydrostatic gradient of the local formations, the influx of water will cease and the reaction will shut down due to a lack of moderator and/or reflector. The power level produced is that which maintains the limiting temperature and pressure conditions. The temperature will be equivalent to the temperature of the groundwater system at the equilibrium pressure which still permits inflow of water to the critical mass. Since the mass flow rate characteristic of groundwater is so low relative to the mean lifetime of a fissile neutron (10^{-7} gm/cm²-s vs. 10^{-3} s), the critical system is maintained in essentially a steady state condition, and an explosive reaction associated with the rapid accumulation of fissile material and moderator is not possible.

6.2.2.2 Temperature Distribution and Power Level. A rough indication of the temperature distribution around a critical mass can be evaluated by equating the change in heat conduction with distance from the energy source to the amount of energy being removed at that distance. The following equation was developed as one method of showing this relationship:

$$k \nabla^2 T(r) = (T(r) - T_o) C_m W / \ell$$

where:

k = thermal conductivity of salt, $\frac{1.24 \text{ cal}}{\text{cm-s-}^\circ\text{C}}$

$T(r)$ = temperature at any radius from the center of the critical mass, ($^\circ\text{C}$)

T_o = the ambient temperature of the salt, ($^\circ\text{C}$)

C_m = specific heat of water, $\left(\frac{1 \text{ cal}}{\text{gm-}^\circ\text{C}}\right)$ $\frac{.15 \times 10^{-6} \text{ gm}}{\text{cm}^2\text{-s}}$

W = mass flow rate of water through a unit cross sectional area,

ℓ = length of the heat transfer surface, (assumed to be 3 meters).

The solution of this equation indicates that the temperature perturbation is reduced by a factor of 10 at a distance of about 30 meters. Hence there should be no direct effect on the population

from a criticality in a deep geological repository. The localized nature of the perturbation also suggests that effects such as thermal expansion of the medium should not be extensive.

The above equation was also used to estimate the power level of the critical system. The power output was calculated assuming a single fuel assembly in a quasi-steady state critical condition. The fissile mass is assumed to be redistributed into a cylinder 3 meters in diameter by 3 meters long. Assuming the repository is at atmospheric pressure the power level is calculated to stabilize at about 1.5 kW. The critical system was assumed to be at a hydrostatic equilibrium with the surrounding geological formations (100 bars). Based on these assumptions, a power level of 4 kW was calculated. While the accuracy of such a calculated value and the validity of the assumptions used may be questioned, the analysis does suggest that a critical reaction in a repository (if it did occur) would be of an extremely low order of magnitude.

An interesting characteristic of a system where the temperature at the periphery of the critical mass is constant is that the power output is independent of the volume. This is true because production of power is controlled by the rate at which the salt media can transfer heat away from the critical mass. (If the temperature of the critical system becomes too great, the water will be driven away and the reaction will shut down.) This means that although the analysis was performed for a critical event involving a single fuel assembly, it is equally applicable to an event involving several waste canisters. The total power output from any given volume of critical material is constant.

6.2.3 Potential Radiological Impact

6.2.3.1 Fission Product Buildup. As indicated above, the direct effects of a critical event are localized and will remain within the repository. Harm to future populations can only come from the eventual release of radionuclides from such a reaction to the biosphere. The mechanisms causing such a release are not considered. However, the risk to future populations is addressed by comparing the radioactive nuclide inventory caused by a criticality with radionuclides originally placed in the repository as spent fuel or waste.

There are two groups of biologically significant nuclides that would be generated by a critical system: fission products and actinides. Assuming a fissile mass stays critical indefinitely, the fission product inventory will eventually stabilize. This is due to the relatively short half-life of fission product nuclides. The actinides, on the other hand, will continue to increase (but at an increasingly slower rate) because of the very long life of these transuranics.

The increase in fission products can be described by the following expression:

$$\frac{P}{\lambda} (1 - e^{-\lambda t})$$

where:

P is the power level

λ is the decay constant

t is the time since the system went critical

A half-life of 30 years reasonably characterizes the fission products. With this assumption, a four kilowatt critical source operating indefinitely approaches a fission product inventory associated with an integrated power of about 40 MWD. This is less than 0.4 percent of the original fission product inventory associated with only one spent fuel assembly. Such an increase in the total repository inventory is trivial and will not add measurably to the risk to future populations.

6.2.3.2 Actinide Buildup. The buildup of actinide elements is not proportional to the power, but to the rate of non-fission neutron absorption in heavy metals. These metals would act as neutron poisons in a fissile system. Among the prerequisites for achieving criticality in the waste is the

redistribution of fissile material without the presence of neutron poisons by some undefined (but highly improbable) natural event. The high improbability of this happening plus the high neutron leakage that would occur from the ensuing concentrated critical mass suggest that the production rate of actinides would be low at any power level. However, ignoring the above facts, it is still evident that more than 100,000 years would be required for a four kilowatt critical source to generate the original actinide inventory associated with a single spent fuel assembly.

A nonrigorous comparison of a critical event in a repository can be made with the naturally occurring criticality that happened about two billion years ago at Oklo, Gabon, Africa.⁽³⁾ The similarities between the two systems are probably as close as any that could be produced by either analytical or experimental simulation. At Oklo, naturally occurring uranium, which at that time was about 3 percent U-235, produced a critical system when it came into contact with water. The Oklo System operated for 10^5 to 10^6 years at a power of 10 to 100 kilowatts throughout several hundred cubic meters of ore. Measurements at the site have shown that the actinides produced during this reaction have not migrated far from the site of production.

It is conservative to speculate that a criticality in a repository producing 4 kilowatts would not last longer than 10^6 years. In this time the actinide inventory of the repository (ignoring radioactive decay) would be increased by an amount equivalent to about 10 spent fuel assemblies. Since a repository can be expected to contain an inventory far in excess of 10 fuel assemblies, this increase is relatively small. In addition, the measurements at Oklo indicate that any actinides so produced are not susceptible to migration to the biosphere. Therefore, the increase in risk to future generations due to an increase in the actinide inventory is judged to be negligible.

REFERENCES FOR SECTION 6

- (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) D. G. Brookins, "Geochemical Constraints on Accumulation of Actinide Critical Masses from Stored Nuclear Waste in Natural Rock Repositories," ONWI-17, December 1978.
- (3) G.A. Cowan, "A Natural Fission Reactor," *Scientific American*, 235, 36 (1976)

Section 7

ASSESSMENT FOR RADIOLOGICAL IMPACT

TABLE OF CONTENTS

Section		Page
7.1	Assessment Basis and Approach	7-5
7.1.1	Definition of Radiological Impact	7-5
7.1.2	Scope	7-5
7.1.3	Approach	7-5
7.2	Near-Term Assessment — Facilities	7-7
7.2.1	AFR	7-7
7.2.2	P/E Facility	7-7
7.2.3	Transportation System	7-12
7.2.4	Repository	7-15
7.3	Long-Term Assessment — Repository	7-19
7.3.1	General Discussion	7-19
7.3.2	Liquid Breach and Transport of Radionuclides	7-21
7.3.3	Direct Access Drilling	7-21
7.3.4	Catastrophic Breaching Events	7-22
	References for Section 7	7-24

Section 7

ASSESSMENT FOR RADIOLOGICAL IMPACT

The radiation hazard associated with any nuclear operation is always of primary concern. Assessment of radiological impact is, therefore, an important consideration for case comparison in the SFDS. This section presents the aspects of radiological impact as they affect the processing, encapsulation, transportation, storage, and disposal of spent fuel and various types of nuclear waste.

In this assessment, consideration is given to the radiological effects induced in both the near-term and long-term. The near-term assessment considers radiological impacts of each waste processing facility and the transportation steps between facilities, during normal operations and during abnormal events. In the long-term assessment, the radiological consequences that would follow a repository breach at some time in the future are analyzed, although no attempt is made to predict the probability that such a breach will occur.

The data presented in this section are adaptations of pertinent information generated by others that are judged to be appropriate and relevant for the study. The resultant radiation doses developed from these data are summarized in Tables 7-1 and 7-2. It should be noted that the source data used were based on a variety of assumptions made by their authors in analyzing particular subject areas. Adapting such data to this study involved the broad use of engineering reasoning and judgment. The specific doses presented in Tables 7-1 and 7-2, and elsewhere in this section should therefore be recognized as indicative rather than absolute. However, the relative ranking of the study cases in any area or operation assessed, is judged to be reasonably valid. A number of technical considerations for the assessment of radiological impact and a list of technical references are given in Appendix H of Volume 3 as support for this assessment.

Table 7-1 indicates that the radiological impact on occupational workers during normal process operations is greatest for Case 3 followed by the Reference Case, Case 2, and Case 1 respectively. Case 3 dominates because of the number of P/E facility processing steps required and the large amount of secondary waste generated, which must be stored at the repository. The Reference Case follows in impact due to the high doses involved in completely reprocessing the spent fuel. Cases 2 and 1 follow in the order expected because of the succeeding fewer number of process steps required and the subsequent generation of less secondary waste.

The radiation impact on occupational workers due to an abnormal event or accident is greatest for Case 3 (Table 7-1). This is due to the potential release of the high concentration of radioactive nuclides present in this waste form during processing and repository operations. The impact from the other three cases is the same within the accuracy of this study.

The spent fuel cases dominate the radiological impact on non-occupational workers due to normal process operations (Table 7-1). The reason for this is the operation of nine AFR facilities which are required in each of the spent fuel cases. Within the accuracy of the study, Cases 1 and 2 are equivalent in impact while Case 3 is only slightly higher. Note that for all cases the non-occupational dose from the P/E facilities is approximately equivalent to that from transporting the waste, while the dose from the repository is insignificant.

For abnormal events or accidents the potential non-occupational dose is completely dominated by a transportation accident (Table 7-1). The magnitude of the potential dose is judged to be the same for each case.

Table 7-1
SUMMARY OF NEAR-TERM RADIOLOGICAL IMPACT DATA
FOR A 5000 MTHM/YR SYSTEM^(a)

	WHOLE BODY DOSE ^(b)			
	Occupational		Non-occupational	
	Normal (man-rem/yr)	Abnormal (man-rem/yr)	Normal (man-rem/yr)	Abnormal (mrem/event)
Reference Case				
P/E facilities	2,814	426	89	0.27
Transportation system	216	(c)	87	1,100
Repository	902	454	1.5 E-5 ^(d)	65
Case 1				
AFRs	396	132	325	1.9 E-3
P/E facilities	720	125	20	0.27
Transportation system	80	(c)	49	1,100
Repository	1,295	569	5.0 E-6	56
Case 2				
AFRs	396	132	325	1.9 E-3
P/E facilities	1,341	202	41	0.27
Transportation system	97	(c)	52	1,100
Repository	1,188	532	7.7 E-6	56
Case 3				
AFRs	396	132	325	1.9 E-3
P/E facilities	2,328	332	62	0.27
Transportation system	124	(c)	80	1,100
Repository	2,301	978	1.1 E-5	56
Totals				
Reference Case	3,932	880	176	1,165
Case 1	2,493	828	394	1,156
Case 2	3,022	866	418	1,156
Case 3	5,149	1,442	467	1,156

(a) Normal operations assume 9 AFRs, 3 P/E facilities, 1 repository and the interconnecting transportation system. Abnormal events are evaluated based on only one facility or one transportation step.

(b) For comparison the normal background radiation dose rate is 100 mrem/yr or 1.6×10^5 man-rem/yr for a typical area 50 miles in radius in the U.S.A. (based on a U.S.A. population density of 200 people per square mile per WASH-1258⁽⁷⁾ Appendix I).

(c) Not assessed

(d) 1.5 E-5 means 1.5×10^{-5}

The long-term consequence of breaching the repository is presented in Table 7-2. The following conclusions can be drawn:

- The effects of groundwater intrusion depend on the long-lived fission products. Since the amount of fission products in the repository, and/or involved in the event, is nearly the same for all cases, the radiation impact from such an event is the same for all cases. The resultant dose from such an event is also far below background radiation

Table 7-2

LONG-TERM CONSEQUENCES OF REPOSITORY BREACHING

1. Groundwater Intrusion — Depending on the scenario, the whole-body radiation dose to the maximum exposed individual will vary from about 1.0 E-6 to $7.0 \text{ E-4}^{(a)}$ mrem/yr. This is several orders of magnitude below normal background of 100 mrem/yr.

		Radiation Dose			
		Ref. Case	Case 1	Case 2	Case 3
2. Direct Access Drilling —					
Dose to maximum exposed drilling crew member (rem):					
100 years		340	90	100	44
250 years		13	4	4	2
1000 years		5	1	2	1
Whole body dose to maximum exposed individual living downwind of the drilling mud pond (rem):					
100 years		1.6 E-1	4.3 E-2	4.8 E-2	2.4 E-2
250 years		5.7 E-3	1.5 E-3	1.7 E-3	7.4 E-4
1000 years		6.7 E-3	4.6 E-4	5.1 E-4	2.2 E-4
3. Catastrophic event exposing the waste:		At 1000 Years		At 2000 Years	
		Ref. Case	Cases 1, 2&3	Ref. Case	Cases 1, 2&3
Within a 5-Km radius area (mrem/yr)		1.12 E+6	9.85 E+7	6.74 E+5	5.93 E+7
To the nonspecific population (man-rem/yr)		6.51 E+5	6.80 E+7	1.32 E+4	1.38 E+6

(a) 7.0 E-4 means 7.0×10^{-4}

- Direct access drilling within 100 years yields a high dose to the drilling crew (44 rem to 340 rem depending on the case). The maximum exposed individual, living near the drill site, receives a dose approximately equal to background radiation. If drilling is carried out at about 250 years, the dose to the drill crew is about three times that currently permitted per year to an occupational worker. For time periods up to 1,000 years the dose from the Reference Case dominates due to the high concentration of fission products in one canister. The other cases have approximately the same impact, with Case 3 being slightly less at all times. Case 3 is less because of the lower quantity of radioactive waste originally placed in a single container.
- A catastrophic breach of the repository (caused by a volcano or large meteorite impact) releases enough radiation to make the site uninhabitable if the event occurs within the first few thousand years. The dose to a non-specific population is about four times background if the event occurs at 1000 years. The spent fuel cases have much higher radiological impacts from such an event (by about two orders of magnitude) due to the fact that they contain large amounts of uranium and plutonium which have been removed from Reference Case waste

In differentiating between study cases, it should be recognized that normal background radiation is about 1.6×10^5 man-rem/yr for a typical area 50 miles in radius in the U S A (based on a U S A population density of 200 people per square mile per WASH-1258 Appendix I⁽⁷⁾) In comparison the levels of radiation identified in this study are extremely small During normal waste processing and storage operations, the release of radiation is negligible because of design features and licensing constraints When assessing the possible radiation effects due to an accident, this study considers the Design Basis Accident applicable to the particular facility being considered The release from such an event is also relatively small Such severe accidents are postulated normally to establish equipment design classifications and the likelihood of their occurrence is judged almost impossible In performing this analysis, however, the a priori assumption is made that such accidents will indeed occur The same conservatism also applies to the long-term radiological risk assessment where the consequence analysis assumes that some very low probability mechanism has breached the repository

7.1 ASSESSMENT BASIS AND APPROACH

7.1.1 Definition of Radiological Impact

For purposes of this study, radiological impact is defined as the potential radiation dose which would be received by an individual or a population segment as a result all processes and operations needed to entomb the waste and the long-term effects of isolating the waste in a deep salt repository. The radiological assessment is a relative measure of its potential radiological impact.

7.1.2 Scope

The scope of the assessment includes three general areas; near-term occupational, near-term non-occupational, and long-term impacts. These are treated as separate assessments within this section and no attempt is made to combine their results. In the near-term, radiological impacts are estimated for:

- Occupational personnel due to normal operations and abnormal events or accidents that could occur during these operations, and
- The non-occupational population as a result of normal operations and abnormal events or accidents

In the long term, the consequences to the general population of breaching the repository are assessed as a result of:

- Water intrusion, with subsequent transport of radionuclides along an aquifer to the biosphere
- Direct access drilling while exploring for minerals, and
- Catastrophic events that would expose the entire repository contents to the biosphere

7.1.3 Approach

The potential radiation dose that any person or population segment can receive is a complex function of several important and diverse parameters. The evaluation must cover numerous operations throughout several facilities and transportation steps in the system, as well as long-term isolation in a repository. The object in performing this assessment is to evaluate these factors in order to differentiate between the study cases. In performing the assessment, the significant parameters that affect radiation impact were studied in detail. These include, but are not limited to:

- The inventory of nuclides contained in the waste that are biologically hazardous (i.e., source terms)
- The nature of process operations where occupational personnel normally receive some amount of direct exposure to radiation and where some radioactive effluents are normally released to the environment
- Operations such as transporting radioactive materials and decommissioning facilities where there is a reasonable potential for an accidental release of radioactivity to the environment
- The susceptibility of radioactive particles of respirable size (< 10 microns) to entrainment and transport by air
- The quantity of waste containers handled or shipped, and
- The size of the population affected

After considering these parameters, several were found to apply equally to each case. In presenting the assessment, only those parameters that provide a significant basis for differentiating between the cases are discussed. This method was chosen to avoid the detailed and potentially confusing presentation of complicated data that are not necessary in making case comparisons.

The assessment methodology utilizes radiological impact data for the waste management facilities applicable to each of the basic assessment areas shown in Tables 7-1 and 7-2. The discussion regarding each facility is presented to facilitate case differentiation. The near-term assessment is made first, in Subsection 7.2, with the facilities covered in the following order:

- AFR
- P/E facility
- Transportation system
- Repository

The long-term assessment of the repository is presented in Subsection 7-3.

Data on the absolute radiological impact are generally available in the literature for the Reference Case waste but are not always available for the spent fuel cases. The spent fuel cases can, however, be approximated by appropriately proportioning Reference Case data. In the Reference Case, approximately 99.5% of the uranium and plutonium is removed from the spent fuel during reprocessing and is reused as reactor fuel. For the spent fuel cases, therefore, the inventories of uranium and plutonium and their decay products are assumed to be 200 times those of the Reference Case.

Note that the assessment does not include a breach of safeguards at any of the facilities or systems investigated, or the radiological impact induced by a criticality event. These are considered separately in Sections 5 and 6 of this report.

7.2 NEAR-TERM ASSESSMENT — FACILITIES

7.2.1 AFR

AFRs accommodate the spent fuel in the SFDS for a period of nine years. During this time, the rates of radioactive decay and thermal heat generation subside, making the fuel assemblies more acceptable for disposal in a repository. The SFDS assumes that nine AFRs, each with a capacity of 5000 metric tons of heavy metal, are operating during equilibrium fuel cycle conditions.

Tables 7-3 and 7-4 present the AFR radiation dose data developed from a DOE spent fuel storage study.⁽¹⁾ The dominant consideration in differentiating between the cases is simply whether or not an AFR is required by the case definition. Cases 1, 2, and 3 each require the use of AFRs. The radiological impacts of the AFR are identical for all three cases and, therefore, do not differentiate between the three. The Reference Case does not require an AFR because spent fuel is transferred directly from the reactor to the P/E facility. (Note, however, that the high-level waste generated by processing operations in the Reference Case is held at the P/E facility for the same nine-year period although it is then in an encapsulated form.)

7.2.1.1 Occupational Impacts. The radiological impact to workers from an AFR is less than the impact from other more active nuclear facilities during normal operation. This is due to the relatively benign nature of operations at an AFR. The impact on workers due to an accident at an AFR is also less than at other process facilities. Accidents at AFRs are usually the result of mishandling a fuel assembly. The benign nature of such accidents limits the number of assemblies involved and hence, the amount of radioactivity released.

7.2.1.2 Non-Occupational Impacts. The non-occupational radiological impacts from the normal operation of an AFR come predominantly from the ground deposition of airborne particulates and the release of fission product gases (such as ^{85}Kr , ^3H , and ^{129}I) along with the activation product ^{14}C in the form of CO_2 . Estimates of the dose accumulated by the yearly releases from an AFR are shown in Table 7-3. Noting that the level of average background radiation within a 50 mile radius is about 1.6×10^5 man-rem/yr, Table 7-3 shows that the radiological effect of an AFR is extremely small. The cumulative impact of nine AFRs is still very low with respect to other facilities required for processing nuclear wastes.

As shown in Table 7-4, the impact due to a potentially severe accident (Design Basis Tornado in this case) is quite small compared to other process steps. This, again, is due to the nature of the potential accidents that can occur. The release of basin water with subsequent consumption by the public was not considered a credible accident scenario in this analysis.

7.2.2 P/E Facility

The P/E facilities are designed:

- For the Reference Case, to provide complete processing of spent fuel including separation and recycle of uranium and plutonium, removal of fission gases, volume reduction and vitrification of high-level waste
- For Case 1, to encapsulate and overpack spent fuel assemblies and prepare them for disposal at the geologic repository
- For Case 2, to remove end fittings, chop and voloxidize the fuel assemblies, remove tritium and a high percentage of the fission gases, and encapsulate the chopped fuel in preparation for geologic disposal, and

Table 7-3
ANNUAL RADIATION DOSE TO OCCUPATIONAL
AND NON-OCCUPATIONAL PERSONNEL DURING
NORMAL OPERATIONS OF AFRs^(a)

	Radiation Dose (man-rem/yr)	
	Nine AFRs	One AFR
Occupational		
Workers at the AFR	396	44
Non-Occupational		
AFR normal releases integrated over 50-mile radius area	316 ^(b)	35
Releases during cask unloading	9 ^(c)	1
Total annual non-occupational dose within 50-mile radius	325	36

(a) Based on nine government-owned AFRs that store 5000 MTHM of spent fuel, each, for nine years. Data developed from DOE/EIS-0015-D⁽¹⁾

(b) Developed by proportioning world dose to local dose (50-mile radius)

(c) Average per year includes the accumulated effect of hazardous nuclides for 100 years

Table 7-4
RADIATION DOSE TO OCCUPATIONAL AND
NON-OCCUPATIONAL PERSONNEL DUE TO ACCIDENTS
AND ABNORMAL EVENTS AT AN AFR

Occupational ^(a)	132 man-rem/yr
Non-Occupational ^(b)	
Maximum Dose — Body	0.0019 mrem/accident
— Bone	0.0090 mrem/accident
Maximum Dose Risk — Body	1.9×10^{-8} mrem/yr
— Bone	9.0×10^{-8} mrem/yr

(a) Based on abnormal handling event

(b) Based on the Design Basis Tornado for most severe public exposure
Data source: Developed from DOE/EIS-0015-D⁽¹⁾

- For Case 3, to provide complete processing as in the Reference Case except that uranium and plutonium are not separated but are included in the vitrified waste

For this study, a somewhat simplifying assumption is made that the magnitude of radiological releases is directly proportional to the number of operational steps needed to process the waste for each case. This is assumed for both normal operations and for postulated abnormal or accidental events. It is also assumed that the number of personnel needed for a specific process is the same for all cases. Case differentiation is simplified because the number of process steps varies greatly between the study cases. Thus, the total occupational man-rem dose increases with the number of process operations in each case. Note, however, that the study does not attempt to differentiate between the relative potential dangers associated with each particular process step.

7.2.2.1 Occupational Impacts. The radiological impacts from the P/E facility on occupational personnel, presented in Table 7-5, are determined by estimates of the man-rem dose calculated for each operation. This table is based on data from the Exxon Nuclear Fuel Recovery and Recycling Center PSAR.⁽²⁾ The parameters involved in the calculations that allow differentiation between the cases are:

- The number of operations needed to process the waste
- The number of personnel per operation, and
- The occupancy frequency within different radiation zones within the facility during process operations

Because the Reference Case requires complete chemical processing and production of PuO_2 and UF_6 , the overall impact of the Reference Case P/E facility is the largest of all cases, with Case 3 nearly as high. Cases 1 and 2 are significantly less due to fewer process steps.

7.2.2.2 Non-Occupational Impacts. The differences in non-occupational impacts from the P/E facility for each case can be assessed by considering:

- The magnitude of the potential offsite release from each process step, and
- The degree to which the waste is dispersed once it leaves the facility

Table 7-6 indicates the doses calculated from Barnwell Nuclear Fuel Plant data.⁽³⁾ The calculation assumes the magnitude of the release is proportional to the number of process steps required by each case. Since the purpose of the study is to differentiate between cases, the degree of dispersion is considered to be the same for all cases.

Non-occupational doses due to postulated abnormal events have an extremely low probability of occurrence. The Barnwell SAR⁽³⁾ presents data for this type of exposure on the basis of the type of accident. The resultant doses, applied to the SFDS cases, are presented in Tables 7-7 and 7-8. The data indicate a much greater potential accident dose for the Reference Case, mainly because of the number of process steps involved. This is followed in decreasing order by Cases 3, 2 and 1, as would be expected.

Table 7-5
OCCUPATIONAL DOSE DURING NORMAL AND ABNORMAL OPERATION OF
A SINGLE P/E FACILITY^(a)

Process Units	Radiation Dose (man-rem/yr)							
	Reference Case		Case 1		Case 2		Case 3	
	Normal Operation	Abnormal Operation	Normal Operation	Abnormal Operation	Normal Operation	Abnormal Operation	Normal Operation	Abnormal Operation
Spent Fuel Receiving	10	2	10	2	10	2	10	2
Spent Fuel Handling and Storage	16	20	—	—	—	—	—	—
Transfer and Shipment of Processed Waste (See Transportation Steps)	36	40	36	40	36	40	36	40
Offgas Recovery and Storage and Fission Gas Release	55	55	—	—	55	55	55	55
Decommissioning (prorated over operating years)	13	12	1	1	7	6	13	12
Encapsulation Unit Maintenance	12	21	12	21	12	21	12	21
Process Unit Maintenance	36	18	—	—	—	—	36	18
Failed Fuel Handling	54	40	54	40	54	40	54	40
Mechanical Processing	48	6	—	—	48	6	48	6
Hulls Volume Reduction	69	6	—	—	69	6	69	6
TRU Volume Reduction	90	15	90	15	90	15	90	15
Chemical Processing	120	63	—	—	—	—	120	63
Uranium Separation — Conversion	115	42	—	—	—	—	—	—
Plutonium Separation — Conversion	104	50	—	—	—	—	—	—
Waste Solidification (Calcination, Vitrification)	122	36	—	—	—	—	122	36
Support/Overhead	38	—	38	7	66	11	111	18
Totals (man-rem/yr)	938	426	241	126	447	202	776	332
Total Facility Personnel	618	297	189	75	318	139	572	260
Average Dose (rem/yr)	1.5	1.4	1.3	1.7	1.4	1.5	1.4	1.3

(a) Data source: Developed from Exxon NFRRC - PSAR⁽²⁾

Table 7-6
NON-OCCUPATIONAL DOSE DURING
NORMAL OPERATIONS AT A SINGLE P/E FACILITY^(a)

	Ref. Case	Dose (man-rem/yr)		Case 3
		Case 1	Case 2	
Whole Body	29.8	6.7	13.6	20.5
Bone	47.3	10.6	21.6	32.6
Thyroid	134.0	30.2	61.2	92.3
Skin	329.0	74.0	150.2	226.7

(a) Integrated over a 50-mile radius area.
Data source: AGNS-BNFP SAR⁽³⁾

Table 7-7
NON-OCCUPATIONAL DOSE DUE TO ACCIDENTS
AT A P/E FACILITY

Operation or Location	Accident Scenario	Affected Organ	Maximum Off-Site Dose (mrem) ^(a)			
			Ref. Case	Case 1	Case 2	Case 3
Spent fuel receiving	Fuel handling accident ^(b)	Thyroid	31.6	31.6	31.6	31.6
Offgas recovery and storage	Breach of offgas confinement	Thyroid	0.29	—	0.29	0.29
Chemical processing and waste solidification	Loss of decontamination recovery and waste treatment	Thyroid	308	—	—	308
"	Explosion in LAW concentrator	Thyroid	0.75	—	—	0.75
"	Explosion in HAW concentrator	Bone	0.196	—	—	—
U and Pu separation	Solvent fire in Pu extractor	Bone	0.335	—	—	—
"	Solvent fire in co-decontamination cycle	Lung	111	—	—	—
"	Explosion in Pu concentrator	Bone	677	—	—	—

(a) Integrated over a 50-mile radius area

Data source: Developed from AGNS-BNFP SAR⁽³⁾

(b) This accident produces the worst whole body dose, 0.271 mrem. (See Table 7-8)

Table 7-8

**SUMMARY OF POTENTIAL OFFSITE DOSES BY
POSTULATED ACCIDENTS AT THE BARNWELL NUCLEAR FUEL PLANT^(a)**

Cases to Which Accident is Applicable	Accident Scenario	Maximum Off-Site Dose (mrem) at Site Boundary ^(b)						
		Whole Body	Thyroid	Bone	Kidney	Liver	G.I. Tract	Lung
Ref.	Explosion in the HAW Concentrator	1.34 E-2	5.88 E-2	1.96 E-1	4.04 E-2	8.45 E-2	4.40 E-3	9.17 E-2
Ref.	Solvent Fire in Pu Extraction Cycle	7.58 E-3	0.0	3.35 E-1	3.41 E-2	3.21 E-2	1.84 E-6	5.28 E-3 ^(c)
Ref.	Explosion in Pu Product Concentrator	1.53 E-1	0.0	<u>6.77 E+0</u>	<u>6.90 E-1</u>	<u>6.53 E-1</u>	3.71 E-5	1.08 E-1
Ref.	Solvent Fire in Co-Decontamination Cycle	7.58 E-3	1.45 E-4	2.12 E-1	6.43 E-2	1.80 E-2	8.49 E-2	<u>1.11 E+0</u>
Ref., 3	Explosion in the LAW Concentrator	3.00 E-4	7.46 E-2	4.03 E-3	1.06 E-3	1.21 E-3	1.00 E-4	1.92 E-2
Ref., 3	Loss of Decontamination in Acid Recovery and Waste Treatment	4.48 E-3	3.08 E-5	1.58 E-1	1.85 E-2	2.67 E-2	2.14 E-4	8.69 E-3
Ref., 1,3	Breach of Process Offgas Confinement	5.09 E-4	2.86 E-1	8.19 E-4	1.43 E-3	8.47 E-4	1.65 E-4	3.48 E-2
Ref., 1,2,3	Fuel Handling Accident	<u>2.71 E-1</u>	<u>3.16 E+1</u>	3.05 E-1	3.72 E-1	3.08 E-1	<u>2.33 E-1</u>	6.37 E-1

(a) Taken from AGNS-BNFP SAR⁽³⁾

(b) Underlined values indicate accident releases resulting in the highest consequences.

(c) 5.28 E-3 means 5.28×10^{-3}

7.2.3 Transportation System

The transportation of radioactive material associated with each study case includes the movement of the following:

- 1) Spent fuel assemblies (after one year of storage) are transferred from the reactor to an AFR for Cases 1-3, or directly to a P/E facility for the Reference Case.
- 2) Spent fuel assemblies (10 years out of the reactor) are moved from an AFR to a P/E facility for Cases 1-3.
- 3) Spent fuel assemblies, either intact, as chopped assemblies, or in a vitrified matrix, are moved from the P/E facility to the repository for Cases 1-3. Encapsulated high-level reprocessed waste is transported from the P/E facility to the repository for the Reference Case.

- 4) Low and intermediate level wastes generated in various process operations are transported from the P/E facility to shallow land burial and/or the repository for all cases considered in the study. The quantity of material generated is least in Case 1 and increases in Cases 2, 3, and the Reference Case, respectively. The Reference Case, which involves full fuel reprocessing, generates by far the largest amount of radioactive waste.

In assessing the radiological impact, the quantity and type of waste transported, type of transporter used and the distance traveled allow for differentiating between the cases. The assessment includes the resultant doses for both occupational and non-occupational personnel.

7.2.3.1 Occupational Impacts. The vehicles used for transport include both truck and railcar. The occupational dose is completely dominated by the truck transport. This is because the driver receives a constant dose rate while enroute due to the close proximity to the waste. This is not true for personnel operating a train. (For normal operations, this assessment assumes the occupational dose induced by rail transport to be negligible.) The total dose received by a truck driver depends on the time enroute and hence, on the distance traveled. (Note that in this study, the exposure obtained during all loading and unloading operations is considered part of the facility impact.)

Table 7-9 contains the occupational dose due to normal transportation operations. This was developed from information in the WIPP EIS.⁽⁴⁾ This dose is greatest in the Reference Case because:

- 1) The P/E facility is assumed to be 300 miles further than the AFR. Hence, shipping waste directly to the P/E facility (as would occur in the Reference Case option) would contribute a greater dose to the truck drivers.
- 2) All spent fuel assemblies shipped from the AFR to the P/E facility are assumed to come by rail (i.e., there is no dose commitment attached to such shipments).
- 3) The amount of low and intermediate level waste shipped by truck from the P/E facility to shallow land burial and/or the repository is greatest in the Reference Case by a factor of at least four.

The other cases, ranked in decreasing order of radiological impact, are Case 3, Case 2, and Case 1. All of the spent fuel cases have approximately the same impact within the accuracy of this study.

To breach a transportation canister would require so severe an accident that it would in all probability kill the occupational personnel of concern. Therefore, no radiological assessment was performed for this category. Personnel that survive are assumed to be located some distance from the source of radiation release. These are assessed as part of the non-occupational population.

7.2.3.2 Non-Occupational Doses. Under normal operating conditions, the non-occupational impact of radiation due to transportation operations is small. Any such small doses are obtained by people walking near vehicles containing radioactive waste, etc. The impact breakdown by case is the same as that for occupational personnel with the Reference Case the worst followed by Case 3, Case 2, and Case 1, respectively. Table 7-10 estimates the magnitude of these doses. These data were also developed from the WIPP EIS.

For abnormal events (accidents) the radiological impact on non-occupational personnel will depend on the following:

- The total mileage that is traveled. (The probability of an accident is assumed to be proportional to this value), and
- The quantity and form of the waste being transported, which will specify the nuclide content and susceptibility to dispersion from the scene of the accident

Table 7-9
OCCUPATIONAL DOSE DURING NORMAL TRANSPORTATION
OF WASTES BETWEEN FACILITIES^(a)

	Assumed Distance (miles)	Ref. Case	Dose (man-rem/yr)		
			Case 1	Case 2	Case 3
Reactor to AFR	400	—	38.6	38.6	38.6
Reactor to P/E Facility	700	67.3	—	—	—
AFR to P/E Facility	700	—	0.2	0.2	0.2
AFR to Shallow Land Burial	400	—	4.6	4.6	4.6
P/E Facility to Repository	1,100	107.4	32.8	49.5	76.5
P/E Facility to Shallow Land Burial	700	20.5	3.8	3.8	3.8
P/E Facility to Recycle Facilities	400	<u>21.1</u>	<u>—</u>	<u>—</u>	<u>—</u>
Total		216.3	80.0	96.7	123.7

(a) Data source: Developed from WIPP EIS⁽⁴⁾

Table 7-10
NON-OCCUPATIONAL DOSE DURING NORMAL
TRANSPORTATION OF WASTES BETWEEN FACILITIES^(a)

	Assumed Distance (miles)	Ref. Case	Dose (man-rem/yr)		
			Case 1	Case 2	Case 3
Reactor to AFR	400	—	18.6	18.6	18.6
Reactor to P/E Facility	700	32.2	—	—	—
AFR to P/E Facility	700	—	5.5	5.5	5.5
AFR to Shallow Land Burial	400	—	0.4	0.4	0.4
P/E Facility to Repository	1,100	46.3	24.0	26.3	55.1
P/E Facility to Shallow Land Burial	700	3.8	0.7	0.7	0.7
P/E Facility to Recycle Facilities	400	<u>4.4</u>	<u>—</u>	<u>—</u>	<u>—</u>
Total		86.7	49.2	51.5	80.3

(a) Data source: Developed from WIPP EIS⁽⁴⁾

As is shown in Table 7-11, spent fuel provides by far the worst non-occupational radiological impact. This is due to the total number of trips that must be made and the extent to which the fuel can be dispersed. Because all cases begin by transporting spent fuel from the reactor, they all tend to be approximately the same. The Reference Case and Case 3, however, are somewhat less susceptible to dispersion in the air following an accident. This is offset, though, by the need to transport significantly more radioactive waste by-products from the P/E facility than are created by Cases 1 and 2.

7.2.4 Repository

The repository is used for storage and long-term geologic isolation of nuclear wastes. During its operational life (i.e., as the repository is being filled), this facility has a low radiological impact on the public because of the benign nature of its operation. Various studies have indicated that the consequences of breaching the repository in the distant future are also small, unless a catastrophic natural event occurs within a few thousand years. This section presents the radiological impact on both occupational and non-occupational personnel in the near-term and for the non-occupational population in the long-term.

7.2.4.1 Occupational Impacts. The radiological impact to occupational workers due to repository operations is assessed by considering:

- The number of waste containers handled
- The number of people involved in the handling operations, and
- The length of time personnel must occupy higher radiation zones

Table 7-12 shows the estimated radiological impact of repository operations. These data were developed by comparing probable repository handling operations with handling and process operations at a modern reprocessing plant⁽²⁾ and estimating probable doses from the considerations listed above. For normal operations, Case 3 provides the largest quantity of spent fuel containers and low and intermediate level processing by-product containers placed in the repository. Hence, it has the highest relative radiological impact. For the conditions analyzed in SFDS, Cases 1, 2, and the Reference Case have only about half the impact of Case 3.

The occupational dose obtained as a result of abnormal occurrences is also given in Table 7-12. For this study, the dose is proportional to the number of waste containers handled, the form of the waste, and the hazard index of the nuclides present. Again, Case 3 yields the largest relative impact because it has the greatest number of containers (and also the greatest number of people). Cases 1, 2, and the Reference Case have only about half the impact of Case 3. The Reference Case has the least impact due to the low quantities of uranium and plutonium present and the complex form of the waste.

7.2.4.2 Non-Occupational Impact. The non-occupational radiological impact due to normal repository operations is dominated by the release of surface contaminants from contact-handled (CH), low-level waste, containers. The resultant doses to the public are given in Table 7-13. This table assumes that two percent of the containers have the maximum surface contamination allowed by shipping regulations. All of the surface contaminants from these containers are assumed to be released through HEPA filters into the air.

Table 7-11

**MAXIMUM DOSE TO A NON-OCCUPATIONAL
INDIVIDUAL DUE TO A SINGLE TRANSPORTATION ACCIDENT^(a)**

Scenario^(c)	Dose Commitment (rem)^(b)		
	Whole Body	Bone	Lung
CH TRU rail	0.012	0.49	0.025
CH TRU truck	0.014	0.590	0.029
RH TRU	0.00003	0.0003	0.000007
Spent Fuel	1.1	1.2	0.30

(a) Individual is assumed to be one-half mile from the accident.

(b) Taken from WIPP EIS⁽⁴⁾

(c) CH means contact handled (low level waste)

RH means remote handled (intermediate and high level waste)

TRU is transuranic.

Table 7-12
OCCUPATIONAL DOSE DURING NORMAL AND ABNORMAL OPERATIONS
AT A REPOSITORY^(a)

Process Units	Dose (man-rem/yr)							
	Reference Case		Case 1		Case 2		Case 3	
	Normal Operation	Abnormal Operation	Normal Operation	Abnormal Operation	Normal Operation	Abnormal Operation	Normal Operation	Abnormal Operation
Spent Fuel/Modified S.F. Waste Receiving	—	—	110	58	86	46	202	104
HLW Unit Transfer and Handling — Above Ground	132	92	—	—	17	12	39	27
LLW Unit Transfer and Handling — Above Ground	5	6	23	15	36	23	52	36
HLW Unit Handling and Emplacement — Below Ground	268	185	219	153	195	136	215	147
LLW Unit Handling and Emplacement — Below Ground	2	2	21	13	34	23	51	36
Retrievability Phase Waste Removal	54	28	290	116	229	92	526	210
Offgas Recovery and Release	55	33	2	2	23	13	55	33
Decommissioning (prorated over operating years)	6	5	23	16	22	15	40	28
Repository Maintenance	41	19	44	23	44	21	85	42
Damaged Waste Canister — Handling and Processing	70	28	61	22	63	26	133	52
Mining & Drilling — For Retrieval	54	28	296	136	235	110	535	248
Surface Waste Management	41	28	23	15	23	15	23	15
Support/Overhead	<u>174</u>	<u>—</u>	<u>183</u>	<u>—</u>	<u>181</u>	<u>—</u>	<u>345</u>	<u>—</u>
Totals (man-rem/yr)	902	454	1,295	569	1,188	532	2,301	978
Total Facility Personnel	1,000	300	1,068	352	1,045	332	1,985	603
Average Dose (rem/yr)	0.9	1.5	1.2	1.6	1.1	1.6	1.2	1.6

(a) Data developed from radiation data in Exxon NFRRC PSAR⁽²⁾

Table 7-13
NON-OCCUPATIONAL DOSE DURING NORMAL
OPERATIONS AT A REPOSITORY^(a)

Dose (man-rem/yr)	Ref. Case	Case 1	Case 2	Case 3
Whole body	1.52 E-5	4.98 E-6	7.69 E-6	1.11 E-5 ^(b)
Bone	6.12 E-4	2.00 E-4	3.10 E-4	4.48 E-4
Thyroid	1.47 E-8	4.81 E-9	7.44 E-9	1.08 E-8

(a) Data source: Developed from preliminary WIPP safety studies⁽⁵⁾

(b) 1.11 E-5 means 1.11×10^{-5}

A Design Basis Accident or abnormal occurrence at a repository has only a negligible impact on non-occupational personnel. All of the cases provide waste forms that are not readily dispersible. This is especially true of the complex waste in Case 3 and the Reference Case. Table 7-14 shows the non-occupational doses possible from a Design Basis Accident at a repository. For Cases 2 and 3, the SFDS also assumes that gas bottles containing ⁸⁵Kr, ¹²⁹I, ³H, and ¹⁴C obtained from processing the spent fuel are stored in the repository. If one of these bottles were broken open and the entire contents escaped from the facility, the resultant dose to an individual would be of the order given in Table 7-14 for Case 1.

Table 7-14
NON-OCCUPATIONAL DOSE FROM
DESIGN BASIS ACCIDENT AT A REPOSITORY^(a)

	Whole Body	Dose (rem)	
		Bone	Lung
Reference Case ^(b)	0.065	1.1	0.110
Case 1	0.056	1.8	0.093
Case 2	0.056	1.8	0.093
Case 3 ^(c)	0.056	1.8	0.093

(a) The Design Basis Accident assumes the hoist fails and the waste canister drops down the mine shaft. This is based on preliminary WIPP safety studies.⁽⁵⁾

(b) Assumes that the experimental high level reprocessed waste assumed for WIPP is equivalent to the Reference Case waste.

(c) This study assumes that the complexed form of the Case 3 waste is compensated for by the increased curie activity so that the total release remains approximately constant.

7.3 LONG-TERM ASSESSMENT — REPOSITORY

It is impossible to accurately predict the probability of naturally occurring events causing a breach in the repository in the far distant future. The SFDS assessment does not attempt to perform a probability analysis for such events. While the consequences of such events are analyzed in this study, their occurrence is considered unlikely.

The assessment of long-term radiological impact considers breaching of the repository via three mechanisms:

- Liquid breach by water in an overlying or underlying aquifer with the subsequent transport of leached radionuclides through the aquifer to the biosphere
- Directly drilling through a waste container while exploring for minerals, and
- Catastrophic events that would completely open the repository and expose the waste

7.3.1 General Discussion

A comparison of cases indicates that the radiological impact of spent fuel is much worse than Reference Case waste. This is due to the higher content of uranium and plutonium. However, it should be pointed out that the rate at which these elements migrate in solution through the geologic medium is extremely low.

The magnitude of the hazard to future populations depends on the time frame in which the repository is breached. As time passes, the radionuclides, contained in the waste, decay. Eventually, the radioactive content reaches a level equal to the amount of radioactivity given off by naturally occurring uranium ore. For the waste types considered, about 1000 years is needed to reach this level. The potentially most hazardous element in the repository throughout long periods of time is plutonium (and its daughters) because of its extremely long half-life and toxic properties. Plutonium is contained in significant amounts in spent fuel with trace amounts also present in Reference Case waste. The decay of spent fuel is shown graphically in Figure 7-1. The following discussion presents information relative to breaching the repository and which hazardous nuclides are predominant as a function of time:

- **<100 Years** — This is the time period of greatest radioactivity because of the presence of fission products (which have not yet decayed) in combination with the long-lived actinides. The fission products ^{90}Sr and ^{137}Cs and the actinides ^{239}Pu and ^{241}Am dominate the waste during this time. No breach of the repository is expected during the first hundred years because of the engineered containment and the resistance of the waste to leaching
- **100-1000 Years** — A breach of the repository by catastrophic events or direct drilling before 1000 years could lead to significant doses especially from the spent fuel cases. During this period, nuclides are kept within the geologic media by the resistance of the waste to leaching, the impermeability of the salt, and the long distance that nuclides would have to traverse to get to the biosphere. The actinides ^{239}Pu and ^{241}Am are dominant during this time
- **10,000-1,000,000 Years** — The dominant nuclides during this period are ^{229}Th , ^{226}Ra , and ^{237}Np . ^{239}Pu is also dominant for the first 200,000 years. By one million years after entombment all of these levels have decayed to only a few curies per container.

NOTE:
Waste is emplaced in
repository 10 years
after removal from
reactor

Figure 7-1.
Radioactive Decay of Spent Fuel

7.3.2 Liquid Breach and Transport of Radionuclides

Over long periods of time, all waste will leach if groundwater breaches the repository. Hence, in the long-term the initial form of the waste is unimportant. The dominant factor affecting the radiological impact is the time required for the radionuclides to reach the biosphere. This time is dependent on:

- The transmissivity, i.e., the rate at which a given nuclide will pass in solution through the geological media
- The distance the waste must travel, and
- The flow rate of the water

It becomes, in the end, a race between the rate at which biologically harmful nuclides are being moved to the biosphere and the rate at which they are decaying. For a given salt repository, the estimated transport times are about 5000 years for the fastest traveling radionuclides. Nuclides such as uranium and plutonium have a very low coefficient of transmissivity and tend to remain nearly in place.

When the liquid transporting the nuclides reaches the biosphere (at about 5000 years) it is dominated completely by the two long-lived fission products ^{99}Tc and ^{129}I . These nuclides yield a dose to the maximum exposed individual that is less than one percent of background. Since the fission product inventory of the repository is the same for all four study cases, this breaching event does not help to differentiate between the cases.

7.3.3 Direct Access Drilling

The direct access drilling scenario is extremely conservative in that it postulates the following sequence of events:

- 1) Knowledge of the repository is lost.
- 2) There is an economic incentive to explore in the area of the repository (i.e., minerals of economic value are thought to be present).
- 3) Drilling is carried out at the repository site.
- 4) Direct drilling intercepts the highest concentration of radionuclides.
- 5) The material brought to the surface is not recognized as radioactive and is left untreated and exposed.

The non-occurrence of any step in this sequence would either negate the scenario or mitigate its consequences.

There are two segments of the population that may be exposed: the drill crew and persons living downwind from the pond containing contaminated drilling mud. As can be seen in Table 7-15, the site geologist, who is nearest the waste for the longest period of time, receives the highest dose. The dose from Reference Case waste is by far the worst due to the quantity of fission products contained. In terms of absolute magnitude, the worst dose is at 100 years. An individual living downwind from the contaminated drill mud is exposed to particulates entrained in the air. Table 7-15 indicates that the maximum exposed individual receives a dose that is approximately equal to background. Again, Reference Case waste yields the worst dose.

Table 7-15
LONG-TERM RADIOLOGICAL CONSEQUENCES
OF A NON-CATASTROPHIC BREACH OF THE REPOSITORY^(a)

1. Direct access drilling. Site geologist exposed at 1 meter for 1 hour

Time of Drilling after Emplace- ment (yr)	Ref. Case ^(a)	Dose (rem)		
		Case 1	Case 2	Case 3
100	340.0	90.0	100.0	43.7
250	13.4	3.5	3.9	1.7
1,000	5.3	1.4	1.6	0.7

2. Maximum exposed individual living downwind of the drilling mud pond

Time of Drilling after Emplace- ment (yr)	Organ	Ref Case ^(b)	50 Year Dose Commitment for 1 Year Intake (rem)		
			Case 1	Case 2	Case 3
100	Lung	2.67 E+0	7.00 E-1 ^(c)	7.89 E-1	3.43 E-1
	Bone	1.26 E+0	3.30 E-1	3.72 E-1	1.62 E-1
	Whole Body	1.64 E-1	4.30 E-2	4.84 E-2	2.37 E-2
250	Lung	5.35 E-3	1.40 E-3	1.57 E-2	6.90 E-4
	Bone	1.05 E-1	2.74 E-2	3.09 E-2	1.35 E-2
	Whole Body	5.72 E-3	1.50 E-3	1.69 E-3	7.40 E-4
1,000	Lung	2.59 E-3	6.79 E-4	7.65 E-4	3.33 E-4
	Bone	4.96 E-2	1.30 E-2	1.47 E-2	6.40 E-3
	Whole Body	6.65 E-3	4.56 E-4	5.14 E-4	2.24 E-4

(a) Data source: Developed from WIPP environmental and safety studies^{(4) (5)}

(b) Exact data regarding HLW was not available. These numbers were estimated by considering the impact of the fission product inventory which dominates out to 1000 years. The actinide inventory, which will slowly become predominant as time passes, was ignored.

(c) 7.00 E-1 means 7.00×10^{-1}

7.3.4 Catastrophic Breaching Events

The non-radiological consequences of a catastrophic event far outweigh the radiological consequences. The probability of an event such as a volcano or meteorite causing a breach in the repository is diminishingly small. However, the ensuing radiological release from such an event is significant if the repository is breached within the first few thousand years. Although a catastrophic breach is not expected, the results of such a breach are included in the assessment of completeness. Table 7-16 lists the resultant doses for a breaching event that exposes all of the repository waste 1,000 and 2,000 years in the future. The three spent fuel cases (which are indistinguishable from each other) dominate the radiation impact of a catastrophic breach. This is because the repository will contain large amounts of uranium and plutonium which are present in much smaller quantities in Reference Case waste.

Table 7-16
RADIOLOGICAL IMPACT DUE TO A CATASTROPHIC
EVENT CAUSING A BREACH OF THE REPOSITORY
WITH THE SIMULTANEOUS RELEASE OF ALL WASTE^(a)

	Total Body Dose			
	1000 Years		2000 Years	
	Ref. Case	Cases 1-3	Ref. Case	Cases 1-3
Within an area of 5-Km radius from the site (mrem/yr)	1.12 E+6 ^(b)	9.85 E+7	6.74 E+5	5.93 E+7
To the nonspecific population (man-rem/yr)	6.51 E+5	6.80 E+7	1.32 E+4	1.38 E+6

(a) Data source: Developed from Logan⁽⁶⁾

(b) 1.12 E+6 means 1.12×10^6

REFERENCES FOR SECTION 7

- (1) "Draft Environmental Impact Statement for Storage of U.S. Spent Power Reactor Fuel," (DOE/EIS-0015-D, U.S. Department of Energy, August 1978.
- (2) "Nuclear Fuel Recovery and Recycling Center Preliminary Safety Analysis Report," Docket No. 50-564, Exxon Nuclear Company, Inc., September 1977.
- (3) "Barnwell Nuclear Fuel Plant Safety Analysis Report," Docket No. 50-332, August 1970.
- (4) "Draft Environmental Impact Statement for the Waste Isolation Pilot Plant, "DOE/EIS-0026-D, U.S. Department of Energy, April 1979.
- (5) Duncan, D. S., "Safety Considerations of the WIPP Design," presented at the 87th meeting of the American Institute of Chemical Engineers, Boston, Mass., August 18-22, 1979.
- (6) Logan, S.E. and Berbano, M.C., "Development and Application of a Risk Assessment Method for Radioactive Waste Management," EPA-520/6-78-005, U.S. Environmental Protection Agency, June 1978.

Section 8

ASSESSMENT FOR RETRIEVABILITY

TABLE OF CONTENTS

Section		Page
8.1	Assessment Basis and Approach	8-2
8.1.1	Definition of Retrievability	8-2
8.1.2	Measurement of Retrievability	8-2
8.1.3	Methodology for Developing the Assessments	8-4
8.2	Assessment Considerations	8-5
8.2.1	Retrievability Subsystems	8-5
8.2.2	Basis for Current Approach	8-6
8.3	Assessment by Cases	8-7
8.3.1	Reference Case	8-8
8.3.2	Case 1	8-9
8.3.3	Case 2	8-9
8.3.4	Case 3	8-10
8.3.5	Summary of the Four Cases	8-10

Section 8

ASSESSMENT FOR RETRIEVABILITY

An efficient waste canister retrieval system is required as a contingency system for the successful long-term disposal of high level waste (HLW) and spent fuel in a geologic repository. In this section, the retrieval capability for the four cases is evaluated by conducting an assessment similar to that made for technical feasibility (Section 4). The conclusions can be summarized as follows:

- In general, the retrievability assessment does not significantly differentiate between cases
- A five-year retrievability period for the systems as described in this study is feasible in all cases
- Case 2 (voloxidized chopped fuel bundle) is probably the least desirable spent fuel form if retrieval of waste canisters should occur, because of the nature of the canister contents
- The number of canisters to be retrieved directly impacts the mechanics of the retrieval operation and its radiological protection aspects

Technical considerations, which support this assessment for retrievability, are included in Appendix I of Volume 3 of this report. Technical references are also cited in that Appendix.

8.1 ASSESSMENT BASIS AND APPROACH

8.1.1 Definition of Retrievability

For assessment purposes, retrievability is defined as the ability to extract and return the spent fuel and HLW canisters from geologic storage to the entrance of the repository surface facility during the initial 5-year repository demonstration period. In this study, the retrieval of emplaced canisters is considered only in conjunction with an initial repository demonstration or validation program, which is discussed in Appendix D 2.8 of Volume 3. Other aspects of retrievability, such as canister removal for subsequent reprocessing and recovery of energy values, are not included in this assessment.

The intent of the retrievability assessment is to produce a numerical measure of the confidence that certain emplaced canisters can be retrieved satisfactorily. Designated canisters considered in the four cases for potential retrieval are:

- HLW canisters — Reference Case
- Spent fuel canisters — Cases 1, 2, and 3
- Canisters of end fittings — Reference Case and Cases 2 and 3
- Canisters of cladding hulls — Reference Case and Case 3

8.1.2 Measurement of Retrievability

The measure of retrievability for a given study case is a weighted average confidence index, A_r , reflecting the degree to which the retrievability system and its operations for that case are expected to perform satisfactorily.

This measure is defined by the equation

$$A_r = \sum_m (A_m W_m) \quad (8-1)$$

where

r = The retrievability system for a given case

m = A retrievability subsystem within the repository

A_m = The degree of technical development and implementation for each subsystem, m

W_m = The relative contribution of each subsystem, m to the repository as a whole

8.1.2.1 Technical Development, A_m Except for radiological protection and material quantity, A_m depends on the degree of previous successful demonstration of the retrievability subsystems required. The values of A_m and their meanings are given in Table 8-1. A scale of 1 to 5 is used indicating high to low confidence in retrievability, respectively. The values are derived from experience and are discussed in Subsection 8.3.

Table 8-1

VALUES OF TECHNICAL DEVELOPMENT RATING, A_m , FOR RETRIEVABILITY

A_m	Meaning
1	Well known operation with long history of plant operation
2	Operation with all technical problems resolved. Minor improvements are possible through research
3	Well known operation that requires continued improvement and research. No large scale "hot" operating experience at a commercial facility up to the present date
4	A sound theoretical operation that needs additional research and development and small-scale demonstration
5	A new operating concept that has not yet been developed through the design state

Table 8-2

VALUES OF COMPLEXITY FACTOR, C_m , FOR RETRIEVABILITY

C_m	Meaning
1	A subsystem consisting of a single component with contact operation
2	A subsystem consisting of several components with contact operations
3	A subsystem having a single component with remote operation
4	A complex subsystem consisting of several components with remote operations

Table 8-3

VALUES OF RELATIVE IMPORTANCE FACTOR, R_m , FOR RETRIEVABILITY

R_m	Meaning
1	A noncritical subsystem
2	A critical subsystem that is necessary for safe operation and/or whose failure results in extended downtime

8.1.2.2 Relative Contribution, W_m . W_m is a weighting factor. Each W_m was developed to reflect, as nearly as possible, the following two factors: subsystem complexity and the relative importance of the subsystem to the repository. For each retrievability subsystem, W_m is defined by the relationship:

$$W_m = \frac{C_m R_m}{\sum_m (C_m R_m)} \quad (8-2)$$

where:

C_m = A measure of the complexity of the subsystem

R_m = A measure of the relative importance of the subsystem

The values C_m and R_m and the meanings of these values are given in Tables 8-2 and 8-3.

8.1.3 Methodology for Developing the Assessments

The following methodology is used in developing the retrievability assessment for a given case:

- (1) Identify all subsystems within the retrievability system. The subsystems may be physical components of the repository or performance aspects of components. They are identified in such a way that:
 - The sum of the subsystems constitutes the whole of the retrievability system
 - If a given step is a subsystem in one case, it is also a subsystem in other cases where it occurs
- (2) Using the 1 to 5 scale defined in Table 8-1, determine the A_m of each subsystem of the retrievability system.
- (3) Using the guide provided in Subsection 8.1.2.2, determine the relative contribution, W_m , of each subsystem of the retrievability system. These contributions are based on the technical considerations for retrievability which are found in Appendix I of Volume 3.
- (4) Combine the results of Steps 1, 2, and 3 using Equation 8-1 and obtain the degree of technical development and implementation, A_r , for the retrievability system for each case.

The results obtained in Step 4 provide a basis for comparing the degree of confidence in retrieving radioactive waste canisters for each case. A summary comparison of the four cases is presented in Subsection 8.3.5.

8.2 ASSESSMENT CONSIDERATIONS

8.2.1 Retrievalability Subsystems

In this assessment, the retrievalability system is divided into the following retrievalability subsystems for each of the four study cases analyzed:

- Sleeve integrity
- Canister integrity
- Transfer/overpack cell
- Decontamination/corrosion removal cell
- Repository ventilation system
- Radiological protection
- Monitoring and surveillance system
- Quantity of retrievable material
- Room closure

The measure of retrievalability is determined by examining the individual numerical values of A_m , C_m , and R_m for each of the subsystems, and combining them into an A_r value as discussed in Subsection 8.1.

8.2.1.1 Degree of Technical Development and Implementation, A_m . Components and subsystems in new facilities are likely to be the most reliable if they have been operating in other commercial or government facilities, and hence they inspire a high level of confidence. Components and systems under development and testing in laboratories and pilot plants rate a lower degree of confidence.

The degree of technical implementation of a component or subsystem reflects its state of development. Some subsystems do not need implementation in a "hot" pilot plant because they are proven, standard subsystems in the nonnuclear world and can be readily evaluated without such additional development. The A_m of such proven subsystems is 1.

8.2.1.2 Degree of Technical Complexity, C_m . The complexity factor is assigned in light of the required application for each subsystem. Many of the retrieval operations are done by remote handling and control. These operations are significantly more complex than contact operations, where the operator can easily make necessary adjustments. Simple operations are rated low (e.g., C_m of 1); the more difficult operations are rated higher (e.g., C_m from 2 to 4).

8.2.1.3 Relative Importance, R_m . The relative importance of a subsystem is measured by its impact on the overall operation. $R_m = 1$ for a relatively unimportant subsystem and 2 for an important subsystem. For example, loss of ventilation in the storage area could cause a shutdown of operations until temperatures can be brought back to normal working levels. Thus, loss of ventilation is assigned an R_m of 2. R_m , when combined with the complexity factor, gives appropriate weight to the component or subsystem within the total assessment.

8.2.2 Basis for Current Approach

In each of the four cases studied, an attempt is made to perform an objective analysis of the criteria that affect feasibility. Factors essential to the proper assessment of each case are assigned numerical values although it should be recognized that the scales used are, of necessity, somewhat arbitrary. Retrievalability in this study is considered to have five stages of technical development, four stages of complexity, and two degrees of relative importance. The rating scale for each of these is intended to be generally linear, although each is subject to interpretation. A retrievalability subsystem with a rating of 1 is more feasible than one with a rating of 2, but it may not be necessarily twice as feasible.

8.3 ASSESMENT BY CASES

This subsection presents the assessment of retrievability for the study cases using the bases and methodology described in Section 8.1. The retrievability assessment evaluates the nine retrievability subsystems identified in Subsection 8.2.1.

The selected values for technical development, A_m , relative importance, R_m , and complexity, C_m , for the components or subsystems are based on the definitions outlined in Subsection 8.1 and applied as follows:

The principal functions of the **sleeve** are the same in all cases and have been demonstrated in pilot plant operations but not in commercial plants; therefore, A_m is assigned a value of 3. R_m is assigned a value of 2 since the primary objective is to eliminate canister entrapment. C_m is assigned a value of 1 since only a simple component is involved.

The technical feasibility of maintaining **canister integrity** is effectively the same for all cases and is assigned an A_m value of 3. Because the canister is a single-component item requiring remote operation for movement, it has a C_m value of 3 for the Reference Case and Cases 1 and 3. But, because the failure of a canister in Case 2 would cause infinitely more problems, C_m for that case is assigned a value of 4. The R_m of the canister is small ($R_m = 1$) when the material is in solid form, but significantly more important ($R_m = 2$) for the chopped fuel form in Case 2.

The **transfer/overpack cell** from the operational standpoint is the same for all cases and assigned a value of $A_m = 3$. Since this cell plays a significant role in retrievability, it is assigned an R_m of 2. In the Reference Case, Case 2, and Case 3, there is a variety of canister types to be handled, and C_m is assigned a value of 4; in Case 1, which involves a single canister type, C_m is assigned a value of 3.

The **decontamination/corrosion removal cell** has the same rating as the transfer cell and is assigned an A_m of 3. R_m and C_m values for this cell are lower than for the transfer cell because their impact on the cell's ability to perform is not as significant. Values of $R_m = 1$ and $C_m = 3$ are assigned for all cases.

Ventilation systems in mining operations are well established and can be adequately designed to meet the requirements of a geologic repository for radioactive materials. Their A_m is, therefore, 1. In each of the four cases, ventilation represents a major subsystem ($R_m = 2$), and C_m is the same as in a single contact system ($C_m = 1$).

Radiological protection is rated differently. Because the assessment is based on the potential dose to the public, the larger the quantity of radioactive material to be moved, the greater the risk. Also, there is an inherent potential of fission gas leakage from spent fuel canisters in Case 1. Therefore, A_m for the four cases, on a scale of 1 to 5, is as follows:

Reference Case	Case 1	Case 2	Case 3
$A_m = 1$ (lowest potential)	$A_m = 3$	$A_m = 2$	$A_m = 4$

In each of the four cases, R_m is rated high ($R_m = 2$), and C_m is the same as in a single remote system ($C_m = 3$).

Monitoring and surveillance is essentially the same for all cases. Since improvements are usually required, A_m is assigned a value of 2. R_m is rated as minor ($R_m = 1$), and C_m is the same as in a single remote system ($C_m = 3$).

The **quantity of material** to be retrieved is ranked by a simple consideration of the range of quantities with a range of 1 to 4, as follows:

Reference Case	Case 1	Case 2	Case 3
6,328 canisters	10,200	9,605	20,858
$A_m = 1$	$A_m = 2$	$A_m = 2$	$A_m = 4$

R_m is ranked high ($R_m = 2$), and the C_m is the same as in a single contact system ($C_m = 1$).

In all four cases, some trimming of the rooms may be required to prevent **room closure** before retrieval operations can be started. A_m is 2 for all cases, R_m is considered low ($R_m = 1$), and C_m is the same as in remote multi-operations ($C_m = 4$).

8.3.1 Reference Case

The retrievability assessment for the Reference Case is presented in Table 8-4.

Table 8-4
RETRIEVABILITY ASSESSMENT FOR THE REFERENCE CASE

Subsystem m	A_m	R_m	C_m	W_m	$A_m W_m$
Sleeve integrity	3	2	1	0.06	0.18
Canister integrity	3	1	3	0.09	0.27
Transfer/overpack cell	3	2	4	0.24	0.72
Decontamination/ corrosion cell	3	1	3	0.09	0.27
Ventilation system	1	2	1	0.06	0.06
Radiological protection	1	2	3	0.18	0.18
Monitoring and surveillance	2	1	3	0.09	0.18
Material quantity	1	2	1	0.06	0.06
Room closure	2	1	4	0.12	0.24
Totals				1.00	2.16

$$A_r = 2.2$$

8.3.2 Case 1

The retrievability assessment for Case 1 is presented in Table 8-5.

Table 8-5
RETRIEVABILITY ASSESSMENT FOR CASE 1

Subsystem m	A _m	R _m	C _m	W _m	A _m W _m
Sleeve integrity	3	2	1	0.06	0.18
Canister integrity	3	1	3	0.10	0.30
Transfer/overpack cell	3	2	3	0.19	0.57
Decontamination/ corrosion cell	3	1	3	0.10	0.30
Ventilation system	1	2	1	0.06	0.06
Radiological protection	3	2	3	0.19	0.57
Monitoring and surveillance	2	1	3	0.10	0.20
Material quantity	2	2	1	0.06	0.12
Room closure	2	1	4	0.13	0.26
Totals				1.00	2.56

$$A_r = 2.6$$

8.3.3 Case 2

The retrievability assessment for Case 2 is presented in Table 8-6.

Table 8-6
RETRIEVABILITY ASSESSMENT FOR CASE 2

Subsystem m	A _m	R _m	C _m	W _m	A _m W _m
Sleeve integrity	3	2	1	0.05	0.15
Canister integrity	3	2	4	0.21	0.63
Transfer/overpack cell	3	2	4	0.21	0.63
Decontamination/ corrosion cell	3	1	3	0.08	0.24
Ventilation system	1	2	1	0.05	0.05
Radiological protection	2	2	3	0.16	0.32
Monitoring and surveillance	2	1	3	0.08	0.16
Material quantity	2	2	1	0.05	0.10
Room closure	2	1	4	0.11	0.22
Totals				1.00	2.50

$$A_r = 2.5$$

8.3.4 Case 3

The retrievability assessment for Case 3 is presented in Table 8-7.

Table 8-7
RETRIEVABILITY ASSESSMENT FOR CASE 3

Subsystem m	A _m	R _m	C _m	W _m	A _m W _m
Sleeve integrity	3	2	1	0.06	0.18
Canister integrity	3	1	3	0.09	0.27
Transfer/overpack cell	3	2	4	0.24	0.72
Decontamination/ corrosion cell	3	1	3	0.09	0.27
Ventilation system	1	2	1	0.06	0.06
Radiological protection	4	2	3	0.18	0.72
Monitoring and surveillance	2	1	3	0.09	0.18
Material quantity	4	2	1	0.06	0.24
Room closure	2	1	4	<u>0.12</u>	<u>0.24</u>
Totals				1.00	2.88

$$A_r = 2.9$$

8.3.5 Summary of the Four Cases

The retrievability assessment summary for all four cases is presented in Table 8-8.

Table 8-8
RETRIEVABILITY ASSESSMENT SUMMARY
A_mW_m

Subsystem m	Reference Case	Case 1	Case 2	Case 3
Sleeve integrity	0.18	0.18	0.15	0.18
Canister integrity	0.27	0.30	0.63	0.27
Transfer/overpack cell	0.72	0.57	0.63	0.72
Decontamination/ corrosion cell	0.27	0.30	0.24	0.27
Ventilation system	0.06	0.06	0.05	0.06
Radiological protection	0.18	0.57	0.32	0.72
Monitoring and surveillance	0.18	0.20	0.16	0.18
Material quantity	0.06	0.12	0.10	0.24
Room closure	<u>0.24</u>	<u>0.26</u>	<u>0.22</u>	<u>0.24</u>
A _r	2.2	2.6	2.5	2.9

Within the accuracy of this assessment criteria, the indicated differences between the cases are not considered significant. This is especially true of sleeve integrity, repository ventilation system, monitoring and surveillance system, and room closure considerations, which make essentially no differentiation between cases.

The dominant subsystems in the assessment are the quantity of material to be retrieved and radiological protection. The number of canisters maintained in a retrievable mode impacts directly on both of these aspects of the retrievability assessment. The canister integrity subsystem also impacts against Case 2 because the (hazardous) contents of the canister would be so easily dispersed compared to the other spent fuel/waste forms.

Section 9
ECONOMIC ASSESSMENT

TABLE OF CONTENTS

Section		Page
9.1	Background	9-2
9.2	Approach to the Economic Assessment	9-3
9.3	Bases of the Economic Assessments	9-4
9.4	Assessment by Cases	9-8
9.4.1	The Reference Case	9-8
9.4.2	Case 1	9-8
9.4.3	Case 2	9-11
9.4.4	Case 3	9-11
9.4.5	Comparison of Cases	9-11
9.4.6	Sensitivity to Discount Rate	9-15
9.5	Approximate Charges in Commercially Operated Systems	9-17
	References for Section 9	9-19

Section 9

ECONOMIC ASSESSMENT

This assessment compares the four alternative spent fuel disposal systems in terms of their respective total real costs to the nation, without regard to possible charges to users for the services. The expenditures of resources in these systems vary greatly from year to year over the system lifetimes. Using present value analysis, the levelized average cost in \$/MTHM is calculated for each system. These are the costs which, if allocated to each MTHM at the time it is received for disposal, would be equivalent to the varying cost streams actually experienced during the period. Levelized costs are calculated using discount rates of zero, five, ten and fifteen percent.

Results are summarized in Table 9-5, which appears later in this section. The costs of the Reference Case are more than offset by the values of recycled UF_6 and PuO_2 obtained, so that this system could operate at a substantial net gain to the economy. Cases 1 and 2 have approximately the same costs which are, in turn, significantly lower than those of Case 3. Owing to heavy initial capital investments the Reference Case is strongly affected by the discount rate, its costs per MTHM more than doubling as the rate goes from zero to 15 percent. The other cases are not greatly affected by the choice of discount rates.

9.1 BACKGROUND

This assessment, like the technical assessment, assumes that the U.S. government will choose from among the alternative spent fuel disposal systems that system which will best serve the interests of the nation. To facilitate comparison of the four systems, the economic assessment for each system evaluates the total real costs, and expresses these in dollars/MTHM.

This analysis does not address the question of charges that the government might impose on utilities for carrying out the spent fuel disposal function, but it does calculate the resources of personnel and materials that would be required to operate the disposal systems. These resources would be committed solely to spent fuel disposal and would not be available for production of other goods and services of value to the nation. It is the giving up of such other goods and services that constitutes the economic cost of spent fuel disposal. Among economists, this is known as the "opportunity cost" of disposing of spent fuel, a definition of costs that is commonly used to evaluate water resource development projects and other proposed public investments and that is considered to be the most appropriate definition on which to base the present analysis.

Since the choice of a system will be made by the government in the national interest, it is appropriate to base the economic assessment of the systems on total real costs to the nation as a whole. The system having the lowest total costs per MTHM is economically the most desirable system. The way in which these costs are borne by the various members of society is a separate issue that should be considered apart from the choice of system. Similar options for cost sharing exist for all systems; therefore, these options are not considered in the present assessment. Cost sharing is determined by such factors as the form of ownership of the facilities, the methods of financing, and government policy with respect to charging utilities for spent fuel disposal services. In earlier studies, these variables were analyzed in order to calculate the amount that the government might charge to utilities under existing guidelines.^{(1),(2),(3)} But the methodology of these studies differs fundamentally from the present assessment.

Potential charges for spent fuel disposal services are influenced by the ownership of facilities (public or private), which in turn affects the taxes to be paid. Charges are also influenced by the way in which facilities are financed — the debt/equity ratio, interest rates for borrowed funds, and the desired rate of return on equity capital. Therefore, taxes and financial arrangements (considered by economists as transfer payments among members of the economy rather than as a consumption of real resources) play an important role in an analysis of potential charges. However, as taxes and financial arrangements do not affect the amount of real resources (i.e., the number of personnel and the materials) required to implement the several systems of spent fuel disposal, they are not relevant to a comparison of the economic costs of the alternative systems and are not included in the assessment in Subsections 9.3 and 9.4.

Nevertheless, a brief analysis is made and is discussed in Subsection 9.5 to determine the approximate rates that might be charged by a commercial venture for handling the materials in each of the four cases. This analysis is provided for information and reference only and is not considered to be an appropriate basis for national policy comparison or choice of a preferred case.

9.2 APPROACH TO THE ECONOMIC ASSESSMENT

The total costs (made up of initial capital costs, ongoing operating costs, and final decommissioning costs) of spent fuel disposal are incurred as a stream of expenditures over a period of many years. For convenience in making the economic comparison, all of these costs are reduced to a single present worth value at year 0. The spent fuel processed by the system also represents a stream — in this case, of material — which is processed during the period for which the analysis is made. This material quantities stream is also discounted to year 0, using the same discount rate applied to the stream of costs in order to provide an appropriate basis for the economic cost assessment. The levelized average cost in \$/MTHM over the period of the analysis is obtained by dividing the present worth of the cost stream by the present worth of the material quantities stream.

The levelized average cost in \$/MTHM represents that cost which, if charged to each unit of material at the time it is received for disposal in the system, would be equivalent to the varying cost streams actually experienced over the period of the analysis.* It provides a sound basis for comparing the alternatives. In actual practice, after a case has been selected for implementation, a fee schedule would probably be developed which would relate the charges to the conditions of ownership and financing. Fees would probably be adjusted from time to time in response to price changes in the economy that would affect operating and maintenance costs.

* The levelized average cost is calculated with respect to a specified discount rate which represents the annual return that could be obtained from the use of resources for other purposes

9.3 BASES OF THE ECONOMIC ASSESSMENTS

The following definitions and assumptions are used in developing the economic cost per MTHM in the four systems analyzed.

Each of the four disposal systems is defined to include all the facilities and functions necessary to dispose of spent fuel from the time it leaves storage at the reactor, 1 year after being removed from the reactor core. Figure 9-1 is a generalized flow chart illustrating the steps involved in the disposal process. Some of the systems analyzed do not require that the fuel pass through all the steps shown.

It is assumed that the AFR, the P/E facility, and the repository will not be colocated in any of the four systems analyzed. However, the possibility of colocation is addressed as a study variation in Subsection 10.5.

The following facilities and functions are treated as cost centers:

- AFR
- P/E facility
- Repository
- Transportation
 - Reactor to AFR
 - Reactor to P/E facility
 - AFR to P/E facility
 - AFR to shallow land burial
 - P/E facility to repository
 - P/E facility to fuel fabrication/enrichment facility
 - P/E facility to shallow land burial

The total economic cost per MTHM is the sum of economic costs per MTHM for all of the facilities and functions constituting the system.

It is assumed that spent fuel disposal services will be needed over a long period of years exceeding the physical life of all of the original facilities required in each of the systems. The analysis includes the costs properly allocable to the entry of spent fuel into each disposal system during the first 40 years of operation. Some of these costs are, of course, incurred after the 40th year of operation. Since the analysis is designed to aid in the selection of a system to be implemented in the future, it is appropriate to include costs from the beginning of construction and covering all operations in order to allow for differences among the systems with respect to the timing of construction of the component facilities.

Forty years of spent fuel receipts are analyzed because this is the approximate effective working life of the P/E facility, and because prolongation of the period of analysis would have little effect on the results of the present value calculations. Although it is possible that new technology for handling spent fuel will become available before the end of the 40 years, the possibility that this technology will result in the replacement of the facilities in the systems is not included in the analysis. Since nuclear facilities are difficult to deactivate, the incentive to declare an existing installation obsolete is probably less in the nuclear field than in other industries, and a 40-year working life is not unreasonably long.

Just as it is assumed that the need for disposal services will continue beyond the life of any original facility, so it is assumed that facilities not physically worn out or filled to capacity after 40 years of fuel receipts have been disposed of, will have a remaining useful life. Accordingly, the costs of such facilities are allocated both to the period of analysis and to the remaining useful life.

Figure 9-1.
Economic Analysis Diagram

The study is further based on the assumption that spent fuel will enter each system at the rate of 5,000 MTHM/yr over the 40-year period of receipts analyzed. This assumption reflects a segment of the industry as it is likely to exist early in the next century. Any changes in this capacity are not expected to significantly affect the economic costs per MTHM.

Costs that have been incurred prior to the selection of a system for implementation, such as those for research and development, are not included as economic costs. These are sunk costs representing resources that have been expended and that cannot be recovered for reuse in the economy. They are not changed by the choice of a system for implementation, and are therefore not relevant to the selection process. (Present guidelines for calculating government charges for spent fuel disposal require recovery by the government of the monetary costs that have been incurred for such prior activities. However, this is a policy with respect to cost sharing, rather than a factor in comparing the economic costs of alternative choices.) All costs, except government administrative costs, that are incurred in further developing and implementing a system after it is selected, are included in the estimates shown for each system in the analysis.

Tables 9-1 through 9-4 include a column headed "Estimated Costs Incurred (\$ Millions)." These data summarize in very brief form the total estimated costs of various elements, such as capital costs, operating costs, decommissioning costs, transportation lease costs, and energy value credits. These costs are taken from Appendices B, C, D, and E, in Volume 3, which provide facility descriptions and cost estimates for AFRs, P/E facilities, repositories, and transportation systems. It should be borne in mind that all cost estimates have been developed from preconceptual designs and at summary estimating levels and as such are considered to be subject to a range of uncertainty of +75 percent, -25 percent. Accordingly, the economic costs are also subject to an accuracy within a similar range.

In the present-value calculations, the capital cost of each facility is entered as a total constructed cost in the last year of construction work. The total includes an allowance for interest during construction, reflecting results experienced on heavy construction projects. This allowance is made in lieu of a detailed study of probable construction schedules and expenditure rates that would permit dividing the costs among the years of the construction period. Such a detailed approach is not warranted in view of the wide range of uncertainty in the basic estimates at this time.

The data available as a basis for operating cost estimates are very limited, because there has been no direct experience with the kind of facilities required for spent fuel disposal. Whenever possible, staff levels are developed by shift at each facility, and costs are estimated on the basis of staffing levels and rates and estimated material operating costs for the facility. The total average cost per employee is cross-checked with the results of similar estimates by other economists, where they exist.

Transportation costs are based on cask rental charges by carriers. Owing to the low volume of current shipments, the available transportation facilities are under-utilized. When the volume reaches 5,000 MTHM/yr, better utilization should be possible. Although this increase in production should be reflected in lower costs per MTHM, no attempt to estimate such cost reductions is made in the present study.

Government overhead costs that would be incurred in connection with the spent fuel disposal system are not included in the analysis. These costs, which would be considered economic costs to the nation, are assumed to be the same for all of the systems and would therefore not affect the comparison of the systems.

All cost estimates are expressed in dollars reflecting conditions prevailing in January 1979. Although the four systems generally have similar manpower and materials requirements, they could be somewhat differently affected by future price changes for these entities because of

variances in procurement schedules for each system. However, there is no reliable basis at present on which to project the divergence of future prices from the general price level. Rather than introduce a potential distortion in the comparisons by attempting such a prediction, it is considered more prudent to assume that all systems will be similarly affected. Thus, the impact of future price changes is not included in this analysis.

A discount rate is required to calculate present values in deriving the economic cost estimates for the four systems. The appropriate discount rate for evaluation of projects undertaken in the public interest has been a subject of controversy for many years. A seminar of economists was held to address this particular subject and to consider the U.S. Nuclear Regulatory Commission Report, Generic Environmental Statement on Mixed Oxide Fuels. The report of that seminar and the research conducted for this study confirm that there is still no general agreement on this issue. A discount rate of 10 percent is used to calculate the cost estimates shown in Tables 9-1 through 9-4. The analysis is also carried out with discount rates of 5 percent and 15 percent. Curves fitted to the average costs per MTHM, calculated with these discount rates, are used later to show the sensitivity of the estimated economic costs to the discount rate. It is clear that the choice of discount rate has little effect on the economic comparison of the four systems, except that low rates make Case 3 more costly relative to the other cases.

Formulas that embody the end-of-year convention are used in calculations of present values and levelized annual average costs per MTHM. That is, all costs incurred, and spent fuel received, during a year are treated as if they occurred at the end of the year. With this convention, the end of year zero (0) represents the present. Computations could be carried out on the assumption of expenditures and receipts distributed throughout the year. However, with the discount rates and time periods involved in the present analysis, the difference in results would be small. Further, the range of uncertainty in all estimates and in any assumptions with respect to the distribution of expenditures and receipts does not warrant the added complexity of computation.

9.4 ASSESSMENT BY CASES

Estimates of the economic costs of the four cases covered by the study are given in this subsection. The flow of materials is briefly described for each case; more detailed descriptions of the cases appear in Section 3 and in Appendices B, C, D, and E. Total costs and average economic costs per MTHM are shown for each of the facilities and functions involved in each system. January 1979 dollars and a discount rate of 10 percent are used as a basis of the study. Each case is analyzed separately, followed by a summary comparison table. The sensitivity of average total economic costs per MTHM, to changes in the discount rate, are discussed in Subsection 9.4.6.

9.4.1 The Reference Case

In this case, 1-year-old spent fuel is assumed to be shipped from the reactor to the processing/encapsulation (P/E) facility, where it is reprocessed immediately for recovery of uranium and plutonium, which are shipped to a uranium enrichment plant and a mixed oxide (MOX) fuel fabrication facility, respectively. TRU low level waste (LLW), intermediate level waste (ILW), and cladding hulls are encapsulated and shipped directly to the geologic repository at this time;* non-TRU LLW is shipped to shallow land burial. The high level waste (HLW) is calcined, vitrified, and stored at the P/E facility for 9 years; then it is shipped to the repository for final disposal. The 2,000-acre repository can handle the spent fuel received over a 40-year period before it becomes filled with waste materials equivalent to 200,000 MTHM. (The actual time by which the repository becomes filled is 39.3 years, but 40 years has been used in this analysis, an approximation that has a negligible effect on accuracy.) For the first 9 years, the repository will be operated only to receive hulls, ILW, and LLW for emplacement. It will be operated during years 10 to 14 to make possible the retrieval of HLW canisters, if that should become necessary.

At the end of their useful lives, all facilities will be decommissioned, as discussed elsewhere in this report. The estimated economic costs for the Reference Case are shown in Table 9-1.

9.4.2 Case 1

Cases 1, 2, and 3 do not involve the recovery of energy components from the spent fuel. In these cases, the spent fuel is subject to preparation for storage in a geologic repository. In Case 1, the spent fuel is shipped from the reactor at age 1 year to an away-from-reactor (AFR) storage facility where it is held until it is 10 years old. Nine AFRs regionally located for convenient access from the reactors are provided. At age 10 years, the spent fuel is shipped to a P/E facility for encapsulation; it is then shipped immediately to the geologic repository, together with some TRU and non-TRU ILW. Non-TRU LLW from the AFRs and the P/E facility is shipped to shallow land burial as it is generated.

The repository will be operated during the first 5 years of its life in a manner that will permit retrieval of the spent fuel, if that should become necessary. The repository in this case has a capacity for 17 years of spent fuel receipts; therefore, a second and third repository will be required to accommodate the spent fuel received during the 40-year period of fuel receipts that was selected for analysis. The third repository will be only partially filled by fuel received during the 40-year period, and part of its costs are, therefore, allocated to remaining useful life.

At the end of their useful lives, all facilities will be subjected to decommissioning, as discussed in Appendix D. The estimated economic costs for Case 1 are shown in Table 9-2.

* It would be possible to provide storage for these wastes at the P/E facility for the first nine years of operation and to postpone construction of the repository until the ninth year. Storage at the P/E facility would not be required once the repository is available. The cost of this alternative system was not estimated.

Table 9-1
AVERAGE LEVELIZED ECONOMIC COSTS
REFERENCE CASE
(Based on January 1979 Dollars)

Cost Element	Useful Life (years)		Project Years in Which Costs Are Incurred	Estimated Costs Incurred (\$ million)	Present Worth Factor at 10% Discount Rate	Present Worth of Estimated Costs (\$ million)	Levelized Unit Costs (\$ MTHM) ^(a)
	Total	Used Up in Period Analyzed					
Transportation							
Reactor to P/E facility (spent fuel elements)	Renewable Leases		1-40	93	9 779	905	18,500
P/E facility to repository	"						
Cladding hulls and miscellaneous	"		1-40	41	9 779	396	8,100
TRU LLW	"		1-40	Nil ^(d)	9 779	1	Nil
TRU and non-TRU ILW	"		1-40	4	9 779	40	800
HLW	"		10-49	19	4 147	78	1,600
P/E facility to shallow land burial (non-TRU LLW)	"		1-40	1	9 779	9	200
P/E to fuel recycle (UF ₆ and PuO ₂)	"		1-40	2	9 779	23	500
Subtotal Transportation						1,452	29,700
P/E facility							
Capital costs	40	40	0	4,230	1 000	4,230	86,600
Operating costs—fuel receiving and processing period			1-40	190	9 779	1,830	37,300
Operating costs—HLW storage and shipping only			41-49	7	0 127	1	Nil
Decommissioning costs			50	480	0 009	4	100
Subtotal P/E facility						6,065	124,000
Repository							
Capital costs	40	40	0	940	1 000	940	19,200
Operating costs—retrievable mode for cladding hulls, etc			1-5	71	3 791	267	5,500
Operating costs—nonretrievable mode for cladding hulls, etc			6-9	43	1 968	85	1,700
Operating costs—retrievable mode for HLW, nonretrievable mode for other materials			10-14	84	1 608	135	2,800
Operating costs—nonretrievable mode for all materials			15-49	58	2 539	147	3,000
Decommissioning costs			50	230	0 009	2	Nil
Subtotal Repository						1,576	32,200
Total costs						9 093	185,900
Credit for recycled UF ₆ ^(b)			1-40	(650)	9 779	(6,360)	(130,000)
Credit for recycled Pu ^(c)			1-40	(1,050)	9 779	(10,270)	(210,000)
Net costs (profit)						(7,537)	(154,000)

(a) Present worth of estimated costs divided by present worth of 5,000 MTHM/yr for 40 years, calculated as $5,000 \times 9.779 = 48,895$

(b) $5,000 \text{ MTHM/yr} \times 1,400 \text{ kg UF}_6 \text{ per MTHM} \times \$63 \text{ per kg} = \$650 \text{ million per year}$

(c) $5,000 \text{ MTHM/yr} \times 7,000 \text{ gr Pu per MTHM} \times \$30 \text{ per gr} = \$1,050 \text{ million per year}$

(d) Nil denotes an insignificant value.

Table 9-2
AVERAGE LEVELIZED ECONOMIC COSTS
CASE 1
(Based on January 1979 Dollars)

Cost Element	Useful Life (years)		Project Years in Which Costs Are Incurred	Estimated Costs Incurred (\$ million)	Present Worth Factor at 10% Discount Rate	Present Worth of Estimated Costs (\$ million)	Levelized Unit Costs (\$/MTHM)(a)
	Total	Used Up in Period Analyzed					
Transportation							
Reactor to AFR (spent fuel elements)	Renewable Leases		1-40	65	9 779	633	13,000
AFR to P/E facility (spent fuel elements)	"		10-49	54	4 147	226	4,600
AFR to shallow land burial (non-TRU LLW)	"		1-49	Nil(d)	9 906	2	Nil
P/E facility to repository	"						
Spent fuel elements	"		10-49	108	4 147	447	9,200
TRU LLW	"		10-49	Nil	4 147	Nil	Nil
TRU and non-TRU ILW	"		10-49	1	4 147	5	100
P/E facility to shallow land burial (non-TRU LLW)	"		10-49	Nil	4 147	1	Nil
Subtotal Transportation						1,314	26,900
AFRs							
Capital costs	40	40	0	2,160	1 000	2,160	44,200
Operating costs			1-49	45	9 906	446	9,100
Decommissioning costs			50	171	0 009	1	Nil
Subtotal AFRs						2,607	53,300
P/E facility							
Capital costs	40	40	9	1,017	0 424	431	8,800
Operating costs			10-49	93	4 147	387	7,900
Decommissioning costs			50	83	0 009	1	Nil
Subtotal P/E facility						819	16,700
Repository							
Unit one							
Capital costs	17	17	9	950	0 424	403	8,200
Operating costs—fuel retrievable mode			10-14	137	1 608	220	4,500
Operating costs—nonretrievable mode			15-26	84	1 794	15	3,100
Decommissioning costs			27	235	0 076	18	400
Unit two							
Capital costs	17	17	26	950	0 084	80	1,600
Operating costs in nonretrievable mode			27-43	84	0 673	57	1,200
Decommissioning costs			44	235	0 015	3	100
Unit three							
Capital costs	17	6	43	950	0 009(b)	9	200
Operating costs in nonretrievable mode			44-49	84	0 072	6	100
Decommissioning costs			61	235	0 002(c)	Nil	Nil
Subtotal Repository						947	19,400
Total costs						5,687	116,300

(a) Present worth of total costs divided by present worth of 5,000 MTHM/yr for 40 years, calculated as $5,000 \times 9.779 = 48,895$.

(b) Present worth factor for total capital costs incurred in 43rd year is 0 017. Allocation made to operating years 44-49 in ratio of present value of tonnage handled in those years to present value of tonnage handled in repository lifetime, years 44-60, which is.

$$0.017 \times \frac{0.072}{0.133} = 0.009$$

(c) Present worth factor for total decommissioning costs incurred in year 61 is 0.003. Allocated to years 44-49, as in Note (b) above.

$$0.003 \times \frac{0.072}{0.133} = 0.002$$

(d) Nil denotes an insignificant value.

9.4.3 Case 2

The flow of materials in Case 2 is the same as described for Case 1. Some differences in the method of treatment and encapsulation affect the costs of processing, shipping, and storage at the repository. The repository for Case 2 can accommodate the spent fuel received during a 19-year period, and three such repositories will be required. The third repository will be only partially filled at the end of the 40-year period of analysis. Also, as in Case 1, the first repository will be operated to permit the retrieval of emplaced materials during the first 5 years of disposal of each class of material. Estimated economic costs are shown in Table 9-3.

9.4.4 Case 3

The flow of materials in this case is also the same as in Case 1. Case 3 differs from the previous cases with respect to treatment and encapsulation of the spent fuel, which causes further differences in shipping, treatment, and storage costs. The repository design for this case can accommodate the spent fuel received over a period of 17 years, as in Case 1; therefore, three such repositories are required. The first repository will be operated to permit retrieval of emplaced materials during the first 5 years of disposal for each class of material. Estimated economic costs are shown in Table 9-4.

9.4.5 Comparison of Cases

The economic costs estimated for each of the four cases are summarized by major cost centers in Table 9-5. In addition to the costs calculated for the 10% discount rate used as a basis in the study, this table also shows the economic costs for three other discount rates which were analyzed for the sensitivity investigation discussed in Subsection 9.4.6. For the Reference Case, where uranium and plutonium energy contents are recovered from the spent fuel, the net economic value of these materials is credited against the costs of the case. The estimated credit values are shown in Table 9-1.

A value of \$30 per gram of fissile material was developed for recovered plutonium in the Reference Case. That figure is based on an indifference analysis of uranium and plutonium fuel cycle costs. In this analysis, uranium fuel cycle costs are equated to plutonium fuel cycle costs. At the point that plutonium and uranium fuel cycle costs are equivalent, the utility is indifferent to the fuel cycle used in the reactors. It was assumed in this study that the plutonium is recycled in LWRs and not in breeder reactors. Cost factors used in the analysis include:

- \$43 per lb of U_3O_8
- \$110 per separative work unit
- \$500 per kg heavy metal recycle fabrication
- \$5 per kg of uranium hexafluoride conversion
- \$125 per kg U for fuel fabrication
- 70 percent plant factors
- 10 percent discount rate

Owing to the extensive processing required for the recovery of the uranium and plutonium, the economic cost for the Reference Case is the highest of the four cases. However, these costs are more than offset by the value of the recovered materials, which results in a large negative cost, or profit, for the Reference Case.

Table 9-3
AVERAGE LEVELIZED ECONOMIC COSTS
CASE 2
(Based on January 1979 Dollars)

Cost Element	Useful Life (years)		Project Years in Which Costs Are Incurred	Estimated Costs Incurred (\$ million)	Present Worth Factor at 10% Discount Rate	Present Worth of Estimated Costs (\$ million)	Levelized Unit Costs (\$ MTHM) ^(a)
	Total	Used Up in Period Analyzed					
Transportation							
Reactor to AFR (spent fuel elements)	Renewable Leases		1-40	65	9 779	633	12,900
AFR to P/E facility (spent fuel elements)	,		10-49	54	4 147	226	4,600
AFR to shallow land burial (non-TRU LLW)	,		1-49	Nil ^(d)	9 906	2	Nil
P/E facility to repository	,						
Spent fuel	,		10-49	54	4 147	222	4,600
Cladding hulls and miscellaneous	,		10-49	16	4 147	66	1,400
TRU LLW	,		10-49	Nil	4 147	Nil	Nil
TRU and non-TRU ILW	,		10-49	1	4 147	5	100
P/E facility to shallow land burial (non-TRU LLW)	"		10-49	Nil	4 147	1	Nil
Subtotal Transportation						1,155	23,600
AFRs							
Capital costs	40	40	0	2,160	1 000	2,160	44,200
Operating costs			1-49	45	9 906	446	9,100
Decommissioning costs			50	171	0 009	1	Nil
Subtotal AFRs						2,607	53,300
P/E facility							
Capital costs	40	40	9	1,400	0 424	594	12,100
Operating costs			10-49	103	4 147	428	8,800
Decommissioning costs			50	117	0 009	1	Nil
Subtotal P/E facility						1,023	20,900
Repository							
Unit one							
Capital costs	19	19	9	940	0 424	399	8,200
Operating costs—fuel retrievable mode			10-14	108	1 608	174	3,600
Operating costs—nonretrievable mode			15-28	67	1 940	130	2,700
Decommissioning costs			29	230	0 063	14	300
Unit two							
Capital costs	19	19	28	940	0 069	65	1,300
Operating costs in nonretrievable mode			29-47	67	0 580	39	800
Decommissioning costs			48	230	0 010	2	Nil
Unit three							
Capital costs	19	2	47	940	0 002 ^(b)	2	Nil
Operating costs in nonretrievable mode			48-49	67	0 019	1	Nil
Decommissioning costs			67	230	— ^(c)	—	—
Subtotal Repository						826	16,900
Total costs						5,611	114,700

(a) Present worth of total costs divided by present worth of 5,000 MTHM/yr for 40 years calculated as $5,000 \times 9\ 779 = 48,895$

(b) Present worth factor for total capital costs incurred in 47th year is 0 011. Allocation made to operating years 48-49 in ratio of present value of tonnage handled in those years to present value of tonnage handled in repository lifetime, years 44-66, which is

$$0.011 \times \frac{0\ 019}{0\ 084} = 0\ 002$$

(c) Present worth factor for total decommissioning costs incurred in year 67 is 0 002. Allocated to years 48-49, as in Note (b) above

$$0\ 002 \times \frac{0\ 019}{0\ 084} = 0\ 0004$$

Considered negligible and not included in table

(d) Nil denotes an insignificant value

Table 9-4
AVERAGE LEVELIZED ECONOMIC COSTS
CASE 3
(Based on January 1979 Dollars)

Cost Element	Useful Life (years)		Project Years in Which Costs Are Incurred	Estimated Costs Incurred (\$ million)	Present Worth Factor at 10% Discount Rate	Present Worth of Estimated Costs (\$ million)	Levelized Unit Costs (\$ MTHM) ^(a)
	Total	Used Up in Period Analyzed					
Transportation							
Reactor to AFR (spent fuel elements)	Renewable Leases		1-40	65	9 779	633	13,000
AFR to P/E facility (spent fuel elements)	,		10-49	54	4 147	226	4,600
AFR to shallow land burial (non-TRU LLW)	„		1-49	Nil ^(d)	9 906	2	Nil
P/E facility to repository	,						
Spent fuel	,		10-49	192	4 147	797	16,300
Cladding hulls and miscellaneous	,		10-49	40	4 147	168	3,400
TRU LLW			10-49	Nil	4 147	Nil	Nil
TRU and non-TRU ILW	,		10-49	3	4 147	12	300
P/E facility to shallow land burial (non-TRU LLW)	,		10-49	Nil	4 147	<u>1</u>	<u>Nil</u>
Subtotal Transportation						1,839	37,600
AFRs							
Capital costs	40	40	0	2,160	1 000	2,160	44,200
Operating costs			1-49	45	9 906	446	9,100
Decommissioning costs			50	171	0 009	<u>1</u>	<u>Nil</u>
Subtotal AFRs						2,607	53,300
P/E facility							
Capital costs	40	40	9	3,000	0 424	1,272	26,000
Operating costs			10-49	233	4 147	967	19,800
Decommissioning costs			50	333	0 009	<u>3</u>	<u>100</u>
Subtotal P/E facility						2,242	45,900
Repository							
Unit one							
Capital costs	17	17	9	1,240	0 424	526	10,800
Operating costs—fuel retrievable mode			10-14	283	1 608	455	9,300
Operating costs—nonretrievable mode			15-26	147	1 794	264	5,400
Decommissioning costs			27	305	0 076	23	500
Unit two							
Capital costs	17	17	26	1,240	0 084	104	2 100
Operating costs in nonretrivable mode			27-43	147	0 673	99	2,000
Decommissioning costs			44	305	0 015	5	100
Unit three							
Capital costs	17	6	43	1,240	0 009 ^(b)	11	200
Operating costs in nonretrievable mode			44-49	147	0 072	11	200
Decommissioning costs			61	305	0 002 ^(c)	<u>1</u>	<u>Nil</u>
Subtotal Repository						1,499	30,600
Total costs						8,137	167,400

(a) Present worth of total costs divided by present worth of 5,000 MTHM/yr for 40 years, calculated as $5,000 \times 9\ 779 = 48,895$.

(b) Present worth factor for total capital costs incurred in 43rd year is 0 017. Allocation made to operating years 44-49 in ratio of present value of tonnage handled in those years to present value of tonnage handled in repository lifetime years 44-60, which is

$$0.017 \times \frac{0.072}{0.0133} = 0.009$$

(c) Present worth factor for total decommissioning costs incurred in year 61 is 0 003. Allocated to years 44-49 as in Note (b) above.

$$0.003 \times \frac{0.072}{0.0133} = 0.002$$

(d) Nil denotes an insignificant value

Table 9-5
AVERAGE LEVELIZED ECONOMIC COSTS FOR
FOUR CASES OF SPENT FUEL DISPOSAL ^(a)
(\$/MTHM)

Cases	Discount Rate (Percent)			
	0	5	10	15
Reference Case				
Transportation	32,000	30,800	29,700	29,200
P/E facility	61,865	87,200	124,000	164,800
Repository	20,725	24,500	32,200	41,300
Total cost (rounded)	114,590	143,000	186,000	235,000
Credit value of recycled UF ₆	(130,000)	(130,000)	(130,000)	(130,000)
Credit value of recycled PuO ₂	<u>(210,000)</u>	<u>(210,000)</u>	<u>(210,000)</u>	<u>(210,000)</u>
Net profit (rounded)	(225,000)	(198,000)	(154,000)	(105,000)
Case 1				
Transportation	45,600	34,100	26,900	22,300
AFR	22,700	34,900	53,300	74,100
P/E facility	24,100	19,800	16,700	14,000
Repository	<u>32,100</u>	<u>25,400</u>	<u>19,400</u>	<u>15,400</u>
Total cost (rounded)	125,000	114,000	116,000	126,000
Case 2				
Transportation	38,000	29,200	23,600	20,100
AFR	22,700	34,900	53,300	74,100
P/E facility	28,200	24,000	20,900	17,900
Repository	<u>26,700</u>	<u>22,000</u>	<u>16,900</u>	<u>13,800</u>
Total cost (rounded)	116,000	110,000	115,000	126,000
Case 3				
Transportation	70,800	50,400	37,600	29,500
AFR	22,700	34,900	53,300	74,100
P/E facility	63,300	53,000	45,900	39,000
Repository	<u>51,000</u>	<u>40,000</u>	<u>30,600</u>	<u>24,200</u>
Total cost (rounded)	208,000	178,000	167,000	167,000

(a) Calculated from January 1979 prices with 10 percent discount rate and 40-year study period.

Cases 1, 2, and 3 have no recovered materials and are comparable in terms of achieving permanent disposition of the spent fuel in geologic storage. Cases 1 and 2 have almost the same total costs, while Case 3 is substantially more costly owing to greater costs for all elements except the AFRs.

9.4.6 Sensitivity to Discount Rate

As noted above, the basic economic analysis was made with a 10 percent discount rate with results detailed in Tables 9-1 to 9-4. Similar economic calculations were also made for each case, using discount rates of 15, 5 and zero percent. The average levelized economic costs per MTHM for each case for the four discount rates are tabulated in Table 9-5 and plotted graphically in Figure 9-2. Smooth curves were fitted to the plotted points to show the approximate variations of economic costs over a wide range of discount rates.

As Figure 9-2 reveals, changing the discount rate does not alter the relative costliness of the cases in any significant way except to make Case 3 more costly than the Reference Case at low rates. Transportation costs are significantly higher in Case 3 than in the other cases, but since they are incurred over a long period, those in the distant future become relatively more important as the discount rate is lowered. Repository capital and operating costs are also relatively high in Case 3. Since construction of the repositories is completed in project years 9, 26, and 43, the present value of their costs is highly sensitive to the discount rate. The operating costs extend over a long period and are similarly affected.

Figure 9-2.
System Average Levelized Economic Cost \$/MTHM
for Four Cases at Various Discount Rates

9.5 APPROXIMATE CHARGES IN COMMERCIALLY OPERATED SYSTEMS

It is of interest to compare the four cases as if the facilities in each case were built and operated by a private enterprise charging a fee per MTHM for spent fuel disposal. Such an enterprise would have to cover its financial costs plus an acceptable return on its financial investment. It would pay taxes that were excluded from the economic analysis just described, and its acceptable return on investment, after taxes, would be higher than the 10 percent used as the basis for Tables 9-1 through 9-4.

The following assumptions are used to determine charges in a privately owned spent fuel disposal system:*

- 15 percent return on investment, after taxes
- 100 percent equity capital
- 20-year life of facilities for financial analysis
- 50 percent overall income tax rate
- 2.5 percent property tax on original capital investment
- 0.5 percent insurance premium on original capital investment
- Straight line depreciation for tax purposes
- Working capital equal to 6 months' operating costs

The assumption of 20 year life for financial analysis is interpreted to mean that the enterprise bases its evaluation on 20 years of spent fuel receipts for which a uniform charge per metric ton of heavy metal is levied. The physical facilities in the disposal system last longer than 20 years, and some of the fuel received during the 20 years cannot be finally disposed of until the 29th year, after which facilities are decommissioned and the system shut down. It is assumed that the private enterprise evaluates the investment opportunity at the outset on the basis of no spent fuel receipts after the 20th year and termination of the enterprise by dismantling and decommissioning by the end of the 30th year. The facility will collect sufficient revenues in the first 20 years to pay the remaining costs.

Cash flow analyses reflecting the foregoing assumptions are conducted for each of the four systems. The approximate uniform charge per metric ton of heavy metal needed to provide a return of 15 percent of invested capital, after taxes, is determined for each case by trial and error. These charges are summarized below.†

Ref. Case	\$ 95,000/MTHM (with U, Pu credits)
Ref. Case	\$435,000/MTHM (without U, Pu credits)
Case 1	\$350,000/MTHM
Case 2	\$330,000/MTHM
Case 3	\$480,000/MTHM

* These assumptions were recently used by ORNL for estimating costs for fuel refabrication plants (4)

† Owing to the characteristics of the flow of funds in the enterprise, there are times when not all of the capital is required continuously. Such unutilized funds have been assumed to be deposited in risk-free accounts earning interest at 5 percent per annum until the funds are again required for operations. The net after tax return of the venture has been calculated at 15 percent per annum. Potential reductions in income taxes in the last 10 years due to operating losses, are not included in the analysis.

This so-called typical commercial economic analysis is not considered to be an appropriate basis for comparing or selecting a preferred case. Nor is it meant to imply that a system wholly privately owned is likely, or even possible. It is provided simply for information and as reference material for the interested reader.

REFERENCES FOR SECTION 9

- (1) "Preliminary Estimates of the Charge for Spent Fuel Storage and Disposal Services," U.S. Department of Energy, DOE/ET-0055, July 1978.
- (2) "Sensitivity of the Federal Fee for Managing Spent Fuel to Financial and Logistical Variations," Battelle Memorial Institute, Pacific Northwest Laboratory, June 1978.
- (3) "Economics of National Waste Terminal Storage, Spent Fuel Pricing Study," Y/OWI/SUB-78/42512/2, TRW Energy Systems Planning Division, May 1978.
- (4) "Barnwell Nuclear Fuels Plant Applicability Study," Volume 3 Appendices, DOE/ET-0040/3, U.S. Department of Energy, March 1978.

Section 10
STUDY VARIATIONS

TABLE OF CONTENTS

Section		Page
10.1	Approach to the Study Variations	10-1
10.2	Time Factor Variations	10-3
10.2.1	Age of Spent Fuel Leaving Reactor	10-3
10.2.2	Age of Spent Fuel at Time of Emplacement in the Repository	10-4
10.2.3	Retrievability Time Period Options	10-8
10.3	Canister Design Variations	10-14
10.3.1	Canister Void Fillers	10-14
10.3.2	Overpacks and Thicker Wall Canisters	10-19
10.3.3	Number of Spent Fuel Assemblies per Canister	10-20
10.4	Repository Design Variations	10-23
10.4.1	Room Cooling During Extended Retrievability Period	10-23
10.4.2	Horizontal/Diagonal Emplacement of Canisters	10-24
10.4.3	Combination Spent Fuel/HLW Repository	10-25
10.4.4	Non-Salt Repository	10-26
10.5	Colocation Options	10-30
10.5.1	Colocation of 3 AFRs and 1 P/E Facility	10-30
10.5.2	Colocation of 1 P/E Facility and 1 Repository at 3 Sites	10-34
10.5.3	Colocation of 3 AFRs, 1 P/E Facility, and 1 Repository	10-37
10.6	Spent Fuel Disassembly Option	10-42
10.6.1	Design Impact	10-42
10.6.2	Assessment Area Impact	10-43
10.7	Application of Study Variations	10-46
	References for Section 10	10-47

Section 10

STUDY VARIATIONS

10.1 APPROACH TO THE STUDY VARIATIONS

To define the spent fuel disposal systems of the various study cases, it was necessary to make many assumptions. These assumptions are tabulated in Appendix A. Since the design of the spent fuel disposal system is at a preconceptual stage for this study and since system optimization studies, as such, have not been performed, some of the assumptions represent arbitrary choices.

The concept of study variations (or sensitivity studies) is introduced into the SFDS so that some of the alternative choices can be examined. The study variations are divided in this section into the following categories:

- Time factor variations
- Canister design variations
- Repository design variations
- Colocation options
- Spent fuel disassembly option

Each of the variations is described, with respect to the established study basis for Case 1, in the following terms:

- **Scope.** The nature of the variation as it relates to the conditions assumed for the spent fuel disposal system of Case 1 is described
- **Design impact.** The principal impact of the variation on the design of the facilities and transportation steps of Case 1 is analyzed
- **Assessment area impact.** The impact of each variation, with respect to each of the six assessment areas, is analyzed

The following observations or conclusions can be made as a result of this analysis of study variations:

- (1) Colocation options (10.5) have inherent advantages from both the Safeguards and Radiological Impact perspectives. They considerably limit the opportunities for theft and sabotage, and they also reduce radiation exposure to personnel by elimination of onsite and offsite handling operations.
- (2) Canister design variations (10.3) — especially void filler options — and the spent fuel disassembly option (10.6) have merit and warrant further consideration at the time a specific spent fuel disposal case is designed or implemented.
- (3) Of the repository design variations (10.4) horizontal/diagonal emplacement of canisters warrants consideration at the time a repository design is being prepared. The evaluation should be made then almost entirely on an economic trade-off basis.
- (4) Room cooling during extended retrievability is, of course, coupled with retrievability periods longer than five years. It should be more fully assessed if longer retrievability periods are seriously considered.

- (5) Time factor variations (10.2) are more hypothetical than practical. When a real spent fuel or waste disposal program is planned, clear-cut time values will be known (or specified) about age of fuel leaving reactors and the time of emplacement, as well as retrievability time periods. The analyses in Subsection 10.2 describe in general terms the anticipated impacts of these variations.
- (6) Combinations of some of the study variations can have important impacts on a future spent fuel disposal system, which were not addressed in this section. They should be taken into account when planning a final disposal system. Colocation and time factor variations, for example, when applied in certain ways, can have significant combined effects not obvious in the individual analyses.

10.2 TIME FACTOR VARIATIONS

10.2.1 Age of Spent Fuel Leaving Reactor

The age of spent fuel leaving the reactor storage pools is assumed as 1 year for the Spent Fuel Disposal Study (SFDS). This assumption is based on the design basis storage capacity of spent fuel storage pools at the reactors. This 1-year age assumption may be too short under present-day conditions. Many of the reactors in operation and under design have increased, or are planning to increase, their storage capacity for spent fuel.

In this variation, a study is made of the effect of having spent fuel leave the reactor storage pools after 5 years and after 10 years. The expansion of storage capacity at the reactors will result in increased capital and operating costs to the operator. Since reactors are not within the scope of the SFDS, the cost impact of increased spent fuel storage capacity at the reactors would be to defer cost allocations for away-from-reactor storage facilities (AFRs).

10.2.1.1 Design Impact. The impact of this variation on the design of various facilities and transportation systems is discussed in the following subsections and summarized in Table 10-1.

AFR. Since the age of spent fuel at emplacement in the repository is assumed to be 10 years in the SFDS, spent fuel is stored at the AFR for 9 years. If the fuel is stored for longer periods at the reactor, it reduces the storage time requirements at the AFR. Longer storage periods, 5 and 10 years, at the reactor reduce the thermal load on the AFRs. The major impact of the variations is the reduced number of AFRs required, with the corresponding reduction in capital, operating, and decommissioning costs (Table 10-1).

Table 10-1
DESIGN IMPACT SUMMARY
AGE OF FUEL LEAVING THE REACTOR

Item	Case 1	Variations	
	1 Year	5 Years	10 Years
AFR			
Age of fuel arriving at the AFR, yr	1	5	—
Age of fuel leaving the AFR, yr	10	10	—
Number of AFRs, 5,000 MTHM capacity each	9	5	0
AFR throughput, MTHM/yr each	555	1,000	0
P/E facility	Base	No change	No change
Repository	Base	No change	No change
Transportation			
Distance from reactor, miles	400	550	700
Spent fuel age, yr	1	5	10
Distance from AFR, miles	700	700	—
Spent fuel age, yr	10	10	—
Distance from P/E facility	1,100	1,100	1,100
Encapsulated fuel age, yr	10	10	10

P/E facility. This variation has no impact on the design of the P/E facility.

Repository. This variation has no impact on the design of the repository.

Transportation. If the spent fuel is stored for 5 years at the reactors, 5-year-old fuel is shipped from the reactors to the AFRs instead of the 1-year-old fuel. The average distance between the reactors and the AFR also changes owing to the reduced number of AFRs in this variation. It is possible to ship more 5-year-old spent fuel per shipment than 1-year-old spent fuel, which results in overall transportation cost savings.

If the spent fuel is stored for 10 years at the reactors, no AFRs are required. Therefore, the spent fuel is shipped directly to the P/E facility instead of shipping the 1-year-old fuel first from the reactors to the AFR, and then the 10-year-old fuel from the AFR to the P/E facility. This approach results in an overall reduction in the total transportation cost and the cask fleet required. The impact of these variations on the transportation cost is shown in Table 10-2.

10.2.1.2 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in this subsection and summarized in Table 10-2.

Technical feasibility. This variation does not affect the technical feasibility of Case 1.

Safeguards assessment. This variation does not significantly affect the safeguards assessment of the individual facilities and the transportation system. In the case of transportation from reactors, the specific activity of the spent fuel is slightly lower, but the total activity per shipment may still be of the same order of magnitude. Within the accuracy of the SFDS assessment, the risk from sabotage of spent fuel during transportation is assumed to be the same even though slightly fewer shipments will be required. The assessment risk from theft is not affected, as the materials involved in this variation are the same as for Case 1.

Criticality assessment. No impact.

Radiological assessment. This variation does not influence the radiological assessment of the individual facilities and the transportation system. In the case of transportation from reactors, the specific activity of the spent fuel is slightly lower, but the total activity per shipment may be of the same order of magnitude. Because there will be slightly fewer shipments, the radiological impact will be slightly less than for Case 1. When the spent fuel is stored for 10 years in the reactor, AFRs are not required. This will result in a decrease in the overall radiological impact from Case 1.

Retrievability assessment. No impact.

Economic assessment. The AFR and the transportation step from the reactor affect the overall economic assessment (expressed as \$/MTHM) of Case 1 as shown in Table 10-2.

10.2.2 Age of Spent Fuel at Time of Emplacement in the Repository

The age of spent fuel at the time of emplacement in the repository is assumed to be 10 years for the SFDS. The impact of changing this from 10 years to 5 years, and then to 25 years, is studied in this variation.

10.2.2.1 Design Impact. The impact of this variation on the design of facilities and the transportation systems of Case 1 is discussed in the following subsections and summarized in Table 10-3.

Table 10-2
ASSESSMENT IMPACT SUMMARY
AGE OF FUEL LEAVING THE REACTOR

Item	Case 1	Variations	
	1 Year	5 Years	10 Years
Technical feasibility	Base	No change	No change
Safeguards assessment	Base	No change	No change
Criticality assessment	Base	No change	No change
Radiological assessment	Base	Decrease ^(a)	Decrease
Retrievability assessment	Base	No change	No change
Economic assessment			
AFR, \$/MTHM	53,300	30,700 ^(b)	— ^(c)
P/E facility, \$/MTHM	16,700	16,700	16,700
Repository, \$/MTHM	19,400	19,400	19,400
Transportation from reactor, \$/MTHM	12,900	8,200	4,600
From AFR, \$/MTHM	4,700	4,700	—
From P/E facility, \$/MTHM	9,300	9,300	9,300
Total, \$/MTHM	116,000	89,000	50,000

(a) Any decrease would be slight.

(b) This includes cost of 5 AFRs only. The increased cost of storage at the reactors for 5 years is not included in this assessment.

(c) There are no AFRs required in this case. The increased cost of storage at the reactors for 10 years is not included in this assessment.

Table 10-3
DESIGN IMPACT SUMMARY
AGE OF SPENT FUEL AT TIME OF EMPLACEMENT IN THE REPOSITORY

Item	Case 1	Variations	
	10 Years	5 Years	25 Years
AFR			
Age of fuel arriving at the AFR, yr	1	1	1
Age of fuel leaving the AFR, yr	10	5	25
Number of AFRs, 5,000 MTHM each	9	4	24
AFR throughput, MTHM/yr each	555	1,250	210
P/E facility	Base	No change	No change
Repository			
Heat content, kW/can.	0.55	0.93	0.38
Thermal loading, kW/acre	60	101	41
Pillar spacing, ft	132	132	130
Canister pitch, in.	64	64	64
Repository life, estimated yr	17	18	18
Transportation			
From AFR			
Spent fuel shipment, PWR/BWR	14/34	11/27	Not Determined
From P/E facility			
Encapsulated fuel shipment, PWR/BWR	9/27	8/24	Not Determined

AFR. Since the age of spent fuel at emplacement in the repository is assumed to be 10 years in Case 1, the spent fuel is stored for 9 years at the AFR, with 9 AFRs required for Case 1. When the age of spent fuel at emplacement in the repository is changed to 5 years and 25 years, the spent fuel is stored for 4 years and 24 years, respectively, at the AFRs, and the number of AFRs required is 4 and 24, respectively. This results in different capital, operating, and decommissioning costs for the two variations.

P/E facility. This variation has no impact on the design of the P/E facility.

Repository. The thermal characteristics of the waste canisters affect the layout, extraction ratios, and thermal loading of the repository. A detailed analysis of the thermal aspects of the repository design is described and referenced in Appendix D (Subsection D.2.1). The age of spent fuel at emplacement affects the heat content of the spent fuel canisters. The results of the analysis are summarized below:

- The maximum temperature of the spent fuel cladding is well below 200°C, which is the limiting near-field condition for emplacing 5-year-old and 25-year-old spent fuel
- The areal thermal loading must be limited to 101 kW/acre for spent fuel 5 years old at emplacement (0.93 kW/can.) and to 41 kW/acre for spent fuel 25 years old at emplacement (0.38 kW/can.) as compared to 60 kW/acre for spent fuel 10 years old at emplacement (0.55 kW/can.). These loadings provide essentially the same MTU/acre as in Case 1. This ensures that the total energy deposited in the formation over a long time frame is nearly the same and, therefore, results in similar surface uplifts as for Case 1 (far-field effect)

- The optimum extraction ratios, pillar widths, room dimensions, and assemblies per canister are the same for both the 5-year-old and 25-year-old spent fuel at emplacement. For the 25-year-old spent fuel at emplacement option, the pillar spacing is, however, reduced from 132 to 130 feet. The geometry of emplacement is assumed to be identical for both options.

The repository capital and operating costs for the two options of this variation are the same as for Case 1.

Transportation. The design and capacity of the cask for transporting spent fuel depend on the age of the spent fuel. In the variations, the transported fuel is 5 years old and 25 years old; in Case 1, it is 10 years old. The casks for transporting spent fuel from the AFR to the P/E facility and encapsulated fuel from the P/E facility to the repository are, therefore, affected. In general, the quantity of spent fuel per shipment is less for 5-year-old fuel and more for 25-year-old fuel, as compared to that for 10-year-old fuel. Thus, the overall transportation cost increases for 5-year-old fuel and decreases for 25-year-old fuel, as compared to Case 1.

10.2.2.2 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections and summarized in Table 10-4.

Table 10-4
ASSESSMENT IMPACT SUMMARY
AGE OF SPENT FUEL AT TIME OF EMPLACEMENT
IN THE REPOSITORY

Item	Case 1	Variations	
	10 Years	5 Years	25 Years
Technical feasibility	Base	No change	No change
Safeguards assessment	Base	No change	No change
Criticality assessment	Base	No change	No change
Radiological assessment	Base	Slightly higher	Slightly lower
Retrievability assessment	Base	Less desirable	More desirable
Economic assessment			
AFR, \$/MTHM	53,300	23,800	Not determined
P/E facility, \$/MTHM	16,700	16,700	Not determined
Repository, \$/MTHM	19,400	30,700	Not determined
Transportation from reactor, \$/MTHM	12,900	18,500	Not determined
From AFR, \$/MTHM	4,700	9,500	Not determined
From P/E facility, \$/MTHM	9,300	16,800	Not determined
Total \$/MTHM	116,000	116,000	

Technical feasibility. This variation does not affect the technical feasibility of Case 1

Safeguards assessment. This variation does not affect the safeguards assessment of the individual facilities and the transportation system. In the case of transportation from the AFR and the P/E facility, the specific activity of the spent fuel is slightly lower for 25-year-old fuel and slightly higher for 5-year-old fuel, but, the total activity per shipment in each case is still of the same order of magnitude. Therefore, the safeguards assessment of transportation is not affected, within the accuracy of the SFDS assessment.

Criticality assessment. No impact

Radiological assessment. This variation affects the radiological assessment of the individual facilities and the transportation system as discussed below:

- In the case of transportation from the AFR and the P/E facility, the specific activity of the spent fuel is slightly lower for 25-year-old fuel, and slightly higher for 5-year-old fuel, but the total activity per shipment is of the same order of magnitude. However, because the number of shipments may increase with 5-year-old fuel, the potential overall population exposure would probably increase slightly. The reverse is true for 25-year-old fuel.
- The storage of spent fuel for 5 or 25 years in the AFR does not in itself change the overall radiological assessment for Case 1, within the accuracy of the SFDS assessments.

Retrievability assessment. Retrievability may be slightly easier to accomplish, with the spent fuel being emplaced at the lower temperature (greater age of the fuel). Therefore, the 25-year-old fuel is slightly more desirable and the 5-year-old fuel is slightly less desirable from a retrievability point of view.

Economic assessment. The overall impact on the economic assessment (expressed as \$/MTHM) for the 5-year variation is shown in Table 10-4. Similar costs for the 25-year case were not specifically determined. AFRs, because so many are required, will be the predominant influence. The bottom line number may be on the order of twice that of Case 1.

10.2.3 Retrievability Time Period Options

10.2.3.1 Scope. Case 1 utilizes a 5-year period during which emplaced fuel elements are retrievable. As a measure of sensitivity of the study to this Case 1 time period, two additional time periods are considered as a part of this study variation. These periods, 10 and 25 years, represent reasonable extensions of time that may eventually be selected as retrievable time periods. The active operating life of the Case 1 repository (i.e., the time required to fill it to design capacity) is between 10 and 17 years, depending on the retrievability time period — which varies from 5 to 25 years.

The 25-year retrievability option requires the repository to be designed and maintained in an operational mode for up to 25 years from the date the first canister is emplaced. If a decision to retrieve the spent fuel were made near the end of the 25-year retrievability period, it could take another 10 years to recover all the stored canisters before the mine could be backfilled and decommissioned. As a result, with 25-year retrievability, the repository, including the aboveground facilities, shaft, and ventilation system, etc., would be designed for an operating life of about 35 years, compared to the 10 years required for filling the repository. For the 10-year retrievability period, the corresponding repository design life would be about 20 years.

At the end of the 10- and 25-year retrievability period, the empty spaces left, due to reduced repository loading of the 48 and 36 kW/acre during the retrievability period, will be filled to achieve an average loading of 60 kW/acre for the variations (same as that for Case 1).

10.2.3.2 Design Impact. The impact of this variation on the design of various facilities and transportation systems of Case 1 is discussed in the following subsections and summarized in Table 10-5

Table 10-5
DESIGN IMPACT SUMMARY
RETRIEVABILITY TIME PERIOD OPTIONS

Item	Case 1	Variations	
	5 Years	10 Years	25 Years
AFR	Base	No impact	No impact
P/E facility	Base	Thicker canisters	Thicker canisters
Repository			
Retrievability period, yr	5	10	25
Repository loading density			
During retrievability period, kW/acre	60	48	36
After retrievability period, kW/acre	60	60	60
Life of first repository, yr	17	17	10
Active life of facilities at repository, yr	17	20	35
Transportation	Base	No impact	No impact

Effect on AFR. No impact

Effect on P/E facility. This variation does not impact the design of the P/E facility, the design of the canisters is affected by the retrievability time period selected. Figure I-1 (in Appendix I of Volume 3) shows the added thickness of material (sacrificial thickness) required for carbon steel, stainless steel, inconel, and glass-on-steel for various time periods up to 25 years. The thickness of the canisters increases with the increase in retrievability time periods to provide for increased corrosion allowances. This increases the operating cost of the P/E facility.

Effect on repository. This variation affects the design of the repository in the following areas:

- Repository loading for Case 1 with a 5-year retrievability period is 60 kW/acre. The repository loading is reduced to 48 and 36 kW/acre for 10- and 25-year retrievability periods, respectively, and is expected to result in the same near-field effects as those experienced with a 60 kW/acre loading density in 5 years (there is no room cooling planned during the basic retrievability period).
- For the 10-year retrievability variation, the spaces for canister emplacement are drilled for an average loading density of 60 kW/acre. During the first 10 years, these spaces are only partially utilized to maintain a repository loading of 48 kW/acre. At the end of 10 years, these empty spaces will be filled and the loading after 10 years is continued at 60 kW/acre.
- For the 25-year retrievability variation, storage wells for canister emplacement are drilled for an average loading density of 60 kW/acre. The spaces are, however, utilized partially to achieve a repository loading of 36 kW/acre. The repository will be filled in about 10 years, but will be maintained in a standby status for another 15 years. At the end of the 25-year retrievability period, the empty spaces in the repository will be filled to achieve an average loading density of 60 kW/acre.

- The average active life of all above and below ground facilities of the repository is required to be 20 years and 35 years, respectively, to accommodate the retrievability operation, as compared to 17 years for Case 1 with 5-year retrievability
- Extra surveillance and monitoring are required during longer than 5-year retrievability options. Also, frequent checks and dry runs may be required to demonstrate the retrievability of the waste especially during the 25-year retrievability period
- Additional equipment may be required for removing canisters from salt for longer retrievability periods of 10 or 25 years. Some additional facilities for repacking of damaged canisters may also be required

Effect on transportation. No impact.

10.2.3.3 Assessment Area Impact. The impact of this variation with respect to each assessment area is discussed in the following subsections and summarized in Table 10-6.

Table 10-6
ASSESSMENT IMPACT SUMMARY
RETRIEVABILITY TIME PERIOD OPTIONS

Item	Variations	
	10 Years	25 Years
Technical feasibility	No impact	No impact
Safeguards assessment	No impact	No impact
Criticality assessment	No impact	No impact
Radiological assessment	Minor increase	Greater increase
Retrievability assessment	(See Table 10-7)	
Economic assessment	Not determined	

Technical feasibility. The longer retrievability periods require more complex retrieval equipment and procedures. The impact of these periods is discussed under retrievability assessment. This variation has no other impact on the technical feasibility.

Safeguards assessment. This variation has no significant impact on the safeguards assessment of individual facilities. Stricter physical protection, material control, and accounting requirements may be imposed during retrievability periods, especially the 25-year retrievability period. The overall assessment of safeguards for Case 1 is not, however, significantly affected.

Criticality assessment. Since the same form of nuclear material is emplaced in the repository as in Case 1, this variation has no impact on the retrievability assessment of Case 1.

Radiological assessment. The radiation exposure to the operating personnel may increase slightly, as compared with Case 1, during receiving and emplacement of additional encapsulated fuel from offsite to increase the loading density to 60 kW/acre of the repository from 48 for 10-year retrievability and 36 kW/acre (for 25-year retrievability). The integrated radiological impact of maintaining the repository for longer periods of time, however, will increase but is not expected to affect the radiological assessment significantly.

Retrievability assessment. The assessment impact of this retrievability time period extension to 10 and 25 years utilizes the assessment basis approach presented in Section 8.

The nine retrievability subsystems selected are as follows:

- Sleeve integrity
- Canister integrity
- Transfer/overpack cell
- Decontamination/corrosion removal cell
- Repository ventilation system
- Radiation safety during retrieval
- Repository monitoring and surveillance system
- Repository loading density
- Repository room closure considerations

The selection of these subsystems offers a direct comparison with Case 1 on the basis of degree of technical development and implementation, A_m . The values for technical development and implementation for the retrievability subsystems are based upon the nature and scale of previous demonstrations. One exception to this is in the case of the radiation safety subsystem. In this subsystem, the aspects of the degree of protection to the public and the degree of protection to the facility operators are introduced into the degree of technical development and implementation. A postulated scale for A_m (1 through 5) is presented in Table 8-1. This scale for degree of technical development and implementation for subsystems ranges from a value of A_m of 1 for successful operating history, to 5 for design only, based on analysis. The assignment of appropriate index or number values A_m for retrievability subsystems is presented in Table 10-7. The supporting rationale for the assignment of each number value is presented in the following paragraphs.

Sleeve integrity ($A_m = 4$). The principal functions of the sleeve are to reduce corrosion of the waste container, and to allow in-situ tests on the canister and/or sleeve during the retrievability period as discussed in Appendix D.

Although short term exposure of material under a simulated repository conditions has been done, longer exposure periods have been assigned a lower level of development. (Value of 3 in Case 1 and value of 4 in this variation.)

Canister integrity ($A_m = 4$). Integrity of the canister is designed to provide confinement for the 10 and 25 year storage and retrieval life. The criteria for a retrievable canister design include canister environment (external), structural considerations, canister materials, and internal corrosion.

The canister is currently envisioned as being fabricated from standard commercial pipe with appropriate end closures. The current view is that this effort will be within existing engineering design and analysis technology. However, longer exposures of the canister to potential salt environments, caused by salt or brine penetrating between sections of the sleeve, make canister integrity slightly less demonstrated than in Case 1 (hence, $A_m = 4$ rather than 3).

Transfer/overpack cell ($A_m = 3$). The proposed repository design incorporates transfer/overpack cells in the subsurface as well as in the surface facilities. Although there is a slightly greater probability of canister failure or entrapment occurring as storage time increases, A_m was given a value of 3, as it was in Case 1.

Decontamination cell ($A_m = 4$). Contamination and corrosion are removed in the decontamination cell. The removal of radioactive products of corrosion is common to both cases, although penetration of corrosion increases with time. A slightly "less demonstrated" rating is given to this variation compared to Case 1.

Table 10-7
DEGREE OF TECHNICAL DEVELOPMENT
AND IMPLEMENTATION
 A_m

Subsystem	Case 1	Study Variation
Sleeve integrity	3	4
Canister integrity	3	4
Transfer/overpack cell	3	3
Decontamination cell	3	4
Repository ventilation system	1	2
Radiation safety during retrieval	1	3
Repository monitoring and surveillance system	2	2
Repository loading density	1	1
Repository room closure considerations	2	2

Note: A minimum score of 1 represents a very high degree of development and implementation of a subsystem. A maximum score of 5 represents a low degree of development and implementation.

Repository ventilation system ($A_m = 2$). The ventilation of repositories has been studied by Kaiser Engineers^{(5),(6)} and Stearns-Roger^{(4),(7),(8),(9),(10)} and the detailed design of ventilating and airconditioning systems is well established in the nuclear industry. While no major differences exist in the state of development for this study variation, the time required to cool will increase somewhat as temperatures will be increasing over the first 25 years.

Radiation safety ($A_m = 3$). The radiation safety considerations for the SFDS are presented in Section 7. 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 which are encountered during the storage operation would occur during retrieval if the canister is intact. As the probability of having failed canisters increases with time, a higher dose commitment may occur during retrieval of the longer stored canisters.

Repository monitoring and surveillance system ($A_m = 2$). Devices installed to monitor the repository and collect experimental data during the retrievability period are the same, regardless of the time of retrievability.

Repository loading density ($A_m = 1$). SAI has estimated the allowable loading density for emplacement of 10-year-old waste and retrieval of 20-year-old waste to be 48 kW/acre (Appendix D, Subsection D.2.1). They have estimated that emplacement of 10-year-old waste and retrieval of 35-year-old waste should not exceed a loading density of 36 kW/acre at time of emplacement. No differences in development or implementation exists between Case 1 and

these variations, since the only change would be to emplace more wastes in the storage rooms when thermal loading can be increased to 60 kW/acre.

Repository room closure considerations ($A_m = 2$). As the retrievability period is extended, thermal loading is reduced; hence, either Case 1 or its variation should result in equivalent room closures over their retrieval periods.

Economic assessment. The major impact of the longer retrievability time periods is the operating cost of the repository. For the 10-year and 25-year retrievability periods the operating cost will increase due to the following:

- The higher level of surveillance and monitoring required during the longer retrievability period
- The cost associated with maintaining the repository in a stand-by operating mode for a longer period of time after the repository is filled

10.3 CANISTER DESIGN VARIATIONS

10.3.1 Canister Void Fillers

As indicated in the description of the P/E facility for Case 1, the canisters containing spent fuel are assumed to be filled with helium gas. The object of this variation is to consider and compare several alternatives to the helium filler in these canisters.

The use of fillers has been promoted to enhance some of the properties of the package, such as higher heat transfer coefficient, the ability to further inhibit nuclide migration, and to provide added resistance to accidents, including criticality. Further calculations are required to predict the effect of filler selection on the fuel temperature and the near- and far-field effects on the repository. Compatibility of the filler with the fuel assembly cladding has to be investigated. Those investigations are beyond the scope of the SFDS; therefore, only a brief description of their probable affects are presented in this section.

10.3.1.1 Types of Fillers Considered. The following fillers are considered in this study variation:

Gases	Solids	Granular
Air	Lead	Lead shot
Nitrogen	Aluminum	Sand
Argon	Glass	Glass frit

Gaseous Fillers. Gas fillers in most cases are relatively less expensive than solid fillers and are easier to handle. Their disadvantages are that they offer no physical stability, they do not contribute to overall canister strength, and they provide relatively little heat transfer.

Solid Monolithic Fillers. Metal encapsulation of spent fuel assemblies has been demonstrated on a full-scale basis with spent fuel assemblies in the United States, Russia, France, and Canada.⁽¹⁾ The packaging concept consists of placing an assembly in the canister, filling the entire canister with molten metal, and then sealing and cooling the canister.

The advantages of using a low-melting metal as a filler are:

- High heat transfer coefficient
- Retardation of radionuclide migration within the canister
- Increased canister strength

Some of the disadvantages are:

- Increased cost of materials and facilities
- Increased weight to handle heavier canisters
- Consumption of additional resource materials

Solid Granular Fillers. The encapsulation of spent fuel assemblies in a granular matrix is a simpler operation than adding molten filler and may be a more economical way to provide other desired properties to the spent fuel canister. Although granular materials may not have the thermal properties of solid materials, by proper selection it may enhance the heat transfer to the repository environment. Granular filler materials are expected to perform the following functions effectively:

- Reduce temperature of the fuel assembly cladding
- Provide physical stability and support for the fuel assembly
- Maintain the chemical stability of the fuel assembly cladding
- Provide an additional barrier to the migration of nuclides

Table 10-8 ranks some selected canister filler materials as compared to the helium filler in Case 1.

Table 10-8
COMPARISON OF CANISTER FILLER MATERIALS

Item	Gaseous Fillers				Monolithic Solid Fillers			Granular Solid Fillers		
	Base He	Air	N ₂	Ar	Lead	Al	Glass	Lead Shot	Sand	Glass Frit
Reduce temperature of the fuel assembly	0	-1	-1	-1	+2	+2	+1	+1	-1	+1
Provide support to the fuel assembly	0	0	0	0	+2	+2	+2	+1	+1	+1
Provide chemical stability with the assembly	0	-1	0	0	-1	-1	0	-1	0	0
Provide additional barrier for migration of the nuclides	0	0	0	0	+2	+2	+2	+1	+1	+1

Note:

- +2 Improvement
- +1 Minor improvement
- 0 No change from helium fill
- 1 Minor impairment
- 2 Impairment

10.3.1.2 Design Impact. The impact of changes in the filler material on the design of the facilities and the transportation system is discussed in this subsection and summarized in Table 10-9.

AFR. This variation does not affect the design of the AFR since fuel is not encapsulated until processed through a P/E facility.

P/E facility. If gases other than helium are used, some helium still has to be added to enable the leak testing to be performed following welding of the canister lid. The impact on the design and operation of the facility is an additional step to provide injections of two gases in each canister.

If monolithic solids are used, the design of the P/E facility is modified to provide equipment for metal melting and pouring into the canister. Canister handling equipment may need modification to handle the increased weight and the facilities may need to be enlarged; also, electrical and other services have increased load requirements.

The extent of facility modifications also depends on the physical and chemical properties of the

Table 10-9
DESIGN IMPACT SUMMARY
CANISTER VOID FILLERS

Item	Gaseous Fillers				Monolithic Solid Fillers			Granular Solid Fillers		
	He	Air	N ₂	Ar	Lead	Al	Glass	Lead Shot	Glass Sand	Glass Frit
AFR	Base	No change			No change			No change		
P/E facility	Base	No change			More complex			More complex		
Repository	Base	No change			More complex handling equipment			More complex handling equipment		
Transportation										
From reactor	Base	No change			No change			No change		
From AFR	Base	No change			No change			No change		
From P/E facility	Base	No change			New cask designs			New cask designs		

filler. High-density fillers, such as lead, impact more on the handling equipment than aluminum or glass, but, at the same time, require less energy for the melting process. On the other hand, the melting of glass requires very high temperatures and large amounts of energy for melting.

The introduction of molten lead or aluminum alloys in steel canisters utilizes existing technology and equipment. Experimental encapsulation of defective fuel assemblies in lead has been accomplished in the United States;⁽¹⁾ remote operations on a large scale basis have not been tried. Studies have been made for a facility design to encapsulate discharged fast reactor assemblies in an aluminum silicon alloy matrix.⁽²⁾

If granular solids are used, the facility design impact is increased somewhat to provide additional equipment and services, such as storage and metering systems for the granular material, and vibrating equipment for canister filling.

Repository. If other gaseous fillers are used, the rate of heat transfer within the canister may be reduced since the thermal conductivity of helium is at least 5 times greater than that of air or nitrogen, and 8 times greater than that of argon; however, the impact of these fillers on the repository is not significant. The limiting thermal conditions, in the repository, are determined by the properties of the nearby salt, not by the thermal considerations within the storage well.

If monolithic solid fillers are used, the thermal conductivity within the metal-filled canisters is an order of magnitude larger than that of helium-filled canisters. In spite of this, the impact on the repository design is insignificant because, as mentioned in the preceding paragraph, the very near field within the storage well is not controlling the repository thermal considerations.

The licenseability of the repository may be enhanced by the use of solid fillers, due to the increased safety and stability of the canister contents.

On the negative side, equipment such as cranes, trucks, and other canister handling tools may have to be larger and stronger. Yet, the casks and shield plugs may be somewhat lighter because of the lower dose rate from the canister.

If solid, granular, fillers are used, they provide increased confinement and stability to the canister contents.

The increased weight of the canister requires the use of larger and stronger canister handling tools, cranes, and trucks as compared to the helium-filled canisters.

Transportation. The gas fillers do not affect the design of the transportation system of Case 1.

The solid monolithic and granular fillers in the canisters make the encapsulated spent fuel canisters heavier than those filled with helium gas. Those fillers reduce the shielding required for the encapsulated fuel cask. The overall impact may be a lower total weight of system as solid fillers bring the shielding closer to the source. A new cask may need to be developed for each solid and granular filler. Since cask designs for these filler materials have not been developed within the scope of SFDS, the difference in cost for transporting encapsulated fuel from the P/E facility to the repository with various solid and granular fillers, as compared to those with helium filler, cannot be determined.

10.3.1.3 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is shown in Table 10-10 and is discussed below.

Table 10-10
ASSESSMENT IMPACT SUMMARY
CANISTER VOID FILLERS

Item	Gaseous Fillers			Monolithic Solid Fillers			Granular Solid Fillers		
	He	Air	N ₂ Ar	Lead	Al	Glass	Lead Shot	Glass Sand	Frit
Technical feasibility	Base	No change		Some reduction			Some reduction		
Safeguards assessment	Base	No change		No change			No change		
Criticality assessment	Base	No change		Improvement			Improvement		
Radiological assessment	Base	No change		Improvement			Improvement		
Retrievability assessment	Base	No change		Improvement			Improvement		
Economic assessment	Base	Little change		Most costly			Little change		

Technical feasibility. A slight reduction in feasibility for solid fillers is postulated because of the need to develop and demonstrate methods for filling.

Safeguards assessment. No impact.

Criticality assessment. The presence of solid and granular fillers may provide an additional barrier to the leaching out of the fissile material and may reduce the risk of a criticality event in the repository. The gaseous fillers, other than helium, have no impact on the criticality assessment of Case 1.

Radiological assessment. The presence of solid and granular fillers may provide an additional barrier to the selective leaching of some radionuclides. This may tend to reduce the long-term radiological impact of the repository. The gaseous fillers (other than helium) have no impact on the radiological assessment of Case 1.

Retrievability assessment. The solid, and possibly the granular, fillers present an additional barrier to corrosion of the spent fuel assemblies. Therefore, the spent fuel may be retrievable over a longer period. The weight of the solid fillers may, however, add slightly to the difficulty of retrieving these canisters. The gaseous fillers, other than helium, have no impact on the retrievability assessment of Case 1.

Economic assessment. The use of other fillers increases overall costs in the following areas:

- Cost of filler material as compared to the cost of helium
- Cost of additional equipment at the P/E facility for using fillers other than helium
- Cost of transporting heavier canisters from the P/E facility to the repository for solid and granular fillers

These additional costs have not been calculated and, therefore, the economic assessment impact cannot be quantified.

10.3.2 Overpacks and Thicker Wall Canisters

As described in Appendix C, Subsection C.3.1.2, the P/E facility for Case 1 is designed to package the PWR and BWR assemblies in cylindrical canisters. Overpacks are not routinely required for the encapsulated fuel assemblies at the P/E facility; however, the need for overpacks may arise if the canisters develop leaks prior to disposal at the repository.

Both the impact of adding overpacks and the impact of increasing wall thickness of the primary canisters are studied in this variation. Generally, these changes increase operating costs, but reduce the potential corrosion problems.

10.3.2.1 Design Impact. The impact of these variations on the design of facilities and the transportation systems is discussed in the following subsections and summarized in Table 10-11.

Table 10-11
DESIGN IMPACT SUMMARY
OVERPACKS AND THICKER WALL CANISTERS

Item	Impact of Thicker Canisters	Impact of Overpacks
AFR	No impact	No impact
P/E facility	Slightly more difficulty in handling empty canisters	Requires additional processing area; higher operating costs
Repository	No impact	Requires oversized storage wells to accommodate
Transportation	No impact	Slight derating of cask capacity possible

AFR. These variations do not impact the design of an AFR, as encapsulation capability is not provided at the AFR.

P/E facility. The P/E facility would have to be modified to add overpacking capabilities. The use of thicker wall canisters increases the cost of the canisters and the cost and difficulty of welding, heat treating, and testing operations on the canister closure joint. The use of overpacks increases the operation costs.

Repository. Uncertainty in the number of canisters needing overpacks affects the repository design. Most likely, a conservative number of oversized wells will be required in each room which can accommodate either single canisters or an overpacked canister.

The use of thicker wall canisters or overpacked canisters decreases the rate of heat transfer from the waste, with a corresponding increase in the waste temperature. Calculations must verify that the 200°C limiting temperature for the fuel assembly cladding is not exceeded; otherwise, thermal conditions in the repository are not impacted.

Transportation. The use of overpacks or thicker canisters may affect the design of cask internals and heat removal systems. Available cask designs may be adapted with some economic penalty because of reduced cask capacity.

10.3.2.2 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections and summarized in Table 10-12.

Table 10-12
ASSESSMENT IMPACT SUMMARY
OVERPACKS AND THICKER WALL CANISTERS

Item	Variation
Technical feasibility	No impact
Safeguards assessment	No impact
Criticality assessment	No impact
Radiological assessment	Improved
Retrievability assessment	Improved integrity enhances retrievability
Economic assessment	Unfavorable relative to material costs, potential derating of the transportation system, and higher operating and capital costs at the P/E facility

Technical feasibility. No impact.

Safeguards assessment. No impact.

Criticality assessment. No impact.

Radiological assessment. The use of thicker wall canisters and overpacks reduces the radiation dose at the canister surface and increases the safety factor against accidents.

Retrievability assessment. Since the canisters with overpacks are heavier, the cost of retrieving them may be slightly higher. On the other hand, the use of thicker wall canisters reduces the potential corrosion problem and provides greater integrity during the retrievability period.

Economic assessment. If the overpacks and canisters are used, the economic impact affects the following:

- The cost of overpacks
- The additional cost of thicker metal in canisters and the increased transportation cost of the added weight
- The penalty of derating the capacity of existing transportation system designs

10.3.3 Number of Spent Fuel Assemblies Per Canister

As described in Appendix C, Subsection C.3.1.2, the number of assemblies placed in a canister at the P/E facility is either 1 PWR or 3 BWR fuel assemblies. The Spent Fuel Handling and Packing Program⁽³⁾ of Rockwell Hanford Operations (RHO) was one of the documents used to develop the bases for the SFDS. This variation studies the impact of placing 2 PWR or 6 BWR spent fuel assemblies in the canister.

10.3.3.1 Design Impact. The impact of this variation on the design of various facilities and transportation systems of Case 1 is discussed in the following subsections and summarized in Table 10-13.

Table 10-13
DESIGN IMPACT SUMMARY
NUMBER OF SPENT FUEL ASSEMBLIES PER CANISTER

Item	Variation
AFR	No change
P/E facility	Minor changes in canister loading and storage
Repository	Considerable; would have to reestablish excavation and emplacement geometries
Transportation	Favorable; greater densification of the radioactive source should reduce the overall weight of shielding and tend to increase the capacity and efficiency of system

AFR. The design of the AFR is not affected by this variation as the encapsulation is done in the P/E facility.

P/E facility. The design is affected only in a limited way. Most of the changes occur in the encapsulation equipment and in the modifications to storage pool racks to accommodate bigger canisters.

Repository. The repository is affected because of the change in the heat content of the larger and heavier canisters, and also because of the spacing of rooms, pillars, and canisters. Repository acceptance criteria may also have to be revised to accommodate the higher heat content and physical changes in the new canisters to accommodate 2 PWR or 6 BWR assemblies.

Transportation. This variation only affects the transportation of encapsulated spent fuel from the P/E facility to the repository. Since either 2 PWR or 6 BWR assemblies are packaged per canister, new casks need to be developed. Although the impact of this variation on the transportation cost is not calculated in this analysis, the overall impact should be increased efficiency of the transportation system and thus reduced unit costs.

10.3.3.2 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections and summarized in Table 10-14.

Table 10-14
ASSESSMENT IMPACT SUMMARY
NUMBER OF SPENT FUEL ASSEMBLIES PER CANISTER

Item	Variation
Technical feasibility	No impact
Safeguards assessment	No impact
Criticality assessment	No impact
Radiological assessment	Less impact in the near term
Retrievability assessment	Negative impact
Economic assessment	Favorable reduction in unit costs at the P/E facility

Technical feasibility. No impact.

Safeguards assessment. No significant impact. In the case of transportation of encapsulated spent fuel from the P/E facility to the repository, a lesser number of shipments may be involved. The total activity per shipment is still expected to be of the same order of magnitude and, therefore, the assessment for sabotage is not affected. The assessment for theft is also not affected as the materials involved in this variation are the same as for Case 1.

Criticality assessment. A spent fuel canister with 2 PWR or 6 BWR assemblies contains twice the fissile mass as compared to Case 1. Even though twice as much material is available, the same conditions are required to form a critical mass as in Case 1, and the same sequence of events needs to take place. Therefore, this variation is considered to have equally low risk of criticality when compared to Case 1.

Radiological assessment. Increasing the number of assemblies packaged per canister reduces the number of shipments and hence reduces the potential exposure of the population due to transportation. In a similar manner, decreasing the number of canisters decreases the radiological impact to repository personnel for both normal and abnormal operations and also reduces the impact on the public of accidents.

Retrievability assessment. Retrievability may be slightly more difficult to accomplish with a higher heating rate in the canister. The salt-to-sleeve gap may have been reduced, and the potential for sleeve entrapment may be enhanced. The additional weight of the canister may have an insignificant impact on retrievability.

Economic assessment. The use of larger-sized canisters containing 2 PWR or 6 BWR assemblies has a favorable impact on the cost of canisters and the cost of transportation of encapsulated assemblies from the P/E facility to the repository.

10.4 REPOSITORY DESIGN VARIATIONS

10.4.1 Room Cooling During Extended Retrievability Period

10.4.1.1 Scope. The SFDS assumes a retrievability period of 5 years. In Subsection 10.2.3, longer retrievability periods (10 and 25 years) are analyzed. In this study variation, the probability of longer retrievability periods is investigated further. Since the retrievability period is only a temporary state for the repository, the basic design has to meet the far-field, long-term requirements. To allow for short-term control of near-field effects (i.e., room closure), room cooling is considered as an alternative to reduce thermal loading during longer retrievability periods.

10.4.1.2 Design Impact. The impacts of this variation on the design of the various facilities and transportation systems are discussed in the following subsections.

Effect on AFR. None.

Effect on P/E facility. None.

Effect on transportation. None.

Effect on repository. The retrievability of waste packages is related to room closure (near-field effects). The room closure is dependent on several parameters; four of which are:

- Geological depth of repository
- Barrier pillar geometry
- Extraction ratio
- Thermal load

For a given geological depth, the thermal loading of the repository is fixed based on far-field effects (temperature and displacement of surface). The extraction ratio is minimized via thermal calculations; therefore, 3 of the 4 parameters are fixed due to other than near-field effects.

Several modeling studies indicate that the barrier pillars (massive pillars situated at regular intervals in the repository) concept may help in reducing room closure. These results are inconclusive as yet.

Since retrievability is a temporary period in the repository life, temporary reduction in room closure rates can be achieved by reduction of the thermal loading. This can be done in two ways:

- Limited emplacement of waste canisters during the retrievability period; reentry and continuing emplacement, to full design capacity, during the post-retrievability period
- Emplacing to full thermal load design and providing a cooling system for reduction of effective thermal load during the retrievability period. No room cooling is to be provided during the post-retrievability period

The first option is not studied in this variation. The second option allows for the original (Case 1, SFDS) design of the repository to remain unchanged during longer (10 and 25 years) retrievability periods. A cooling system is needed in this option to remove a portion of the generated heat so that the salt retains only 48 or 36 kW/acre loading, respectively.

10.4.1.3 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed below.

Technical feasibility. Existing technology.

Criticality. None

Radiological. Slight increase Additional system complexity and maintenance requirements may increase the exposure of repository personnel

Retrievability. Allows for longer retrievability periods, as required

Economics. Negative impact because of adding and maintaining the cooling/ventilation system during the retrievability period

10.4.2 Horizontal/Diagonal Emplacement of Canisters

10.4.2.1 Scope. In the SFDS, emplacement of the waste canisters is assumed to be in vertical wells in the floors of the repository rooms This assumption is an historically developed concept, based on accepted and available simple technology Waste packages are lowered into the wells and plugged from above This approach establishes physical limitations on the emplacement equipment since the vehicles must avoid passing over the drilled holes in the floor

In this variation, the impact of other than vertical emplacement is being studied, potential savings can be realized by reducing the size of the storage rooms with larger emplacement capacity

10.4.2.2 Design Impact. The impact of this variation on the design of the various facilities and transportation systems is discussed in the following subsections

Effect on AFR. None

Effect on P/E facility. None

Effect on transportation. None

Effect on repository. Drilling horizontal/diagonal holes in the walls improves ease of movement of the emplacing and drilling vehicles All operations can be executed from nonrail vehicles, providing more flexible maneuvering, independent of salt creep effects Room height can be reduced as less headroom is needed than for vertical emplacement equipment The room width is determined by the transfer cask length

Versatile emplacement configurations can be accommodated such as several canisters in tandem in each hole or horizontal emplacement of drilled HLW canisters in an upper row and ILW canisters in a diagonally drilled lower row The emplacement density can achieve 4 rows per room, as compared with 3 or less rows per room utilizing the floor of the storage rooms The thermal load limitations determine the loading density in the mine in either the vertical or the horizontal approach

Heat distribution from horizontal/diagonal emplacement may be better as the heat is transmitted directly to the pillars A detailed thermal analysis is required in order to determine allowable heat loading

10.4.2.3 Assessment Area Impact. The impact of this variation, with respect to each assessment area is discussed below

Technical feasibility. Some modifications are necessary for the drilling, sleeving, and emplacement equipment, basically, all seem feasible

Safeguards. None

Criticality. None

Radiological. None

Retrievability. The horizontal/diagonal emplacement mode has the disadvantage, during the retrievability period, of being additionally affected by creep. The orientation of the emplacement canister may change, thereby complicating seeking and retrieving operations.

Economics. Potential cost savings can be realized by less mining, including reduced headroom height, possible increased emplacement density, and elimination of rail installation.

10.4.3 Combination Spent Fuel/HLW Repository

10.4.3.1 Scope. The repository is assumed to have been designed to accommodate canistered spent fuel (Case 1, SFDS). Since governmental policy may change, the possibility exists that a decision to reprocess spent fuel may be pending during the repository operating lifetime. Two main scenarios can be envisioned:

- Retrieving all emplaced spent fuel for reprocessing and converting all of the repository to accept high-level waste (HLW) and other wastes from fuel reprocessing
- Sealing the emplaced spent fuel rooms and converting the rest of the repository to accept HLW and other reprocessing wastes

It is assumed that the changeover to accept HLW and other wastes from reprocessing could occur at any time during the life of the repository. The changeover period should span a few years in which the system will be rearranged. Another retrievability period may have to be provided to allow for HLW repository demonstration.

10.4.3.2 Design Impact. The impact of this variation on the design of various facilities and the transportation system is discussed in the following subsections.

Effect on AFR. None.

Effect on P/E facility. The P/E facility for encapsulating spent fuel (Case 1) must be upgraded to a full reprocessing facility (as discussed in Reference Case, SFDS). If an existing reprocessing facility were already in use as an encapsulation facility, it might be reconverted to the original design and purpose to accommodate reprocessing.

Effect on transportation. Shipping casks must be designed and constructed as required in the Reference Case analysis of the SFDS (i.e., adding HLW and hulls casks, providing for plutonium oxide transportation to fabrication plants, etc.).

Effect on repository. The surface facilities designed for encapsulated spent fuel are capable of handling the HLW (shorter canisters). Certain modifications are required, such as adjusting the surge storage cell capacity, or providing for special storage racks. The operation of the facility has to be adjusted to the new throughput of the HLW and associated waste canisters. In the underground development, modifications of waste emplacement rows, hole drilling patterns, and pillar width (due to change in thermal characteristics of emplaced canisters) are expected. The repository lifetime will be longer (i.e., more storage capacity) with the conversion to the HLW option. If the spent fuel portion of the mine is isolated and sealed, the rest of the repository can be converted to HLW disposal. The capacity and life of the repository is estimated as follows:

- About 28 rooms (with associated areas) or 99 acres of the 1,700-acre underground storage area is used per year for spent fuel emplacement (Appendix D, Subsection D.1.4)
- When HLW is emplaced in the repository, only about 11 rooms with associated areas, or 43 acres, are filled per year (Appendix D, Subsection D.3.1)

As an example, the variation in the useful life of a combined spent fuel and HLW repository would have, after the first 8 years of spent fuel emplacement, about 792 acres filled with 40,000 MTHM,

while 908 acres would remain available for accepting an additional 105,000 MTHM over a 23-year period. Under those conditions, the 1,700-acre repository would accept a total of 145,000 MTHM, with 29 years of useful life.

Since the HLW canisters and transfer cask are 5 ft 6 in. shorter than those used for spent fuel, the height of the emplacement gantry crane can be reduced 3 feet.

Based on the above changes, the amount of excavated salt from the storage area drops from 1,170,000 tons/yr for the spent fuel to 450,000 tons/yr for the vitrified HLW.

10.4.3.3 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections.

Technical feasibility. Since this variation involves application of the Reference Case and Case 1 technologies, technical feasibility is as discussed in both cases.

Safeguards. The safeguards assessment involves aspects from both the Reference Case and Case 1.

Criticality. As per Case 1 and the Reference Case.

Radiological. As per Case 1 and the Reference Case.

Retrievability. If repository demonstration is not satisfied during the encapsulated spent fuel retrievability period, a further retrievability period may be required for HLW emplacement.

Economics. The impacts on economic considerations are potential savings because of less mining, increased costs of upgrading the P/E facilities, and the increased costs of redesigning casks, etc., but in any case, the SFDS Reference Case is the limiting constraint.

10.4.4 Non-Salt Repository

10.4.4.1 Scope. A study was performed to evaluate the effects on the assessment of Case 1 of alternate geologic media. The repository assumed for Case 1 is in salt formation. However, other geologic media (i.e., granite, basalt, and shale) have been proposed for repositories.

Disposal in geologic formations seems to be the most available solution for a large percentage of the nuclear waste produced by the power industry. The role and importance of the different rock parameters involved in the confinement of radioactivity, the three different geological media, i.e., granite, basalt, and shale, are briefly reviewed in the following paragraphs.

Salt is a leading candidate for a repository. Shale and related rock types may be the least desirable media for storing waste. Shale has a number of positive attributes such as low permeability, good sorptive characteristics, and low solubility in water. However, thermally-induced changes at elevated temperatures (above 100°C), in turn, increase the permeability and porosity of shale and, thus, tend to increase radionuclide transport rates. Also, inhomogeneities in shale rock affect its structural characteristics and present difficulties in mining and keeping the tunnels open.

Granite must be considered a potential geologic medium for a repository because it has the following characteristics:

- Very good strength and structural stability
- Very low water content
- Good sorptive properties
- Good thermal properties
- Very low permeability

While granite is sensitive to fractures that affect permeability, it would be very difficult to locate a repository in any rock mass without fractures. The construction and operating costs are higher in granite because of the characteristics of crystalline homogenized rock.

Basalt must be considered a favorable geologic medium for spent fuel waste disposal. Basalt shows a very high confining capacity because of its low porosity, low permeability, high mechanical strength, and tolerance of high thermal loading.

10.4.4.2 Design Impact. The impact of this variation on the design of the various facilities and transportation systems is discussed in the following subsections.

Effect on AFR. None.

Effect on P/E facility. None.

Effect on transportation. None.

Effect on repository. Typical media properties are listed in Table 10-15 and classified according to their relative potential as isolation media. These properties of the media affect repository capacity, spent fuel placement geometry, rate of migration of radionuclides, repository temperature, and drilling methods. Each of the properties listed are briefly discussed and evaluated in the follow paragraphs.

Table 10-15
TYPICAL MEDIA PROPERTIES⁽¹³⁾

Item	Case 1	Alternative Geologic Media		
	Salt	Granite	Shale	Basalt
Rock mechanical strength	Poor	Very good	Good	Very good
Porosity	Very good	Very good	Poor	Very good
Permeability	Very good	Very good	Very good	Very good
Solubility	Poor	Negligible	Negligible	Negligible
Thermal conductivity	Very good	Good	Good	Poor
Heat capacity	Very good	Good	Good	Good
Chemical properties	Vary greatly; range of possible chemical reactions between spent fuel and rocks is great			

Granite and basalt, in general, exhibit a high compressive strength with that property having a major impact in the design and cost of a repository. Shales are subject to slacking (loss of strength due to drying and wetting) and require larger pillars and more support than granite and basalt. Salt exhibits plasticity under stress and at high temperatures; it may need an extensive support system depending on the state of stress and wall temperatures. Unsupported granites and basalts can fail in underground openings by slabbing, and rock bolts are used to stabilize this buckling type of failure. However, the cost for stabilizing granites and basalts is lower than the costs involved for stabilizing shales.

Rock hardness and strength affect the mining of the rock. Rock strength increases the drilling time. The cost is proportional to drilling and excavating time. Since shale is weaker and softer than basalt and granite it is cheaper to excavate because drilling rates would be faster or, conversely, less energy is expended to achieve the desired rate of excavation.

Salt is an impermeable rock, but is highly soluble in water. Any geologic or other mechanism that may cause water to reach salt could potentially result in a breach in the waste containment barrier. Shale, granite, and basalt are all relatively impermeable rock materials. The groundwater, both in terms of moisture content within rock and the overall regional flow patterns of circulatory groundwater have effects on certain design features of the repository. These effects are much more pronounced in shales than in basalt and granite. Specifically, the presence of groundwater would significantly affect repository shaft design.

The depth of the repository affects the overall cost because deeper shafts and vertical transport increase the costs for hauling excavated rocks. Also, as the shafts become deeper, more equipment is needed to provide increased ventilation capacity, increased water pumping capacity, etc. Finally, because rock pressure increases with the depth below ground surface, the weaker rock, such as shale and salt, may require some pillar supports to provide the same structural stability as a repository in stronger rocks such as granite or basalt.

The heat that is released from the emplaced spent fuel and its induced thermal stresses could have a very significant impact on the design and cost of the repository. The heat induced into the formation depends upon the reactor exposure and subsequent decay of the spent fuel placed in the repository. In addition, very-near-field temperatures may depend upon details of the design of the spent fuel package and storage well. The temperatures reached at any location within the formation depend upon the thermal properties of the formation material. The lower thermal conductivity of granite dictates that less waste be stored per canister in order to meet thermal restrictions regarding maximum waste form and/or canister temperatures. The temperature histories of the room and pillar regions of the repository are strongly influenced by the physical properties of the host geologic formation. The thermal conductivity, mass density, and heat capacity of the host rock determine the rate of heat dissipation throughout the formation. It has been analytically determined that the thermally induced stresses in hard rocks like granite and basalts are small compared to the strength of the rock in the pillars. Since shale and salt are generally weaker than both granite and basalt, the combined stresses due to heat and gravity may require additional supports, increased spacing between canisters, etc.; these alterations would increase the cost of construction and operation of the repository.

These phenomena can be minimized and/or be avoided by the proper engineering design, but at a cost which could be considerable. The consideration of other effects has a significant impact on the design. The canister stress, temperature, and corrosion effects are difficult to predict; it is postulated that in a non-salt rock formation, even low concentrations of chlorides in circulating groundwater may cause corrosion. Furthermore, if sleeves are used, they must be sufficiently thick to allow for corrosive effects and also remain intact during the entire retrievability phase. These sleeves are a substantial fraction of repository costs; therefore, further analysis regarding their requirements for emplacement in each type of rock formation is warranted. Also, it is obvious that the control of this groundwater becomes a major objective, and thus an additional cost factor in the design. The repository designs in granite formations need to include an engineered groundwater handling system to remove this water.

10.4.4.3 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections.

Technical feasibility. Available technology; therefore, no impact.

Criticality. No impact.

Radiological. The rock surrounding the canister is irradiated by the wastes. The degree of radiation depends on the shielding provided by the canisters and other materials that may be necessary for support or protection of the canisters. Other aspects to be assessed include ascertaining

that all pathways of radioactive wastes to the biosphere are identified and that complete isolation is assured. The most likely mechanism for the release of radionuclides is through dissolution and transport in groundwater, and once this takes place, factors such as radioactive decay, dispersion, and mixing determine the concentration of radionuclides moving toward points of potential human access. Hydrogeologic information applicable to these problems is currently not fully available, thus, a significant limitation exists relative to the ability to assess radiological impact.

Safeguards. Physical protection and safeguards requirements for repository facilities are specified in the Code of Federal Regulations (10 CFR Parts 105, 70, and 73). In general, after placement in the geologic repository, the waste is essentially unavailable for theft or dispersion through sabotage, and a successful intrusion is highly unlikely because of the operational controls over entry and the physical security provided at the access points in the surface facility. No significant differences are expected in the physical protection requirements among acceptable repositories in salt, granite, basalt, and shale.

Retrievability. The retrievability concept requires that the waste be retrievable for a minimum of 5 years, with approximately the same effort, and at approximately the same rate, at which it was placed into the repository. During the time waste is retrievable, the repository rooms must remain open, the openings must stand firm, and the spent fuel canisters must remain structurally intact. In general, these performance criteria impact the capital and operational costs of the repository. For example, gradual creep in soft rocks, such as exists in some shales and salt during the operating lifetime of the repository, can reduce the size of the openings, thus threatening the retrievability options. In granite and basalt formation, spalling, slabbing, and rock falls may result from high rock pressure, waste heat, and seismic events. Such occurrences could inhibit retrievability.

Economics. Preliminary construction, operating, and unit costs developed for repositories in granite, shale, basalt, and salt for several basic fuel cycles are based on the review of existing pertinent literature. The brief review of these costs is limited to the handling and storage of canistered spent fuel elements.

The higher construction costs reported for granite and basalt reflect the difficulty in mining these rocks. The operating costs for a granite and basalt repository are significantly higher than costs for shale and salt, for the same reason. Another significant item in the operating cost difference is the higher cost of drilling storage wells in granite and basalt. A review of the unit repository costs shows that disposal in salt is the least expensive, the next least expensive medium is shale, and the next, granite. Disposal in basalt is the most expensive alternative. This brief cost comparison, based on available sources, is but an example of what could be achieved with current technology. Further process refinements and improvements could bring costs down.

10.5 COLOCATION OPTIONS

The following assumptions are made regarding the location of the facilities of Case 1. All facilities, including the AFRs, the P/E facilities, and the repository are independently located, with the average distance between the reactors and the AFRs being 400 miles, the average distance between the AFR and P/E facility 700 miles, and the average distance between the P/E facility and the repository 1,100 miles.

The effect of colocating one or more of these facilities at the same site is being assessed. The three considered colocation options are:

- Colocation of 3 AFRs* and 1 P/E facility at each of three sites
- Colocation of 1 P/E facility and 1 repository at each of three sites
- Colocation of 3 AFRs*, 1 P/E facility, and 1 repository at each of three sites

10.5.1 Colocation of 3 AFRs and 1 P/E Facility

In this variation, 3 AFRs are assumed to be colocated with 1 P/E facility at each of three sites, as shown in Figure 10-1. In addition to the basic assumptions, the following assumptions are also made for this variation:

- Three AFRs are located on the same site with each P/E facility, a total of three such sites exists for a total storage/processing capacity of 5,000 MTHM/year
- Only 1 operating repository (2,000 acres) exists at a time with an emplacement capacity of 5,000 MTHM/year
- The average distance between the AFR and the shallow land burial is 700 miles

10.5.1.1 Design Impact. The impact of this variation on the design of the facilities and the transportation system of Case 1 is discussed in the following subsections and summarized in Table 10-16.

AFR. Since 3 AFRs are colocated with each P/E facility at the same site, some common facilities can be eliminated or consolidated including the following:

- Overhead and support service components of the operating cost of AFRs can be reduced
- The radwaste and water treatment systems of each AFR can be consolidated resulting in some savings
- The spent fuel receiving areas of each AFR can be consolidated, resulting in some savings
- The spent fuel shipping areas of each AFR can be eliminated
- The physical security, material control, and accounting systems of the 3 AFRs can be integrated with those of the P/E facility, resulting in some savings

P/E facility. Since each P/E facility is colocated with 3 AFRs, some of the common functions between the AFRs and the P/E facility can be consolidated including the following:

- The spent fuel receiving area and the spent fuel pool at the P/E facility can be eliminated
- The radwaste system of the P/E facility can be consolidated with that of the AFR, resulting in some savings
- Some utilities and some site work can be reduced for the P/E facility
- The physical security, material control, and the accounting system of the AFRs and the P/E facility can be consolidated, resulting in some savings

* In this subsection the references to three AFRs being colocated on the same site does not mean three separate facilities; rather it implies a single facility incorporating the capacity of three AFRs.

Table 10-16
DESIGN IMPACT SUMMARY
COLOCATION OF THREE AFRs AND ONE P/E FACILITY

Item	Case 1	Variation
AFR		
Number at each site	1	3
Number of regional sites	9	3
Capacity of each, MTHM	5,000	5,000
P/E facility		
Number at each site	1	1
Number of regional sites	3	3
Total capacity, MTHM/year	5,000	5,000
Repository		
Number at each site	1	1
Number of regional sites	1	1
Size of repository, acres	2,000	2,000
Total capacity, MTHM/year	5,000	5,000
Transportation		
Average distance from reactor to AFR, miles	400	700
Average distance from AFR to P/E facility, miles	700	0
Average distance from P/E facility to repository, miles	1,100	1,100
Average distance from AFR to shallow land burial, miles	400	700
Average distance from P/E facility to shallow land burial, miles	700	700
Cask fleet required for spent fuel, road	13	17
Cask fleet required for spent fuel, rail	171	149
Cask fleet required for TRU and non-TRU waste, road	10	10

Repository. No impact.

Transportation. The transportation step, from the AFR to the P/E facility, is eliminated. Further, since all the reactors are shipping spent fuel to 3 locations (rather than 9 in Case 1), the average distance between the reactors and the AFR and the shallow land burial increases. The cask fleet required for transporting spent fuel by road and rail changes from 13 to 17 (road) and from 171 to 149 (rail). The road fleet increases because the distance from the reactor to the AFR increases. The decrease in rail fleet size is due to the elimination of the transport phase from the AFR to the P/E facility. The cask fleet required for TRU and non-TRU waste remains unchanged.

10.5.1.2 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections and summarized in Table 10-17.

Figure 10-1.
Option I – AFRs
and P/E Facility – Colocated

Table 10-17

**ASSESSMENT IMPACT SUMMARY
COLOCATION OF THREE AFRs AND ONE P/E FACILITY**

Item	Case 1	Variation
Technical feasibility assessment	Base	No impact
Safeguards assessment	Base	No impact
Criticality assessment	Base	No impact
Radiological assessment	Base	Significant improvement
Retrievability assessment	Base	No significant impact
Economic assessment		
AFR, \$/MTHM	53,300	51,100
P/E facility, \$/MTHM	16,700	15,700
Repository, \$/MTHM	19,400	19,400
Transportation		
From reactor, \$/MTHM	12,900	18,500
From AFR, \$/MTHM	4,700	—
From P/E facility \$/MTHM	9,300	9,300
Total, \$/MTHM	116,000	114,000

Technical feasibility assessment. No impact.

Safeguards assessment. This variation reduces the total number of sites having nuclear material from 13 (in Case 1) to 4. It also eliminates one transportation step. The reduction in the number of facilities and transportation steps reduces the potential of theft and sabotage. The risk from sabotage and theft in each facility is based on the highest risk (or rating) of the material handled in that facility for the purposes of the SFDS. Therefore, within the scope of the SFDS, there is no impact on safeguards assessment due to this variation.

Criticality assessment. No impact.

Radiological assessment. The reduction by nine sites and one transportation step handling nuclear material has a beneficial effect on the radiological assessment of this variation. The combination of operations at colocated sites reduces the overall personnel requirements and the integrated dose commitment of operating personnel. The elimination of a transportation step reduces the radiological impact on personnel and the public.

Retrievability assessment. No impact.

Economic assessment. The economic impact of this variation is due to reduced capital and operating costs for the AFRs and the P/E facility and also the elimination of one transportation step. The latter is partially offset by a greater shipping distance.

10.5.2 Colocation of 1 P/E Facility and 1 Repository at 3 Sites

In this variation, 1 P/E facility is colocated with 1 repository at each site, as shown in Figure 10-2. There are three such sites with a total processing capacity of 5,000 MTHM/year. The following assumptions are made for this variation:

- Nine AFRs are regionally located, as in Case 1
- Three P/E facilities are regionally located. The total processing capacity of the P/E facilities is 5,000 MTHM/year
- There are 3 operating repositories (2,000 acres each) at a time with a total emplacement capacity of 5,000 MTHM/year
- The average distance between the reactors and the AFR is 400 miles
- The average distance between the AFR and the P/E facility or repository is 700 miles

10.5.2.1 Design Impact. The impact of this variation on the design of the facilities and the transportation system of Case 1 is discussed in the following subsections and summarized in Table 10-18.

Table 10-18
DESIGN IMPACT SUMMARY
COLOCATION OF A P/E FACILITY AND A REPOSITORY

Item	Case 1	Variation
AFR		
Number at each site	1	1
Number of regional sites	9	9
P/E facility		
Number at each site	1	1
Number of regional sites	3	3
Repository		
Number at each site	1	1
Number of regional sites	1	3
Transportation		
Average distance from reactor to AFR, miles	400	400
Average distance from AFR to P/E facility, miles	700	700
Average distance from P/E facility to repository, miles	1,100	0
Average distance from AFR to shallow land burial, miles	400	400
Average distance from P/E facility to shallow land burial, miles	700	700
Cask fleet required for spent fuel, road	13	13
Cask fleet required for spent fuel, rail	171	88
Cask fleet required for TRU and non-TRU waste, road	10	3

Figure 10-2.
Option II – P/E Facility and Repository – Colocated

AFR. No impact

P/E facility. Since each P/E facility is colocated with 1 repository at the same site, this variation has the following impacts

- The surge capacity and the shipping area of the P/E facility can be eliminated
- The physical security, material controls, and accounting systems of the P/E facility and the repository can be integrated, resulting in some savings
- Some utilities and site work for the P/E facility can be reduced

Repository. In this variation, 3 repositories of 2,000 acres each are assumed with a total emplacement rate of 5,000 MTHM/year for the 3 repositories. This results in lower operating costs for each repository. Since each repository is colocated with 1 P/E facility, the following areas of the repository are affected

- The receiving area and surge storage can be eliminated
- The radwaste handling facilities of the repository can be eliminated. The radwaste generated at the repository can be handled in the P/E facility
- The physical security, material controls, and accounting systems of the P/E facility and the repository can be integrated, resulting in some savings
- Some utilities and site work for the repository can be reduced

Transportation. The transportation step, from the P/E facility to the repository, is eliminated. The cask fleet required for spent fuel is 13 road and 88 rail casks, as compared to 13 road and 171 rail casks for Case 1. The cask fleet required for TRU and non-TRU waste also reduces from 10 to 3 road casks.

10.5.2.2 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections and summarized in Table 10-19.

Table 10-19

**ASSESSMENT IMPACT SUMMARY
COLOCATION OF A P/E FACILITY AND A REPOSITORY**

Item	Case 1	Variation
Technical feasibility assessment	Base	No impact
Safeguards assessment	Base	No impact
Criticality assessment	Base	No impact
Radiological assessment	Base	Significant impact reduction
Retrievability assessment	Base	No significant impact
Economic assessment		
AFR, \$/MTHM	53,300	53,300
P/E facility, \$/MTHM	16,700	16,700
Repository, \$/MTHM	19,400	35,400 ^(a)
Transportation		
From reactor, \$/MTHM	12,900	12,900
From AFR, \$/MTHM	4,700	4,700
From P/E facility, \$/MTHM	9,300	—
Total, \$/MTHM	116,000	123,000

(a) In this variation, three repositories are assumed to be regionally located and operating at the same time, as compared with one repository at a time for Case 1.

Technical feasibility. No impact.

Safeguards assessment. This variation reduces the total number of sites handling nuclear material from 13 (in Case 1) to 12. It also eliminates one transportation step. The reduction in the number of facilities and transportation steps reduces the theft and sabotage potential. However, the SFDS assumes the risk from sabotage and theft in each facility to be based on the higher risk (or rating) of the material handled in that facility. The apparent advantage, therefore, does not effectively impact on the safeguards assessment related to this variation.

Criticality assessment. No impact.

Radiological assessment. The reduction of one site and one transportation step handling nuclear material has significantly reduced the radiological assessment of this variation. This variation is not expected to result in as significant a reduction as the AFR-P/E-facility colocation variation.

Retrievability assessment. No impact.

Economics assessment. The economic impact of this variation is due to the change in cost of transportation and of the repository. Three repositories are operating at the same time in this variation as compared to the 1 repository in Case 1. Also, one transportation step, from the P/E facility to the repository, is eliminated with an attendant economic impact.

10.5.3 Colocation of 3 AFRs, 1 P/E Facility, and 1 Repository

In this variation, 3 AFRs, 1 P/E facility, and 1 repository are assumed to be colocated, as shown in Figure 10-3. For this variation, the following assumptions are made:

- Each AFR has a storage capacity of 5,000 MTHM. Three AFRs are located on the same site, with each P/E facility and repository. Three such sites exist
- The 3 repositories are regionally located with a total emplacement capacity of 5,000 MTHM/year
- The average distance between the reactors and the AFR is 700 miles

10.5.3.1 Design Impact. The impact of this variation on the design of the facilities and the transportation system of Case 1 is discussed in the following subsections and summarized in Table 10-20.

AFR. Since 3 AFRs are colocated with each P/E facility and each repository at the same site, these common facilities can be eliminated or consolidated:

- The radwaste and water treatment systems of each AFR can be consolidated, resulting in some savings
- The spent fuel receiving areas of each AFR can be consolidated, resulting in some savings
- The spent fuel shipping areas of each AFR can be eliminated
- The physical security, material control, and accounting systems of the 3 AFRs can be integrated with those of the P/E facility and the repository, resulting in some savings

P/E facility. Since each P/E facility is colocated with 3 AFRs and 1 repository at the same site, this variation has the following impacts:

- The spent fuel receiving area and spent fuel pool at the P/E facility can be eliminated
- The radwaste system of the P/E facility can be consolidated with that of the AFR and the repository, resulting in some savings
- Some utilities and site work can be reduced for the P/E facility

Table 10-20
DESIGN IMPACT SUMMARY
COLOCATION OF THREE AFRs, A P/E FACILITY, AND A REPOSITORY

Item	Case 1	Variation
AFR		
Number at each site	1	3
Number of regional sites	9	3
Capacity of each, MTHM	5,000	5,000
P/E facility		
Number at each site	1	1
Number of regional sites	3	3
Total capacity, MTHM/yr	5,000	5,000
Repository		
Number at each site	1	1
Number of regional sites	1	3
Size of each repository, acres	2,000	2,000
Total capacity, MTHM/yr	5,000	5,000
Transportation		
Average distance from reactor to AFR, miles	400	700
Average distance from AFR to P/E facility, miles	700	0
Average distance from P/E facility to repository, miles	1,100	0
Average distance from AFR to shallow land burial, miles	400	700
Average distance from P/E facility to shallow land burial, miles	700	700
Cask fleet required for spent fuel, road	13	17
Cask fleet required for spent fuel, rail	171	66
Cask fleet required for TRU and non-TRU waste, road	10	3

- The physical security, material control, and accounting system of the AFRs, the P/E facility, and the repository can be consolidated, resulting in some savings
- The surge capacity and the shipping area of the P/E facility can be eliminated

Repository. In this variation, 3 repositories of 2,000 acres each are assumed, with an emplacement rate of 5,000 MTHM/year for the 3 repositories. This results in a lower operating cost for each repository. Also, since each repository is colocated with 1 P/E facility and 3 AFRs, the following areas of the repository are affected:

- The receiving area and the surge storage can be eliminated
- The radwaste handling facilities of the repository can be eliminated. The radwaste generated at the repository can be handled in the P/E facility

Figure 10-3.
Option III – AFRs
P/E Facility, and Repository – Colocated

- The physical security, material control, and accounting systems of the P/E facility, the AFRs, and the repository can be integrated, resulting in some savings
- Some utilities and site work for the repository can also be reduced

Transportation. The transportation steps, from the AFR to the P/E facility and from the P/E facility to the repository, are eliminated in this variation. The cask fleet required for spent fuel is 17 road and 66 rail casks as compared to 13 road and 171 rail casks for Case 1. The cask fleet required for TRU and non-TRU waste also decreases from 10 to 3 road casks.

10.5.3.2 Assessment Area Impact. The impact of this variation, with respect to each assessment area, is discussed in the following subsections and summarized in Table 10-21.

Table 10-21
ASSESSMENT IMPACT SUMMARY
COLOCATION OF THREE AFRs, A P/E FACILITY, AND A REPOSITORY

Item	Case 1	Variation
Technical feasibility assessment	Base	No impact
Safeguards assessment	Base	No impact
Criticality assessment	Base	No impact
Radiological assessment	Base	Significant improvement
Retrievability assessment	Base	No significant impact
Economic assessment		
AFR, \$/MTHM	53,300	51,100
P/E facility, \$/MTHM	16,700	15,700
Repository, \$/MTHM	19,400	35,400 ^(a)
Transportation		
From reactor, \$/MTHM	12,900	18,800
From AFR, \$/MTHM	4,700	—
From P/E facility, \$/MTHM	9,300	—
Total	116,000	121,000

(a) In this variation, three repositories are assumed to be regionally located and operating at the same time, as compared with one repository at a time for Case 1.

Technical feasibility assessment. No impact.

Safeguards assessment. This variation reduces the total number of sites handling nuclear material from 13 (in Case 1) to 3. It also eliminates two transportation steps. The reduction in the number of facilities and transportation steps significantly reduces the potential of theft and sabotage. However, since SFDS assumes the risk from sabotage and theft in each facility, this assessment is based on the highest risk (or rating) of the material handled in that facility. This significant benefit does not effectively impact the safeguards assessment for this variation.

Criticality assessment. No impact.

Radiological assessment. Although there may be additional local impacts to the public due to the abnormal events at a given site, reducing the number of sites and transportation steps is expected to have an overall significant beneficial effect on near-term and long-term radiological impact.

Retrievability assessment. No impact.

Economic assessment. The economic impact of this variation is due to the reduced capital and operating costs for the AFRs, the P/E facility, and the elimination of two transportation steps, but is more than offset by increased costs for three repositories instead of one.

10.6 SPENT FUEL DISASSEMBLY OPTION

This option is similar to Case 1, except the canister contains active spent fuel rods only, instead of complete spent fuel assemblies. In this option the end fittings are removed from the fuel elements, the elements are disassembled, the fuel rods are bundled together, and then sealed into canisters. The canister size is chosen so that a bundle with an acceptable heat load would fill the canister to capacity. These canisters are to be emplaced in the repository 10 years after discharge from the reactor, and a 5-year retrievability period is assumed.

The design data for this spent fuel disassembly option is drawn from the Science Applications, Inc. study report on storage room design parameters.⁽¹¹⁾ The basic considerations are as follows:

Repository thermal load = 60 kW/acre

Heat content of spent fuel = 2.86 watts/rod

Spent fuel canister size = 12.75 inches O.D. (was assumed to be ample to reach clad temperature limitation; however, diameter could have been larger)

10.6.1 Design Impact

The impact of this variation on the design of various facilities and transportation systems is discussed in the following subsections and summarized in Table 10-22.

Table 10-22
DESIGN IMPACT SUMMARY
SPENT FUEL DISASSEMBLY OPTION

Item	Case 1	Variation
AFR	Base	No impact
P/E facility	Base	Some additional facilities
Repository		
Canister spacing, in.	64	73
Thermal load, kW/acre	60	60
Storage rooms, per yr	28	16
Transportation		
Reactor to AFR	Base	No change
AFR to P/E facility	Base	No change
P/E facility to repository		
Encapsulated spent fuel	Base	New casks
TRU and non-TRU waste	Base	No change
Fuel assembly fittings	None	800
Encapsulated spent fuel	10,200	5,100

10.6.1.1 AFR. No impact.

10.6.1.2 P/E Facility. The P/E facility design for this option is more complex than the facility required for Case 1. Additional features needed beyond those described for Case 1 are as follows:

- Fuel disassembly pool or dry hot cell for inspection and assembly breakdown
- Fuel rod packaging area for bundling of the fuel rods
- Assembly fittings disposal area for processing of the remaining nonfuel-bearing items (end fittings, guides, tie rods, etc.)
- Assembly fittings canisters area for welding, testing, inspection, and shipping of waste canisters

The basic difference between this option and Case 1 is the reduction in the number of spent fuel canisters and the addition of canisters for assembly fittings. The reduction in number of spent fuel canisters is due to the increase in the number of fuel rods per canister. The canister for this option contains the equivalent of 2 PWR or 6 BWR assemblies per canister.

10.6.1.3 Repository. The repository design for this option, as compared with Case 1, requires the following feature changes:

- Spent fuel canister hole spacing in the storage rooms is increased from 64-inch pitch to 73-inch pitch
- The number of storage rooms required per year is decreased from 28 to 16, based on the configuration shown in Figure 10-4
- The minimum extraction ratio for 20-foot wide rooms is 7.6 percent

The thermal effects are extensively discussed in the Science Applications study report on repository room design.⁽¹¹⁾ Figure 10-4 shows a typical layout for the storage room and canister hole sizes.

10.6.1.4 Transportation. This option affects the transportation of encapsulated fuel from the P/E facility to the repository. New casks need to be developed for the encapsulated fuel containing 2 PWR or 6 BWR spent fuel rods, as well as for the assembly fittings. The impact of this variation on the transportation cost is not calculated in this analysis.

10.6.2 Assessment Area Impact

The impact of this variation, with respect to each assessment area, is discussed in the following subsections and summarized in Table 10-23.

10.6.2.1 Technical Feasibility. While spent fuel disassembly has not been done on a production basis, studies have not shown technical reasons why this task could not be carried out. Methodology for completing this task is extensively covered in a report by Combustion Engineering, Inc.,⁽¹²⁾ which describes destructive and nondestructive types of disassembly.

The additional processing of the fuel assembly fittings presents no technical obstacles; the technology for chopping and canning of these items is available. The emplacement in the repository is essentially the same as for Case 1, and consequently, technology is available to achieve the goals of the repository.

Table 10-23
ASSESSMENT IMPACT SUMMARY
SPENT FUEL DISASSEMBLY OPTION

Assessment Area	Variable
Technical feasibility	No change
Safeguards	Favorable; some impact but not apparent by the SFDS approach
Criticality	No change
Radiological	Probably no change
Retrievability	No change
Economics	Significantly less repository development (not quantified)

In comparing the overall technical feasibility of this option with Case 1, the following considerations apply:

- Some added degree of difficulty because of the additional processing of fuel assemblies, but the design remains technically feasible
- Main advantage is the reduced number of canisters to be transported to the repository

10.6.2.2 Safeguards Assessment. The transportation of spent fuel represents the highest risk from sabotage or theft in Case 1 (see Subsection 5.3.3). Comparing shipments of encapsulated spent fuel for this option and Case 1 requires approximately 50 percent less shipments for this option. Therefore, a less significant risk factor exists, compared to Case 1, using the risk of theft or sabotage for the AFR, the P/E facility, and the repository, which remains unchanged from Case 1.

Since the risk due to sabotage or theft in any component is based on the highest risk (or rating) of the material handled in that component, the overall risk of rating for this option is the same as for Case 1, within the scope of the SFDS. Since the overall rating for a case is based on the weakest link principle, the overall rating for this variation remains the same as for Case 1.

10.6.2.3 Criticality Assessment. Even though the spent fuel canisters in this study variation (with the equivalent of 2 PWR or 6 BWR assemblies) contain twice the fissile mass of the Case 1 canisters, the formation of a critical mass would require that the same conditions occur as in Case 1, and the same sequence of events would need to take place. Therefore, this variation is considered to have an equally low risk of criticality when compared to Case 1.

10.6.2.4 Radiological Impact. The radiological impact assessment of this variation is essentially the same as the radiological impact assessment for Case 1. Disassembly of the fuel increases the short-term radiological impact at the P/E facility, but this impact will probably be offset by the decrease in radiological impact due to fewer subsequent handling, transportation, and emplacement operations.

Figure 10-4.
Geological Repository Spent Fuel
Waste Emplacement Schematic During Retrieval Period
Fuel Pin Close Spacing

10.7 APPLICATION OF STUDY VARIATIONS

In addition to the conclusions drawn in the assessments of the individual study variations in this section, there are two significant observations to be made about the concepts analyzed.

The first is that each study variation was analyzed in this section independently of any others and was contrasted with the established study basis for Case 1. No combined effects were considered. In future planning it is quite likely that an optimum system will include design criteria aspects drawn from several of the study variations. The ground work has been prepared in this study for such combinations to be applied and evaluated.

The second is that some of the study variation concepts will almost certainly be used in a future optimum spent fuel disposal scenario. It was not the intent of this study to represent the three study cases themselves as ultimate real disposal systems. Rather they were defined to provide an uncomplicated basis for making primary evaluations, and the study variations were introduced to assess changes in those case definitions.

REFERENCES FOR SECTION 10

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- (3) "Spent Fuel Receiving and Packaging Facility, Conceptual Design Report," Rockwell International, RHO-CD-506 Draft, September 1978.
- (4) "Preliminary Conceptual Design Description for National Waste Repository No. 1," Stearns-Roger Engineering Company, EY-77-C-05-5367, November 1977.
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- (11) "Storage Room Design Parameters for the ONWI Spent Fuel Disposal Study: Nine Case Studies in Salt," Science Applications Inc., ONWI/78/Sub-E513-00413-1, March 23, 1979.
- (12) "Report on Fuel Element Assembly-Disassembly Methods," Combustion Engineering, Inc., Power Systems for AGNS, September 1978.
- (13) "Draft Environmental Impact Statement — Management of Commercially Generated Radioactive Waste," U.S. Department of Energy, DOE/EIS-0046-D, April 1979.