128
~~11/17/78~~
OR 743 UC-10
MASTER MASTER

Analytical Methodology and Facility Description Spent Fuel Policy

August 1978

U.S. Department of Energy
Assistant Secretary for Energy Technology

DISCLAIMER

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

DISCLAIMER

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

Analytical Methodology and Facility Description Spent Fuel Policy

NOTICE

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

August 1978

U.S. Department of Energy
Assistant Secretary for Energy Technology
Washington, D.C. 20545

This document is
PUBLICLY RELEASABLE
Bary Steels

Authorizing Official
Date: *7-18-07*

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Available from:

National Technical Information Service (NTIS)
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161

Price: Printed Copy: \$11.00
Microfiche: \$ 3.00

FOREWORD

In October 1977, a Presidential policy on the interim management of spent fuel was announced. Under this policy, the Federal government would offer to take title to and provide interim storage for spent fuel from U. S. power reactors and possibly from some foreign reactors. Three generic environmental impact statements (GEISs) on domestic fuels, foreign fuels, and storage charges are being prepared to provide environmental input into decisions on whether, and if so, how this spent fuel storage policy should be implemented. This report provides background information for two of these environmental impact statements: *Storage of U. S. Spent Power Reactor Fuel* and *Storage of Foreign Spent Power Reactor Fuel*. It includes the analytical methodology used in the GEISs to assess the environmental effects and description of the facilities used in the two GEISs.

Other related environmental reviews which provided input to this report include *Draft Generic Environmental Impact Statement on Handling and Storage of Spent Fuel from Light Water Reactor Power Reactors* (NUREG 0404) and *Light Water Reactor Fuel Reprocessing and Recycling* (ERDA-77-75).

CONTENTS

I. Characteristics of LWR Fuel I-1

A. LWR Fuel I-1

A.1 Description of an LWR Fuel Assembly I-1

A.2 Characteristics of Spent LWR Fuel I-3

A.2.1 Radionuclides I-7

A.2.1.1 Fission Products I-7

A.2.1.2 Transuranics I-7

A.2.1.3 Activation Products I-13

A.2.2 Heat Generation I-13

References for Section I,A I-14

B. HWR Fuels I-15

B.1 Description of a CANDU Fuel Assembly I-15

B.2 Characteristics of Spent CANDU Fuel I-15

B.2.1 Radionuclides I-15

B.2.1.1 Fission Products I-22

B.2.1.2 Transuranics I-22

B.2.1.3 Activation Products I-22

B.2.2 Heat Generation I-22

References for Section I,B I-23

II. Generic Facility Description II-1

A. Storage of Spent Fuel in Independent Spent Fuel Storage (ISFS) Water Basins II-1

A.1 Design Criteria II-4

A.1.1 Activity Containment and Integrity II-5

A.1.2 Radiation Protection II-6

- A.2 Operations at ISFS Water Basins II-6
 - A.2.1 Handling and Storage II-6
 - A.2.1.1 Cask-Carrier Handling II-6
 - A.2.1.2 Cask Processing II-6
 - A.2.1.3 Underwater Handling and Storage II-8
 - A.2.1.3.1 Fuel Unloading II-8
 - A.2.1.3.2 Spent Fuel Storage Basins II-8
 - A.2.1.3.3 Spent Fuel Loading and Shipping II-8
 - A.2.2 Criticality Hazard and Control II-9
- A.3 Facility Description II-9
 - Throughput and Capacity II-11
 - Fuel II-11
 - Water Pools and Basins II-11
 - Confinement and Waste Management II-12
 - Personnel Protection II-12
 - A.3.1 Cask-Carrier Handling II-14
 - A.3.2 Cask Processing II-14
 - A.3.2.1 Cask Offload-Load Area II-16
 - A.3.2.2 Cask Cooling and Washdown II-17
 - A.3.2.3 Cask Decontamination II-17
 - A.3.3 Underwater Handling and Storage II-17
 - A.3.3.1 Cask Unloading Pool II-18
 - A.3.3.2 Storage Basins II-20
 - A.3.3.3 Transfer Aisles II-21
 - A.3.3.4 Crane Maintenance Areas II-21
- A.4 Support Systems II-21
 - A.4.1 Cooling Water Systems II-22
 - A.4.1.1 Closed Loop Primary Cooling Water System II-22
 - A.4.1.2 Secondary Cooling Water System II-23
 - A.4.1.3 Emergency Cooling Water System II-23

A.4.2	Basin Water Cleanup	II-24
A.4.3	Heating, Ventilation, and Air Conditioning Systems	II-24
A.4.4	Radioactive Waste Generation and Treatment	II-25
A.4.4.1	Evaporation	II-28
A.4.4.2	Immobilization of Wastes	II-28
A.4.4.2.1	Cement Immobilization	II-28
A.4.4.2.2	Bitumen Immobilization	II-29
A.4.4.3	General Trash and Combustible Waste Treatment	II-31
A.4.4.4	Failed Equipment and Noncombustible Waste Treatment	II-33
A.4.4.5	Off-Gas System	II-33
A.5	Facility Operation	II-35
A.5.1	Operating Manpower	II-36
A.5.2	Consumable Material	II-37
A.5.3	Waste Generation	II-37
A.5.3.1	Secondary Radioactive Waste	II-37
A.5.3.2	Nonradioactive Waste	II-41
References for Section II,A		II-44
B.	Geologic Storage in Salt Formation	II-46
B.1	Design Bases	II-47
B.2	Operation at Geologic Storage Facilities	II-51
B.2.1	Spent Fuel Handling Operations	II-51
B.2.2	Reprocessing Waste Handling Operations	II-53
B.2.3	Low-Level Waste Handling Operations	II-54
B.2.4	Mining Operations	II-54
B.3	Facility Description	II-56
B.3.1	Surface Facilities	II-57
B.3.1.1	Canistered Waste Building	II-57
B.3.1.2	Low-Level Waste Building	II-58
B.3.1.3	Support Buildings	II-58
B.3.1.4	Salt Tailings Storage Warehouse	II-61

- B.3.1.5. Salt Tailings Disposal Area II-61
- B.3.2 Underground Facilities II-62
 - B.3.2.1 Shafts and Hoists II-62
 - B.3.2.1.1 Canistered Waste Shaft II-62
 - B.3.2.1.2 Low-Level Waste Shaft II-62
 - B.3.2.1.3 Men and Materials (M&M) Shaft II-63
 - B.3.2.1.4 Ventilation Exhaust Shaft II-63
 - B.3.2.2 Mine Facilities II-64
 - B.3.2.2.1 BWR Spent Fuel Mine II-64
 - B.3.2.2.2 PWR Spent Fuel Mine II-65
 - B.3.2.2.3 CANDU Spent Fuel Mine II-66
 - B.3.2.2.4 Reprocessing Waste Mine II-66
 - B.3.2.2.5 Low-Level Transuranic Waste Mine II-67
 - B.3.2.3 Ventilation Systems II-67
 - B.3.2.3.1 Underground Ventilation Systems II-68
 - B.3.2.3.2 Ventilation of Surface Structures II-68
- B.4 Facility Operation II-69
 - B.4.1 Operating Manpower II-69
 - B.4.1.1 Subsurface Manpower II-70
 - B.4.1.1.1 Mining and Backfilling II-70
 - B.4.1.1.2 Spent Fuel or Waste Transport and Burial II-70
 - B.4.1.2 Surface Manpower II-71
 - B.4.2 Consumable Material II-71
 - B.4.3 Thermal Releases II-73

References for Section II,B II-77

- C. Spent Fuel Reprocessing and Mixed Oxide Fuel Fabrication II-78
 - C.1 Design Bases II-79
 - C.2 Operation at Fuel Reprocessing - MOX Plants II-82
 - C.2.1 Fuel Reprocessing II-82

C.2.2	Production of Mixed Oxide Fuels for LWR Reactors	II-84
C.2.3	Waste Treatment and Interim Storage	II-85
C.2.3.1	Liquid Waste	II-86
C.2.3.2	Solid Waste	II-87
C.2.4	Storage of Plutonium	II-88
C.3	Facility Description	II-88
C.3.1	Fuel Receiving and Storage	II-90
C.3.2	Fuel Reprocessing Plant	II-91
C.3.3	Uranium Conversion Facility (UF ₆ Plant)	II-96
C.3.4	Plutonium Conversion Facility	II-96
C.3.5	Mixed Oxide (MOX) Fuel Fabrication Facility	II-102
C.4	Support Systems	II-104
C.4.1	Building Ventilation Systems	II-105
C.4.2	Electrical	II-106
C.4.3	Water Supply	II-106
C.4.4	Sewage	II-106
C.4.5	Communications and Alarms	II-106
C.4.6	Fire Protection	II-106
C.4.7	Maintenance	II-107
C.5	Facility Operation	II-107
C.5.1	Operating Manpower	II-108
C.5.2	Consumable Materials	II-108
C.5.3	Waste Generation	II-108
	High-Level Waste (HLW)	II-108
	Cladding Waste	II-111
	Intermediate-Level Waste (ILW)	II-111
	Low-Level Transuranic Waste (LLW-TRU)	II-111
	Low-Level Wastes (LLW)	II-112
	References for Section II,C	II-113

III. Transportation III-1

- A. Federal, State, Local and International Regulatory Requirements for Spent Fuel Shipping Casks III-2
 - A.1 Federal Regulations III-3
 - A.1.1 NRC Regulations III-3
 - A.1.2 DOT Regulations III-4
 - A.1.3 ICC Regulations III-6
 - A.1.4 Pending Regulatory Issues III-6
 - A.2 State and Local Regulations III-7
 - A.3 International Regulations III-8
- B. Description of Existing Spent Fuel Casks III-9
 - B.1 Truck Casks III-11
 - B.2 Rail Casks III-13
 - B.3 Casks for Ships III-16
- C. Future Spent Fuel Casks III-19
- D. Spent Fuel Cask Handling Operations III-20
 - D.1 Operations at Storage Basins, Ports, and the Disposition Facilities III-20
 - D.2 Transportation Routing III-20
- E. Packages and Casks for Radioactive Wastes III-21
 - E.1 Wastes from Storage Basins III-21
 - E.2 Wastes from Reprocessing of Spent Fuel III-22
- F. Generic Casks III-23
 - F.1 Land Transport III-23
 - F.2 Maritime Transportation III-25

References for Section III,C III-26

IV. Assumptions and Models Used to Assess Environmental Effects From Storage Facilities IV-1

- A. Meteorology IV-1
 - A.1 Releases From Normal Operations IV-1
 - A.2 Accidental Releases IV-3

- B. Local Dose IV-3
 - B.1 Maximum Individual Dose IV-3
 - B.2 80-Kilometer Radius Population Dose IV-3
 - B.3 Uptake-Dose Models IV-5
 - B.3.1 Tritium IV-5
 - B.3.2 Carbon-14 IV-6
 - B.3.3 Krypton-85 IV-6
 - B.3.4 Iodine-129 IV-6
 - B.3.5 Other Nuclides and Pathways IV-8
 - B.3.5.1 External Exposure to Gamma-Emitters Deposited on the Ground IV-8
 - B.3.5.2 Inhalation IV-8
 - B.3.5.3 Foodstuffs IV-10
 - B.3.5.4 Direct External Exposure IV-10
- C. United States Dose IV-12
 - C.1 Atmospheric Concentrations and Population Distribution IV-12
 - C.2 Tritium IV-12
 - C.3 Carbon-14 IV-12
 - C.4 Krypton-85 IV-12
 - C.5 Iodine-129 IV-13
 - C.6 Other Nuclides IV-13
- D. Worldwide Dose IV-13
 - D.1 Population Growth IV-13
 - D.2 Tritium IV-13
 - D.3 Carbon-14 IV-14
 - D.4 Krypton-85 IV-14
- E. Local and Regional Dose — Foreign Countries IV-15
- References for Section IV IV-17

V. Assumptions and Methodology Used to Assess Environmental Effects of Transportation V-1

A. Demographic Model V-2

B. General Assumptions V-4

B.1 Thermal and Radiological Characteristics V-4

B.2 Modes of Spent Fuel Shipments V-6

B.2.1 Transshipment From Domestic Reactor Basin to Other Reactor Basins V-7

B.2.2 Domestic Reactor Basin to ISFS Basin V-7

B.2.3 Domestic Reactor Basins to Disposition Facility V-7

B.2.4 ISFS Basin to Disposition Facility V-8

B.2.5 Disposition Facility or ISFS Basin to Fuel Reprocessing Plant (FRP) V-8

B.2.6 Reactor Basins to Spent Fuel Reprocessing Plant (FRP) V-9

B.2.7 Foreign Ports to U.S. Ports V-9

B.2.8 U.S. Ports to ISFS Basins V-10

B.2.9 U.S. Ports to Disposition Facility V-10

B.3 Categories of Waste Shipments V-10

B.3.1 ISFS Basins to Land Burial Site and Disposition Facility V-10

B.3.2 FRP-MOX to Land Burial Sites or Disposition Facilities V-11

B.4 Abnormal Events During Transportation V-11

B.4.1 Transportation Accident Statistics V-12

B.4.1.1 Land Transportation V-12

B.4.1.2 Maritime Transportation V-14

B.4.2 Mitigating the Consequences of Transportation Accidents V-14

B.4.3 Fuel Element Failure During Transportation V-15

B.4.4 Postulated Accidents for Shipments of Spent Fuel V-16

B.4.5 Postulated Accidents for Shipments of Reprocessing Wastes V-25

B.4.5.1	High-Level Wastes	V-25
B.4.5.2	Cladding Wastes	V-26
B.4.5.3	Intermediate-Level Wastes	V-31
B.4.5.4	Low-Level Transuranic (LLW-TRU) Wastes	V-31
B.4.5.5	Low-Level Wastes	V-31
B.5	Transportation Workers	V-34
B.5.1	Personnel Requirements	V-34
B.5.2	Injuries and Fatalities in Transportation Accidents	V-35
C.	Methods Used to Assess Environmental Effects of Transportation	V-36
C.1	Summary	V-37
C.2	Dosimetry Methods for Normal Transport	V-39
C.2.1	Dose to Persons Surrounding the Transport Route as Shipment Passes	V-40
C.2.2	Dose to Population During Shipment Stops	V-45
C.2.3	Dose to Crewmen	V-46
C.2.4	Dose to Persons in Vehicles Sharing the Highway with the Shipment	V-47
C.2.4.1	Dose to Persons Traveling in the Opposite Direction	V-47
C.2.4.2	Dose to Persons Traveling in the Same Direction as the Shipment	V-50
C.2.5	Parameters for Normal Transportation Calculations	V-52
C.3	Assumptions for Calculation of Maximum Individual Dose and Consequences of Transportation	V-53
C.3.1	Normal Transportation	V-53
C.3.2	Transportation Accidents	V-54
C.4	Occupational Exposure — Transportation Workers	V-56
C.5	Nonradiological Releases	V-56
C.6	Thermal Effects	V-56
	References for Section V	V-58

VI. Radiological Health Effects VI-1

References for Section VI VI-7

Appendix — Glossary of Terms and Abbreviations A-1

LIST OF FIGURES

I-A-1	Typical PWR Fuel Rod	I-2
I-A-2	PWR Fuel Assembly	I-5
I-A-3	BWR Fuel Assembly	I-6
I-B-1	CANDU Fuel Bundle for Pickering Reactor	I-16
II-A-1	G. E. Morris Operation — Fuel Pool	II-2
II-A-2	G. E. Morris Operation — Fully Loaded Storage	II-3
II-A-3	Spent Fuel Handling and Storage	II-7
II-A-4	Plot Plan for Water Storage Facility for Unpackaged Spent LWR Fuel	II-10
II-A-5	Cask and Fuel Handling Building — Fuel Storage Pool Layout	II-13
II-A-6	Stainless Steel Storage Basket — PWR Assembly	II-19
II-A-7	Stainless Steel Storage Basket — BWR Assembly	II-19
II-A-8	Heat Dissipation System	II-23
II-A-9	Ventilating Air Flow Requirement	II-26
II-A-10	Radioactive Waste from Spent Fuel Handling	II-27
II-A-11	Process Flow Diagram for Cement Immobilization	II-29
II-A-12	Process Flow Diagram for Bitumen Immobilization	II-30
II-A-13	Incineration Process	II-32
II-A-14	Flow Diagram for Treatment of Failed Equipment and Noncombustible Waste	II-34
II-A-15	Flow Diagram for Off-Gas System	II-35
II-A-16	ISFS Operating Staff	II-36

II-A-17	Fossil Fuel Consumption	II-38
II-A-18	Secondary Radioactive Waste Generation	II-39
II-A-19	Secondary Radioactive Waste Generation	II-40
II-A-20	Nonradioactive Liquid Waste Generation Rate	II-41
II-A-21	Nonradioactive Liquid Waste Generation	II-42
II-A-22	Nonradioactive Solid Waste Generation	II-43
II-B-1	Spent Fuel Handling Operation	II-52
II-B-2	Schematic Flow Diagram — Low-Level Waste Building	II-55
II-B-3	Plot Plan for Repository Surface Facilities	II-56
II-B-4	Underground Repository	II-57
II-B-5	Canistered Waste Building — Plans and Sections	II-59
II-B-6	Low-Level Waste Building — Plans and Sections	II-60
II-B-7	Retrievable Burial	II-65
II-B-8	Schematic Mine Ventilation	II-69
II-B-9	Transporter Manpower	II-71
II-B-10	Surface Staff	II-72
II-C-1	Flow Diagram for Mixed Oxide Fuel Cycle	II-80
II-C-2	Simplified Flow Diagram for Reprocessing Plant	II-81
II-C-3	Plot Plan for Collocated Fuel Reprocessing and MOX Plant	II-89
II-C-4	Ventilation System for Fuel Reprocessing Plant	II-94
II-C-5	High-Level Liquid Waste Solidification	II-97
II-C-6	Flow Diagram for UF ₆ Facility	II-98
II-C-7	Flow Diagram for Plutonium Product Facility	II-100
II-C-8	Ventilation and Off-Gas System for Plutonium Product Facility	II-101

II-C-9	Simplified Flow Diagram for MOX Plant	II-103
II-C-10	Collocated FRP-MOX Operating Staff	II-109
II-C-11	Fossil Fuel Consumption	II-110
III-1	NLI 10/24 Irradiated Fuel Shipping Cask	III-14
III-2	NLI 10/24 Cask and Carrier	III-15
III-3	GE IF 300 Irradiated Fuel Shipping Cask	III-17
III-4	GE IF 300 Cask and Carrier	III-18
IV-1	X/Q as a Function of Distance Averaged Over All Sectors	IV-2
IV-2	Population Growth with Time	IV-4
V-1	Dose Received by an Individual as a Shipment Passes	V-40
V-2	Dose to Person Living Along the Transport Route	V-42
V-3	Dose to Persons in Vehicles Sharing the Highway with the Shipment	V-47

LIST OF TABLES

I-A-1	Characteristics of Typical LWR Fuel Assemblies	I-4
I-A-2	Major Fission Product Activity in Aged PWR Spent Fuel	I-8
I-A-3	Major Transuranic Activity in Aged PWR Spent Fuel	I-10
I-A-4	Major Activation Products in Aged PWR Fuel	I-11
I-A-5	Effect of Decay on Thermal Power of Irradiated Fuel	I-12
I-B-1	Characteristics of Typical CANDU-HWR Fuel Assemblies	I-16
I-B-2	Major Fission Product Activity in Aged CANDU Spent Fuel	I-17
I-B-3	Major Transuranic Activity in Aged CANDU Spent Fuel	I-19
I-B-4	Major Activation Products in Aged CANDU Fuel	I-20
I-B-5	Effect of Decay on Thermal Power of Irradiated CANDU Fuel	I-21
II-A-1	Cask Process Times	II-15
II-A-2	Minimum Number of Cask Handling Locations at a Fuel Storage Facility	II-16
II-B-1	Geologic Repository Thermal Loading Limit	II-49
II-B-2	Thermal Characteristics of Spent Fuel	II-50
II-B-3	Parameters for Heat Generation Function — Spent PWR Fuel	II-75
II-B-4	Heat Generation Rate of Spent Fuel, Function Versus ORIGEN Code	II-76

II-C-1	Retention Factors for Radionuclides in FRP-MOX Plant	II-95
III-1	Licensed and Available Shipping Casks for Current Generation LWR Spent Fuel	III-10
IV-1	Relative Concentration and Deposition Fractions	IV-3
IV-2	Thyroid Doses from Unit Concentration of ^{129}I in Air	IV-7
IV-3	Dose Rate and Integrated Dose Conversion Factors from Deposited Radionuclides	IV-9
IV-4	50-Yr Dose Conversion Factors for Inhalation of Radionuclides, rem/ μCi Inhaled	IV-9
IV-5	50-Yr Dose Commitment Factors for Ingestion of Radionuclides, rem/ μCi Ingested	IV-11
IV-6	United States and Foreign Country Population Densities	IV-16
V-1	Demographic Model for 1984	V-3
V-2	Radioactive Properties of Spent Fuel Normalized to 0.5-Year Cooled Fuel	V-4
V-3	Thermal Loading for Shipping Casks	V-5
V-4	Rail and Truck Accident Probabilities	V-13
V-5	Estimated Activity Available for Release in Penetration of Fuel Rod Cladding	V-19
V-6	Spent Fuel Rail Cask — Minor and Moderate Accidents	V-19
V-7	Spent Fuel Rail Cask — Severe Accidents	V-20
V-8	Spent Fuel Rail Cask — Extra Severe or Extreme Accidents	V-21
V-9	Spent Fuel Truck Cask — Minor and Moderate Accidents	V-22
V-10	Spent Fuel Truck Cask — Severe Accidents	V-23

V-11	Spent Fuel Truck Cask — Extra Severe and Extreme Accidents	V-24
V-12	Solidified High-Level-Waste Rail Cask — Minor and Moderate Accidents	V-27
V-13	Solidified High-Level-Waste Rail Cask — Severe Accidents	V-28
V-14	Solidified High-Level-Waste Rail Cask — Extra Severe to Extreme Accidents	V-28
V-15	Cladding Waste Rail Cask — Minor, Moderate or Severe Accidents	V-29
V-16	Cladding Waste Rail Cask — Extra Severe or Extreme Accidents	V-30
V-17	Low-Level TRU Waste Containers — Minor, Moderate or Severe Accidents	V-32
V-18	Low-Level TRU Waste Containers — Extra Severe or Extreme Accidents	V-33
V-19	Risks of Injury or Fatality from Accidents During Transportation	V-36
V-20	Population and Transportation Workers Dose — Normal Transport of a Spent Fuel Cask	V-38
V-21	Comparative Population and Transportation Worker Dose for Different Spent Fuel Casks	V-38
V-22	Parameters for Calculating Radiation Doses During Normal Transportation	V-52
V-23	Emission Factors for Operation of Transportation Equipment	V-57
VI-1	Radiological Dose-Effect Conversion Factors	VI-5

I. CHARACTERISTICS OF LWR FUEL

The new Federal policy on storage of spent reactor fuel does not exclude any type of fuel, but two fuel types are expected to predominate: light water reactor (LWR) and heavy water reactor (HWR) fuels, the latter particularly from foreign reactors.

A. LWR FUEL

Most power reactors in the United States are light water reactors. The LWR fuel is metal-clad uranium dioxide (UO_2) in which the readily fissionable uranium-235 (^{235}U) has been enriched to a maximum 3 to 4%. The balance of the uranium consists primarily of nonfissionable uranium-238 (^{238}U). Two main types of light water reactors are pressurized water reactors (PWR) and boiling water reactors (BWR). These reactor types have somewhat different fuel designs.

A.1 Description of an LWR Fuel Assembly

Nuclear fuel for light water cooled and moderated power reactors consists of short cylinders (pellets) of high-fired (sintered) uranium dioxide (UO_2). The sintered fuel pellets are hard, strong, and insoluble in water. Depending upon the specific reactor design, these pellets are 0.75 to 1.25 cm (0.30 to 0.49 in.) in diameter and about 1.5 cm (0.59 in.) long.

A stack of these pellets about 366 to 376 cm (144 to 148 in.) long is loaded and hermetically sealed in a zirconium alloy cladding tube.* This unit, called a fuel rod (Figure I-A-1), is a strong but flexible structure. A fuel rod for a pressurized water reactor (PWR) is about 1 cm (0.39 in.) in diameter and 380 cm (150 in.) long and consists of about 2 kg (0.91 lb) of slightly enriched uranium in oxide form. A fuel rod for the boiling water reactor (BWR) is about 1.4 cm (0.55 in.) in diameter and 400 cm (157 in.) long and consists of about 3 kg (1.4 lb) of enriched uranium in the oxide form.

* Zirconium alloy (Zircaloy), which is highly resistant to corrosion in water, is currently used for cladding; in some early fuel element designs stainless steel was used.

FIGURE I-A-1. Typical PWR Fuel Rod¹

Fuel rods are assembled into bundles in a square array. Each rod is spaced and supported by grid structures and end pieces, which are also highly resistant to corrosion. Although similar in design, fuel assemblies used in PWRs and BWRs differ significantly in size and the quantity of fuel they contain (Table I-A-1).

A typical PWR assembly (Figure I-A-2) contains 264 fuel rods in a square array (17×17 positions); reactivity control rods, instrumentation, and burnable-poison elements are also in the array. A PWR assembly has no cover or coolant channel, but it is secured by top and bottom nozzle fixtures and spring-clip grids which are spaced along the length of the element cluster. A typical PWR assembly contains about 460 kg (1012 lb) of uranium. A complete fuel assembly weighs about 670 kg (1470 lb), has a cross section of 22×22 cm (8.7×8.7 in.), and is about 4 m (13.1 ft) long.

In a typical present-day BWR assembly (Figure I-A-3), the fuel rods are arranged in a 7×7 rod square array, but advanced fuels will contain 8×8 arrays. The individual rods are secured by upper and lower tie plates and spacer grids placed at intervals along the element cluster. The entire assembly is covered by a thin Zircaloy sheet-metal box called a fuel channel. The fuel channel is normally reused after an assembly is withdrawn. A typical BWR assembly contains about 190 kg (418 lb) of uranium. The total weight of a complete assembly is about 280 kg (616 lb) (excluding fuel channel); the cross section is 14×14 cm (5.5×5.5 in.), and the length is about 4.5 m (14.8 ft).

A.2 Characteristics of Spent LWR Fuel

After an LWR fuel assembly is irradiated, it normally shows no outward physical change other than the presence of a thin film of corrosion product. However, within the fuel rods, two changes occur:

- The UO_2 pellets fracture because of thermal stress.
- The composition of the rods changes because of irradiation.

The nuclear reactions during irradiation are associated with fission and transmutation which produce a large number of radioactive nuclides. These new nuclides are contained in the structural matrix of the fuel, in the annular gap surrounding the fuel pellets, and in the hardware components of the fuel assembly.

TABLE I-A-1

Characteristics of Typical LWR Fuel Assemblies¹

	<u>PWR</u>	<u>BWR</u>
Overall assembly length, m	4.059	4.470
Cross section, cm	21.4 × 21.4	13.9 × 13.9
Fuel element length, m	3.851	4.064
Active fuel height, m	3.658	3.759
Fuel element OD, cm	0.950	1.252
Fuel element array	17 × 17	8 × 8
Assembly total weight, kg	668.6	279
U/assembly, kg	461.4	188.7
UO ₂ /assembly, kg	523.4	214.1
Zircaloy/assembly, kg	129.7 ^a	56.7 ^b
Hardware/assembly, kg	15.5 ^c	8.2 ^d
Total metal/assembly, kg	145.2	64.9

^a. Includes Zircaloy control-rod guide thimbles.

^b. Includes Zircaloy Fuel-rod spacers.

^c. Includes 10-kg SS nozzles and 5.5-kg *Inconel**-718 grids.

^d. Includes SS tie plates and negligible amount of *Inconel* springs.

* Trademark of Huntington Alloys, Inc.

FIGURE I-A-2. PWR Fuel Assembly¹

FIGURE I-A-3. BWR Fuel Assembly¹

New fuel is relatively harmless and can be handled and shipped as a standard commercial product, but spent fuel is highly radioactive and produces considerable heat. Therefore, spent fuel must be cooled and shielded. The cooling and shielding requirements of the spent fuel decrease with time, as a result of the natural process of radioactive decay.

The radioactive nuclide inventories and thermal powers of spent fuel from a typical large PWR were calculated by using the ORIGEN code² (Tables I-A-2 through I-A-5). Irradiation exposure and specific power were assumed to be 33,000 megawatt-days thermal per metric ton of uranium (MWD/MTU) and 30 megawatts per metric ton (MW/MTU), respectively. Composition of BWR spent fuel would be similar, but the radioactive nuclide inventory and thermal power would be less because of lower average burnup and specific power. For conservatism, the generic environmental impact statement on the Spent Fuel Policy^{3,4} assesses the environmental effects of spent LWR fuel management by using fission product inventories and thermal powers (per MTU) calculated for PWR conditions.

A.2.1 Radionuclides

A.2.1.1 Fission Products

During reactor operation, a portion of the uranium initially present in the fuel and some of the plutonium generated *in situ* is fissioned; the fissioning process results in the formation of fission products and the release of neutrons and energy as radiation. The fission products consist of two groups of elements: 1) elements which have atomic weights somewhat less than half that of uranium, and 2) elements which have atomic weights somewhat more than half that of uranium. Some of these fission product elements are stable and indistinguishable from elements found in nature; some are unstable (radioactive) and seek a stable state by radioactive decay. A measure of their instability is the half-life of each radionuclide. These half-lives vary from a small fraction of a second to millions of years. Table I-A-2 lists the major fission products present in spent fuel up to 30 years after discharge from the reactor.

A.2.1.2 Transuranics

During reactor operation, a portion of the uranium present also undergoes nuclear transmutation by the absorption of neutrons. These transmutation products constitute a group of heavy nuclides known as transuranic elements (elements above uranium in the periodic table, i.e., elements with atomic numbers greater than 92). Major transuranics in the spent fuel are listed in Table I-A-3.

Major Fission Product Activity in Aged PWR Spent Fuel
(Curies per MT of Uranium Charged to Reactor)

Radionuclide	Years After Discharge							
	0.5	1.0	2.0	4.0	6.0	8.0	10.0	30.0
^3H	5.0×10^2	4.8×10^2	4.6×10^2	4.1×10^2	3.7×10^2	3.3×10^2	2.9×10^2	9.5×10^1
^{85}Kr	1.1×10^4	1.1×10^4	1.0×10^4	8.8×10^3	7.8×10^3	6.8×10^3	6.0×10^3	1.7×10^3
^{89}Sr	6.4×10^4	5.6×10^3	4.3×10^1	-	-	-	-	-
^{90}Sr	7.7×10^4	7.6×10^4	7.5×10^4	7.1×10^4	6.8×10^4	6.4×10^4	6.1×10^4	3.7×10^4
^{90}Y	7.7×10^4	7.6×10^4	7.5×10^4	7.1×10^4	6.8×10^4	6.4×10^4	6.1×10^4	3.7×10^4
^{91}Y	1.1×10^5	1.3×10^4	1.7×10^2	-	-	-	-	-
^{95}Zr	2.0×10^5	2.8×10^4	5.7×10^2	2.4×10^{-1}	-	-	-	-
$^{95\text{m}}\text{Nb}$	4.2×10^3	6.0×10^2	1.2×10^1	-	-	-	-	-
^{95}Nb	3.8×10^5	6.0×10^4	1.2×10^3	5.3×10^{-1}	-	-	-	-
^{99}Tc	1.4×10^1	1.4×10^1	1.4×10^1	1.4×10^1	1.4×10^1	1.4×10^1	1.4×10^1	1.4×10^1
^{103}Ru	5.0×10^4	2.0×10^3	3.4×10^0	-	-	-	-	-
$^{103\text{m}}\text{Rh}$	5.0×10^4	2.0×10^3	3.4×10^0	-	-	-	-	-
^{106}Ru	3.8×10^5	2.7×10^5	1.3×10^5	3.4×10^4	8.5×10^3	2.1×10^3	5.4×10^2	-
^{106}Rh	3.8×10^5	2.7×10^5	1.3×10^5	3.4×10^4	8.5×10^3	2.1×10^3	5.4×10^2	-
^{123}Sn	3.2×10^3	1.2×10^3	1.5×10^2	2.7×10^0	-	-	-	-
^{125}Sb	7.6×10^3	6.7×10^3	5.2×10^3	3.1×10^3	1.9×10^3	1.1×10^3	6.6×10^2	3.9×10^0
$^{129\text{m}}\text{Te}$	1.4×10^3	3.4×10^1	-	-	-	-	-	-
^{129}Te	8.9×10^2	2.1×10^1	-	-	-	-	-	-
^{129}I	3.7×10^{-2}	3.7×10^{-2}	3.7×10^{-2}	3.7×10^{-2}	3.7×10^{-2}	3.7×10^{-2}	3.7×10^{-2}	3.7×10^{-2}
^{131}I	1.3×10^{-1}	-	-	-	-	-	-	-

TABLE I-A-2, Continued

<i>Radionuclide</i>	<i>Years After Discharge</i>							
	<i>0.5</i>	<i>1.0</i>	<i>2.0</i>	<i>4.0</i>	<i>6.0</i>	<i>8.0</i>	<i>10.0</i>	<i>30.0</i>
^{134}Cs	2.0×10^5	1.7×10^5	1.2×10^5	6.3×10^4	3.2×10^4	1.6×10^4	8.3×10^3	9.6×10^0
^{137}Cs	1.1×10^5	1.0×10^5	1.0×10^5	9.8×10^4	9.4×10^4	9.0×10^4	8.6×10^4	5.4×10^4
$^{137\text{m}}\text{Ba}$	1.0×10^5	9.8×10^4	9.6×10^4	9.2×10^4	8.8×10^4	8.4×10^4	8.0×10^4	5.0×10^4
^{141}Ce	2.8×10^4	5.7×10^2	2.3×10^{-1}	-	-	-	-	-
^{144}Ce	7.2×10^5	4.6×10^5	1.9×10^5	3.2×10^4	5.3×10^3	8.9×10^2	1.5×10^2	-
^{144}Pr	7.2×10^5	4.6×10^5	1.9×10^5	3.2×10^4	5.3×10^3	8.9×10^2	1.5×10^2	-
^{147}Pm	9.7×10^4	8.5×10^4	6.5×10^4	3.8×10^4	2.3×10^4	1.3×10^4	7.9×10^3	4.0×10^1
$^{148\text{m}}\text{Pm}$	1.9×10^3	9.5×10^1	2.3×10^{-1}	-	-	-	-	-
^{151}Sm	1.2×10^3	1.2×10^3	1.2×10^3	1.2×10^3	1.2×10^3	1.2×10^3	1.2×10^3	9.9×10^2
^{154}Eu	6.7×10^3	6.6×10^3	6.3×10^3	5.8×10^3	5.3×10^3	4.8×10^3	4.4×10^3	1.9×10^3
^{155}Eu	6.1×10^3	5.0×10^3	3.4×10^3	1.6×10^3	7.4×10^2	3.4×10^2	1.6×10^2	-
Total Fission Product Activity	3.8×10^6	2.2×10^6	1.2×10^6	5.9×10^5	4.2×10^5	3.5×10^5	3.2×10^5	1.8×10^5

TABLE I-A-3

Major Transuranic Activity in Aged PWR Spent Fuel
(Curies per MT of Uranium Charged to Reactor)

<i>Radionuclide</i>	<i>Years After Discharge</i>							
	<i>0.5</i>	<i>1.0</i>	<i>2.0</i>	<i>4.0</i>	<i>6.0</i>	<i>8.0</i>	<i>10.0</i>	<i>30.0</i>
^{238}Pu	2.8×10^3	2.8×10^3	2.8×10^3	2.8×10^3	2.7×10^3	2.7×10^3	2.6×10^3	2.3×10^3
^{239}Pu	3.2×10^2	3.2×10^2	3.2×10^2	3.2×10^2	3.2×10^2	3.2×10^2	3.2×10^2	3.2×10^2
^{240}Pu	4.7×10^2	4.7×10^2	4.7×10^2	4.7×10^2	4.7×10^2	4.7×10^2	4.7×10^2	4.7×10^2
^{241}Pu	1.0×10^5	9.8×10^4	9.4×10^4	8.5×10^4	7.7×10^4	7.0×10^4	6.4×10^4	2.5×10^4
^{241}Am	1.7×10^2	2.4×10^2	4.0×10^2	6.8×10^2	9.4×10^2	1.2×10^3	1.4×10^3	2.6×10^3
^{243}Am	1.7×10^1	1.7×10^1	1.7×10^1	1.7×10^1	1.7×10^1	1.7×10^1	1.7×10^1	1.7×10^1
^{242}Cm	1.5×10^4	6.8×10^3	1.4×10^3	7.2×10^1	1.0×10^1	7.2×10^0	7.0×10^0	6.4×10^0
^{244}Cm	2.2×10^3	2.2×10^3	2.1×10^3	1.9×10^3	1.8×10^3	1.6×10^3	1.5×10^3	7.1×10^2
Total Transuranic Activity	1.2×10^5	1.1×10^5	1.0×10^5	9.1×10^4	8.4×10^4	7.7×10^4	7.0×10^4	3.1×10^4

TABLE I-A-4

Major Activation Products in Aged PWR Fuel
(Curies per MT of Uranium Charged to Reactor)

<i>Radionuclide</i>	<i>Years After Discharge</i>							
	<u>0.5</u>	<u>1.0</u>	<u>2.0</u>	<u>4.0</u>	<u>6.0</u>	<u>8.0</u>	<u>10.0</u>	<u>30.0</u>
$^{14}\text{C}^a$	6.6×10^{-1}	6.6×10^{-1}	6.6×10^{-1}	6.6×10^{-1}	6.6×10^{-1}	6.6×10^{-1}	6.5×10^{-1}	6.5×10^{-1}
^{54}Mn	1.6×10^2	1.1×10^2	4.6×10^1	8.6×10^0	1.6×10^0	3.1×10^{-1}	-	-
^{55}Fe	1.7×10^3	1.5×10^3	1.2×10^3	6.8×10^2	4.0×10^2	2.3×10^2	1.4×10^2	6.6×10^{-1}
^{58}Co	1.6×10^3	2.6×10^2	7.6×10^0	-	-	-	-	-
^{60}Co	5.9×10^3	5.5×10^3	4.8×10^3	3.7×10^3	2.9×10^3	2.2×10^3	1.7×10^3	1.2×10^2
^{63}Ni	5.5×10^2	5.5×10^2	5.4×10^2	5.4×10^2	5.3×10^2	5.2×10^2	5.1×10^2	4.4×10^2
^{95}Zr	4.1×10^3	5.8×10^2	1.2×10^1	-	-	-	-	-
^{95}Nb	7.7×10^3	1.2×10^3	2.5×10^1	1.1×10^{-2}	-	-	-	-
^{125}Sb	3.8×10^1	3.4×10^1	2.6×10^1	1.6×10^1	9.4×10^0	5.6×10^0	3.4×10^0	-
^{125m}Te	1.6×10^1	1.4×10^1	1.1×10^1	7.9×10^0	4.0×10^0	2.8×10^0	1.4×10^0	-
Total Activation Product Activity	2.2×10^4	9.8×10^3	6.7×10^3	5.0×10^3	3.8×10^3	3.0×10^3	2.4×10^3	5.7×10^2

a . Generated in UO_2 by the reaction of neutrons with small amounts of nitrogen-14 and oxygen-17 in the oxides.

TABLE I-A-5

Effect of Decay on Thermal Power of Irradiated Fuel^a

Operating Power = 30 MW/MTU

Exposure = 33,000 MWD/MTU

Time After Discharge, Years	Thermal Power		
	Watts/MTU	Watts/Assembly	
		PWR	BWR
0.5	18,300	8,400	3,450
1.0	11,000	5,080	2,080
2.0	5,900	2,700	1,110
4	2,700	1,250	510
6	1,800	830	340
8	1,450	670	270
10	1,290	600	240
30	790	360	150

^a. BWR fuel assumed to be irradiated under PWR conditions for conservatism in assessing environmental effects of spent fuel management.

A.2.1.3 Activation Products

Elements in the metallic structural components of reactor fuels also undergo transmutation during reactor operation. Table I-A-4 lists the radionactive nuclides in these materials. Contribution to total radioactivity of the fuel assembly from this source is minor, being at least two orders of magnitude less than that of the fission products in the fuel.

In addition to the activation products described above, carbon-14 (^{14}C) is produced during the operation of light water reactors by the reaction between neutrons and small amounts of nitrogen-14, carbon-13, and oxygen-17 in the reactor core. In fuel elements, the major source of ^{14}C is from the reaction of neutrons with the approximately 25 ppm nitrogen impurity in the oxide fuel. For this level of nitrogen, the ^{14}C produced in fuel irradiated in a PWR is about 0.66 Ci/MTU (Table I-A-4).

A.2.2 Heat Generation

During radioactive decay, an unstable nuclide reaches a more-stable energy state by the ejection of alpha and/or beta particles and/or the release of energy in the form of electromagnetic radiation. The absorption by matter of such particles and energy rays produces heat. The rate of heat generation in spent fuel after reactor shutdown depends upon the quantities of radionuclides present and their decay rates. Table I-A-5 shows heat generation for spent fuel that has been exposed to 33,000 MWD/MTU of reactor operation at a specific power of 30 MW/MTU.

REFERENCES FOR SECTION I,A

1. W. H. Baker and F. D. King. *Technical Data Summary, Spent Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant*. DPSTD-AFCT-77-7, Savannah River Laboratory, E. I. du Pont de Nemours and Co., Aiken, SC (August 1977).
2. M. J. Bell. *ORIGEN - The ORNL Isotope Generation and Depletion Code*. USAEC Report ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, TN (1973).
3. *Generic Environmental Impact Statement - Storage of U. S. Spent Power Reactor Fuel*, DOE/EIS-0015D.
4. *Generic Environmental Impact Statement - Storage of Foreign Power Reactor Fuel* (in preparation).

B. HWR FUELS

The predominant HWR fuel is CANDU (CANada Deuterium Uranium) fuel. CANDU fuel rods are constructed of natural uranium as UO_2 . The UO_2 is contained in a metal sheath. Each CANDU bundle is designed to facilitate onpower refueling. New CANDU fuel is charged to the reactor while the reactor is operating at normal power level by automatic refueling machines. These automatic refueling machines move the fuel bundle in a series of progressive steps through the reactor as burnup proceeds.

B.1 Description of a CANDU Fuel Assembly

Nuclear fuel for the CANDU pressurized heavy water power reactors is natural uranium in the form of short cylinders (pellets) of sintered uranium dioxide (UO_2). Depending upon the specific reactor design, these pellets are about 1.2 to 1.4 cm (0.47 to 0.55 in.) in diameter and 1.6 to 2.0 cm (0.63 to 0.79 in.) long.

A stack of these pellets about 50 cm (19.7 in.) long is loaded and hermetically sealed in a Zircaloy sheathing tube with end caps. This tubular unit, called a fuel rod, usually contains about 0.5 to 0.7 kg (1.1 to 1.5 lb) of natural uranium.

Fuel rods are assembled into a circular bundle about 8 to 10 cm (3.1 to 3.9 in.) in diameter and 50 cm (16.8 in.) long. A 28-rod assembly with each rod spaced and supported by Zircaloy-bearing pads and structural end plates are shown in Figure I-B-1. Fuel assemblies for typical CANDU reactors contain about 14 to 20 kg (30.8 to 44.0 lb) of uranium (Table I-B-1).

In the typical CANDU reactor, 12 or 13 fuel bundles are loaded into each of several hundred horizontal pressure tubes fixed in the reactor core. Pressurized heavy water circulates within each pressure tube around the individual fuel rods to remove the heat generated by reactor operation.

B.2 Characteristics of Spent CANDU Fuel

B.2.1 Radionuclides

The radioactive nuclide inventories and thermal powers of spent fuel from a typical HWR reactor were calculated by using the ORIGEN code¹ (Tables I-B-2 through I-B-5). For the calculations used to generate these tables, irradiation exposure and

FIGURE I-B-1. CANDU Fuel Bundle for Pickering Reactor²

TABLE I-B-1

Characteristics of Typical CANDU-HWR Fuel Assemblies³

	<i>Douglas Point HWR^a</i>	<i>Pickering HWR^b</i>	<i>Bruce HWR^c</i>
Assembly length, cm	49.5	49.5	49.5
Assembly diameter, cm	8.2	10.2	10.2
Number of fuel rods in assembly	19	28	37
Fuel rod OD, cm	1.52	1.52	1.31
Fuel Pellet OD, cm	1.43	1.43	1.22
U/assembly, kg	13.5	19.8	18.3
UO ₂ /assembly, kg	15.3	22.5	20.8
Zircaloy/assembly, kg	-	~2.0	-

a. Nominal 200 MWe (megawatts, electrical) plant design.

b. Nominal 540 MWe plant design.

c. Nominal 750 MWe plant design.

TABLE I-B-2

Major Fission Product Activity in Aged CANDU Spent Fuel
(Curies per MT of Uranium Charged to Reactor)

Radionuclide	Years After Discharge							
	0.5	1.0	2.0	4.0	6.0	8.0	10.0	30.0
^3H	9.4×10^0	9.1×10^0	8.6×10^0	7.7×10^0	6.9×10^0	6.2×10^0	5.5×10^0	1.8×10^0
^{85}Kr	2.6×10^3	2.5×10^3	2.4×10^3	2.1×10^3	1.8×10^3	1.6×10^3	1.4×10^3	4.0×10^2
^{89}Sr	5.2×10^4	4.6×10^3	3.5×10^1	-	-	-	-	-
^{90}Sr	1.7×10^4	1.7×10^4	1.7×10^4	1.6×10^4	1.5×10^4	1.4×10^4	1.4×10^4	8.4×10^3
^{90}Y	1.7×10^4	1.7×10^4	1.7×10^4	1.6×10^4	1.5×10^4	1.4×10^4	1.4×10^4	8.4×10^3
^{91}Y	9.1×10^4	1.1×10^4	1.4×10^2	-	-	-	-	-
^{95}Zr	1.7×10^5	2.4×10^4	4.8×10^2	2.0×10^{-1}	-	-	-	-
$^{95\text{m}}\text{Nb}$	3.5×10^3	5.0×10^2	1.0×10^1	-	-	-	-	-
^{95}Nb	3.2×10^5	5.0×10^4	1.1×10^3	4.3×10^{-1}	-	-	-	-
^{99}Tc	3.7×10^0	3.7×10^0	3.7×10^0	3.7×10^0	3.7×10^0	3.7×10^0	3.7×10^0	3.7×10^0
^{103}Ru	4.5×10^4	1.8×10^3	3.1×10^0	-	-	-	-	-
$^{103\text{m}}\text{Rh}$	4.5×10^4	1.8×10^3	3.1×10^0	-	-	-	-	-
^{106}Ru	2.0×10^5	1.4×10^5	7.0×10^4	1.8×10^4	4.4×10^3	1.1×10^3	2.8×10^2	-
^{106}Rh	2.0×10^5	1.4×10^5	7.0×10^4	1.8×10^4	4.4×10^3	1.1×10^3	2.8×10^2	-
^{123}Sn	2.4×10^3	8.8×10^2	1.2×10^2	2.0×10^0	-	-	-	-

TABLE I-B-2, Continued

Radionuclide	Years After Discharge							
	0.5	1.0	2.0	4.0	6.0	8.0	10.0	30.0
^{125}Sb	4.1×10^1	3.5×10^1	2.7×10^1	1.7×10^1	1.0×10^1	6.0×10^0	3.5×10^0	2.1×10^{-2}
$^{129\text{m}}\text{Te}$	1.3×10^3	3.1×10^1	-	-	-	-	-	-
^{129}Te	8.2×10^2	2.0×10^1	-	-	-	-	-	-
^{129}I	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}
^{131}I	1.2×10^{-1}	-	-	-	-	-	-	-
^{134}Cs	2.1×10^4	1.8×10^4	1.3×10^4	6.4×10^3	3.2×10^3	1.6×10^3	8.4×10^2	1.0×10^0
^{137}Cs	2.8×10^4	2.7×10^4	2.7×10^4	2.5×10^4	2.4×10^4	2.3×10^4	2.2×10^4	1.4×10^4
$^{137\text{m}}\text{Ba}$	2.6×10^4	2.5×10^4	2.5×10^4	2.4×10^4	2.3×10^4	2.2×10^4	2.1×10^4	1.3×10^4
^{141}Ce	2.5×10^4	5.0×10^2	2.0×10^{-1}	-	-	-	-	-
^{144}Ce	3.5×10^5	2.2×10^5	9.2×10^4	1.6×10^4	2.6×10^3	4.4×10^2	7.4×10^1	-
^{144}Pr	3.5×10^5	2.2×10^5	9.2×10^4	1.6×10^4	2.6×10^3	4.4×10^2	7.4×10^1	-
^{147}Pm	5.9×10^4	5.2×10^4	4.0×10^4	2.3×10^4	1.4×10^4	8.1×10^3	4.8×10^3	2.4×10^1
$^{148\text{m}}\text{Pm}$	6.4×10^2	3.1×10^1	7.5×10^{-2}	-	-	-	-	-
^{151}Sm	4.6×10^2	4.6×10^2	4.6×10^2	4.5×10^2	4.4×10^2	4.4×10^2	4.3×10^2	3.7×10^2
^{154}Eu	6.9×10^2	6.8×10^2	6.5×10^2	6.0×10^2	5.5×10^2	5.0×10^2	4.6×10^2	1.9×10^2
^{155}Eu	1.0×10^3	8.5×10^2	5.8×10^2	2.7×10^2	1.3×10^2	5.8×10^1	2.7×10^1	-
Total Fission Product Activity	2.0×10^6	9.9×10^5	4.7×10^5	1.8×10^5	1.1×10^5	9.0×10^4	7.9×10^4	4.5×10^4

TABLE I-B-3

Major Transuranic Activity in Aged CANDU Spent Fuel
(Curies per MT of Uranium Charged to Reactor)

Radionuclide	<i>Years After Discharge</i>							
	<u>0.5</u>	<u>1.0</u>	<u>2.0</u>	<u>4.0</u>	<u>6.0</u>	<u>8.0</u>	<u>10.0</u>	<u>30.0</u>
^{238}Pu	1.0×10^2	1.0×10^2	1.0×10^2	1.0×10^2	1.0×10^2	1.0×10^2	9.8×10^1	8.4×10^1
^{239}Pu	1.6×10^2	1.6×10^2	1.6×10^2	1.6×10^2	1.6×10^2	1.6×10^2	1.6×10^2	1.6×10^2
^{240}Pu	2.8×10^2	2.8×10^2	2.8×10^2	2.8×10^2	2.8×10^2	2.8×10^2	2.8×10^2	2.8×10^2
^{241}Pu	2.7×10^4	2.7×10^4	2.6×10^4	2.3×10^4	2.1×10^4	1.9×10^4	1.8×10^4	6.8×10^3
^{241}Am	3.2×10^1	5.3×10^1	9.5×10^1	1.7×10^2	2.4×10^2	3.1×10^2	3.6×10^2	7.1×10^2
^{243}Am	1.2×10^0	1.2×10^0	1.2×10^0	1.2×10^0	1.2×10^0	1.2×10^0	1.2×10^0	1.2×10^0
^{242}Cm	1.1×10^3	5.2×10^2	1.1×10^2	5.3×10^0	6.7×10^{-1}	4.5×10^{-1}	4.4×10^{-1}	4.0×10^{-1}
^{244}Cm	4.8×10^1	4.7×10^1	4.6×10^1	4.2×10^1	3.9×10^1	3.6×10^1	3.4×10^1	1.6×10^1
Total Transuranic Activity	2.9×10^4	2.8×10^4	2.6×10^4	2.4×10^4	2.2×10^4	2.0×10^4	1.8×10^4	8.0×10^3

TABLE I-B-4

Major Activation Products in Aged CANDU Fuel
(Curies per MT of Uranium Charged to Reactor)

Radionuclide	<i>Years After Discharge</i>							
	<u>0.5</u>	<u>1.0</u>	<u>2.0</u>	<u>4.0</u>	<u>6.0</u>	<u>8.0</u>	<u>10.0</u>	<u>30.0</u>
$^{14}\text{C}^a$	4.6×10^{-1}	4.6×10^{-1}	4.6×10^{-1}	4.6×10^{-1}	4.6×10^{-1}	4.6×10^{-1}	4.6×10^{-1}	4.6×10^{-1}
^{54}Mn	6.4×10^1	4.2×10^1	1.8×10^1	3.4×10^0	6.4×10^{-1}	1.2×10^{-1}	-	-
^{55}Fe	1.9×10^3	1.6×10^3	1.3×10^3	7.4×10^2	4.3×10^2	2.5×10^2	1.5×10^2	7.2×10^{-1}
^{58}Co	6.4×10^2	1.1×10^2	3.2×10^0	-	-	-	-	-
^{60}Co	3.9×10^3	3.6×10^3	3.2×10^3	2.4×10^3	1.9×10^3	1.4×10^3	1.1×10^3	7.9×10^1
^{63}Ni	4.8×10^2	4.8×10^2	4.8×10^2	4.7×10^2	4.6×10^2	4.6×10^2	4.5×10^2	3.9×10^2
^{95}Zr	5.9×10^3	8.4×10^2	1.7×10^1	-	-	-	-	-
^{95}Nb	1.1×10^4	1.8×10^3	3.8×10^1	1.1×10^{-2}	-	-	-	-
^{125}Sb	4.0×10^1	3.5×10^1	2.7×10^1	1.6×10^1	9.8×10^0	5.8×10^0	3.5×10^0	-
$^{125\text{m}}\text{Tc}$	1.6×10^1	1.5×10^1	1.1×10^1	6.8×10^0	4.1×10^0	2.4×10^0	1.4×10^0	-
Total Activation Product Activity	2.5×10^4	8.6×10^3	5.0×10^3	3.7×10^3	2.8×10^3	2.2×10^3	1.7×10^3	4.7×10^2

a. Generated in UO_2 by the reaction of neutrons with small amounts of nitrogen-14 and oxygen-17 in the oxides.

TABLE I-B-5

Effect of Decay on Thermal Power of Irradiated CANDU Fuel

Operating Power = 26.3 MW/MTU

Exposure = 8,170 MWD/MTU

<i>Time after Discharge Years</i>	<i>Thermal Power Watts/MTU</i>
0.5	9,100
1.0	4,450
2.0	2,050
4	730
6	420
8	330
10	290
30	180

specific power were assumed to be 8170 megawatt-days thermal per metric ton of uranium (MWD/MTU) and 26.3 megawatts per metric ton (MW/MTU), respectively.

B.2.1.1 Fission Products

Table I-B-2 lists the major fission products present in spent CANDU fuel up to 30 years after discharge from the reactor.

B.2.1.2 Transuranics

Major transuranics in the spent fuel are listed in Table I-B-3.

B.2.1.3 Activation Products

Elements in the metallic structural components of reactor fuels also undergo transmutation during reactor operation. Table I-B-4 lists the radioactive nuclides in these materials.

In addition to the activation products described above, the ^{14}C produced in fuel irradiated in a CANDU fuel is estimated to be about 0.46 Ci/MTU (Table I-B-4). This estimate is based upon the assumption that the nitrogen impurity in the oxide fuel is about 25 ppm (the same is assumed for LWR calculations).

B.2.2 Heat Generation

The rate of heat generation in spent fuel after reactor shutdown depends upon the quantities of radionuclides present and their decay rates. Table I-B-5 shows heat generation for CANDU spent fuel that has been exposed to 8170 MWD/MTU of reactor operation at a specific power of 26.3 MW/MTU.

REFERENCES FOR SECTION I,B

1. M. J. Bell. *ORIGEN - The ORNL Isotope Generation and Depletion Code*. USAEC Report ORNL-2648, Oak Ridge National Laboratory, Oak Ridge, TN (1973).
2. L. R. Haywood, J. A. L. Robertson, J. Pawliw, J. Howieson, and L. L. Bodie. *Fuel for Canadian Power Reactors*. CONF 49/A/156 (September 1971).
3. R. D. Page. *Fuel for CANDU Pressurized Heavy Water Reactors*. Nuclear Engineering International, London, Vol. 19, No. 217, p. 496 (June 1974).

II. GENERIC FACILITY DESCRIPTION

A. STORAGE OF SPENT FUEL IN INDEPENDENT SPENT FUEL STORAGE (ISFS) WATER BASINS

Water pool storage at reactor sites have been used for approximately 30 years and represents well-established technology for storage of unpackaged fuel. This practice is followed at all currently operating commercial LWR reactors and is planned for all future reactors. Most U. S. reactor fuel irradiated to date is stored at the reactor sites in water-cooled basins designed to hold from 50 to 150 MTU of irradiated fuel.¹

In addition to reactor storage basins, independent storage basins have been constructed. The recently designed storage facility at the Barnwell Nuclear Fuel Plant has a capacity of 400 metric tons of spent fuel.² The storage pool at the General Electric Morris Plant has been expanded to approximately 700 MTU for storage of spent fuel, and the basins at Nuclear Fuel Services have a capacity of 260 MTU of spent fuel. The pool was expanded by using stainless steel storage baskets to reduce spacing and by utilizing space previously designated for storing solidified HLW. Figures II-A-1 and II-A-2 show two photographs of the Morris Plant fuel storage pools. These pools are typical of the storage density assumed in the generic ISFS basins.

Water-basin storage of packaged LWR fuel assemblies provides cooling and radiation shielding and allows visual inspection and direct manipulation of the fuel. The storage basin can also serve as a large heat and activity sink under some accident conditions.

A concept presented in *U.S. Nuclear Regulatory Guide 3.24* for a separate facility (i.e., not located at a reactor or fuel-reprocessing site) for the storage of spent fuel suggests the use of water-filled modular basins, with each basin having a capacity of about 500 MTU of LWR spent fuel.³ These basins could also be used to store CANDU fuel. If sized for comparable decay heat and radionuclide content, each CANDU basin would hold ~1700 MTU of CANDU fuel.

The generic spent-fuel storage facility considered in the DOE generic environmental impact statement supporting the Spent Fuel Policy contains stainless-steel-lined concrete structures. The facility is designed to receive, handle, decontaminate, and reship spent fuel casks; to remove irradiated fuel from casks; to place

FIGURE II-A-1. G. E. Morris Operation — Fuel Pool.⁴
Water Depth is 8.7 m (28.5 ft).

FIGURE II-A-2. G. E. Morris Operation — Fully Loaded Storage⁴ Canister Being Transported to a Pool Storage Location. Water Depth is 8.7 m (28.5 ft).

the fuel in a water basin; and to cool and control the quality of the water. The facility is also designed for removing spent fuel from storage basins, loading the spent fuel into shipping casks, decontaminating loaded casks and shipping spent fuel. Modular construction allows facility expansion with a minimum of additional support facilities and services.

Basin water is circulated in closed-loop systems through a filter-demineralizer system to maintain water purity and collect radioactive contaminants. Periodically, the filter-demineralizer system is backwashed and rejuvenated with water and chemicals that are then discharged to the waste evaporator. The water is a confinement barrier should radioactivity escape the fuel. A secondary cooling loop to transfer the heat from the basin water to the environment further minimizes the transfer of radioactive materials to the surroundings. A ventilation system maintains the concentrations of radioactive gaseous contaminants in the facility at a low level.^{3,5}

A.1 Design Criteria

The major consideration in the design and operation of a facility for handling and storing spent fuel is to protect the integrity of the fuel cladding. The cladding is the primary barrier to contain radioactivity and prevent release of radioactive materials. Therefore, the heat generated by decay of fission products must be removed, and the fuel elements must be protected from accidents that could rupture the cladding. The facility is designed and operated to preclude the following credible accidents:³

1. Criticality
2. Personnel Exposure from stored fuel
3. Dropping of heavy loads on fuel
4. Multiple massive ruptures of fuel elements by missiles
5. Complete loss of cooling water
6. Damage to critical equipment by casks

In general, the above objectives can be met, according to ANSI Standard N.305, when⁶

Pools (basins) and associated structures and systems, where fuel is handled outside the cask, are designed to maintain

their functional integrity following any credible natural phenomena (earthquake, tornado, hurricane, flood, drought, etc.) and accidents (cask drop, power loss, equipment failure, etc.), or any credibly related combinations to the extent necessary to prevent excessive release of radioactivity.

A.1.1 Activity Containment and Integrity

The fuel cladding is the initial barrier for confinement during storage. The cladding withstands a far more-severe environment in the reactor than is encountered in a water storage basin even though exposure time may be shorter. A storage facility is designed to protect the fuel cladding against mechanical, chemical, or thermal damage; the storage facility provides for a safe, subcritical arrangement of fuel assemblies and adequate shielding of operating personnel from the fuel assemblies.

At the facility, spent fuel assemblies outside the confines of the shipping cask are normally handled and stored underwater.

Once unloaded from the shipping casks, the spent fuel is stored underwater. The water provides:

- An excellent heat transfer medium for removing decay heat from the fuel and a substantial heat sink under accident conditions that result in loss of cooling water flow to the heat exchanger.
- A transparent radiation shield that allows visual inspection and direct manipulation of the fuel
- Partial containment of fission product gases and essentially full containment of any particulate radioactive material that may escape from a fuel assembly

The basins are designed to retain their watertight integrity for all credible accidents, including the design basis tornadoes and earthquakes. These basins are designed as Category I seismic structures and, as such, designed to 1) resist rupture causing excessive loss of water and 2) support and prevent all massive equipment, such as cranes, etc., from falling into the basins, thus causing damage to the spent fuel during the design basis earthquake. The water shielding the fuel will mitigate the effect of tornadic or other wind-driven missiles.⁷ Because of this protection, the basin roof is of light industrial construction that could blow away in tornadic winds.

A.1.2 Radiation Protection

ISFS basin areas are designed to be routinely accessible; thus, the radiation level during normal operations must be ≤ 0.5 mrem/hr. If the water in the basin is at an activity level of 2×10^{-4} Ci/m³, the radiation level at the surface (air-water interface) would be 0.4 mrem/hr, if equal activity levels of ⁶⁰Co, ¹³⁴Cs, and ¹³⁷Cs are assumed. The radiation contribution from the stored fuel is then limited to 0.1 mrem/hr which requires a water depth above the fuel of about 3.6 m (12 ft) (storing LWR fuel irradiated to 33,000 MWD/MTU at a rate of 30 MW/MTU).

A.2 Operations at ISFS Water Basins

A.2.1 Handling and Storage

The facility to receive and store spent fuel is arranged into the following process areas: cask-carrier handling, cask processing, and underwater handling and storage (Figure II-A-3).

A.2.1.1 Cask-Carrier Handling

Upon receipt of a cask, a thorough safeguards inspection is made of each cask and carrier; the shipment is identified; and the cask is inspected for shipping damage and monitored for radiation level. If inspection shows that normal handling procedures can be followed, the cask carrier will be unloaded. Otherwise, the cask will be sent to the holding area for later unloading by special procedure. Damaged casks, carriers, and peripheral equipment will be repaired when possible in the maintenance area.

After the casks are emptied of fuel, decontaminated, and reloaded onto the carrier, a final inspection is made of the cask carrier to ensure that the shipment meets DOT regulations. Then, the cask carrier is shipped offsite.

A.2.1.2 Cask Processing

After the cask has been removed from the carrier, it is prepared for insertion into the fuel unloading pool. This preparation includes washing the cask to remove road dirt, venting the cask to the off-gas system and cooling and flushing the cask as required. The waste from cask rinsing, flushing, and cooling is collected for treatment. Empty casks are also decontaminated for shipment offsite.

II-7

FIGURE II-A-3. Spent Fuel Handling and Storage

A.2.1.3 Underwater Handling and Storage

Spent fuel is unloaded from casks and stored and reloaded into casks underwater. These process steps are discussed in more detail in the subsequent paragraphs.

A.2.1.3.1 Fuel Unloading

Cleaned and cooled casks are inserted into the unloading pool in a vertical position. When the cask contains LWR fuel, the cask lid is removed; and individual LWR fuel assemblies are removed and transferred to multiple assembly storage baskets in the pool. When the cask contains CANDU fuel, the cask lid is removed; and the baskets containing CANDU fuel are removed and transferred to storage. All spent fuel in storage baskets is handled by remote control under a minimum of 3.6 m (12 ft) of water to shield the operating personnel from the intense radiation emitted from the irradiated spent fuel. If a significant fuel assembly leak is detected, the fuel assembly will be placed in a special container to control release of radioactivity during handling and storage.

Before the empty cask is removed from the pool, it is inspected to ensure that all fuel and nonfuel items have been removed. The cask lid is then replaced on the cask. As the cask is removed from the pool, sprays of high velocity demineralized water remove pool water and contamination from the exterior of the cask. When the cask reaches the parapet level, the head bolts are replaced in the lid; and the cask is transferred to the decontamination area.

A.2.1.3.2 Spent Fuel Storage Basins

The LWR storage basins receive multiple assembly storage baskets containing spent LWR fuel assemblies from the fuel unloading pool. The baskets are stored underwater in racks fastened to the basin floor. The racks are designed to maintain the spacing of the baskets even during extreme natural phenomena.

Basins that store CANDU fuel contain racks to support stacked arrays of CANDU baskets. Baskets of CANDU fuel are stored in the racks four deep. These racks are designed to limit motion of the baskets and possible fuel damage during extreme natural phenomena.

A.2.1.3.3 Spent Fuel Loading and Shipping

A spent fuel shipment from the ISFS basin will follow a reverse sequence to that described in the previous paragraph. The loaded casks will be inspected and monitored to ensure that they meet all DOT regulations before shipment. Any detected deficiencies will be corrected before shipment.

A.2.2 Criticality Hazard and Control

Although no criticality accidents have occurred in spent fuel basins,⁸ the potential for accidental nuclear criticality is considered in handling and storing irradiated fuel assemblies.

Criticality safety throughout the facility is provided either by physically limiting the proximity of one fuel assembly to another or by using neutron absorbing structural material.

There is no requirement to maintain the spacing of CANDU fuel, which is subcritical in light water. The following administrative controls and technical specifications⁶ are applied, relative to LWR fuel:

1. Not more than one assembly will be outside a shipping cask or basket at any given time in a given pool or basin (with the exception of leaking assemblies in critically safe single assembly containers).
2. No fuel-bearing basket will be in the fuel unloading pool while a cask is being handled in or over the unloading pool.
3. Only one loaded basket may be left freestanding at any time in the fuel storage basin.
4. Loaded baskets are stored in basket racks (or other device) except during transfer.
5. The basket cap shall be secured in position during all movement of loaded baskets.
6. The maximum k_{eff} of an infinite array of loaded baskets shall be no greater than 0.95 (based on the unirradiated fuel composition).
7. No loaded basket will be moved over another loaded basket.

Criticality monitors and alarms are located throughout the facility. At least two operating criticality monitors are located in each fuel handling area.

A.3 Facility Description

A facility (Figure II-A-4) for water pool storage of spent LWR fuel receives fuel from PWR or BWR in heavily shielded casks transported either by rail or truck.

Support systems for the storage facility are provided for

- Dissipating heat

- Controlling the quality of water in the pools
- Ventilating the building
- Treating radioactive wastes generated
- Electrical power

FIGURE II-A-4. Plot Plan for Water Storage Facility for Unpackaged Spent LWR Fuel

It is assumed that the facility will be licensed by the Nuclear Regulatory Commission and constructed according to the following design basis.

Throughput and Capacity

- Facilities will be provided to handle 30% of the domestic fuel load in legal weight truck casks and the remaining fuel in rail casks. Facilities will be designed to handle 20% of the foreign LWR fuel in overweight truck casks and the remaining in rail casks. These facilities will also handle all of the CANDU fuel in rail casks.
- LWR spent fuel assemblies are stored in modular basins, each with a capacity of 500 MTU. CANDU fuel is stored in modular basins with ~1700 MTU capacity. The facility is designed to facilitate addition of modular basins as needed.
- Facility throughput and capacity are based upon the requirements of the specific case analyzed in the GEISs.

Fuel

- Fuel not in the confines of a cask will be handled remotely underwater.
- LWR fuel will be stored in poisoned (stainless steel) multiple assembly baskets to permit higher density storage and yet maintain storage that is critically safe. CANDU fuel will be stored in stainless steel baskets that contained the fuel during shipment.

Water Pools and Basins

- Water pools and basins will be concrete and lined with stainless steel.
- Each water pool or basin will be designed so that it can be isolated from the remaining pools or basins to permit drainage of an individual basin.
- Each pool or basin will be directly accessible from any other pool or basin.
- Radioactivity in the water will be maintained at $\leq 2 \times 10^{-4}$ Ci/m³.

- Conductivity of the water will be maintained at 1 to 2 $\mu\text{mho/cm}$.
- The temperature of the pool water will be maintained at $\leq 40^\circ\text{C}$ ($\leq 104^\circ\text{F}$).
- Water pools and basins will be sufficiently deep to handle fuel assemblies up to 4.6 m (15 ft) in length.
- All piping leading to or from the basins will be designed to prevent draining or siphoning of a pool below a safe level. Administrative controls will prevent introduction of potential siphoning drains.

Confinement and Waste Management

- A radioactive waste handling system will be provided to process solid, liquid, and gaseous radioactive wastes from fuel storage.
- The facility will have air locks and a controlled leakage of air into the facility.
- A leakage collection system will be provided for water pools or basins.
- Redundancy in water supply, equipment, and electrical supply will be provided to prevent credible loss of cooling and shielding.

Personnel Protection

- Shielding will be provided to ensure that radiation levels resulting from exposure of plant personnel to radiation will be $< 0.5 \text{ mrem/hr}$ in continuously occupied areas and $< 5 \text{ mrem/hr}$ in intermittently occupied areas.
- Plant personnel safety will be ensured by compliance with applicable NRC, OSHA (29 CFR 17), and industry standards.

The preceding criteria were used to develop a description of the generic facility. The next several sections describe the types and sizes of the facilities assumed in the generic environmental impact statements on the Spent Fuel Policy. These facilities were developed, based upon the flow diagram shown in Figure II-A-3. Figure II-A-5 shows a typical ISFS building designed to receive and ship up to 2500 MTU/year of LWR fuel and to store

FIGURE II-A-5. Cask and Fuel Handling Building — Fuel Storage Pool Layout

3000 MTU. The layout shown in Figure II-A-5 is typical of those assumed in the generic environmental impact statements on the Spent Fuel Policy. The facility can be expanded by adding storage basins, usually in pairs.

A.3.1 Cask-Carrier Handling

The receiving, shipping, and holding areas shown in Figure II-A-3 are unenclosed areas adjacent to the ISFS basin facility. Space is required for at least two days of cask-carrier throughput in the receiving and shipping areas and four days in the holding area. The generic facility is assumed to operate 300 days of each year.

The maintenance area provides facilities and equipment for maintenance and repair of the casks, carriers, and peripheral equipment. An enclosed area large enough to contain an LWT cask carrier, a rail cask carrier, and the offloaded peripheral equipment is provided. Maintenance and repair work is normally performed after fuel has been removed from the cask. Work is limited to routine maintenance and minor repairs except where more-extensive work is required to allow cask processing.

The preparation area is an enclosed work space of standard industrial construction that provides facilities and equipment to prepare the cask-carrier for unloading the cask or for shipping offsite. Space ~24 m (~80 ft) long and 9 m (30 ft) wide is provided for each rail cask carrier in the preparation area. The preparation area also serves as an air lock to control air leakage into the cask processing and fuel storage building.

In the preparation area, peripheral equipment such as crash shields, personnel shields, etc., are removed, checked for contamination, and stored until the cask is emptied. When the cask is emptied, the equipment is replaced; and the cask-carrier is readied for offsite shipping. One or more 9 tonnes (10-ton) cranes is provided for handling peripheral equipment.

A.3.2 Cask Processing

The structure housing the cask processing area is constructed of heavily reinforced concrete to a height of 7.6 m (25 ft) above pool water level. The additional structure [to a height of about 18 m (60 ft)] and roof are constructed of insulated metal and designed to withstand the 100-year recurrent wind but not the design basis tornado. Two large overhead cranes, each with a load

capacity of approximately 113 tonnes (125 tons), are provided for cask transfers. Other smaller cranes are provided to handle auxiliary equipment.

The cask-processing times used in this report are based upon experience at the General Electric Morris Facility. The expected and upper limit values from a log normal distribution of the process times are shown in Table II-A-1.

TABLE II-A-1

Cask Process Times

Area	<i>Time, hr</i> <i>LWT Cask^a</i>		<i>GE IF-300 Cask</i>	
	<i>Expected</i>	<i>Upper Limit^b</i>	<i>Expected</i>	<i>Upper Limit^b</i>
Preparation-Offload	4.0	8.0	4.9	9.1
Cooldown	4.0	8.0	4.9	9.1
Fuel Unloading	2.8	7.2	7.0	11.4
Decontamination	3.3	11.0	14.0	22.4
Reload-Delay (Preparation-Offload)	4.0	10.5	6.1	13.3
TURNAROUND TIME	18	45	37	65

a. Legal weight truck (LWT) casks such as NAC-1, NFS-4, NLI 1/2 (see Section III, Table III-1).

b. Upper limit includes 95% of all values.

The LWT casks shown in Table II-A-1 or similar casks are the casks expected to be used for transporting domestic LWR fuel under the Spent Fuel Policy. In addition to the GE IF-300 cask, the NLI 10/24 and possibly the TN-12 rail casks may be used for both domestic and foreign LWR fuel. These later casks transport more fuel per shipment than the GE IF-300 cask. Cask process times are unavailable for the NLI and TN rail casks, and this analysis assumes these casks can be unloaded and reloaded in the same time per unit weight of fuel. Shipment of foreign LWR spent fuel assumes the use of overweight trucks (OWT), casks which handle about three times the amount of fuel handled in LWT casks. Since handling data on these casks are also not available, this analysis assumes the handling time for OWT casks is similar to LWT casks per unit weight of fuel handled. CANDU fuel will be received in

rail casks similar to the NLI 10/24; and, due to the lower radiation levels and high density of the CANDU fuel, these casks will transport a little more than twice the amount of LWR fuel per load. This analysis also conservatively assumes the CANDU fuel handling time per unit weight of fuel is the same as that listed in Table II-A-1. With the assumptions listed above, the number of handling locations can be determined from Table II-A-2.

TABLE II-A-2

Minimum Number of Cask Handling Locations at a Fuel Storage Facility

Spent Fuel, MTU/year ^a	500	1000	1500	2000	2500	3000	6000
Casks/day, Rail	0.4	0.8	1.1	1.5	1.8	2.2	4.4
Casks/day, LWT	1.4	2.7	4.0	5.4	6.7	8.0	16.0
Handling Location	<u>Number</u>						
Preparation Area and Cask Offload-Load Area	1	2	2	3	4	4	8
Cask Cool and Washdown Area	1	1	1	2	2	2	4
Fuel Unloading Pools	1	1	1	2	2	2	4
Cask Decontamination Area	1	1	2	2	2	3	5

a. Assumes 300 days operation of the facility at full capacity.

The number of handling locations is calculated assuming that cask location can process either LWT casks, OWT casks, or rail casks. If any location were limited to a single cask type, additional handling locations could be required.

The limiting size for a single handling facility is judged to be about 3000 MTU/year for the assumed rail/truck split. Larger facilities would be hampered by limitations on movement of cranes handling both casks and fuel baskets and by difficulty in achieving a layout for material flow in a larger facility, etc.

A.3.2.1 Cask Offload-Load Area

The cask offload-load area provides a space 30 m (100 ft) long by 9 m (30 ft) wide for each cask carrier in the area.

After the cask is removed from the carrier, the carrier is washed, if necessary. Wash water is collected in a hold tank for subsequent radiation level checks to determine whether additional treatment of the water is required.

A.3.2.2 Cask Cooling and Washdown

The cask cooling and washdown area facilities and equipment which will occupy a below-grade area of 5.5 x 5.5 m (18 x 18 ft) are provided for each cask. This area will allow the necessary access to the cask. The floor is designed to direct wash water to a collection sump. A special energy absorbing pad is provided to protect cask and floor in the event a cask is dropped. Design prevents tipping of a cask while it is being unloaded or loaded. All piping in the area is mechanically shielded to protect it from damage caused by inadvertent contact with a cask.

Casks are washed to remove the road dirt before transfer to the fuel unloading pool. Wash water is collected for treatment. After washdown, the cask is vented to the facility off-gas system, and the primary coolant is checked for temperature and radioactivity. Depending upon the need, the cavity of the cask is cooled and flushed to reduce temperature and radioactivity levels to levels acceptable for transfer to the fuel unloading pool. Cooling and flush water is either collected for treatment or circulated through a filter deionizer system.

A.3.2.3 Cask Decontamination

The decontamination area provides facilities and equipment to reduce surface contamination of casks to acceptable levels for reshipment. The area is below grade. A space of 5.5 x 5.5 m (18 x 18 ft) is provided for each cask. This space allows necessary access to the cask for whatever decontamination procedure is selected. The floor is designed to direct wash solutions to a collection sump. A special energy absorbing pad is provided. Portable work platforms provide access to the top and the sides of the cask.

A.3.3 Underwater Handling and Storage

The structure housing the pools and basins for underwater handling and storage of the fuel is heavily reinforced concrete above grade to a height of 25 feet above the pool water level. Additional superstructure and roof are constructed of insulated metal designed as described in A.3.2 of this section. All water-

filled pools and basins are structures below grade which are constructed of heavily reinforced concrete and are lined with stainless steel.

Pool or basin water is maintained at a temperature of $\leq 40^{\circ}\text{C}$ to reduce loss to evaporation and to retard algae growth. The radioactivity in this water is maintained at $\leq 2 \times 10^{-4} \text{ Ci/m}^3$.

The pools and basins are provided with a high-flow emergency water supply as a backup to maintain water level, if the normal makeup system is insufficient. Pipe leading into and out of the pools and basins is installed in such a manner as to prevent draining or siphoning below safe levels. Strict administrative controls will prevent the introduction of potential siphon drains. In the event of a gross leak, any pool or basin can be isolated.

A.3.3.1 Cask Unloading Pool

The cask unloading pool provides facilities and equipment for underwater transfer of spent fuel from the shipping cask to storage baskets holding either 9 BWR assemblies or 4 PWR assemblies or 1.9 MTU CANDU fuel. Figures II-A-6 and II-A-7 show the storage baskets for PWR and BWR fuel. The baskets are designed to maintain critically safe spacing of the fuel assemblies.

The LWR storage baskets are made of stainless steel to reduce the spacing otherwise required for criticality safety. The baskets for PWR assemblies (assumed to be Combustion Engineering⁹ design) are designed for safe storage of unirradiated fuel with an enrichment of 3.4 wt % ^{235}U . The fuel is placed in 22.8-cm² (9-in.²) tubes with 0.63-cm (1/4-in.) wall thickness. The center-to-center spacing of the 2 × 2 array is 3.6 cm (12 in.). The basket design for BWR assemblies is based upon designs for the Barnwell¹⁰ and Morris¹¹ facilities. The basket consists of an array of 16.5-cm² (6.5-in.²) tubes with 0.63 cm (1/4-in.) wall thickness and center-to-center spacing of 20.3 cm (8 in.). If further study shows that the spacings are not conservative for all fuel types, stainless steel with nuclear poison can be used in the baskets; and the same overall size can still be maintained.

The basket for CANDU fuel storage is the shipping basket. This basket has not yet been designed; but in concept, it is a 101 cm (40 in.) diameter × 101 cm (40 in.) high basket that will contain 96 CANDU fuel elements in two stacks. The basket will be constructed of stainless steel. There is no requirement to maintain the spacing of CANDU fuel, which is subcritical in light water.

Dimension, inches

A	24
B	12
C	9
Length	~180

FIGURE II-A-6. Stainless Steel Storage Basket — PWR Assembly¹²

Dimensions, inches

A	24
B	8.0
C	6.5
Length	~180

FIGURE II-A-7. Stainless Steel Storage Basket — BWR Assembly¹²

Each unloading pool provides space for one cask in the vertical position, fuel storage baskets with protective support racks, and a shelf to store the cask lid. A crane with a load capacity of 9 tonnes (10 tons) is available for handling the cask lid and for removing fuel assemblies from the cask. A separate crane is used to transfer loaded baskets to the storage pools. The pools are sufficiently deep to maintain an adequate biological shield over the fuel assemblies as they are drawn from the cask. Each unloading pool covers an area of about 46 m² (500 ft²) and is about 15.2 m (50 ft) deep. A special energy absorbing pad is provided to protect cask and pool integrity in the event a cask is dropped.

Removable nonfuel components which may be received with some fuel assemblies (such as control rod clusters and burnable poison rod clusters) are removed from the assemblies and loaded into the special containers for disposal at a permanent storage facility.

A.3.3.2 Storage Basins

Each LWR storage basin is sized to hold fuel assemblies containing about 500 MTU. A storage basket used in the basin will hold either 1.69 MTU in BWR assemblies or 1.84 MTU in PWR assemblies. With the expected distribution of 2 PWR/1 BWR, 168 PWR baskets and 112 BWR baskets are used in each basin. Each storage basket is 61 cm² (24 in.²). If 2 inches are assumed on each side for support racks (a system similar to the G. E. Morris Storage Facility is used), the total basin area for 500 MTU is about 124 m² (1340 ft²). Depth of each storage basin is about 9.1 m (30 ft). Total volume of the basin is 1,140 m³ (40,200 ft³). Volume of water in a fully loaded basin is about 1,000 m³ (35,600 ft³).

Each CANDU storage basin is sized to hold ~1700 MTU of CANDU fuel. These baskets contain 1.91 MTU of CANDU fuel, and each are stacked four deep in retaining racks that are attached to the basin floor. Water depth in these basins is assumed to be about 9.1 m (30 ft). Total volume of the basin is 2,300 m³ (80,800 ft³). Volume of water in a fully loaded basin is about 2,040 m³ (72,000 ft³).

The volume of water in the storage basins is normally maintained at a temperature <40°C (104°F) and it represents a substantial heat sink. The external cooling system for the water of a storage basin that is filled with spent fuel to its design capacity may be out of service temporarily without creating an unacceptable hazard. At the maximum heat load of 1.35 MW (500 MTU, 4 years old) for a single isolated basin with LWR fuel [water volume = 1,010 m³ (35,600 ft³)], about 50 hours would be required before the basin water starts to boil at 100°C (212°F). An

additional 210 hours would be required to expose fuel through evaporation [initial water coverage of 3.6 m (12 ft)]. Thus, sufficient time is provided to make emergency repairs or to hook up emergency cooling equipment. A much longer time would be required to boil CANDU fuel storage basin. The maximum heat load in a CANDU basin would be essentially the same as that in an LWR spent fuel basin if the CANDU basin were filled with ~1700 MTU of CANDU fuel cooled 4 years, but the water volume would be much larger [2,040 m³ (72,000 ft³)] compared with 1,010 m³ (35,600 ft³) in the LWR basin.

A leakage collection system collects and returns to the basins any water which may leak past the liner. Leakage collection is accomplished by means of a system of interconnecting channels or depressions in the concrete behind the liner. These channels collect leakage and guide it to sumps. Liquid level detection instruments at each sump alarm at preset liquid levels to warn of leakage into the channels behind the liners. Float-actuated sump pumps return this leakage to the deionized water storage tank through the basin water filter deionizer system. If the sump pumps fail to operate and the water level rises, a high-level alarm is actuated.

A.3.3.3 Transfer Aisles

Transfer aisles are provided for moving storage baskets from fuel unloading pools to storage basins. These aisles are 1.8 m (6 ft) wide and about 9.1 m (30 ft) deep — sufficiently deep to provide at least 3.6 m (12 ft) of shielding water above the fuel during movement through the aisles.

A.3.3.4 Crane Maintenance Areas

Two crane maintenance areas are provided to maintain the cask crane and basket cranes. Each of these areas consists of a 6.1-m (20-ft) wide section (see Figure II-A-5) that provides an adequate working area to perform the necessary repairs to the cranes.

A.4 Support Systems

The support systems described in this section include:

- Water cleanup and cooling
- Heating, ventilation, and air conditioning systems
- Radioactive waste treatment

In addition to these process-related facilities, the ISFS basin facility includes support facilities and activities not directly associated with spent fuel handling. These facilities include a chemical laboratory, personnel check stations, a counting room, a change room, a maintenance shop, storage rooms, a battery-charge area, and personnel office space.

The support systems are housed in a building which occupies an area about the same as that for fuel handling and storage and is 6.1 to 7.6 m (20 to 25 ft) high. The structure that houses systems for treatment of pool water and radioactive wastes is assumed to be an integral part of the storage facility and is constructed to withstand design basis earthquakes and tornadoes (Figure II-A-4 as items 9 and 10). Those parts of the waste treatment area of the building where potential for contamination is high are lined with stainless steel. All leakage from this area is diverted to a collection sump that is returned to waste evaporation.

The heating, ventilation, and air conditioning (HVAC) systems and personnel offices (shown in Figure II-A-4 enclosed by dotted lines) are housed in structures of standard industrial construction.

A.4.1 Cooling Water Systems

Heat rejection requirements of the facility are met with circulating cooling water systems that dissipate the heat to the environment. The cooling system (Figure II-A-8) is composed of a closed loop primary cooling water system, the secondary cooling water system, and the emergency cooling water system.

A.4.1.1 Closed Loop Primary Cooling Water System

The primary cooling water system consists of a closed loop arrangement with a heat exchanger where heat from process equipment is transferred to the secondary cooling water system. This arrangement provides a positive barrier between potential leaks in the process equipment and the environment. This barrier reduces the potential for release of radioactivity to the environment through the heat dissipation system.

The closed loop system consists of two primary cooling water systems. These systems serve

- The fuel storage basins
- The process building heating, ventilating, and air conditioning systems

FIGURE II-A-8. Heat Dissipation System

A.4.1.2 Secondary Cooling Water System

The secondary cooling water system transports process and radioactive decay heat from the closed loop primary cooling water systems to cooling towers. Chilled water is pumped from the cooling towers through the heat exchangers. The chilled water receives heat from the primary system and returns to the cooling towers where the heat is removed by evaporating part of the secondary coolant to the atmosphere. The cooling tower (mechanical induced draft variety) is designed to meet the fuel handling and storage cooling water requirements during normal operation. Since water is evaporated, the total dissolved solids content of the remaining secondary cooling water is limited by continuous blowdown. Calcium and magnesium are kept in solution by controlling the pH of the cooling water. Makeup water is introduced into the tower to compensate for water lost by evaporation, blowdown, and drift.

A.4.1.3 Emergency Cooling Water System

The emergency cooling water system consists of a large pond and a special distribution system. This system supplies cooling

water for emergency utilities and system operation. The pond and distribution system are designed to function in the event of loss of normal cooling as the result of extreme natural phenomena.

A.4.2 Basin Water Cleanup

A basin water cleanup system consisting of filtration and deionization facilities is provided to

- Maintain the normal water activity at $<2 \times 10^{-4} \text{ Ci/m}^3$
- Rapidly reduce the activity level in any basin or pool to normal in the event of an unusual release of radioactive material to the basin or pool from leaking fuel

The system consists of a prefilter, deionizer, and an afterfilter.

The prefilter, constructed of a porous stone filter media, is precoated with diatomaceous earth. The prefilter is backflushed periodically, and the precoat and filter cake are sent to the evaporator for waste volume reduction.

The deionizer is an anion-cation exchange column that removes ionic contamination that cannot be filtered out. When the deionizer is fully loaded, it is regenerated; and the regenerating waste solution is transferred to the evaporator for volume reduction.

An afterfilter (a leaf-type cartridge) removes any resin particles that break through the deionizer. The afterfilter is cleaned online by rotating the cartridge. The waste is flushed to the evaporator for volume reduction.

Normal basin water cleanup system provides for a flow of 300 gpm for each 500 MTU stored (or ~ 1700 MTU of CANDU fuel). Additional flow capacity is provided for rapid cleanup of basin water in the event of an unusual fuel failure. (This additional capacity is equivalent to 500 gpm for each 3000 to 4000 MTU of LWR fuel.)

A.4.3 Heating, Ventilation, and Air Conditioning Systems

The building structures and ventilation systems provide for confinement of radioactive materials and ensure that personnel exposure is maintained as low as reasonably achievable. In addition, the heating, ventilation, and air conditioning (HVAC) system, as a whole, maintains an environment conducive to work. The HVAC systems for the fuel storage facility are designed to

supply properly conditioned air to operational areas, to ensure that the air is restricted to prescribed flow paths, and then to discharge the air through a stack to the environment.

Air is supplied to the ventilation systems by conventional supply units with integral filters, heating coils, cooling coils, and fans. Air from these units is distributed to the operational areas of the facility through ductwork. Air exhausted from operating areas is discharged to the outside through a stack about 45 cm (150 ft) high.

At the rates shown in Figure II-A-9, the ventilation portion of the system helps control contamination and functions continuously under normal and accident conditions. The ventilation system is designed for once-through airflow. Controlled inleakage of air is provided by air locks at entrances to the facility. Air will flow from areas with lesser potential for contamination to areas with higher potential for contamination.

A.4.4 Radioactive Waste Generation and Treatment

The operation of spent nuclear fuel storage basins generates liquid, solid, and gaseous radioactive wastes, which must be collected, treated, packaged, stored, and disposed of. The sources of radioactive contamination are primarily fission and activation products from the fuel surface and defects in the fuel cladding. Figure II-A-10 outlines the operational flow of a storage facility as it relates to generation of radioactive wastes. Systems provided for treatment of these wastes are discussed in this section.

The major volumes of liquid and semiliquid wastes requiring treatment are the filter sludges, ion-exchange regeneration solutions from the water treatment system (A.4.2 of this section), and water-detergent solutions used in cask and equipment decontamination. These liquids or semiliquids are sent to the evaporator where they are concentrated. The evaporator concentrate is a slurry which is sent to the waste solidification system for solidification with an agent such as cement or bitumen. The evaporator overheads are discharged directly to the atmosphere through the facility stack.

The solid radioactive wastes include ventilation filters, rags, clothing, plastic, paper, wood, rubber, failed small equipment, and similar items. Where possible, the volume of the material is reduced by either incineration or compaction. This compacted waste is then incorporated into a fixing agent such as cement or bitumen as discussed previously.

FIGURE II-A-9. Ventilating Air Flow Requirement

FIGURE II-A-10. Radioactive Waste from Spent Fuel Handling

A.4.4.1 Evaporation

Evaporation is used to concentrate a solution or slurry before solidification. For evaporation, a vapor-liquid de-entrainer is used to provide a high decontamination factor between the concentrate and overheads. The decontamination factor for evaporation is commonly defined as the ratio of the radionuclide concentration in the evaporator bottoms to the corresponding radionuclide concentration in the overhead vapor. These decontamination factors normally fall within the range of 10^4 to 10^5 for a single stage of evaporation and de-entrainment.

The most significant factors influencing the attainable decontamination factors are the evaporator type and design, the design and performance of associated liquid droplet de-entrainment devices, and the volatilities of feed solution radionuclides. The decontamination factor for radioactive iodine and ruthenium are low because they are often present as volatile chemical compounds. No tritium decontamination is achieved during evaporation; tritium contained in the liquid waste behaves like hydrogen in water. For most radionuclides, subsequent stages of condensation and evaporation can be used to obtain the desired decontamination factor.

A.4.4.2 Immobilization of Wastes

Although no regulations require the radioactive wastes from the ISFS to be immobilized before shipment from the ISFS, the GEISS on the Spent Fuel Policy assume this waste will be immobilized. The method of immobilizing the waste has not been selected, but it will be compatible with land burial as low-level waste. Two candidate processes are discussed in this section. Space will be provided in the ISFS basin conceptual design for one of these means of immobilization.

A.4.4.2.1 Cement Immobilization

Immobilization of radioactive wet wastes in cement (Figure II-A-11) involves mixing the wastes with dry cement ingredients, placing the mixture into drums, and allowing it to harden to a solid waste. Cement immobilization of radioactive wastes has been widely used in the United States. A variety of cementation technologies have been developed, including in-drum mixers, drum tumblers, and in-line mixers, each of which is described in ERDA-76-43.¹¹ For use in this present study, a drum-tumbling system has been selected based upon the following reasons:

- Both liquid and dry wastes may be immobilized without altering the commercially available technology.
- The wastes are mixed inside the drums, thus preventing external solidification of the waste-cement mixture.

FIGURE II-A-11. Process Flow Diagram for Cement Immobilization

The major process components are the following:

- Waste feed system
- Drumming station
- Bridge crane
- Cement feed system
- Inspection and decontamination station
- Control console

A.4.4.2.2 Bitumen Immobilization

Immobilization of radioactive wet wastes in bitumen (Figure II-A-12) involves the mixing of the waste (both solid and liquid form) with liquid bitumen or asphalt binder in an extrusion evaporator. The waste-bitumen is heated to a temperature of above 100°C (212°F) to evaporate the free water, thus reducing the waste volume. The heated waste-bitumen is then extruded into a 210-L (55-gal) drum where it cools and solidifies.

FIGURE II-A-12. Process Flow Diagram for Bitumen Immobilization

The use of bitumen to immobilize radioactive wastes has been well demonstrated, largely through extensive operating experience in Europe. Several types of bitumen immobilization processes have been developed as discussed in ERDA-76-43.¹¹ A continuous screw extruder process has several advantages, including:

- The screw extruder bitumenization process operates at lower temperatures and with shorter residence times than the batch.
- The process uses well-demonstrated technology.
- The process is commercially available in the United States.

The main components of the bitumen immobilization facility are the following:

- Waste feed system
- Bitumen feed system
- Screw extruder/evaporator
- Filling and capping stations
- Inspection and decontamination stations
- Container transfer cart
- Bridge cranes
- Control modules

A.4.4.3 General Trash and Combustible Waste Treatment

Solid radioactive wastes may be incinerated or compacted before packaging or may be packaged without treatment. Equipment includes filter compaction presses, an x-ray scanning device, a heavy duty shredder, assay equipment, a dual-chamber controlled-air incinerator, an off-gas scrubbing system, and a scrubbing-solution-blowdown concentrator. A system for incineration of combustible waste is described as follows.

The incineration process is described in the simplified flow diagram in Figure II-A-13. During feed preparation for the incinerator, boxed feed is examined with an x-ray scanner to detect high-density materials. Feed containing such materials is sorted. High-density combustibles are shredded, as are wooden filter frames after filter media has been removed in a filter

FIGURE II-A-13. Incineration Process

media removal and pelletizing press. Pelletized filter media and noncombustibles are packaged in 210-L (55-gal) drums and disposed of as described in A.4.4.4 of this section. Sorted and shredded combustibles are repackaged; and this waste, along with waste not requiring sorting, is assayed and sent to temporary storage in combustible waste boxes. From the storage area, waste boxes are placed in the ram feeder, which feeds the controlled-air incinerator. The incinerator residue is packaged in 210-L (55-gal) drums and disposed of as described in A.4.4.4 of this section.

The incinerator is a dual-chamber incinerator that has limited air supply to the lower chamber to maintain quiescent burning and an excess of air plus additional natural gas in the upper chamber to thoroughly burn the evolved gases. The off-gas from the secondary chamber is sent through a high energy gas-scrubbing system for cooling and for removal of acidic gases and moisture before being sent through HEPA filters and to the off-gas system or the facility stack. Scrubbing solution blowdown is concentrated about sixfold and sent, along with the incinerator ash, to the wet waste and particulate solids immobilization facility. Water vapor from the concentrator is sent to the facility off-gas system.

The scrubbing system consists of a quench column for gas cooling, a venturi scrubber for particulate removal, a packed column for acidic gas removal, a condenser for moisture removal, a demister, a reheater, and roughing and HEPA filters. Scrub solution is recycled and is passed through heat exchangers for cooling and filtering to remove solids in the process.

A.4.4.4 Failed Equipment and Noncombustible Waste Treatment

Failed equipment and other noncombustible waste are prepared for shipment to land burial or geologic repository. This usually involves some decontaminating, disassembly and packaging of the material. Figure II-A-14 is a schematic flow diagram showing treatment procedures. Metal is the primary constituent of failed equipment and noncombustible waste, but substances such as glass and concrete may also be present. Noncombustible scrap includes small hand tools, used light bulbs, wire ends, electrical panels, metal scraps, and glass wastes from contaminated areas, heating, ventilating, and air conditioning components and accessories within a possibly contaminated system which cannot be economically decontaminated.

A.4.4.5 Off-Gas System

The off-gas from the cask venting and cool-down system, leaking fuel in the spent fuel storage basin, and the radwaste treatment systems is treated in a system incorporated as an integral part of the storage facility. A flow diagram for this off-gas system is shown in Figure II-A-15. Major pieces of process equipment are as follows:

<i>Name</i>	<i>Function</i>
Scrubber de-entrainer	Remove particles from process off-gas
De-entrainer	Remove water droplets and waste from gas stream
Heater	Increase gas stream temperature to 45°C (113°F)
HEPA and prefilter	Remove particles from gas stream
Heater	Increase temperature of gas stream to 150°C (302°F)
Iodine adsorber	Remove iodine from gas stream
Iodine backup adsorber	Retain iodine breaking through on iodine adsorber

Radioactive gaseous emissions from off-gas stream are collected and routed to the off-gas scrubber. Most of the particles from the stream are removed in the scrubber where water flows into the gas stream. Two mist eliminators protect the high efficiency filter (HEPA) and iodine adsorbers from entrained water droplets.

FIGURE II-A-14. Flow Diagram for Treatment of Failed Equipment and Noncombustible Waste

FIGURE II-A-15. Flow Diagram for Off-Gas System

An off-gas stream which exits the top of the scrubber and de-entrainers is heated to 45°C (113°F) by an electric heater, passed through a prefilter and fiberglass (HEPA) filter to remove any remaining particles, heated further to 150°C (302°F), and finally sent through the iodine recovery system containing either a silver zeolite adsorber or impregnated charcoal. Heating the air stream prevents condensation of water vapor in the downstream components.

The HEPA filters remove particles from the gas stream that might have passed through the scrubber. Elemental iodine and methyl iodide are absorbed on the packed silver-zeolite adsorber or impregnated charcoal bed. The gas stream leaving the iodine recovery is continuously monitored for radioactivity and combined with the ventilation air and sent to the ventilation stack.

The particulate decontamination factor (DF) is assumed to be 10^4 for the combined prefilter and HEPA filter. The decontamination factor for the iodine recovery system is assumed to be 10^3 .

A.5 Facility Operation

This section describes manpower requirements and facility operating parameters that contribute to the environmental effects described in the generic environmental impact statements on the Spent Fuel Policy.

A.5.1 Operating Manpower

The operating staff at an ISFS facility varies over the lifetime of the facility and depends upon the projected work load. In ISFS facilities envisioned in the GEISs on Spent Fuel Policy, three different types of operation, i.e., receiving, storing, and removing are assumed. The manpower requirements during receiving and removing spent fuel is higher than during storage. Figure II-A-16 shows the average operating staff over the entire life of the ISFS for facilities designed for receiving and unloading rates of 500, 1000, and 2500 MTU/year. As can be seen, the operating staff is also a function of total basin capacity.

Approximately 75% of the manpower given in Figure II-A-16 is associated with operation of the facility and handling of the fuel. This manpower receives radiation exposure during operation of the facility. The remaining 25% of the staff personnel is assumed not to be exposed to radiation during facility operation.

FIGURE II-A-16. ISFS Operating Staff

A.5.2 Consumable Material

The major consumable material used in an ISFS facility that affects the environment is coal consumption. Coal is used to generate the electricity and the process steam needed to operate the facility. The coal consumption is shown in Figure II-A-17.

Periodically, the diesel motors for emergency power are started and tested to ensure their operation in case of an emergency. These diesel motors are run on the average of two hours each month. The amount of diesel fuel consumed is also shown in Figure II-A-17.

A.5.3 Waste Generation

A.5.3.1 Secondary Radioactive Waste

The amounts of waste generated during operation of the generic ISFS facility are given in Figures II-A-18 and II-A-19. About 95% by volume of the radioactive wastes generated at the facility is solid material and includes ventilation filters, rags, protective clothing, plastic, wood, rubber, failed equipment, and similar equipment. This waste contains about 0.04 Ci of fission products and activation products/MTU of spent fuel received.

Wet wastes arise primarily from operation of the water treatment and decontamination system operations and consist of filter sludges, ion-exchange regeneration solutions, and detergent solutions that are concentrated in the evaporator. These wastes have been solidified as described in A.4.4.2 of this section. These wastes comprise about 5% of the volume of the total wastes from the ISFS facility but contain ~98% of the radionuclide or ~2 Ci/MTU.

All wastes generated during operation of the facility are assumed to contain less than 10 nCi/g of transuranic (TRU) nuclides. TRU isotopes that are released to the basin water are retained in the basin as crud and are removed during decontamination and decommissioning of the facilities.

In addition to the liquid and solid waste generated by operation of an ISFS basin and (shown in Figures II-A-18 and II-A-19 for each 500-MTU basin), 1.7×10^{-2} Ci/year of radionuclides is expected to be released as particulate material along with the gaseous waste released from the facility. This release is based upon operating experience at a DOE basin facility. The activity is expected to be distributed as follows: $^{55}\text{Fe} = 4\%$, $^{60}\text{Co} = 5\%$, $^{90}\text{Sr} = 1\%$, $^{134}\text{Cs} = 12\%$, and $^{137}\text{Cs} = 78\%$.

FIGURE II-A-17. Fossil Fuel Consumption

FIGURE II-A-18. Secondary Radioactive Waste Generation

FIGURE II-A-19. Secondary Radioactive Waste Generation

A.5.3.2 Nonradioactive Waste

Nonradioactive liquid wastes, consisting of chemical and sanitary streams generated at the ISFS facility, are monitored, collected, and treated, if necessary, prior to discharge. The extent of monitoring and treatment is dependent upon the type of liquid effluent and government regulations. Sanitary wastes (see Figure II-A-20) are discharged over land and through a spray irrigation network after pretreatment. All other liquid wastes (sewer discharges containing chemical wastes and cooling tower blowdown) are discharged to the evaporation pond. Chemical wastes released to the chemical sewer are assumed to be equal in generation rate to sanitary wastes and are shown in Figure II-A-21. Cooling tower blowdown for both the storage facility and steam plant is shown in Figure II-A-21. The quantity of normal solid plant refuse is shown in Figure II-A-22.

FIGURE II-A-20. Nonradioactive Liquid Waste Generation Rate

FIGURE II-A-21. Nonradioactive Liquid Waste Generation

FIGURE II-A-22. Nonradioactive Solid Waste Generation

REFERENCES FOR SECTION II,A

1. *Oconee Nuclear Station Units 1, 2, and 3, Final Safety Analysis Report.* Duke Power Company, Charlotte, NC (May 1969).
2. *Draft Environmental Statement Related to the Operation of the Barnwell Fuel Receiving and Storage Station.* U.S. Nuclear Regulatory Commission, Division of Materials and Fuel Cycle Facility Licensing, Washington, DC (May 1975).
3. U.S. Nuclear Regulatory Guide, 3.24, *Guidance on the License Application, Siting, Design, and Plant Protection for an Independent Spent Fuel Storage Installation.* U.S. Nuclear Regulatory Commission, Division of Materials and Fuel Cycle Facility Licensing, Washington, DC (December 1974).
4. A. B. Johnson, Jr. *Behavior of Spent Nuclear Fuel in Water Pool Storage.* BNWL-2256, Battelle Pacific Northwest Laboratories, Richland, WA (September 1977).
5. "Spent Fuel Storage." *Standard Review Plan.* Section 9.1.2, U.S. Nuclear Regulatory Commission, Division of Materials and Fuel Cycle Facility Licensing, Washington, DC (February 1975).
6. *Design Objectives for Highly Radioactive Solid Material Handling and Storage Facilities in a Reprocessing Plant.* ANSI N 305, American National Standards Institute, Inc., New York, NY (March 1975).
7. *Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel.* NUREG 0404, U.S. Nuclear Regulatory Commission, Washington, DC (March 1978).
8. *Safety Evaluation Report Related to Operation of Barnwell Fuel Receiving and Storage Station, Allied General Nuclear Services.* NUREG 0009, Docket 70-1729, U.S. Nuclear Regulatory Commission, Office of Nuclear Materials and Safeguards, Washington, DC (January 1976).
9. R. D. Cyboron, R. S. Harding, and F. Bevilacqua. "High Capacity Storage of Nuclear Fuel Assemblies." Paper presented at ANS Winter Meeting, San Francisco, CA, November 16-21, 1975.

10. *Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle.* ERDA-76-43, U.S. Energy Research and Development Administration, Division of Nuclear Fuel Cycle and Production, Washington, DC (May 1976).
11. *Safety Evaluation Report for Morris Operational Fuel Storage Expansion.* NEDO-20825, General Electric Company, San Jose, CA (March 1975).
12. W. H. Baker and F. D. King. *Technical Data Summary, Spent Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant.* DPSTD-AFCT-77-7, Savannah River Laboratory, E. I. du Pont de Nemours and Company, Aiken, SC (August 1977).

B. GEOLOGIC STORAGE IN SALT FORMATION

A demonstration of using salt as a repository for highly radioactive solid wastes was provided by the pilot repository Project Salt Vault. This pilot repository, in which irradiated reactor fuel elements were used to simulate the wastes, was operated in a completely retrievable manner from 1965 to 1968. Details of the design and operation of this facility are provided in Reference 1. Since April 1967, the Federal Republic of Germany has operated a radioactive waste repository at the Asse Salt Mine. Low- and intermediate-level wastes are permanently buried at this facility. No preparations for handling high-level or TRU wastes were made at the Asse mine. A summary of design and operating experiences for this German facility is provided in Reference 2.

Discussed in this section are facilities conceptualized solely for environmental analyses in support of the Spent Fuel Policy. Selection of a salt repository in these environmental analyses is not to infer that salt is either the most likely or optimum alternative for a geologic repository, but only that it is one which has received the most study. Neither the alternatives to a geologic repository nor the alternative host materials for geologic repositories are discussed in the GEISS on the Spent Fuel Policy since they are to support that policy, not to determine which form of geologic repository should be constructed. The type of repository will not significantly affect the environmental effects being considered. An environmental impact statement on commercial waste management is being prepared, and it will discuss these matters.

This repository is assumed to receive encapsulated spent fuel elements, reprocessing waste, and low-level transuranic wastes (LLW-TRU). The characteristics of spent fuel elements are described in Section I of this report, and their encapsulation is described in this section. The reprocessing wastes are described in Section II., C. of this report. The terms used for reprocessing and LLW-TRU wastes are given in the Glossary.

The generic concept utilizes the short room and pillar layout in the repository underground areas and allows use of the natural salt formation to provide ceiling support and mine stability. Several other concepts are currently being studied, and no decision has been reached regarding the concept that will finally be used for the repository for commercial waste. These choices affect the economics of this storage more than storage safety. Vertical burial of waste canisters in the salt floor is also assumed since it simplifies handling and burial of waste and simplifies the

problems associated with retrievability of this waste if that is required. Horizontal or angle burial, while decreasing ceiling height requirements, raises difficult engineering questions in assuring retrievability.

The salt excavated from the mine must be handled and stored to minimize the environmental impact. Preliminary studies³ indicate that several methods of storing the excavated salt may have acceptable environmental impacts. Alternatives considered include controlled surface storage, industrial use, and environmental blending. Industrial use would entail onsite or offsite processing of the salt and its sale in the industrial chemicals market. Environmental blending would involve either ocean disposal or deep-well injection as a potential means of dealing with the mined salt. The controlled surface storage concept may still require rail transport to an off-site surface storage facility, because an acceptable site for a repository would not necessarily be an acceptable site for permanent storage of excess excavated salt. For impact evaluations in the GEIS on Spent Fuel Policy, it will be assumed that excavated salt storage will be onsite, although onsite tracks and equipment have been included to allow transport offsite. Selection of this onsite surface storage concept does not imply that this alternative has significant advantages over others considered, but selection of this alternative was made because its effects are best known.

B.1 Design Bases

The underground portion of the generic repository occupies a potential area of 810 ha (2000 acres). Spent fuel or reprocessing waste will emit heat from decay of radionuclides. This radionuclide decay heat has to be transmitted to the surface. This heat dissipates into the host rock formations, flows through other rock units surrounding or overlying this formation, and eventually reaches the atmosphere. The amount of heat placed in a given repository must be limited to ensure such factors as

- Integrity of the host rock and the surrounding rock units
- Acceptable long-term surficial uplift and subsidence
- Acceptable temperature rises in overlying aquifers near the earth surface
- Room and pillar stability throughout the retrievable period or period of operation of the mine
- Integrity and recoverability of the spent fuel or waste canisters throughout the retrievable period

The allowable thermal release rate is dependent upon repository design, retrievability requirements, stratigraphy, and rock and canister properties.

An iterative technique is used to establish allowable thermal loading. It involves spent fuel and/or waste and canister temperature requirements, room and pillar stability analyses, and far-field thermal and rock-mass response analysis. Details of this iterative approach are described in Reference 4. Based upon a given canister heat load and configuration, canister spacings were varied until the stresses in the vicinity of the canister, around the room and in the pillars, were adequate to ensure retrievability of the canister and to ensure that the canister upper temperature limits would not be exceeded. In cases where this limit would be exceeded, the canister heat load was decreased; or the canister arrangement was modified. This evaluation procedure was repeated until maximum allowable canister and local areal thermal loading were established. When a satisfactory local areal thermal load was determined, far-field analyses were conducted for the entire repository to evaluate the impact over the long-time frames of the thermal and mechanical stresses generated by the placement of spent fuel or waste. If these impacts were not acceptable, the heat loads were then modified; and the analysis was repeated.

This iterative analysis results in three thermal loading limits useful for the development of the repository design: 1) a canister loading limit, 2) a local loading limit, and 3) an entire 810 ha (2000-acre) repository loading limit.

The canister thermal loading limit (in units of kW/canister at waste emplacement) must be controlled to keep the waste and canister temperatures below 200°C (390°F) for spent fuel and 375°C (710°F) for high-level waste.⁴ Analyses of the rock stresses in the immediate vicinity of the canister indicate that if these canister loading limits are not exceeded, the integrity of the rock surrounding the canister sleeve configuration is maintained. Results of calculations⁴ show that spent fuel canister loading limits are assured by the loading of only one spent fuel assembly per can. The canister loading limit for high-level waste from reprocessing waste is determined by the amount of high-level waste placed in the canister at the reprocessing facility.

The local thermal loading limit in units of kW/acre at emplacement must be controlled so that rock mechanics considerations indicate that room and pillar stability for the configuration is satisfactory throughout the retrievable period. The term "local loading" refers to the heat generation capacity associated with a single storage room, including the width extending from pillar centerline to pillar centerline. The local loading limits⁴ are given in Table II-B-1.

TABLE II-B-1

Geologic Repository Thermal Loading Limit

Type of Waste ^a Age of Waste	<u>Spent Fuel</u>		<u>Reprocessing Waste</u>	
	5 yr	10 yr	5 yr	10 yr
Local Loading Limit, kW/acre				
5-yr Retrievability	175 ^b	150 ^b	175 ^b	150 ^b
25-yr Retrievability	42 ^c	36 ^b	42 ^c	36 ^b
Total Repository Loading				
kW/acre	100 ^c	60 ^b	(175)×(UF ^e) ^c	(150)×(UF ^e) ^c
Total MW ^d	200	120	-	-

- a. Unless otherwise specifically noted, the term waste in this table and in the following text will be inferred to include either or both spent fuel and reprocessing waste.
- b. Limits from Reference 4.
- c. Limits from direct communications with S. C. Matthews, Office of Waste Isolation (OWI).
- d. For 810-ha (2000-acre) repository.
- e. Utilization factor or % of rooms in mine containing high-level waste.

The total thermal loading limit at waste emplacement for the entire 810-ha (2000-acre) repository must be low enough that the far-field thermal and rock mechanics studies predict that the long-term effects near the earth's surface will be acceptable. Because the far-field effects are determined by the thermal output of the mine for the first thousand years or so, these thermal effects are the long-term integral of the thermal output of the mine. This long-term integral can be expressed as a function of the thermal output of the mine at initial emplacement of the waste. Expressed in this manner, the long-term thermal limits in Table II-B-1 are expressed as kW/acre at emplacement time for the total mine and also the allowed thermal limit for the 810-ha (2000-acre) repository. These over-all loading limits are consistent with both the canister and local thermal loading limits.

As can be seen from Table II-B-1, two local loading limits are given: 1) 5-year retrievable waste, and 2) 25-year retrievable waste. Both of these limits prevent rapid closure of a room that could prevent retrievability of the waste for the specified

retrievability period. Satisfactory closure does not mean, however, that routine surface maintenance will not be required in the room during the retrievability period. For example, during the 5-year retrievability period, scraping the floor of the room after about 2-1/2 years will probably be necessary to maintain ingrowth of salt at a reasonable level. Likewise, in the 25-year retrievability case, the salt intrusion will be comparable to the 5-year retrievability ingrowth, but the ingrowth occurs over 25 years rather than 5 years. Periodic scraping of the walls and floor of the room will be required. About 0.9 m (3 ft) of room closure would be expected during either the 5- or the 25-year retrievability periods.

Spent fuel elements or reprocessing wastes are shipped to the repository in shielded rail casks. Transportation requirements for spent fuel or waste are described in Section III of this report. The thermal characteristic of spent fuel decayed 5 years after discharge from the reactor is specified in Table II-B-2.

TABLE II-B-2

Thermal Characteristics of Spent Fuel

	<i>Thermal Output, watts/canister</i>
BWR	400
PWR	970
CANDU	2000

The facilities at these repositories will be designed to protect the operating personnel and the public from radiation and/or contamination hazards and to contain radioactive materials during both normal operation and emergency or accident situations. Surface facilities containing radioactive materials will be designed to maintain their integrity during natural disasters (i.e., tornadoes, earthquakes, etc.). Subsurface facilities are constructed deep in the salt formation at an assumed 560 m (1800 ft) below the surface to provide immunity from surface activities that could otherwise breach the repository integrity.

The U. S. Government will acquire title to all the land area overlying the excavations; thus, drilling, mining, or other human activities that could conceivably breach the storage formation will be prevented. Selective surface rights for this area might, however, be leased for limited recreational or agricultural activities. During operation of the facility, these recreational or agricultural activities will be excluded within 0.8 km (0.5 mile) of the surface facilities and mineshafts. Additional restrictions on underground activities will be established over an area extending outward about 3.2 km (2 miles) from the excavation. It is planned, where possible,

that these restrictions will be defined by acquisition of perpetual underground rights. (Simple title of the newly purchased land allowing unrestricted use of the land surface is not assumed.)

B.2 Operation at Geologic Storage Facilities

Initially, the repository is assumed to operate in a retrievable mode for spent fuel and waste handling. That is, all spent fuel or reprocessing waste could be removed from the repository if circumstances requiring this action should develop. The assumed period of retrievability is the initial 5 years of repository operation, but the period could be extended. To minimize mine closure during this period and to allow for the possible extension of the retrievable period, the spent fuel storage will be limited to the thermal loading that allows 25 years retrievability (Table II-B-1). When retrievability is no longer required, thermal loading of each room will be increased by emplacement of more waste (up to a 5-year retrievable limit, Table II-B-1) before room closure.

B.2.1 Spent Fuel Handling Operations

All spent fuel arrives at the repository in shielded shipping casks. Facilities involved for spent fuel handling operations include:

- Canistered Waste Building
- Canistered Waste Shaft
- Waste Mine

Individual spent LWR fuel assemblies are removed from their casks in shielded transfer cells, encapsulated in the helium-filled steel canisters after they dry, then lowered to mine level through the Canistered Waste Shaft and transported to the appropriate salt bed chamber with a shielded transporter. Two baskets of CANDU fuel assemblies are removed from the cask in the transfer cell and encapsulated together in a single helium-filled steel canister after drying. This canister is then lowered to the mine level through the Canistered Waste Shaft and transported to the appropriate salt bed chamber in a shielded transporter. The encapsulated spent fuel is then buried in vertical burial holes drilled into the floor of the room. A schematic flow diagram of all operations from receiving and shipping casks and containers to loading canisters into the Canistered Waste Shaft are conducted in the Canistered Waste Buildings (Figure II-B-1). All operations within this building are remotely controlled from adjacent operating rooms by manipulators or automated systems.

FIGURE II-B-1. Spent Fuel Handling Operation

The exterior surface and interior atmosphere of each spent fuel cask is checked for contamination upon receipt at the Canistered Waste Building. If this check indicates exterior surface contamination, the cask is decontaminated. If the interior atmosphere is contaminated, the cask is moved to the overpack transfer cell immediately following the required cooling. Casks with uncontaminated interior atmospheres are taken to transfer cells. As the individual fuel assemblies are removed in the transfer cell, they are visually inspected for shipping damage and smeared to detect possible surface contamination. Spent fuel that is not contaminated or damaged is encapsulated in a helium-filled canister. The packaged assemblies are temporarily stored (about 2 days) and inspected to ensure that the canisters are not leaking helium before transfer to below-ground geologic storage. Spent fuel assemblies that are physically damaged or have surface contamination are returned immediately to their casks, and the casks are resealed and removed to the overpack transfer cell.

The overpack transfer cell receives all casks containing damaged or contaminated fuel. Each fuel element is first placed in a fuel canister and then placed in a second container called an overpack. Empty canisters are stored in the overpack transfer gallery. These canisters are individually positioned and raised to be flush with the floor of the overpack transfer cell above. Once the canister unit is positioned, the cask is opened, and the damaged or contaminated fuel assembly is loaded into the canister. The canister is sealed. The canister containing the failed or contaminated spent fuel is then placed in the second canister called an overpack and it is sealed. This second canister provides the same protection for damaged or contaminated fuel as is provided by the canister for normal spent fuel. These doubly-encapsulated, damaged spent fuel canisters are temporarily stored and inspected to ensure that the overpack is not leaking before transfer to below-ground geologic storage. Remote control equipment is used for all transfer operations. Shipping casks are decontaminated as required at the overpack transfer cell.

B.2.2 Reprocessing Waste Handling Operations

Reprocessing waste (high-level waste, HLW; cladding waste; and intermediate-level waste, ILW) is received into the Canistered Waste Building and is handled in a similar fashion to spent fuel. These canistered wastes are received from the reprocessing plant in shielded casks. A second canister is added to the waste as received only if the first has been damaged and is leaking as received. The unloading and transporting of the waste to a salt chamber is similar to the operation described in B.2.1 of this section.

B.2.3 Low-Level Waste Handling Operations

The basic difference between LLW operations and canistered spent fuel waste operations is that LLW-TRU waste does not require remote or shielded handling. Figure II-B-2 shows a schematic flow diagram of facilities involved in LLW operations. Facilities involved for LLW handling operations include:

- Low-Level Waste Building
- Low-Level Shaft
- Low-Level Mine

All LLW-TRU waste is received in lightly shielded overpacks (or cargo carriers) mounted on rail carriers. The cargo carriers are removed from the carrier and placed on transfer cars in the Unloading and Storage Area of the Low-Level Waste Building.

The exterior surface and interior atmosphere of each cargo carrier is checked for radioactive contamination upon its arrival at the Low-Level Waste Building. Pallets containing the 0.2 to 0.3 m³ 210-L and 302-L (55- and 80-gal) drums and boxes with approximate dimensions of 1.2 × 1.2 × 1.8 m (4 × 4 × 6 ft) are removed from the cargo carriers, smeared to determine surface contaminants, and decontaminated, if required. Although contamination is not likely to be present, facilities are provided to repair and decontaminate drums and boxes that are contaminated or damaged. Cargo carriers and pallets are decontaminated when required. Clean or decontaminated pallets of drums and boxes are transported by forklift to the Low-Level Shaft. After pallets, their contents, and boxes are lowered through the shaft, they are stacked two high in the Low-Level Repository.

B.2.4 Mining Operations

After initial repository construction, use of explosives in the mine is not necessary. Electrically powered mining machines are used to construct the storage rooms with maximum production capability of about 2,200,000 tonnes (2,400,000 tons) of salt per year. Diesel powered load-haul-dump units (LHDs) transport the mined salt to branch corridor conveyors.

Branch corridor conveyors transfer the salt to roof-suspended mainline conveyor systems that transport the salt to 2300-tonne (2500-ton) underground storage silos. An apron feeder beneath the storage silos transfers the salt to the skip hoist for elevation to the surface through the Men and Materials (M&M) shaft.

II-55

FIGURE II-B-2. Schematic Flow Diagram - Low-Level Waste Building⁵

Tailings from initial mining activities will be conveyed directly to the surface storage building where they will be stored in a controlled environment or transported to a surface disposal pit located outside the area fence. After these initial areas are excavated and retrievable storage is no longer required, mining and waste burial activities proceed concurrently. The tailings from these operations are used to backfill rooms as they are filled with wastes. About 60% of the total salt mined will eventually be reused for backfilling operations; the excess 40% will require disposal. Total mined salt from this repository is estimated to be 61,000,000 tonnes (67,000,000 tons).³

Men, materials, and equipment entering or leaving the mine use the M&M main corridor. It is similar to the LLW stations. The M&M Shaft is separated from the corridor by a steel door for personnel safety. The lock separating the shaft from the mine is of fireproof construction and also serves as a refuge for mine personnel in case of fires or other emergency conditions.

B.3 Facility Description

The waste repository consists of surface facilities for waste receiving and handling, mining support, and general operations support and subsurface facilities for waste handling and storage and mined salt removal. Figure II-B-3, a facility plot plan, shows the surface facility layout. Figure II-B-4 shows subsurface facilities.

FIGURE II-B-3. Plot Plan for Repository Surface Facilities⁵

B.3.1 Surface Facilities

The surface facilities described in this section encompass all buildings on the site. The surface facilities shown in Figure II-B-3 and the salt disposal area (not shown on Figure II-B-3 but located just outside the area fence) constitute the only visible evidence of the repository. The site disposal area will eventually occupy an additional 80 ha (200 acres).

B.3.1.1 Canistered Waste Building

The Canistered Waste Building provides the capabilities necessary to receive and handle spent fuel and wastes requiring remote handling. Spent fuel and wastes are shipped to the Canistered Waste Building in shielded containers. In addition to those spaces provided for the handling and processing of canisters before delivery to the mine, space is provided for support functions and activities not directly associated with canister handling. These spaces include a chemical laboratory, a personnel check station, a change

FIGURE II-B-4. Underground Repository⁵

room, a shift foreman's office, and a maintenance and storage room. Access from these spaces to the operating gallery and the cask unloading area is through air locks. In the basement, rooms are provided for mechanical equipment housing, repository-generated waste concentration, and packaging and equipment decontamination. Plans and sections of the Canistered Waste Building are provided in Figure II-B-5.

The Hoist House servicing the Canistered Waste Mine Access Shaft is a separate structure adjacent to the Canistered Waste Building.

B.3.1.2 Low-Level Waste Building

The Low-Level Waste (LLW) Building provides the capabilities necessary to receive and handle all low-level waste shipped to the repository by railroad in closed cargo carriers. In addition to spaces for the handling and processing of pallets before delivery to the mine level, space is provided to support functions not directly associated with LLW handling. These spaces include a counting room, a personnel check station, a change room, an operations office, a health physics office, and a battery charge area. An exhaust fan filter room is provided in a basement below the support area. LLW building plans and sections are provided in Figure II-B-6.

The Hoist House servicing the LLW Mine Access Shaft is a separate structure adjacent to the LLW Building.

B.3.1.3 Support Buildings

Buildings which support the mine operations are discussed in this section. The Mine Operations Building is near the M&M shaft. It contains change room and shower facilities for mining personnel, a dispensary, a personnel check station, a supply storage facility, and office accommodations for all mine-related functions. The Mine Operations Building also contains general facilities for repair and maintenance of noncontaminated plant equipment and automotive equipment.

An open truck loading-unloading dock and an adjacent railroad siding are provided. A full basement is provided under the Mine Operations Building to house the mechanical equipment serving the mine ventilation system, the repository main switch gear and standby power generators, and the central control and monitoring station.

The Repository Administration Building is near the LLW Building. It houses all central administrative personnel associated with the operation of the repository. Office space and a central

FIGURE II-B-5. Canistered Waste Building – Plans and Sections⁵

FIGURE II-B-6. Low-Level Waste Building — Plans and Sections⁵

lunch and conference room are provided for repository management, engineering, accounting, personnel, safety, and payroll functions and for files and records.

Warehouses are designed to serve the needs of the repository for storage space and for a maintenance shop for vehicles not associated with mining operation.

The Gatehouse is at the fenced entrance to the repository. It is the main checkpoint for all employees, visitors, and automotive, truck, and railroad traffic entering and leaving the repository. Annunciator and audible indicators for all alarm systems are installed in the facility and are tied into all communications systems used in the facility.

B.3.1.4 Salt Tailings Storage Warehouse

The tailings storage area is conservatively sized to accommodate 1,200,000 m³ (42,000,000 ft³) of salt. The salt storage area and handling equipment (surge hopper, belt conveyor, and transfer house) are inside a salt storage warehouse which is within a railroad car loop. The rail facilities serve as an area for shipping the tailings to an alternative offsite location for disposal if necessary. The storage area is provided with controlled ventilation which controls the humidity of the air within the salt warehouse to below the critical humidity⁶ so moisture pickup will not occur. The ventilation eliminates caking of the salt due to moisture pickup and drying. The Salt Tailings Storage Warehouse also incorporates features such as a PVC-plastic underliner and runoff control system to prevent environmental contamination from the salt storage. The storage warehouse is also equipped with transfer equipment to transfer salt to the salt disposal area immediately outside the area fence.

B.3.1.5 Salt Tailings Disposal Area

Salt tailings which will not be reused or exceed storage capacity of the salt tailings warehouse will be transferred to a storage area for disposal. The disposal area will be a 9.1-m (30-ft) deep pit excavated adjacent to the area fence and will be lined with PVC-plastic and equipped to control and contain all rainwater. This liner will control and prevent environmental contamination from the salt storage. As salt is received, it will be covered with another PVC-plastic cover, then with 3.0 m (10 ft) of earth cover. The salt tailings disposal area will occupy an area of about 900 × 900 m (2700 × 2700 ft) and when finally completed will stand ~13 m (40 ft) above normal grade.

B.3.2 Underground Facilities

The underground facilities at the repository cover a horizontal area of 810 ha (2000 acres). This area includes three waste mines (BWR, PWR and CANDU, and LLW-TRU). Capability exists for adding mines for HLW, cladding waste, and ILW if they become necessary due to reprocessing. These underground facilities also include all corridors, ventilation ducts, all shafts, underground storage, and receiving stations.

B.3.2.1 Shafts and Hoists

Five shafts are required. These shafts include the Canistered Waste (CW) Shaft, the Low-Level Waste (LLW) Shaft, two Men and Materials (M&M) Shafts, and the Ventilation Exhaust (VE) Shaft. These shafts all differ in size.

B.3.2.1.1 Canistered Waste Shaft

The Canistered Waste Shaft provides a means for transporting the canisters of waste (encapsulated spent fuel, high-level, cladding waste, and intermediate-level) from the Canistered Waste Building to the underground storage levels. This shaft has an inside diameter of 1.8 m (6 ft) and is lined with 0.3 m (1 ft) of reinforced concrete. The top of the shaft forms an integral part of the Canistered Waste Building.

The Canistered Waste Shaft provides access to the PWR and CANDU fuel assembly, the reprocessing waste storage area at one elevation, -530 m (-1740 ft), and to the BWR fuel storage area at a lower elevation, -560 m (-1835 ft). (These two mine levels and the LLW mine, described in the next paragraph, are located so they do not overlay each other.) From shielded receiving stations at these two elevations, the canisters will be transferred to shielded transporter vehicles for transport to the storage sites. The canister loading and unloading of the specially constructed elevator cage used in the Canistered Waste Shaft is by remote control, thus ensuring that the canisters are shielded at all times.

B.3.2.1.2 Low-Level Waste Shaft

The Low-Level Waste Shaft is to provide access for transporting the LLW pallets from the Low-Level Waste Facility to the Low-Level Underground Storage Area. This shaft has an inside diameter of 3 m (10 ft) and is lined with 0.3 m (1 ft) of reinforced concrete. The shaft connects the Low-Level Waste Facility with the underground receiving station at storage level, where units are picked up by forklift trucks and loaded onto a transporter.

B.3.2.1.3 Men and Materials (M&M) Shaft

Each M&M Shaft is designed to:

1. Provide access for men and materials into and out of the different underground storage areas
2. Provide salt hoisting for the mining operation of the total system
3. Serve as the supply duct for introducing ventilation air into the entire mine

To perform these functions, the reinforced concrete-lined shaft, 8 m (26 ft) in diameter, is subdivided into three main compartments (men and materials, mine hoisting, and air supply) and two auxiliary compartments for cage counterweights and for the manway.

The M&M compartment is sufficiently large to accommodate a cage with a capacity for 70 men. The compartment is large enough to permit the lowering and raising of large pieces of equipment either in assembled or partly disassembled form.

The salt hoisting compartment is sufficiently large to accommodate hoisting 5.5 million tonnes (6 million tons) of salt to the surface during the peak mining year.

The air supply compartment has a minimum cross-sectional area of about 18 m^2 (200 ft^2) and is sized for an ultimate flow of $14,000 \text{ m}^3/\text{min}$ ($500,000 \text{ cfm}$). This compartment is completely isolated from all other shaft compartments by steel divider walls.

Two auxiliary compartments are provided adjacent to the M&M compartment. One compartment includes a space for utilities and a manway, which is an emergency exit for all mine personnel. The other auxiliary compartment contains counterweights for the M&M hoist. Neither the manway nor the counterweight compartment is airtight with respect to the M&M compartment.

B.3.2.1.4 Ventilation Exhaust Shaft

Ventilation air in the mine-level area is exhausted through this shaft into filtration systems on the surface. This shaft has an inside diameter of 6 m (20 ft), is reinforced concrete-lined, and also serves for emergency ingress to or egress from the mine. A 110-m (360-ft) stack is used to exhaust the mine ventilation.

B.3.2.2 Mine Facilities

The underground portion of the repository encompasses a total affected gross area of about 810 ha (2000 acres) and is shown schematically in Figure II-B-4. The mine layout and storage sequence design is a conventional room-and-pillar layout. It incorporates the requirements of mine ventilation, mine opening stability, thermal effects, and efficient use of excavated space and of mining and haulage equipment. A modular approach to the mine development allows the simultaneous conduction of mining activities and waste storage operations, where excavation of one panel of storage rooms proceeds without interfering with the conduction of storage operations in the previously excavated panels. All mining activities involved in initial phase repository construction are completed before burying any radioactive wastes.

To ensure shaft stability, a 46-m (150-ft) radius safety zone is designated around each shaft. All travelways, except those leading to the shafts, pass outside this zone.

All storage rooms receiving waste canisters are designed to have only one entry. The one entry provides positive control for heat, ventilation, and exhaust for each room. Rooms for low-level waste storage are designed as an open-grid system to permit easy storage without degrading ventilation performance. Ceiling height for the spent fuel portion of the mine is 7.6 m (25 ft) and the reprocessing waste portion is 5.4 m (18 ft).

B.3.2.2.1 BWR Spent Fuel Mine

BWR spent fuel storage is accommodated at two different levels — elevations of -525 m (-1725 ft) and -550 m (-1800 ft). The largest BWR storage mine is located at the -550-m level. Access to this storage level consists of two BWR main corridors originating from the receiving station at the foot of the Canistered Waste Shaft at -560-m (-1835-ft) elevation. The M&M main corridor to this level runs between these two corridors. The BWR storage at -525-m elevation is serviced by a single BWR main corridor and an M&M main corridor. Each BWR main corridor connects with branch corridors from which extend the storage rooms. Each storage room is 11 m wide × 7.6 m high × 170 m long (36 × 25 × 560 ft). The pillar between two adjacent storage rooms is 13 m (42 ft) wide. In each room, canisters are placed in vertical holes bored into the floor of the room. For terminal storage, 372 BWR fuel canisters which have cooled 5 years are stored in a storage room. This requires three rows of canisters in each room with a 3.8-m (9-ft) pitch between rows. The pitch between canisters within a row will be 1.8 m (6 ft). During initial storage, when the waste is retrievable, only 124 BWR fuel canisters cooled 5 years can be stored in each room. During this time, a canister is placed in every third hole described above, and the pitch between canisters within a room will be 5.4 m (18 ft).

In addition to the fewer number of fuel canisters required for retrievability, the holes used for burial of spent fuel are provided with liner sleeves similar to those shown in Figure II-B-7. Once retrievability is no longer required, the sleeve, shown in Figure II-B-7, will be eliminated for new burial, and the hole containing the encapsulated spent fuel will be backfilled with salt. Waste already buried in this retrievable form will be abandoned with the sleeve in place.

FIGURE II-B-7. Retrievable Burial

The BWR mine areas are located so that no significant heating effects impinge on the operation in the shafts or in the mine areas for other waste types.

B.3.2.2.2 PWR Spent Fuel Mine

The receiving station of the Canistered Waste Shaft, 33.5 m (110 ft) above the mine level receiving facility for BWR canisters, receives PWR canisters. Similar to the BWR mine, its location is chosen so that it does not thermally impact operation in other mine areas. Main corridors for men and materials run parallel with the

BWR main corridor. The main corridor, branch corridor, and storage room arrangements are similar to those of the BWR mine. Each storage room accommodates 123 canisters containing 5-year cooled spent fuel which are arranged in three rows with a 3.7-m (12-ft) pitch between rows and a 4.3-m (14-ft) pitch between canisters within each row. During initial storage, when the waste is retrievable, only 53 PWR fuel canisters cooled 5 years can be stored in each room. During this time, spent fuel canisters are omitted from every other hole and from 16% of the remaining positions to control short-time thermal loading. Canister spacing is a 3.7-m (12-ft) pitch between rows and a 6.7-m (28-ft) pitch between canisters in each row. Storage holes and liner sleeves are similar to those described in "BWR Spent Fuel Mine."

B.3.2.2.3 CANDU Spent Fuel Mine

CANDU spent fuel mine is at the same level as the PWR mine discussed in B.3.2.2.2 of this section. Separate storage rooms will be used to store CANDU fuel.

Main corridors for men and materials run parallel with the BWR main corridor. The main corridor, branch corridor, and storage room arrangements are similar to those of the PWR mine. Each storage room accommodates 56 canisters containing 5-year cooled spent CANDU fuel which are arranged in two rows with a 6-m (19.6-ft) pitch between rows and a 5.3-m (17.4-ft) pitch between canisters within each row. During initial storage, when the waste is retrievable, only 25 CANDU fuel canisters cooled 5 years can be stored in each room. During this time, spent fuel canisters are omitted from every other hole and from 11% of the remaining positions to control short-time thermal loading. Canister spacing is a 6-m (19.6-ft) pitch between rows and an 8-m (26-ft) pitch between canisters in each row. Storage holes and liner sleeves are larger in diameter but shorter than those described in B.3.2.2.1 of this section.

B.3.2.2.4 Reprocessing Waste Mine

Reprocessing waste [high-level waste (HLW), cladding waste, and intermediate-level waste (ILW)] will be stored at the same level as the PWR mine discussed in B.3.2.2.2 of this section. If these wastes are stored in the repository at the same time as spent fuel, they will be stored in separate storage rooms.

The main corridor, branch corridor, and storage room arrangements are similar to those of the PWR mine. Each storage room accommodates 27 canisters of HLW containing 5-year cooled spent fuel. The canisters are arranged in a single row with a 5.5-m (18-ft) pitch between each one.

In addition to HLW, two rows of cladding waste and ILW can be stored in the same HLW room. One row is placed on each side of the HLW central row. These rows are assumed to be 3.7 m (12 ft) from the central HLW row and canister emplaced on 2.7-m (9-ft) pitch.

In the environmental impact statement on the Spent Fuel Policy, it is assumed that reprocessing waste will not require retrievability and thus will be emplaced in a hole in the salt and backfilled with salt.

B.3.2.2.5 Low-Level Transuranic Waste Mine

A main corridor to the LLW-TRU storage area receives LLW-TRU material from the receiving station of the LLW Shaft. The main corridors are connected by branch corridors similar to those in the spent fuel mines except that the rooms will be in an open-grid system. Each room is divided into nine pallet storage areas. These areas, 5 m wide $\times$ 17 m long (16 ft wide $\times$ 48 ft long), accommodate 64 pallets (768 drums) and allow longitudinal aisles 3.4 m (11 ft) on each side to permit convenient entry. Transverse aisles 3 m (10 ft) wide are between the pallet storage areas for easy access. The pallets are stacked two pallets high.

The receiving stations of LLW [at elevation -500 m (-1650 ft)] provide space for transfer of the LLW pallets from the cage to the transporter. The walls, floors, and roof of the LLW station are built of reinforced concrete and are structurally tied into the shaft concrete. An adequate parking area for trucks, personnel rest rooms, and a small workshop is adjacent to the receiving station. The station is isolated from the main entry by a double-door air lock.

B.3.2.3 Ventilation Systems

Fresh air is used to purge the underground portion of the mine and is discharged to the atmosphere after filtration through a 110-m (360-ft) ventilation stack. Salt mines are exceptionally free from explosive gases, but occasionally the mines encounter small pockets of flammable gas that is not sufficient to make control a serious problem.⁷ The main function of ventilation is to provide fresh air for the mine workers, but it also dissipates and discharges smoke and gases. The smoke and gases are created by explosives used during initial construction and internal combustion equipment used during construction and operation of the mine. The mine ventilation is sized to meet state laws for mine ventilation.

B.3.2.3.1 Underground Ventilation Systems

Fresh air enters the underground portion of the repository through the two M&M Shafts by the ventilation supply ducts. The total air supply is 28,000 m³/min (1,000,000 cfm) divided as follows:

- 1400 m³/min to the LLW Mine
- 16,000 m³/min to the PWR, CANDU, and Reprocessing Waste Mine
- 10,000 m³/min to the BWR Mine
- 1400 m³/min to mining service areas

The ventilation system in the PWR Shaft has four fans and four intake filter banks; the BWR Shaft ventilation system has three fans and three filter systems. Each ventilation system has been sized so that one of the fans and/or filter systems can be disconnected if necessary. Supply air flow rate is monitored; an alarm system is activated if flow falls below a preset minimum flow rate.

All exhaust from the repository is through the ventilation exhaust shaft. The upper end of this shaft is connected to six filter trains connected in parallel. Each filter train is capable of exhausting 5600 m³/min (200,000 cfm) and contains roughing filter banks and two HEPA filter banks in series. All but one of the filter trains and their fans are normally in operation while the remaining train of filters and fan may be serviced or held in standby. The standby electrical system is capable of providing power to any two exhaust fans at any time. After filtration, the exhaust air is discharged to the atmosphere through a stack.

A schematic diagram of the repository ventilation system is shown in Figure II-B-8. The repository is maintained at a negative pressure relative to the atmosphere.

B.3.2.3.2 Ventilation of Surface Structures

All surface structures handling radioactive wastes are operated at negative pressures with respect to the atmosphere. Ventilation flows are controlled by pressure differential from areas of low contamination potential to areas of successively higher contamination potential. As indicated previously, the exhaust air passes through a roughing filter bank and two HEPA filter banks in series before being discharged through the 110-m (360-ft) mine ventilation stack. Duplicate filter banks and exhaust fans are provided, with both fans connected to the standby power

distribution system. Each structure has a ventilation system independent of other structures. The ventilation stack is continuously monitored to detect any release of radioactivity or noxious gases in the air stream.

FIGURE II-B-8. Schematic Mine Ventilation⁵

B.4 Facility Operation

This section describes facility operating parameters that contribute to the environmental effects determined in the generic environmental impact statements on the Spent Fuel Policy. These include operating manpower, consumable materials, and waste generation.

B.4.1 Operating Manpower

The operating staff at the geologic repository varies over the lifetime of the facility but is dependent upon the projected workload. Manpower requirements are determined for the work

requirements in the year of receipt, whereas the actual work may be accomplished ahead of the receipt time or later. For example, salt mining manpower will probably be expended before spent fuel is received at the repository; and room backfilling will probably be delayed. This method of manpower determination produced peaks and valleys in manpower which will not be encountered in actual practice but on the average produces reasonable manpower requirement. The manpower requirement for each year is summed over the campaign or duration of operation of the repository to produce the total man-years of effort required for operation of the repository.

B.4.1.1 Subsurface Manpower

Subsurface manpower consists of that associated with salt mining and transport of spent fuel or reprocessing waste and its burial.

B.4.1.1.1 Mining and Backfilling

Mining labor is determined from the amount of salt that requires mining. Based upon 0.27 man-hours/ton of salt mined,⁴ one man can mine 5470 tonnes (6020 tons) of salt/man-year. Backfilling labor is assumed to require 10% of the labor required for the initial excavation.

B.4.1.1.2 Spent Fuel or Waste Transport and Burial

Underground transport and burial labor is determined based upon the number of transporters required to transport and bury the spent fuel or waste. These transporters are heavily shielded casks similar to those described in Reference 8. The casks protect the transporter operator and other mine workers from radiation exposure. The environmental effects determined in the GEIS for the Spent Fuel Policy assume that these transporters operate at an average speed of 3.2 kph (2 mph) and require 22 minutes to receive, discharge, and shield the holes containing the spent fuel or waste. Figure II-B-9 (prepared from information given in Reference 8) shows the manpower required for the transporters.

FIGURE II-B-9. Transporter Manpower

B.4.1.2 Surface Manpower

Surface manpower consists of manpower required to operate all surface facilities associated with the geologic repository. The manpower required for these operations is given on Figure II-B-10. This figure is based upon information contained in Reference 5. Additional manpower is required to encapsulate spent fuel in the spent fuel canister. This step is assumed to require three times the manpower shown for canistered waste handling.

B.4.2 Consumable Material

The major consumable material used in the geologic repository that affects the environment is diesel fuel consumed in salt mining and waste handling and coal used to generate electricity and steam used by the repository facilities.

II-72

FIGURE II-B-10. Surface Staff

It has been estimated⁴ that the machines used to mine the salt will burn 1.7 L (0.46 gal) of diesel fuel per tonne of salt mined. This is equivalent to $1.6 \times 10^{-3} \text{ m}^3$ of diesel fuel burned in the mining machines for every tonne of salt mined. This value was used to assess the environmental effects of the geologic repository in the GEIS on the Spent Fuel Policy. Other diesel oil consumption (i.e., consumed by transporters, emergency power supply testing, etc.) is assumed to be much smaller and is not assumed to make a significant contribution to the total consumption.

Coal consumption required to provide the electricity and steam required by the repository is expected to be 7.2×10^4 tonnes/repository-year. The generic facility consumes 25,000 kW of electrical power⁴ (mostly used in ventilating surface and subsurface facilities) and 90,000 kg/hr of steam.⁴ Most of the steam is used to operate the chilled water refrigeration equipment required for the repository.

In the environmental effects analysis, the steam is assumed to be generated in an onsite coal-fired boiler that operates at a 50% efficiency. The electrical power is assumed to be purchased but produced in a coal-fired power plant with a thermal to electrical efficiency of 35%.

The steam and electric power usage was converted to coal consumption by using the conversion methods used by McBride, et al.⁹ in his analyses of coal-fired power plants.

B.4.3. Thermal Releases

The thermal effects at a geologic repository result from radionuclide decay from stored spent fuel or reprocessing waste and from the energy used at the facility.

The decay heat is released to the salt where it flows to other surrounding and overlying formations and eventually reaches the surface where it is dissipated to the atmosphere. The time of release to man's environment will be delayed by several thousand years due to this slow heat transfer path. The radionuclide decay heat release from the facility will continue for many years. The total heat generated by the plutonium contained in the spent fuel results in a significant difference in the total heat generated by a repository storing spent fuel and one storing reprocessing waste.

A thermal balance computer code was written to calculate the heat released from storage of spent fuel. This code was used to calculate the heat released from a repository, ISFS facility, etc., each year and the cumulative amount of heat released. This code was based upon material balance that included the age of the spent fuel and a simple function which approximates heat generation rate of the spent fuel over a range of 4 to 10,000 years.

The function used to describe the heat generation rate is

$$p(t) = \sum_{k=1}^{23} H_k e^{-\lambda_k t} \quad (1)$$

where

k = isotope index (see Table II-B-3 for list of isotopes included)

H_k = heat generation rate of isotope k at time of discharge from reactor, watts/MTU

λ_k = decay constant for k^{th} isotope, yr^{-1}

and t = spent fuel age

This method shows the heat generation of spent fuel agrees with the heat generation calculated from the ORIGEN code (Table II-B-4).

The thermal releases resulting from energy consumption was also determined. These releases result from energy requirements to ventilate the repository and condition the ventilation air and from the fossil fuel consumed in generating these utilities. As indicated in B.4.2 of this section, the generic repository consumes 25,000 kW of electrical power and 90,000 kg/hr of steam. This is equivalent to 25 MW-yr and 70 MW-yr of heat released per year. If, in addition, the steam is assumed to be generated in an onsite coal-fired boiler that operates with a 50% efficiency and the electrical power is purchased but produced in another coal-fired power plant that operates with a thermal to electrical efficiency of 35%, the power plants produce 210 MW-yr of heat for each repository year.

TABLE II-B-3

Parameters for Heat Generation Function - Spent PWR Fuel

	<i>Nuclide</i>	<i>Nuclide Index, k</i>	<i>Decay Constant, $\lambda k, y^{-1}$</i>	<i>Heat Generation Rate at Discharge, H, w/MTU</i>
Light Elements	⁵⁵ Fe	1	2.6655×10^{-1}	2.554
	⁶⁰ Co	2	1.3194×10^{-1}	98.886
Fission Products	⁸⁵ Kr	3	6.4172×10^{-2}	18.564
	⁹⁰ Sr - ⁹⁰ Y	4	2.4659×10^{-2}	564.19
	¹⁰⁶ Ru - ¹⁰⁶ Rh	5	6.9039×10^{-1}	5674.8
	¹²⁵ Sb - ^{125m} Te	6	2.5609×10^{-1}	41.094
	¹³⁴ Cs	7	3.3808×10^{-1}	2575.2
	¹³⁷ Cs - ^{137m} Ba	8	2.3103×10^{-2}	571.95
	¹⁴⁴ Ce - ¹⁴⁴ Pm	9	8.9103×10^{-1}	9564.4
	¹⁴⁷ Pm	10	2.6428×10^{-1}	56.776
	¹⁵¹ Sm	11	7.9671×10^{-3}	2.191
	¹⁵² Eu	12	5.7642×10^{-2}	0.229
	¹⁵⁴ Eu	13	4.3315×10^{-1}	56.287
	¹⁵⁵ Eu	14	3.8275×10^{-1}	6.190
Transuranics	²³⁸ Pu	15	7.8946×10^{-3}	95.540
	²³⁹ Pu	16	2.8419×10^{-5}	5.503
	²⁴⁰ Pu	17	1.0599×10^{-5}	-2.603
	²⁴¹ Pu	18	4.6210×10^{-2}	-100.035
	²⁴² Pu	19	1.7911×10^{-6}	2.180
	²⁴¹ Am	20	1.6008×10^{-3}	122.988
	²⁴³ Am	21	9.4050×10^{-5}	20.539
	²⁴³ Cm	22	2.4755×10^{-2}	2.620
	²⁴⁴ Cm	23	3.8723×10^{-2}	59.504

TABLE II-B-4

Heat Generation Rate of Spent PWR Fuel, Function versus ORIGEN Code

Spent Fuel Age, years	Heat Generation Rate, watts/MTU		Percent Error
	Function ^a	Actual ^b	
3	3758	3782	-0.6
4	2700	2703	-0.1
5	2128	2125	+0.1
6	1798	1798	0.0
8	1457	1460	-0.2
10	1288	1293	-0.4
15	1082	1082	0.0
20	964	964	0.0
30	792	792	0.0
40	661	661	0.0
50	558	557	+0.2
60	476	475	+0.2
80	358	358	0.0
100	281	281	0.0
200	144.9	145.2	-0.2
300	111.4	111.0	+0.4
400	94.1	94.1	0.0
500	81.9	82.0	-0.1
600	72.5	72.5	0.0
700	64.9	64.9	0.0
800	58.6	58.6	0.0
900	53.2	53.2	+0.1
1,000	48.7	48.7	+0.1
1,200	41.6	41.6	-0.1
1,500	34.2	34.2	0.0
2,000	27.3	27.3	0.0
3,000	21.8	21.8	+0.1
6,000	17.1	17.1	±0.1
8,000	15.1	15.1	0.0
10,000	13.4	13.4	0.0
20,000	8.0	8.0	+0.1

^a. As defined by Equation 1 and Table II-B-3.^b. ORIGEN Code values.

REFERENCES FOR SECTION II,B

1. R. L. Bradshaw and W. C. McClain. *Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Waste in Underground Salt Mines*. USAEC Report ORNL-4555, Oak Ridge National Laboratory, Oak Ridge, TN (1971).
2. K. Kuhn and J. Kamstra. "Geologic Isolation of Radioactive Waste in the Federal Republic of Germany and the Respective Programs of the Netherlands," in *Proceedings of the International Symposium on the Management of Waste from LWR Fuel Cycle*, Denver, CO, July 11-16, 1976.
3. *Evaluation of Salt and Mine Rock Disposal*. Y-OWI-SUB-76-16507, prepared by D'Appolonia Consulting Engineers, Inc., Pittsburgh, PA (1976).
4. *Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations*. USERDA Report Y/OWI/TM-35, prepared by Office of Waste Isolation, Oak Ridge, TN (1977).
5. *Waste Isolation Facility Description - Bedded Salt*. Y/OWI/SUB-76-16506, prepared by Parsons, Brinckerhoff, Quade and Douglas, Inc., Office of Waste Management, Oak Ridge, TN (1976).
6. D. W. Kaufman. *Sodium Chloride - The Production and Properties of Salt and Brine*. Reinhold Publishing Corp., New York, NY (1960).
7. A. L. Bock, J. O. Blomeke, W. C. McClain, and B. F. Bottenfield. *Radioactive Waste Repository Project: Technical Status Report for Period Ending September 30, 1971*. USAEC Report ORNL-4751, Oak Ridge National Laboratory, Oak Ridge, TN (1971).
8. *National Waste Terminal Storage Program. Progress Report for Period April 1, 1975, to September 30, 1976*, Office of Waste Isolation, USERDA Report Y/OWI-8, Oak Ridge, TN (1976).
9. J. P. McBride, R. E. Moore, J. P. Witherspoon, and R. E. Blanko. *Radiological Impact of Airborne Effluents of Coal-Fired and Nuclear Power Plants*. USERDA Report ORNL-5315, Oak Ridge National Laboratory, Oak Ridge, TN (August 1977).

C. SPENT FUEL REPROCESSING AND MIXED OXIDE FUEL FABRICATION

In April 1977, the Administration announced a moratorium on commercial reprocessing of spent reactor fuels while studies of ways to avoid proliferation risks are made. In the event a decision is made to go forward with reprocessing, facilities devoted to recovery of fissile material from spent reactor fuel would be constructed in the United States. The Administration also recommended that foreign countries without reprocessing capabilities defer reprocessing until the two-year study on proliferation and nuclear fuel systems (INFEC) is completed. Some foreign countries are already engaged in reprocessing while the decision in other countries with nuclear power capabilities will await the outcome of a reprocessing decision and/or nuclear waste disposal schemes.

In the ongoing assessment programs of alternative fuel cycles with improved proliferation resistance, two basic reprocessing technologies are being assessed for a variety of reactors (i.e., light water cooled, heavy water cooled, high temperature gas cooled and spectral shift). For uranium-plutonium fuel cycles that involve reprocessing, a modification of the Purex process is being evaluated to produce a uranium stream and a combined stream of plutonium and uranium (coprocessing) that also includes a strong gamma emitter (spiking). For thorium fuel cycles, a modification of the Thorex process is being evaluated to produce a stream of uranium-233 denatured with uranium-238. The environmental analysis of these alternative fuel cycles have not been performed, but the proliferation resistance and safeguards features are judged not to increase the environmental impacts over previously analyzed FRP-MOX plants.¹

In this study, calculations of the environmental effects of LWR fuel recycle are based upon the operation of collocated fuel reprocessing-mixed oxide FRP-MOX plants. The conventional Purex process¹ is used in the reprocessing plants to produce separate streams of uranium and plutonium for recycle and waste for disposal. The FRP and MOX plants are assumed to meet the proliferation resistance and safeguards criteria that are current at the time of startup.

Spent fuel reprocessing plants (FRP) have been operated in the United States by the Federal government for over 30 years in connection with the nuclear weapons program. Commercial reprocessing was carried out on a small scale for several years (1966-1971) at West Valley, New York. Reprocessing plants have been built, but never operated, in Morris, Illinois, and Barnwell, South Carolina.

A simplified flow diagram for the mixed oxide fuel cycle is shown in Figure II-C-1. Recovered uranium will be recycled to reactors after re-enrichment by gaseous diffusion, and recovered plutonium will be blended with natural uranium before recycle to reactors. With adoption of plutonium recycle, mixed oxide (MOX) fuel will represent up to 17% of annual plant throughput.

Chemical separation of uranium, plutonium, and wastes is accomplished by liquid solvent extraction techniques in pulse columns,² centrifugal contactors, or mixer-settlers.³ These operations are carried out in heavily shielded structures which are remotely operated. The structures will be designed to protect operating personnel from radiation, to prevent release of radioactive materials to the environment, and to withstand most natural events and process accidents without loss of containment functions. Figure II-C-2 gives a simplified flow diagram of the FRP.

C.1 Design Bases

Reprocessing plants will be designed to withstand design basis earthquakes and other natural phenomena (tornadoes, hurricanes, floods, etc.) as appropriate based upon the location of the plants. Plant auxiliaries include standby diesels for emergency power, fire protection systems, water treatment systems, boilers to produce plant steam, electrical switchgear, and sanitary waste systems. All exhausts from the process are discharged through a 100-m stack after filtration and purification. Primary emphasis in design is given to containment of radioactive materials and to the reduction of release of radioactive materials in liquid and gaseous effluents as low as practical. Structures and equipment are designed to ameliorate the effects of credible accidents of low probability of occurrence such as

- Criticality
- High-level waste concentrator or calciner explosion
- Plutonium concentrator explosion

Other credible accidents (with smaller effects) are

- Solvent fires
- Ion exchange (Pu cycle) fire
- Fuel element drop

FIGURE II-C-1. Flow Diagram for Mixed Oxide Fuel Cycle⁴

FIGURE II-C-2. Simplified Flow Diagram for Reprocessing Plant

Mixed oxide (MOX) fuel fabrication plants are assumed to be collocated with spent fuel reprocessing plants, but in a separate manufacturing facility. Of prime importance in the design of MOX facilities is the confinement to limit release of plutonium. Equipment forms the first level of confinement; the process enclosures (glove boxes, shielded cells, etc.) form the second level of confinement; the manufacturing building forms the third level. The main manufacturing building is designed to withstand earthquakes and other natural phenomena including tornadoes, hurricanes, and floods. The plant complex also includes separate warehouse and administration facilities and field storage tanks for chemicals such as ammonia, nitrogen, and hydrogen and miscellaneous other facilities.

Low probability accidents that could occur in MOX operations are

- Criticality incident
- Fire
- Explosion

Other accidents of higher probability but with low offsite effect are loss of process enclosure window and glove failures. These accidents may result in contamination within the plant, but their offsite effects are very low compared with the accidents listed above.

C.2 Operation at Fuel Reprocessing - MOX Plants

During operation of a reactor, the buildup of fission products and depletion of fissile material requires that one-fourth to one-third of the fuel elements be replaced annually. The discharged fuel elements contain about one-fourth of the ^{235}U that was in the fuel before irradiation, plus any unburned plutonium that was used to enrich the fuel before irradiation or that was produced during irradiation. In this generic description, recovered uranium is recycled to the reactor after spiking with enriched uranium from a gaseous diffusion plant, and the recovered plutonium is blended with natural uranium before recycle to reactors.

C.2.1 Fuel Reprocessing

The processing capacity of fuel reprocessing plants in this generic description varies according to the assumptions used in the various fuel handling-processing options. With the adoption of plutonium recycle through power reactors, mixed oxide (MOX) fuel will represent up to 17% of the fuel recycled through the FRP. The processes at the FRP include (Figure II-C-2):

- Spent fuel receiving and storage
- Fuel element chopping
- Dissolution of fuel in nitric acid
- Purex solvent extraction and partitioning
- UF_6 production from uranyl nitrate by thermal decomposition and anhydrous processing
- High-level liquid waste storage and solidification

Spent fuel assemblies are received at the reprocessing plant by truck or rail in heavily shielded shipping casks. The assemblies are spaced arrays of sealed tubular rods containing UO_2 or MOX pellets. The uranium and plutonium have been partially transformed by irradiation into heavier radionuclides, including americium and curium, and into lighter fission products, including noble gases. The tubes encapsulating the fuel are normally made of Zircaloy, although stainless steel has also been used.

As a first step in reprocessing, fuel elements are chopped into short pieces to expose the oxide. The oxide is leached by hot nitric acid, and the tubing pieces (hulls) are soaked in hot nitric acid and water to dissolve uranium, transuranics, and fission products. After this process, the hulls, which still contain some radioactivity resulting from activation of the Zircaloy and from residual fuel are processed as waste. After adjustments of the nitric acid concentration, the leach solution is processed through solvent extraction and/or ion exchange systems. The solvent extraction step may be carried out in pulse columns,² centrifugal contactors, or mixer-settlers.³ Ion exchange equipment may consist of agitated bed or fixed bed columns of ion exchange resin.⁴ These process steps separate the fission products, uranium, and plutonium (Figure II-C-2).

The purified uranium product is converted to uranium hexafluoride (UF_6) and is shipped to the gaseous diffusion plant for re-enrichment. The process used for converting⁵ uranyl nitrate to UF_6 is described in a later section.

The purified plutonium product is transferred to the MOX fuel fabrication plant for conversion to plutonium oxide (PuO_2). The most probable method for converting plutonium nitrate to plutonium dioxide is oxalate precipitation and calcination (described later).

Waste management operations are expected to result in retention of the high-level wastes with almost no release to the environment. Typically, a concentrated liquid acid waste from the reprocessing plant is held in cooled multibarrier stainless steel tanks on the reprocessing site for a year or more to allow short-lived fission products to decay; the solution is then processed further. Federation regulations⁶ require that the high-level liquids must be converted to solids within five years after their generation and the solids must be transferred to a Federal repository within ten years after their generation.

Reprocessing plants will provide for temporary storage of solid waste materials (hulls and fuel hardware) onsite before shipment to a Federal repository. Other low-specific-activity solid wastes are packaged and may be stored onsite or shipped to commercial burial grounds for long-term storage. Wastes containing transuranics in concentrations exceeding 10 nCi/g will be stored in the reprocessing plant until a Federal repository is available for long-term storage.

Personnel exposure to radiation is controlled by shielding for normal operations and by use of radiation work procedures for maintenance operations. Contamination of air in personnel-inhabited areas is minimized by controlled air flow ventilation from areas of low (or no) contamination to ones of progressively higher contamination.

C.2.2 Production of Mixed Oxide Fuels for LWR Reactors

Recycle of plutonium in light-water-cooled reactors requires production of a mixed uranium dioxide-plutonium dioxide fuel. Mixed oxide (MOX) fuel for light-water-cooled reactors consists of ceramic pellets (<5% PuO₂ with natural UO₂) in Zircaloy tubes. Mixed oxide fuel may be used for forming separate fuel assemblies (PWR concept), or islands of mixed oxide rods surrounded by enriched UO₂ rods in a single assembly (BWR concept). With either concept, other assemblies in the core may be made entirely of enriched UO₂ rods.

MOX fuel elements will be fabricated in a fuel fabrication plant collocated with the reprocessing plant. Processes in the MOX plant include:

- Preparing oxide powder
- Mechanical blending of UO₂ and PuO₂ powders
- Pelletizing, sintering, and grinding of the mixed oxide
- Recovering scrap
- Treating waste to recover plutonium and to prepare the waste for disposal

The UO_2 and PuO_2 powders are mechanically blended and processed to fuel rods in the following steps:

- Blending feed powders for the process powders (these may include PuO_2 , $\text{PuO}_2\text{-UO}_2$, or recycled scrap).
- Pretreating mixed UO_2 and PuO_2 powders by comminution, compaction, and granulation to desired consistency.
- Pelletizing
- Sintering of the pellets
- Grinding to finished dimensions
- Cleaning and drying the pellets
- Loading the pellets into fuel rods, decontaminating the rod ends, and welding the end caps

Multiple levels of confinement are used to limit the release of plutonium from the MOX manufacturing building. The manufacturing building is maintained at negative pressure relative to the outside. Plutonium-handling operations are carried out inside equipment located within process enclosures (glove boxes or shielded cells) maintained at negative pressures relative to the adjacent areas of the manufacturing building. The pressure differentials are maintained so that air flows from noncontaminated areas into areas of potentially higher contamination levels, thus limiting the spread of radioactivity. As in the FRP, equipment forms the first level of plutonium confinement; the process enclosures form the second level of confinement; the manufacturing building forms the third level. To ensure that the required pressure differentials are maintained, spare filter banks, ventilation fans, and controls are provided; and independent emergency power systems are activated automatically in the event of loss of normal plant power.

C.2.3 Waste Treatment and Interim Storage

Wastes containing radioactive isotopes are generated in all segments of the nuclear fuel cycle. Radioisotopes in the wastes from the steps of the uranium feed chain and from the fuel fabrication operations for enriched uranium are primarily those associated with natural uranium. Recycle of uranium increases the ^{236}U content. Radioisotopes in wastes from the mixed oxide fuel fabrication step, from the reactor, and from the reprocessing operations contain man-made isotopes (plutonium and other actinide elements, fission products, and activation products). Most of the radioactive wastes from backend operations are generated in the reprocessing of spent fuels to recover plutonium and uranium.

The high-level waste from the reprocessing of spent fuel is the most significant radioactive waste from the standpoint of potential hazard and difficulty of handling. This waste contains almost all of the fission products, transplutonium elements (actinides of atomic number greater than 94), neptunium, and a small fraction (about 1/2%) of the uranium and plutonium that was initially in the spent fuels. This high-level waste generates substantial heat (3 to 4 kW/MT of LWR fuel processed, 10 years after discharge) and requires cooling. It emits large amounts of potentially hazardous ionizing radiation, and it must be isolated or contained for thousands of years to ensure that significant quantities of the toxic radionuclides do not enter man's environment.

Other wastes that are assumed to be stored are tritium, ^{14}C , ^{129}I , and the radioactive noble gases removed from the gaseous effluents from the fuel reprocessing plant. Currently there are no regulations requiring these wastes to be stored in a repository, but the GEISs on the Spent Fuel Policy assume that wastes will be stored in the geologic repositories.

C.2.3.1 Liquid Waste

After reprocessing, the high-level waste is stored in its acidic liquid form in cooled stainless steel tanks until it is converted to a solid form for longer term storage. Present regulations limit the storage of these liquid wastes to a period of 5 years.

Although storage of waste in tanks is considered an interim measure, waste can be stored in this manner indefinitely if surveillance is continued, with transfer to new tanks if necessary. To date (over 20 years experience), corrosion of stainless steel waste tanks has been negligible; and the lifetime of these tanks have not yet been determined as no leaks have occurred.⁷

In storage systems for acidic liquid wastes, the waste is contained by multiple barriers designed for minimum dependence on people or surveillance devices. However, procedural controls are also provided for continuous monitoring of the storage systems. Required functions such as cooling and surveillance are ensured by backup systems so that acceptable conditions are maintained at all times.

If waste were to leak through the primary stainless steel tank, it would be collected in a secondary container from which it could be transferred to another tank. Retrieval of waste from tank storage is simplified by minimizing the volume of insoluble solids and/or providing agitation in the tank to keep solids from settling.

The function of solidification processing is to convert high-level wastes to a stable solid form. In the process, waste volume may be decreased by a factor of 5 to 10, and secondary wastes may be produced. A facility is required for waste solidification and for treatment of secondary wastes.

Solid waste forms resulting from conversion of high-level wastes will probably be encased in sealed metal canisters for interim storage or transportation to a Federal repository.

Solidification processes that are ready for demonstration produce either a dry powder or granular material (calcine), or a coherent vitrified material (glass). Calcine could also be an intermediate form in the future conversion to glass. Current regulations do not define the type of solidification processes that should be for reprocessing waste. The environmental effects determined in the GEISs on Spent Fuel Policy assume the waste form is the vitrified material.

Temporary storage of solidified high-level waste could consist of either open bins or sealed canisters stored retrievably in a secondary containment structure on the reprocessing plant-site. Solidified high-level waste will be transported to a Federal repository for long-term storage.

C.2.3.2 Solid Waste

Solid radioactive waste from a fuel recycle complex is classified in three categories for processing: 1) miscellaneous waste which includes small items of waste from production and laboratory operations, such as protective clothing, glassware, plastic bags and sheets, decontamination residues, etc.; 2) Zircaloy fuel cladding hulls and stainless steel and *Inconel** fuel assembly components which contain the largest amount of solid waste radioactivity, and 3) obsolete and discarded process equipment, such as vessels, piping, cabinets, and glove boxes.

There are no current regulations that define the form the solid waste from an FRP-MOX must be in before shipment to a disposition facility. In an effort to provide a means of evaluating the environmental effects in the GEISs on the Spent Fuel Policy, the following assumptions were made. Miscellaneous waste will be separated initially in two fractions: those with TRU contents >10 nCi/g and those with TRU contents <10 nCi/g.

- Waste containing >10 nCi/g will be sorted according to combustibility. Combustible TRU waste (>10 nCi/g) will be incinerated to reduce its volume and fixed in a stable matrix. Noncombustible TRU waste (>10 nCi/g) will be doubly encapsulated in durable containers. Both waste types will be shipped offsite to a repository for long-term storage.

* Trademark of Huntington Alloys, Inc.

- Wastes with <10 nCi/g will be packaged and shipped to a commercial burial ground.

Obsolete and damaged process equipment, with TRU nuclides >10 nCi/g, is assumed to be disassembled and decontaminated. Equipment decontaminated to <10 nCi/g will be packaged and stored in a commercial burial ground. Equipment that remains contaminated to >10 nCi/g will be reduced in size and packaged in durable containers for shipment to an offsite repository.

Degraded tributyl phosphate-hydrocarbon solvent will be burned in the TRU waste incinerator. The residue after incineration will be fixed in a stable matrix together with the other ash and doubly encapsulated for shipment offsite.

C.2.4 Storage of Plutonium

During recycle, about two years will elapse between the time plutonium is discharged from a reactor to the time it is reinserted as a mixed oxide assembly. Over much of this period, the plutonium must be stored to await processing, transportation, or recharge. This storage capacity will be provided at the reprocessing and fuel fabrication facilities. If the plutonium is not recycled but is stored for possible future use or eventual disposal, a separate, licensed storage facility will be needed.

C.3 Facility Description

The generic reprocessing and mixed oxide fuel fabrication plants are assumed to be located on the same site, within a fenced area of about 2430 ha (6000 acres).⁸ This provides a 2.4 km (1.5 mi) buffer zone surrounding the plants. A sample layout of the site is shown in Figure II-C-3. The principal operating components of the site are

- Fuel Receiving and Storage Facility
- Fuel Reprocessing Plant
- Uranium Conversion Facility (UF_6 Plant)
- Plutonium Conversion and Mixed Oxide Fuel Fabrication Facility
- Waste Handling and Storage Facility
- Ventilation Filtration Facility and Exhaust Stack

Legend

① Fuel Reprocessing Plant	⑤ Waste Storage Facility
② MOX Fuel Fabrication Plant	⑥ Waste Handling Facility
③ Fuel Receiving Facility	⑦ UF_6 Plant
④ Fuel Storage Facility	⑧ Ventilation Filtration and Stack

FIGURE II-C-3. Plot Plan for Collocated Fuel Reprocessing and MOX Plant

Auxiliary facilities, not shown in Figure II-C-3, are provided for such functions as water and utility supplies, storage of essential materials, treatment and storage of industrial type wastes, maintenance shops, administration offices, etc.

C.3.1 Fuel Receiving and Storage

Fuel reprocessing requires a substantial inventory of spent fuel. The fuel is received in heavily shielded railroad or truck casks. At the fuel receiving and storage facility, casks are inspected, cooled, emptied, decontaminated, and reshipped. The fuel is unloaded underwater to provide both cooling and radiation shielding and is transferred to storage where it is cooled, protected, and, as required, transferred to the fuel reprocessing facility. Fuel is stored in water-filled stainless-steel-lined, reinforced concrete basins. Basins are provided in modular units, each with a storage capacity of 500 MTU.

The facility is designed to preclude the following as credible accidents:

- Criticality
- Exposure of stored fuel through loss of shielding water
- Dropping of heavy loads on fuel
- Multiple massive ruptures of fuel elements by missiles
- Complete loss of cooling water
- Damage to critical equipment by the cask

In addition, the facility:

- Has airlocks and a controlled leakage of air into the facility
- Has a standby filter system for air flowing through the facility in the event of an airborne release of radioactivity
- Is designed so that each 500 MTU pool can be isolated from remaining pools to permit drainage.

The ventilation system for the fuel recovery and storage facility consists of deep-bed fiberglass roughing filters for intake of air (90% efficiency for 0.3 micron-sized particles) and exhaust through HEPA filters to the facility stack.

Pool water is recirculated (in a closed system) through filters and deionizers designed to maintain pool activity at or below 2×10^{-4} $\mu\text{Ci/mL}$. Heat exchangers dissipate decay heat from fuel through cooling water, which in turn, is cooled in cooling towers.

Liquid and solid wastes generated at the fuel receiving and storage facility are stored for disposal with wastes generated in the reprocessing facility.

C.3.2 Fuel Reprocessing Plant

The fuel reprocessing plant is a Category 1, reinforced-concrete structure built on a reinforced-concrete foundation. These structures are proportioned to ensure that they will behave in a ductile manner; that is, under the most severe conditions, the steel reinforcement will yield before the concrete crushes, and the structure will undergo large deformation loads without failure. The major components of an FRP are

- Mechanical cell
- Solvent extraction cells
- Recycle cell
- Waste evaporation cell

This facility releases no liquid effluents of process origin to the environment. Liquid products, waste streams, and process streams are sent to other onsite facilities. Gaseous effluents are scrubbed and filtered before release through the process stack.

A flow diagram for operations in a fuel reprocessing plant is shown in Figure II-C-2. The process begins with receipt of fuel elements in the mechanical cell from the fuel receiving and storage facility. Operations in the mechanical cell are

- Chopping fuel into short lengths
- Leaching the fuel core material with nitric acid (dissolving)
- Preparing the undissolved fuel cladding for disposal (hull handling)
- Preparing the leach solution for solvent extraction
- Separating the uranium and plutonium from the bulk of the fission products by solvent extraction

Before fuel is moved to the mechanical cell, one end of the fuel assembly is removed by chopping. The chopped fuel element is moved, one assembly at a time, to the mechanical cell. Fuel elements are positioned, blown dry of surface moisture, and fed into shears. The shears chop the fuel into 1- to 3-inch lengths. Chopped pieces of fuel fall by gravity through a chute into a dissolver. End fittings are withdrawn from the shear machine and are packaged for cask transfer to the solid waste handling facility. Nitric acid is used to dissolve the fuel core material from the sheared fuel pieces. During dissolving, air is sparged through the dissolver to remove the fission product gases iodine, krypton, and xenon from the dissolver solution. The fission product gases are transferred to the dissolver off-gas treatment system along with NO_x fumes evolved during dissolving. Any remaining fuel traces are dissolved from the cladding hulls by a nitric acid leach cycle followed by a rinse.

Fuel cladding hulls are removed from the dissolver in the perforated dissolver basket, monitored to detect any remaining fuel, dipped into sodium hydroxide to passivate the reactive surface of the Zircaloy hulls, and dried to prevent Zircaloy fires in the hulls during transfer to the solid waste facility.

Uranium solution from the dissolver is collected in a combination receipt-feed adjustment tank. In this tank, nitric acid concentration is adjusted; and the valence of plutonium is adjusted by chemical addition. The solution is then clarified by removing fine particles by centrifugation. The clarified solution is then pumped to the first cycle feed tank. This cycle effects the primary decontamination of uranium and plutonium and separates uranium from plutonium by using a solvent extraction technique. The aqueous stream from this first cycle, containing most of the fission products, is transferred to the waste evaporation cell. The organic phase, containing uranium and plutonium, is scrubbed with nitric acid and is then fed to a pulse column. Valence adjustment causes extraction of plutonium in the aqueous phase and uranium in the organic phase. The plutonium stream is pumped to the plutonium recycle cell (Cell 1) for further purification. The uranium is stripped from the organic phase into very dilute nitric acid stripping solution in another pulse column. This nitric acid solution of uranium is pumped to the uranium recycle cell (Cell 2) for further purification.

The aqueous plutonium stream from the first extraction cycle undergoes two additional solvent extraction cycles in Cell 1. The purpose of these additional cycles is to decontaminate the plutonium of fission products, iron, and other inorganic contaminants. The purified plutonium product stream is transferred to the plutonium conversion facility located in the MOX Fuel Fabrication Facility.

The uranium from the first extraction cycle is transferred to the uranium recycle cell (Cell 2) for further purification in two additional solvent extraction cycles. These cycles decontaminate the uranium of fission products, plutonium, and other impurities. The purified uranium solution from solvent extraction is transferred to the uranium finishing facility for conversion to UF_6 .

The fuel reprocessing plant generates a variety of liquid waste streams containing varying quantities of radioactive constituents. Almost all the liquid wastes are concentrated by evaporation to minimize waste handling and storage requirements. The condensates from these evaporations are processed further to recover nitric acid and to produce water suitable for reuse in the process or for disposal to the environment. Liquid wastes are segregated into three categories before evaporation according to radioactivity content.

- High-level liquid waste
- Intermediate-level liquid waste
- Low-level liquid waste

High-level liquid wastes consist of liquid wastes from head end, the raffinate from first cycle extraction systems, and off-gas scrubber solution from the high-level waste solidification facility. Concentrated wastes are stored in high-level waste tanks before being transferred to the waste solidification facility. Intermediate-level liquid wastes consist of a wide variety of waste types (aqueous wastes from the uranium and plutonium cycles, overheads from the uranium and plutonium evaporators in solvent extraction, decontamination solutions, laboratory wastes, laundry wastes, etc.) which contain no recoverable uranium or plutonium. The concentrated bottoms from intermediate-level evaporators are combined and sent to the intermediate-level solidification unit in the waste solidification facility. Low-level liquid waste consists of overheads from several intermediate-level evaporators. Concentrated bottoms are returned as feed to intermediate-level evaporators and overheads are reused as process water or are superheated and discharged to the exhaust stack.

Three types of ventilation systems are in the fuel reprocessing facility (Figure II-C-4). The building ventilation system provides air flow through the building and maintains pressure differentials between areas to confine radioactive materials within designated process areas. The process off-gas system provides an air flow through process equipment. Process off-gas joins the building ventilation air at the 100-m exhaust stack for discharge to the atmosphere. The dissolver off-gas system vents gas produced during fuel dissolution.

II-94

FIGURE II-C-4. Ventilation System for Fuel Reprocessing Plant

Inlet air for the building ventilation system is filtered, conditioned, and supplied to personnel areas of the reprocessing building such as the control room and offices. Air circulates through the building areas in the direction of increasing potential for release of radioactivity. Air flows into process cells through back-flow dampers and filters that prevent air reversals. Pressure differentials ensure that air flows follow prescribed paths. Radioactive particulates are removed from ventilation air by high-efficiency filters on the discharge vents of each cell.

The vessel off-gas system consists of a condenser, an NO_x absorber, and a fiberglass filter. The flow from this system joins with the flow from the dissolver off-gas system, and the combined flows pass through a mercuric-nitrate acid scrubber, a silver impregnated absorption bed, roughing filter, and a pair of HEPA filters, in series.

The dissolver off-gas system consists of a tritium removal step (voloxidation), a condenser, a NO_x -iodine scrubber, a fiberglass filter, an iodine absorber, and a cryogenic Kr-Xe recovery system. Carbon-14 evolved in dissolution is removed as a step in the Kr-Xe recovery. The combined vessel off-gas and dissolver off-gas pass through a sand filter or a deep-bed fibrous filter before being discharged through the 100-m stack. The overall retention of radionuclides in combined FRP-MOX facilities is shown in Table II-C-1.

TABLE II-C-1

Retention Factors^a for Radionuclides in FRP-MOX Plant

<i>Nuclide</i>	<i>Reprocessing Plant</i>	<i>MOX Plant</i>
^3H	100	-
^{14}C	100	-
^{85}Kr	100	-
^{129}I	200	-
Pu	1×10^{10}	4×10^{10}
Am	5×10^9	4×10^{10}
Other Particulates	5×10^9	-

a. Release fraction = 1/retention factor (complete process including effluent treatment and retention in process streams)

The waste handling facility receives both liquid and solid wastes from lag-storage in the reprocessing facilities. The facility sorts and converts these wastes to forms suitable for long-term storage and ships these forms to an offsite repository or to an onsite storage facility. High-level and intermediate-level wastes are processed in the facility. High-level wastes are calcined and converted to borosilicate glass by an incan melting process (Figure II-C-5); intermediate- and low-level liquid wastes are converted to a concrete form. Solid combustible wastes are incinerated, and the ashes are incorporated into a concrete waste form. Spent fuel hulls from the dissolution step are flattened and loaded into a waste canister. Concrete is then poured into the steel containers to fix the the hulls.

C.3.3 Uranium Conversion Facility (UF₆ Plant)

The uranium conversion facility converts purified uranyl nitrate solution to uranium hexafluoride, a form that can be processed through uranium enrichment facilities (Figure II-C-6). In this process, dilute uranyl nitrate is concentrated by evaporation to prepare a feed for UO₃ production by thermal denitration. The concentrated uranyl nitrate is calcined by thermal denitration to UO₃. The UO₃ is then reduced to UO₂ in a reducing atmosphere of disassociated ammonia (H₂ + N₂). The UO₂ is used as feed material for the hydrofluorination area. In this area, uranium dioxide is reacted with hydrofluoric acid gas to produce uranium tetrafluoride. The UF₄ goes through a second step of fluorination with F₂ to product UF₆. The UF₆ is transferred as a liquid into shipping containers for shipment offsite for enrichment.

Process off-gas from the UF₆ plant contains steam, nitrogen oxide, nitrogen dioxide, nitric acid, oxygen, hydrogen, nitrogen, fluorine, HF, NH₃, and uranium. A filter-scrubber system removes HF, fluorine, and uranium before being discharged to the atmosphere. Other components in the off-gas system are 1) condenser and demister, 2) H₂ burner and flame arrestor, and 3) NO_x scrubber unit and acid recovery tower. Treated off-gases are exhausted to the atmosphere through a stack on the UF₆ plant.

C.3.4 Plutonium Conversion Facility

This facility receives plutonium, in solution, as plutonium tetranitrate in nitric acid and produces plutonium oxide for use in the MOX Fuel Fabrication Facility. The product is calcined crystalline PuO₂ powder suitable for blending with UO₂ powder and fabrication into fuel pellets.

FIGURE II-C-5. High-Level Liquid Waste Solidification

FIGURE II-C-6. Flow Diagram for UF_6 Facility

The Plutonium Conversion Facility shares a building and its services with the MOX Facility. Equipment that handles plutonium in any form is in closed, shielded cabinets operating at negative pressure with respect to personnel areas. Critically safe geometry vessels are used where the presence of concentrated plutonium solutions is credible. Shielding is provided for both neutron and gamma radiation.

The process steps in the Plutonium Product Facility are shown in Figure II-C-7. Plutonium nitrate solutions are received from 1) fresh feed from the solvent extraction cell, 2) internal recycle from within the Conversion Facility, and 3) plutonium recycle from the MOX Facility. The solutions are blended and concentrated in an evaporator and then stored in a lag-storage tank. Concentrated plutonium nitrate is pumped to a precipitator, where the valence is adjusted, and plutonium is precipitated with oxalic acid by using two stages of precipitation. The precipitate slurry is vacuum-filtered with stainless steel filter boats and stainless steel filter frits. The filter cake is washed and aspirated to partial dryness. The filter boat containing the plutonium oxalate is transferred to a furnace where the oxalate is dried and calcined to PuO_2 . The calcined PuO_2 is transported to a filter boat dump station, and the oxide is sampled for accountability and quality control. The product is sealed in cans and placed in a storage vault until transfer to the MOX Facility. Empty filter boats are flushed and returned to filter stations.

The Plutonium Conversion Facility ventilation system (Figure II-C-8) provides ventilating air for operating aisle, laboratory areas, processing areas, the storage area, and glove boxes.⁷ The ventilating system is designed to confine radioactive materials within glove boxes which serve as the secondary confinement barrier for plutonium. (Process equipment provides the primary barrier.) The glove boxes are maintained at negative pressure with respect to surrounding areas to minimize releases of radioactive aerosols, fumes, and vapors into occupied areas of the facility. To the maximum extent practicable, the materials of construction for the ventilation system are fire resistant. The ventilation system is capable of operating during a fire to handle the products of combustion safely. The system is designed to withstand design-basis natural phenomena. Filtered air from the conversion facility flows through an underground concrete duct to the centrally located sand filter and is exhausted through a 100-m stack.

Process off-gases from the Conversion Facility consist of vent gases from process vessels, evaporator off-gas, filter boat vacuum system exhaust, and furnace off-gases. These streams are treated, as appropriate by dehumidification or filtration, followed by HEPA filtration. The treated off-gases join the ventilation air before final filtration in the centrally located sand filter and exhaust to the atmosphere through a 100-m stack.

II-100

FIGURE II-C-7. Flow Diagram for Plutonium Product Facility

FIGURE II-C-8. Ventilation and Off-Gas System for Plutonium Product Facility

C.3.5 Mixed Oxide (MOX) Fuel Fabrication Facility

The purpose of the Mixed Oxide Fuel Fabrication Facility is to fabricate fuel rods from uranium and plutonium oxides for fuel assemblies for use in power reactors. A simplified flowsheet is given in Figure II-C-9. The process involves the following:

- Blending of UO_2 and PuO_2 powders
- Pressing the blend into wafers
- Grinding the wafers into granules
- Pressing the granules into pellets
- Sintering the pellets
- Grinding the pellets to the correct diameter
- Loading the pellets into cladding rods
- Finishing the fuel rods
- Assembling and inspecting rods
- Storing and shipping finished assemblies.

In the blending operation, UO_2 , PuO_2 , and recycled mixed oxides are automatically weighed and discharged into a blender. After blending, the mixed materials are milled to break agglomerates. The milled mixture is then stored in storage silos awaiting the next step in the process.

MOX powder is transported from the storage silos to the wafer press by a vacuum conveyor system as required by production demands. The powder is pressed into short cylindrical wafers. These wafers are discharged onto a conveyor that feeds them to a granulator; the granulator breaks the wafers into granules. The granules pass through a motor-driven screen classification unit to grade the granules into acceptable size particles for the next step. Particles of unacceptable size are recycled to the wafer press.

Granulated powder is fed from a feed hopper into dies in the pellet press. The powder is pressed into pellets of uniform density and size and are moved by a conveyor to a boat loader. Empty boats are loaded with pellets and then placed in lag storage awaiting the next process step.

FIGURE II-C-9. Simplified Flow Diagram for MOX Plant

Loaded pellet boats are automatically moved to sintering furnaces as production demands dictate. The furnace operates continuously at 1700°C (3100°F) in a reducing atmosphere of H₂ and N₂. Boats move continuously by a conveyor through the sintering furnace. After the sintered pellets are inspected and sampled for density measurements, they are placed in lag storage in boats.

Sintered pellets from the furnace storage area are ground to a specified diameter. Ground pellets are inspected, dried, and held for release for fuel rod assembly. Next, fuel tubes, with bottom plugs welded in place, are mechanically loaded with a specified number of pellets. The loaded tube is then decontaminated, and a top end-plug is pressed into place and girth-welded. After inspection and other quality control tests, the loaded tubes are assembled into PWR or BWR fuel assemblies. Completed assemblies are inspected, cleaned, and placed in storage.

The ventilation system of the MOX Fuel Fabrication Facility is similar to the system used for ventilation air in the Plutonium Conversion Facility (Figure II-C-8). These facilities are located in the same building and share a common underground, reinforced concrete exhaust duct which leads to a centrally located sand filter. The ventilation air is exhausted to the atmosphere through a 100-m stack.

C.4 Support Systems⁹

Systems provided to support the FRP-MOX plant under all operating conditions - normal, abnormal, or accident - include the following:

- Ventilation
- Electrical Power
- Compressed Air
- Steam
- Water
- Sewage
- Communications
- Fire Protection

- Maintenance
- Cold Chemical Makeup

The major systems are discussed briefly in the following sections.

C.4.1 Building Ventilation Systems

The ventilation systems are designed to supply properly conditioned air to occupied areas; to ensure that air is confined to prescribed flow paths for confinement and is discharged through a final filter or treatment system and stack; and to ensure that proper monitoring, filtration, and treatment systems are provided. The ventilation systems serve all areas of the facility complex and are designed for continuous service under normal and off-standard operations and accident conditions.

The building structures and ventilation systems provide for confinement of radioactive materials and ensure that personnel exposure is maintained as low as reasonably achievable. The confinement system and barriers limit the movement of radioactive materials to 1) restricted areas so that individuals are not exposed to concentrations of radionuclides in excess of the limits specified in paragraph 20.103 of 20 CFR 20 and 2) unrestricted areas so that releases of radioactive materials are as low as reasonably achievable and do not exceed the annual limit prescribed by paragraph 20.106(a) of 10 CFR Part 20. In the latter case, the confinement systems and barriers are capable of withstanding the effects of design basis accidents, including natural phenomena, and still meet the release criteria.

Air is supplied to the ventilation systems by conventional supply units with integral filters, heating coils, and fans. Air exhausted from operating areas is cleaned and discharged to the atmosphere. Air from areas likely to contain corrosive chemicals is passed through a liquid scrubber. All air from potentially contaminated areas passes through either a roughing filter or a deep-bed fiberglass filter, depending upon the nature of the potential contaminants, plus HEPA filters in series. All components are designed so that they can be tested for operability, efficiency, and functional performance under conditions as close to the design basis as practicable.

Process off-gases are cleaned in engineered treatment systems and combined with ventilation air before the effluent filters. Gaseous effluents are discharged to the atmosphere through stacks which extend to elevations required to provide adequate dispersion. The gaseous effluents are continuously and isokinetically sampled and monitored at the stacks with appropriate recording and alarm devices. Stack condensates are drained into catch tanks where they are sampled and drained to the waste system.

C.4.2 Electrical

The electrical system for the plant supplies electricity for all power, lighting, instrumentation, and communications equipment necessary for operation of the plant during normal and accident conditions. Normal power is purchased offsite and transmitted to the FRP by an electrical substation. Emergency electrical power is generated by diesel-driven emergency generators.

C.4.3 Water Supply

Plant water is supplied from a source that is specific for each site. Water is pumped to a treatment area where it is clarified, filtered, deionized as required, and then pumped to points of usage.

C.4.4 Sewage

Nonradioactive sanitary wastes are treated in a package-type sewage treatment plant of commercial design. Wastes undergo primary and secondary treatment, chlorination, and clarification. Purified liquid effluents from the sewage treatment plant are applied to the land within the exclusion area, and sludge is discharged into sludge pits.

C.4.5 Communications and Alarms

Communication and alarm systems are provided to meet security, safety, and emergency condition requirements. The alarm and communication systems are primarily used for warning operating personnel of hazardous conditions or for requesting assistance from outside agencies in the event of an emergency. In some cases, however, these alarm and communication systems are part of the security system.

C.4.6 Fire Protection

The fire protection system is designed to prevent fires from starting and to minimize damage, particularly to safety-related structures, systems, and components from fires that do occur. Systems that protect Category 1 structures are designed such that system components and structures housing the components would be able to operate during and after a Safe Shutdown Earthquake.

Typical causes of fires that are recognized in the protection system include the following:

- Electrical failures involving equipment and/or wiring

- Thermally hot surfaces, such as heaters, rotating equipment (friction), open flames, etc.
- Pyrophoric metals
- Spontaneous ignition of oil soaked rags, resins, etc.
- Use and handling of flammable materials such as hydrogen, solvents, hydrazine, etc.
- Certain maintenance operations, such as welding and burning, which produce heat
- Use and handling of strong oxidizing agents, such as nitric acid and sodium nitrate
- Thermal heating by radioactive decay

C.4.7 Maintenance

Maintenance is provided by using three basic types of techniques, depending upon the size of equipment and the intensity of the radiation involved.

- 1) Remote maintenance techniques are used to replace large equipment pieces and some of their appurtenances in very high radiation fields. Work is accomplished by the use of hooks and tools suspended from an overhead crane. Personnel entry into the areas is not permitted.
- 2) Semiremote maintenance techniques are used to adjust, repair, or replace small equipment pieces (such as valves, pumps, etc.) in high radiation fields. Work is accomplished by the use of long-handled tools or manipulators working through shielding walls. Personnel entry is normally not permitted but may be possible after extensive decontamination and/or equipment removal.
- 3) Contact maintenance techniques are used in low radiation fields. Work is accomplished by the use of ordinary tools with direct personnel contact.

C.5 Facility Operation

This section describes facility operating parameters that contribute to the environmental effects determined in the generic environmental impact statement on the Spent Fuel Policy. These include operating manpower, consumable materials, and waste generation.

C.5.1 Operating Manpower

The operating staff at a collocated Fuel Reprocessing Plant -MOX Fuel Fabrication Facility will remain relatively constant over the operating life of the facilities and is dependent upon the operating throughput of the plants. Operating manpower requirements for the range of reprocessing rates considered in the generic environmental impact statement on foreign spent fuel storage are given in Figure II-C-10. Radiation workers range from 60 to 65% of the total workers in these facilities.

C.5.2 Consumable Materials

Fossil fuels are used to generate electricity, heat, and process steam for the collocated FRP-MOX plant. Coal is used for electric power operation and generation of steam. Diesel fuel is used for operating emergency power generators. Periodically, these emergency power generators are started and tested to ensure operation in case of emergency. Propane is used to provide process heat for denitration of uranyl nitrate, an intermediate step in the production of UF_6 . Annual fossil fuel consumption for a range of FRP-MOX plant sizes is shown in Figure II-C-11.

C.5.3 Waste Generation

Most of the radioactive waste from fuel reprocessing plants is generated during spent fuel disassembly and separation of uranium, plutonium, and the fission products. The waste, described in this section, considers all waste generated from operations starting with the receipt of spent fuel at the FRP and ending with the packaging for shipment of MOX fuel and uranium hexafluoride. The seven major processing areas considered in the reprocessing complex are shown in Figure II-C-3. Five classes of radioactive wastes generated in these processing areas are described below. Except for fission product krypton, wastes not in the solid form are solidified before disposal.

High-Level Waste (HLW)

High-level wastes generated during the reprocessing of irradiated reactor fuels are assumed to contain essentially all of the nonvolatile fission products, ~0.1% of the iodine, ~0.2% of the uranium and plutonium, and all the other actinides originally contained in the fuel.⁸ These wastes are assumed to be solidified into a monolithic glass which is encapsulated in 3.0-m- (10-ft-) long, 30-cm- (12-in.-) diameter canisters. The glass-to-waste ratios in the vitrified high-level waste are assumed to be adjusted to give volume equivalent to 0.08-m³ (3-ft³)/MTU of reprocessed fuel to meet proposed guidelines on thermal release.¹⁰

FIGURE II-C-10. Collocated FRP-MOX Operating Staff

FIGURE II-C-11. Fossil Fuel Consumption

Cladding Waste

Cladding waste consists of hulls and other hardware components which remain after the fuel is dissolved. The cladding waste is mostly Zircaloy, *Inconel*, and stainless steel.^{1,7,10} In addition to induced radioactivity, the cladding waste is expected to contain approximately 0.05% of the actinides and 0.05% of the nonvolatile fission products of the spent fuel.¹¹ The radioactivity necessitates shielding to protect nearby personnel. The wastes will also contain up to 0.1% of the plutonium originally in the fuel and, therefore, will be TRU wastes.

Several processes are being considered that will immobilize and stabilize these cladding wastes. This analysis assumes that these wastes are fixed by casting in concrete matrix in 3.0-m- (10-ft-) long-, 76-cm- (30-in.-) diameter canisters. The assumed cladding waste volume is $\sim 0.14 \text{ m}^3$ ($\sim 5 \text{ ft}^3$)/MTU of reprocessed fuel.¹¹

Intermediate-Level Waste (ILW)

This category of waste includes all wastes, other than high-level and cladding wastes, which require shielding or other protective action during handling. Fission product tritium, iodine, and carbon-14 that are recovered and fixed are included in this waste. Typically, ILW waste may contain TRU alpha emitters at concentrations $>10 \text{ nCi/g}$ and have surface doses between 10 and 100 mrem/hr. This waste, like the cladding waste, is assumed to be fixed in concrete and to occupy a volume of $\sim 0.7 \text{ m}^3$ ($\sim 25 \text{ ft}^3$)/MTU of reprocessed fuel. It is also assumed to be stored in 3.0-m- (10-ft-) long-, 76-cm- (3-in.-) diameter canisters.

Fission product krypton is recovered during reprocessing and stored in 50-liter gas cylinders at ~ 63 atmospheres pressure. In the compressed state, the krypton occupies about 4 L/MTU of reprocessed fuel.

Low-Level Transuranic Waste (LLW-TRU)

Low-level TRU wastes are defined as those solid materials that contain plutonium or other alpha emitters in concentrations $>10 \text{ nCi/g}$, yet have sufficiently low external radiation levels to allow them to be handled directly without supplementary shielding (surface dose $<10 \text{ mrem/hr}$).¹⁰ Most of this waste is generated during conversion of plutonium to the oxide form and fabrication of MOX fuel.

After incineration of combustibles and disassembly and decontamination where possible of noncombustibles, the LLW-TRU waste is fixed in concrete contained in 210-L (55-gal) drums. The volume of the fixed waste is assumed to be $\sim 2.1 \text{ m}^3$ ($\sim 75 \text{ ft}^3$)/MTU of reprocessed LWR fuel and $\sim 35 \text{ ft}^3$ /MTU of reprocessed CANDU fuel.

Low-Level Wastes (LLW)

Low-level wastes are defined as those solid miscellaneous waste materials that contain beta-gamma emitters and traces of TRU alpha emitters in concentrations $< 10 \text{ nCi/g}$. Most of this waste is generated in the reprocessing facility.

These miscellaneous wastes include general trash and decontamination wastes. Contaminated equipment is disassembled and decontaminated and discarded. Combustible wastes are incinerated, and liquid wastes are evaporated to dryness. Residual combustion ash, evaporation residue and contaminated equipment remaining after decontamination are combined and fixed in concrete in 210-L (55-gal) drums. The volume of the fixed low-level wastes is assumed to be $\sim 1.7 \text{ m}^3$ ($\sim 60 \text{ ft}^3$)/MTU of reprocessed fuel.

REFERENCES FOR SECTION II,C

1. *Generic Environmental Statement, Mixed Oxide Fuel (GESMO)*. NUREG-0002, U. S. Nuclear Regulatory Commission, Washington, DC (August 1976).
2. John F. Flagg, ed. *Chemical Processing of Reactor Fuels*. Academic Press, New York NY (1961).
3. Justin T. Long. *Engineering for Nuclear Fuel Reprocessing*. Gordon and Breach Pub., Inc., New York, NY (1967).
4. *Nuclear Power Growth 1974-2000*, USAEC Report WASH-1139 (74), U. S. Atomic Energy Commission, Office of Planning and Analysis, Washington, DC (1974).
5. *Draft Environmental Statement, UF₆ Plant, Allied General Nuclear Services*. Docket 70-1327, U. S. Atomic Energy Commission, Washington, DC (1974).
6. *U. S. Code of Federal Regulations, Title 10, Part 50, Appendix F*.
7. *Alternatives for Managing Wastes from Reactor and Post-Fission Operations in the LWR Fuel Cycle*. Report ERDA-76-43, Energy Research and Development Administration, Washington, DC (1976).
8. *Light Water Reactor Fuel Reprocessing and Recycling*. Report ERDA-77-75, U. S. Energy Research and Development Administration, Washington, DC (1977).
9. *Nuclear Fuel Recovery and Recycling Center Preliminary Safety Analysis Report*. Docket 50-564, XN-FR-32, Exxon Nuclear Company, Inc., Richland, WA (1976).
10. C. W. Kee, A. G. Croft, and J. O. Blomeke. *Updated Projections of Radioactive Waste to be Generated by the U.S. Nuclear Power Industry*. ERDA Report ORNL-TM-5427, Oak Ridge National Laboratory, Oak Ridge, TN (December 1976).
11. J. O. Blomeke and C. W. Kee. "Projection of Waste to be Generated." CONF-76-0701, *International Symposium of the Management of Waste from LWR Fuel Cycles*, Denver, CO (July 11-16, 1976).

III. TRANSPORTATION

The primary reliance for safety in transporting irradiated nuclear fuels is shipping casks that contain radioactive material. Irradiated nuclear fuel has been safely transported in the United States since the mid-1940s in numerous kinds of shipping casks. The numerous cask designs were developed because of the variety of AEC (ERDA) (DOE) military, research, and commercial nuclear reactors in service. Experience gained in the design and use of these casks, plus comprehensive NRC and Department of Transportation (DOT) regulations,^{1,2} have led to the present generation of LWR shipping casks. These regulations as well as international and state regulations are described in Section III.,A.

All spent LWR fuel transported in the United States and worldwide is currently shipped in heavily shielded casks by truck or rail. Truck casks weigh 23 to 36 MT, (25 to 40 tons) when fully loaded and will normally accommodate one to three PWR fuel assemblies or two to seven BWR fuel assemblies. Currently designed rail casks weigh up to 97 MT (107 tons) when fully loaded and can transport 7 to 12 PWR or 18 to 32 BWR fuel assemblies. Barge shipment has not been used, primarily because of limited suitable waterways and the lack of adequate docking facilities.

In the generic assessment of transportation in environmental impact statements on the Spent Fuel Policy, the spent fuel is assumed to be transported in existing casks designed for spent fuel cooled about 0.5 year. These casks provide more radiation shielding than is required to comply with DOT requirements for transportation operations. When spent fuel that is cooled 5 years is shipped in these casks, the radiation exposure rates from the casks are about 1/6 of the DOT limits.

Future casks may be specifically designed for spent fuel cooled 4 to 5 years or longer, primarily for economic considerations and improved operating efficiency. These casks will probably incorporate less shielding per unit of spent fuel shipped and will have less complicated heat removal systems since they will be required to remove less heat per unit of spent fuel. In addition, an attempt will be made to increase the amount of spent fuel shipped in each shipment. These conditions will result in the radiation exposures that are higher than those estimated in the GEISS on the Spent Fuel Policy, but these exposures will still be within the DOT limits. The increased radiation will be proportional to the capacity and shielding of new casks relative to existing cask designs.

Building of an adequate number of casks of existing design for ISFS startup is estimated to require a lead time of about three years. The lead time to design and build a fleet of casks designed specifically for long cooled fuel will require additional time.

A. FEDERAL, STATE, AND LOCAL AND INTERNATIONAL REGULATORY REQUIREMENTS FOR SPENT FUEL SHIPPING CASKS

Criteria regarding allowable radiation levels, criticality safety, heat dissipation, and release of radioactive materials have been established for spent fuel shipping casks. Cask design must prevent loss or dispersal of spent fuel under normal operating and severe design basis accident conditions.

Overseeing the transportation of radioactive materials in the United States is the responsibility of the Department of Transportation (DOT) and the Nuclear Regulatory Commission (NRC). The interstate Commerce Commission is responsible for economic regulations, but DOT and NRC are primarily responsible for safety impacts regarding shipments of nuclear materials. State requirements are normally auxiliary regulations that pertain to transportation routes and highway load limits or regulations that require additional safety measures.

Regulations issued by the NRC and DOT overlap to some degree. But a memorandum of understanding in 1966 and revised in 1973³ generally delineates the authority of the DOT as setting standards for marking, labeling, shipping safely (radiation levels, temperatures, etc.), regulating shippers and carriers, and approving different packages as suitable for transporting radioactive materials. The authority of the NRC (then AEC) was set forth as reviewing and approving shipping containers for fissile Type B and large quantities of radioactive materials as defined by *Code of Federal Regulations, Title 49, Part 173.393* (49 CFR 173.393).²

The Office of Hazardous Materials of DOT is designated by 49 CFR 173.393 as the competent authority in the United States responsible for administering IAEA transport regulations⁴ for exporting packages containing radioactive materials. DOT requirements for exporting international shipments or moving foreign-made packages in the United States are also listed in 49 CFR 173.393.

For international shipments, the IAEA transport regulations address technical and administrative requirements regarding packaging, labeling, storing, and carrying radioactive materials. The Inter-Governmental Maritime Consultative Organization (IMCO) Code of Practice for Maritime Transport of Nuclear Materials is based upon IAEA transport regulations and is designed to ensure safety to ship, cargo, persons, and the environment.⁵

A.1 Federal Regulations

Major regulations for transporting radioactive materials are set forth in *Code of Federal Regulations, Title 10, Part 20* (10 CFR 20)¹ "Standards for Protection Against Radiation"; Part 71 in (10 CFR 71), "Packaging of Radioactive Materials for Transport and Transportation of Radioactive Material Under Certain Conditions"; and in *Code of Federal Regulations, Title 49, Part 173* (49 CFR 173) "Shippers - General Requirements for Shipment and Packagings."

The packaging and shipping requirements for transporting radioactive materials are a function of the quantity, type, and fissile characteristics of the isotopes being shipped. Radionuclides are classified in one of seven groups according to toxicity and potential hazard during transport. These groups, called "transport groups," are defined in 49 CFR 173.389 and 173.390. Because of the presence of plutonium and other highly toxic isotopes, irradiated nuclear fuel is classified as Transport Group 1, the most restrictive grouping.

In addition to transport group numbers, shipments of fissile material are classified as either Fissile Class I, II, or III, as defined by 49 CFR 173.396. Spent fuel shipments are rated as the most restrictive class - Fissile Class III, which requires special arrangements between the shipper and carrier to ensure nuclear criticality safety during shipment.

A.1.1 NRC Regulations

The general NRC criteria for packaging and shipping radioactive materials are given in 10 CFR 71 Subparts B, C, and D. Because of the large-quantity designation for irradiated fuel shipments, spent fuel casks must also be designed to meet hypothetical accident conditions applied sequentially (10 CFR 71, Appendix B). The hypothetical accidents and the resultant cask response, as set forth, represent a reasonably conservative estimate of actual transportation accident loading inputs and associated response.

10 CFR 71 requires a detailed nuclear safety analysis for a package (cask) to demonstrate that the contents remain in a subcritical configuration during normal and hypothetical accident conditions. Spent fuel cask contents are classified as Fissile Class III shipments, as defined by 49 CFR 173.396. For spent fuel casks, the analysis must show that two casks side by side undergoing the same hypothetical accident conditions and coming to rest in the most reactive geometric arrangement and with close reflection on all sides will still be subcritical (10 CFR 71.40).

Spent fuel casks must be designed to meet hypothetical accident conditions as described in 10 CFR 71 with no loss of containment capability. The risks associated with such accidents have been evaluated, and the chance that physical harm of any significance resulting from radioactivity release or radiation levels that could occur during a rail or truck cask accident is very small.⁶

To show that a cask design will meet the conditions of 10 CFR 71, a detailed safety analysis report for the cask must be submitted to the NRC. This safety analysis contains three basic parts:⁷ 1) general information as a description of the cask, contents, and operational features; 2) technical information, including structural assessment, heat transfer analysis, containment evaluation, shielding, and criticality; and 3) fabrication and operation of the cask, including operating procedures, acceptance tests, maintenance program, and quality assurance. The safety analysis is the principal means for providing information to demonstrate that the cask design meets the requirements of 10 CFR 71. Analyses of cask handling and drop accidents and how they affect the shipping and receiving facilities are covered in the safety analysis reports required for those facilities under 10 CFR 50, "Licensing of Production and Utilization Facilities" or 10 CFR 70, "Spent Nuclear Materials."

NRC regulations for transportation of radioactive materials are also included in 10 CFR 20 and require a receiver to check all casks for transferable contamination within three hours after receipt and to notify the carrier and the NRC immediately if contamination levels above 22,000 dpm/100 cm² are found (10 CFR 20.205). Records must be kept of all such cask surveys and monitorings.

Federal regulations (10 CFR 73.6b) exempt special nuclear materials that are not readily separable from other radioactive materials that have a dose rate >100 rem/hr at a distance of three feet when there is no intervening shielding. Spent fuel and high-level waste shipments are exempted from the physical protection requirements. Spent fuel and high-level wastes are shipped in heavy casks [23 to 97 MT (25 to 107 tons)] and theft of the casks is unlikely. Further, the safety requirements that are intended to enable the casks to withstand severe transportation accidents also protect the cask against acts of sabotage such as small arms, fire, and explosives.

A.1.2 DOT Regulations

DOT regulations for transportation of radioactive materials are given in 49 CFR 170-179. These regulations set the criteria

for radiation levels, surface temperatures, surface contamination levels, bill of lading information, labeling, placarding, shipper certification, accident response, general packaging, and foreign shipments into and from the United States.

DOT regulations specific for different shipping modes are contained in:

- "Carriage by Rail," 49 CFR 174
- "Carriage by Aircraft," 49 CFR 175
- "Carriage by Vessel," 49 CFR 176, and
- "Carriage by Public Highway," 49 CFR 177

Although barge and aircraft shipments of spent fuel are included in the regulations, such shipments are not presently made and are not projected in the environmental impact statements on Spent Fuel Policy.

Spent fuel casks are transported in exclusive-use vehicles and must be filled only by the consignor and opened only by the consignee. Allowable radiation limits (49 CFR 173.393j) for exclusive-use vehicles include the following:

- a. 1000 mrem/hr at three feet from the external surface of the package (cask)
- b. 200 mrem/hr at any point on the external surface of the vehicle (truck or rail car)
- c. 10 mrem/hr at six feet from the external surface of the vehicle
- d. 2 mrem/hr in any normally occupied position in the vehicle.

Because of the large size of the casks used for shipping irradiated fuel, the limiting factor will usually be the radiation level at either three feet from the surface of the cask or six feet from the vehicle.

Although a radiation level of 2 mrem/hr is permitted in a truck cab, the level based upon actual experience is unlikely to exceed 0.2 mrem/hr, because of the distance from the cask and shielding provided by intervening material.⁶

Surface contamination levels must be 220 dis/(min-cm²) or less for beta-gamma (β - γ) radiation, and ≤ 22 dis/(min-cm²) for alpha (α) radiation (49 CFR 173.397). The regulations further

require that in assessing the surface contamination, the average amount of removable contamination (by wiping with an absorbent material) may not exceed 10% of the above limits.

Maximum accessible external surface temperature of a cask transported as an exclusive-use shipment must be 82°C (180°F) or less (49 CFR 173.393e). Heat generated within shipping casks must be dissipated so as not to affect the efficiency of the cask or to damage the contents.

Spent fuel shipments must have shipping papers to identify contents, weight, volume, transport group, list of radionuclides, physical or chemical form, required labels, placards, fissile class, activity, and shipping certificate (49 CFR 172.427).

Regulations require that immediately following any accident, notification must be given to DOT, and a detailed report must be submitted within 15 days (49 CFR 171.15 and 171.16).

A.1.3 ICC Regulations

The Federal agency that has the principal economic regulatory authority over nuclear transportation is the Interstate Commerce Commission (ICC). ICC regulates the rates, charges, and conditions of truck, rail and barge line services operating in interstate and foreign commerce. ICC regulations define three types of carriers: 1) private carriers, which transport their own goods and are exempt from ICC regulation; 2) contract carriers, which selectively transport other people's goods and are subject to limited ICC regulation; and 3) common carriers, which transport goods for the general public in accordance with ICC certificates of public convenience and necessity. Even though transportation safety is primarily the domain of DOT and NRC, some ICC activity may also have safety impacts.

Spent fuel assemblies are transported in casks on legal and overweight trucks and rail systems subject to ICC regulations. The cargo weights and handling aspects of the spent fuel casks are similar to those of equipment used to routinely transport other hazardous materials by truck and rail in routine commerce. Rail transportation of spent fuel and high-level wastes in heavy casks (60 to 90 MT) provides payload to weight factors advantages compared with transportation of smaller casks by truck.

A.1.4 Pending Regulatory Issues

Pending regulatory issues that could affect the establishment of a transportation system for spent fuel and wastes are the following:

- The adequacy of emergency-response planning in the event of accidents.
- Safeguards and security to prevent diversion is of particular importance if the decision is made to reprocess spent fuel.
- Insurance coverage to provide protection against catastrophic losses that might be incurred.
- Labor relations matters pertaining to the transportation industry.

References 8 and 9 discuss these problems and possible solutions.

A.2 State and Local Regulations

State and local regulations on the transport of nuclear materials are normally limited to those requirements for any type of vehicle traffic, such as gross vehicle weight and dimensions. Overweight shipments are sometimes subject to limitations on routing and time of operation. In addition, some states and localities have adopted or are considering regulations on nuclear material shipment to cover the following:¹⁰

- a. Routing restrictions, speed limits, or blanket prohibitions on shipments.
- b. Advance notification of shipments approval by states
- c. Inspection of shipments
- d. Pilot vehicles or escorts
- e. Special training of drivers, additional monitoring personnel, and emergency plans
- f. Requirement of drivers to carry copies of state regulations
- g. Special-use trains
- h. Classification of drivers as radiation workers
- i. Emergency preparedness by state officials

The effect of these restrictions on transportation of spent fuel will vary from state to state and will have to be evaluated on an individual basis. See Reference 8 for additional discussion of existing and proposed state and local legislation.

A.3 International Regulations

Major regulations for transporting radioactive materials internationally are developed and controlled by the IAEA.⁴ These regulations are adopted by almost all international organizations concerned with transportation, and IAEA members use them as the basis of their own regulations.

The IAEA packaging and shipping requirements for transporting radioactive materials are approximately the same as the U.S. Federal regulations and are a function of quantity, type, and fissile characteristics of the radionuclides being shipped. The activity limits for packages specified in IAEA regulations are based upon the radiotoxicity and emitted radiations of the individual radionuclides. The U.S. regulatory approach is the grouping of radionuclides with similar radiotoxicity. Other minor differences exist in the U.S. and IAEA regulations. However, both U.S. and IAEA regulations mutually accept the other's regulations and allow certified packages for shipment both in the U.S. and abroad.

The IAEA regulations, like those of the United States, require shipping packages to be designed to withstand hypothetical accidents with no loss of contents of the package. Testing criteria for these hypothetical accidental conditions differ slightly between the two authorities but overall package designs appear compatible.

The Office of Hazardous Materials (an office under the U.S. Department of Transportation - DOT) ensures that foreign shipments into the U.S. provide the necessary shipping safety. DOT regulations (49 CFR 173.393b²) require the foreign shipper to notify the Office of Hazardous Materials of impending shipments to the U.S. and also to submit a copy of a valid foreign competent authority certificate for the package. If review of these items indicate adequate safety will be ensured, DOT authorizes the shipment.

As a supplement to the above IAEA regulations a code of practice for maritime transport of nuclear materials is included in the Maritime Dangerous Goods Code written by the Inter-Governmental Marine Consultative Organization (IMCO).⁵ The IMCO Maritime Dangerous Goods Code is designed to ensure safety to ship, cargo, persons and the environment. This Code is not legally binding but has been adopted and implemented by the national legislation of many countries.

B. DESCRIPTION OF EXISTING SPENT FUEL CASKS

At present, all spent fuel in the United States is shipped by truck or rail casks. Truck casks are more mobile and much lighter than rail casks. However, truck casks are limited to a loaded weight of about 23 MT (25 tons) because of the maximum 33,300-kg (73,280-pound) load limit (tractor-trailer included) for most highways. Overweight shipments require special permits for most states that will require certain restrictions, such as routings, time of travel, or early notification of shipment. Several transcontinental shipments have been made on an overweight basis where the combined weight of the tractor-trailer and loaded cask was slightly less than 49,000 kg (108,000 pounds). The tractor-trailer combination for the transcontinental shipments was equipped with seven axles to spread the load to comply with axle weight limits in different states.

Existing rail casks weigh 63 to 88 MT (70 to 97 tons) fully loaded but have a capacity of 7 to 12 times greater than legal weight truck casks. The truck casks (which have the average capacity of only 19 kg of heavy metal per metric ton of cask weight compared with 52 kg of heavy metal per metric ton of currently licensed rail casks) have faster turnaround time than rail casks because of greater average transportation speeds and shorter loading and unloading times. Truck casks are needed for those reactors without rail facilities or a nearby rail head. Of the 192 reactors scheduled for operation by 1987, 73% has rail spurs at the site.¹¹

Table III-1 summarizes those casks in the United States that are either presently licensed or known to be in the planning/fabrication stage and are capable of transporting spent LWR fuel. Approximately 12 legal weight truck casks, five overweight truck casks, and six rail casks will be available for transporting spent nuclear fuels in 1979. Future availability of casks is affected by licensing, design time, fabrication problems (lead casting or uranium casting and machining capabilities, procurement times for special items, etc.), quality assurance, capital availability and economic incentive. Fabrication of additional duplicate units of a cask already licensed and with no design changes is estimated to require about 18 months for truck casks and about two years for rail casks.⁸ Designing, testing, evaluating, licensing, and then fabricating the first unit of a new cask may require as much as five years,⁸ dependent upon the complexity or innovativeness of design. An estimated 6 to 8 years would be required to design, test, license, and then fabricate a fleet of newly designed casks.

The goal of NRC is to rule on a license application for transportation containers within 12 months after receipt of the safety analysis report of the proposed cask design.⁷ A complex or innovative design, incomplete SAR or numerous

TABLE III-1

Licensed and Available Shipping Casks for Current Generation LWR Spent Fuel

Cask Designation ^a	Number of Assemblies		Approximate Loaded Cask Weight		Usual Transport Mode	Shielding		Cavity Coolant	Maximum Heat Removal, kW
	PWR	BWR	MT	(Ton)		Gamma	Neutron		
NFS-4 or NAC-1	1	2	23	(25)	Truck	Lead and steel	Borated water and antifreeze	Water	11.5
NLI 1/2	1	2	22	(24)	Truck	Lead, uranium, and steel	Water	Helium	10.6
TN-8	3		36	(40)	Truck ^b	Lead and steel	Borated solid resin	Air	35.5
TN-9		7	35	(38)	Truck ^b	Lead and steel	Borated solid	Air	24.5
IF-300	7	18	63	(70)	Rail ^c	Uranium and steel	Water and antifreeze	Water	76 ^d
NLI 10/24	10	24	88	(97)	Rail	Lead and steel	Water	Helium	97 ^e
TN-12	12	32	97	(107) ^f	Rail	Steel	Borated solid resin	Air	135

a. Cask initials:

NAC = Nuclear Assurance Corp.

NFS = Nuclear Fuel Services, Inc.

NLI = NL Industries (previously National Lead Co.)

TN = Trans Nucleaire

IF = "Irradiated Fuel," symbol used by General Electric Corporation

b. Overweight permit required. The TN-8 and TN-9 casks are also licensed for rail and water shipments.

c. Truck shipment for short distances with overweight permit.

d. Licensed decay heat load is 62 kW.

e. Licensed decay heat load is 70 kW.

f. With a cask body extension for abnormally long fuel, the loaded cask weight is 103 MT (113 tons).

questions by NRC on the design and SAR would likely extend this licensing period. The majority of those casks presently licensed or known to be in the planning/fabrication stage in the United States are discussed in more detail in B.1, B.2, and B.3 of this section. Not included in the discussion are a few other spent fuel casks that are designed for very specific types of fuel and that will not generally handle present-day LWR fuels.

B.1 Truck Casks

NFS-4 (NAC-1). The Nuclear Fuel Services NFS-4 and the Nuclear Assurance Corporation NAC-1 are water-filled 1 PWR/2 BWR* truck casks of about 23 MT (25 tons) loaded weight. Two NFS-4 casks and four NAC-1 casks have been built. Another NAC-1 cask is being built.

The truck cask has an internal cavity 452 cm (178 in.) long by 34 cm (13.4 in.) in diameter, with interchangeable baskets for either PWR or BWR fuel rods. Surrounding the cavity, for gamma shielding and structural strength, is 16.8 cm (6.6 in.) of lead and about 4 cm (1.6 in.) of stainless steel (in several layers). Neutron shielding is provided around the body by 11.4 cm (4.5 in.) of borated water-antifreeze solution. The cask lid is held down with high-strength bolts and sealed with *Teflon*** O-rings. Stainless-steel-encased balsa wood impact limiters are provided around the sides and ends of the cask.

Heat from the contents of the NFS-4 cask is dissipated by convection through the water coolant in the cavity to the inner wall, by conduction to the neutron shield, by convection to the outer wall, and by convection plus thermal radiation to the atmosphere. Maximum heat rejection capacity is 11.5 kW. Maximum design conditions for the inner cavity during normal transport are 174°C at 150 psig. Normal pressure upon receipt is almost always less than 5 psig, however, so the design is quite conservative. Under the maximum fire accident, the cask will withstand 222°C at 948 psig with no loss of containment.

TN-8. The Transnuclear TN-8 is a 36-MT (40-ton) truck cask designed to transport three PWR assemblies in an air atmosphere. This cask will normally travel under overweight restrictions, although several casks could be placed together on a rail car.

* For spent fuel casks, 1 PWR/2 BWR means the cask can accommodate 1 PWR element or 2 BWR elements with the appropriate interchangeable basket.

** Trademark of E. I. du Pont de Nemours and Company.

Two of these casks certified by IAEA are being used in Europe. Two TN-8 casks are being fabricated for use in the United States. The TN-8 cask has an inner cavity length of 427 cm (168 in.) with three separate compartments of 23×23 cm (9.1×9.1 in.) for the individual elements. Surrounding the cavity for gamma shielding is 18.5 cm (7.3 in.) of lead and 6 cm (2.4 in.) of steel (in several layers). Neutron shielding is provided by 15 cm (6 in.) of borated solid resin. Disc-shaped impact limiters constructed of carbon steel and balsa wood are attached to the top and bottom of the cask.

Maximum design heat generation is based upon 35.5 kW. During normal operating, the maximum temperature of the inner shell(s) is 115°C (240°F). Heat rejection is by conduction through the cask body to the outer wall and by convection and thermal radiation from copper cooling fins attached to the outer wall. Pressure is stated to be atmospheric during transport. The cavity design pressure is 110 psig.

A unique feature of the TN-8 cask is a cylindrical shell which is placed around the outside of the cask before it is lowered into a loading or unloading pool. The annular space is filled with clean water during loading. The water and shell, along with other devices to protect the end of the cask, reduces contamination of the cask surface and therefore reduces decontamination time and expense.^{12,13,14}

TN-9. The Transnuclear TN-9 cask is designed for seven BWR rods and weighs 35 MT (38 tons) when loaded. The design features are essentially identical to the TN-8 cask. Trunion location and handling features are identical for the TN-9 and TN-8 casks. Two TN-9 casks certified by IAEA are currently being used in Europe. Three TN-9 casks are being fabricated for use in the United States.

The inner cask cavity of the TN-9 cask is 452 cm (178 in.) long and is divided into seven separate compartments of 23×23 cm (9.1×9.1 in.) for the individual BWR rods. The cask design heat generation is based upon 24.5 kW.¹²⁻¹⁴

NLI 1/2. The National Lead Industries NLI 1/2 cask is a 22-MT (24-ton) helium-filled, 1 PWR/2 BWR truck cask. The cask has a cavity length of 452 cm (178 in.) and a diameter of 34 cm (13.4 in.). An optional 0.6-cm (1/4-in.) thick wall inner container can be used in this cask to contain damaged or failed spent fuel. Use of the liner reduces the cavity diameter to 32 cm (12.6 in.). Shielding around the body is provided by 7 cm of depleted uranium, 5.4 cm of lead, and 3.8 cm of steel (in several layers). Neutron shielding is provided by 12.7 cm of water. Two lids are used to seal the cask at the top. Depleted uranium and steel are used for shielding on the ends.

Maximum design heat rejection for the NLI 1/2 cask is 10.6 kW. Maximum fuel temperature under conditions of normal transport is conservatively estimated at 545°C (1013°F). Normal maximum design pressure is 120 psig when the inner container is used or 22.5 psig when the inner container is not used. Maximum fuel temperature during a fire accident condition is 594°C (1100°F).¹⁵ In 1978, five NLI 1/2 casks will be available in the United States.

B.2 Rail Casks

NLI 10/24. The NL Industries NLI 10/24 cask is an 88-MT (97-ton) helium-filled 10 PWR/24 BWR rail cask. Two casks will be available in 1978.

The cavity is 455 cm (179 in.) long by 114 cm (45 in.) in diameter. The cask has two interchangeable aluminum baskets for use with PWR or BWR fuel. Gamma shielding is provided by 15 cm (6 in.) of lead plus about 5 to 8.6 cm (2 to 3.4 in.) of stainless steel (in several layers). Neutron shielding is provided by 22.8 cm (9 in.) of water. Depleted uranium shielding is used on the cask ends and at strategic locations in the cask wall. Impact structures are used to protect the cask ends and sides. Two closure heads are used. Figure III-1 shows some of the details of the cask and Figure III-2 shows the cask located on a rail car.

Maximum design heat rejection of the NLI 10/24 cask is 97 kW, but all thermal analyses used in the cask licensing were performed for a 70-kW loading. Two auxiliary cooling systems are provided in this cask to circulate water through channels alongside the inner cavity. Auxiliary cooling is not needed for the cask during fire accident conditions. Without auxiliary cooling, and at maximum heat generation rate, the average fuel temperature is 348°C (660°F). Without the cooling system in operation, heat dissipation is by conduction through the body to the neutron shield, by convection to the outer surface, and by convection plus thermal radiation from the finned outer surface to the atmosphere. Maximum fuel temperature during fire accident conditions is 533°C (991°F). Normal cavity pressure during transport is expected to be about 23 psig, with a maximum internal pressure of 105 psig occurring in the fire accident.¹⁶

IF-300. The General Electric IF-300 cask is a 63-MT (70-ton) (when loaded) water-filled rail cask capable of transporting 7 PWR or 18 BWR fuel rods. Four IF-300 casks have been built.

The IF-300 cask is provided with two interchangeable stainless steel baskets, one for each type of fuel. Gamma shielding around the body is provided by a combination of 10 cm (4 in.) depleted uranium clad with stainless steel. Shielding around the cask ends is provided by 7.6 cm (3.0 in.) of depleted uranium clad in stainless steel. Neutron shielding is provided by the water in the cask cavity plus 11.4 cm (4.5 in.) of water contained in an annular stainless steel

III-14

*Drawing Courtesy of
NL Industries Inc.*

FIGURE III-1. NLI 10/24 Irradiated Fuel Shipping Cask

III-15

*Drawing Courtesy of
NL Industries, Inc.*

FIGURE III-2. NLI 10/24 Cask and Carrier

corrugated jacket. The outer wall and ends of the cask are finned for impact protection. Figure III-3 shows some of the details of the cask and Figure III-4 shows the cask and auxiliary cooling equipment located on a carrier.

Maximum design heat rejection for the IF-300 cask is 76.7 kW, but the cask is licensed for 61.5 kW (210,000 BTU/hr) heat load. Cooling is provided by convection to the inner cavity wall; conduction, to the outer neutron shield; and convection, to the corrugated outer wall. Forced air impingement is used to cool the outer shell. During normal operation, the maximum fuel temperature is expected to be 163°C (325°F). If the forced air impingement system is lost, the temperature will rise to a maximum of 239°C (460°F). If water is lost from one of the two neutron shielding water compartments, the water in the remaining compartment will dissipate the heat if cavity coolant does not boil off. In the more remote event that shielding water is lost from both outer compartments, the maximum fuel temperature could reach 788°C (1450°F), but only after all the inner cavity water has boiled off. More than two days have been conservatively estimated to boil off the inner cavity water after shielding water is lost from both outer compartments.¹⁷

TN-12. The Transnuclear TN-12 cask is a rail cask designed to transport 12 PWR or 32 BWR fuel assemblies in an air atmosphere. The design weight of the basic cask is 97 MT (107 tons) loaded; but, with a body extension that accommodates fuel up to 502 cm (198 in.), the loaded cask weighs about 103 MT (113 tons). One cask with a body extension is being fabricated in Europe. A safety analysis report has been submitted to NRC to license the cask for use in the United States.

The basic cask has an inner cavity that is 465 cm (183 in.) long, and the extended cask has a cavity that is 495 cm (195 in.) long. The cask uses a solid stainless steel body for gamma shielding and a borated solid resin for neutron attenuation. Maximum heat rejection is based upon 135 kW. Design pressure is 425 psig.^{12,13}

B.3 Casks for Ships

Rail and truck casks described previously or similar foreign or future licensed casks would be used to transport spent fuel on ships. Because of economic consideration, large rail casks will be preferred for oversea shipment because shipping distances are large. The large rail casks provide the greatest capacity per unit cask weight. If reactor and port facilities are incapable of handling large rail casks, smaller overweight trucks, such as the TN-8 or TN-9 casks,

III-17

FIGURE III-3. GE IF 300 Irradiated Fuel Shipping Cask

Drawing Courtesy of General Electric

FIGURE III-4. GE IF 300 Cask and Carrier

or intermodal casks, such as the IF-300, would be required for transport from other countries to the United States. These different types of casks are transported on or in a wide variety of ships. Rail casks can be mounted on weight-spreading devices on the decks of transport ships. Several transoceanic ships are equipped for roll-on and roll-off facilities which can be used to transport overweight trailers. Shipments of spent fuel have been made across the oceans in holds of dedicated ships provided with load-spreading devices and equipment. These devices and equipment will cool the hold of the ships to provide the comfort of the crew during cargo inspections.¹⁸ Shipments of legal weight truck casks have been made in cargo vessels to a port of call near the destination of the fuel, including about 50 shipments destined for the Receiving Basin for Offsite Fuels at DOE's Savannah River Plant.

C. FUTURE SPENT FUEL CASKS

Existing spent fuel casks are designed for safe shipment of 150- to 180-day cooled spent fuel. The casks have cooling fins and other heat dissipation devices, and in some cases, auxiliary cooling systems (See Figures III-1 through III-4). These complex devices have features that are not necessary for transport of spent fuel cooled 4 years or longer; and they require extra handling and care during loading, unloading, and shipping. The casks meet DOT radiation limits with spent fuel cooled about 0.5 years. When filled with spent fuel cooled 4 years or longer, the radiation levels are at least a factor of 5 less than the DOT radiation limits.

Future casks specifically designed for >4-year cooled spent fuel probably would reduce costs, increase operating efficiency, and decrease turnaround times compared with the casks designed for 0.5-year cooled spent fuel. These casks for long cooled fuel may not require auxiliary heat removal systems and probably would not have fins to radiate heat. The deletion of these heat dissipation features will reduce the cost of fabrication and testing of the casks. A simplified exterior cask structure would decrease loading and unloading times and decontamination efforts. The spent fuel transport capacity of casks, designed for longer cooled spent fuel, may be increased since less heat dissipation would be required. The amount of shielding would probably also be reduced to bring the radiation dose levels outside the casks within allowed radiation levels.

The radiation dose rates, thus population dose commitment, from casks designed for >4-year cooled spent fuels will probably increase relative to those determined if existing casks are used. These increased population dose commitments result from increased capacity and decreased shielding relative to existing casks.

D. SPENT FUEL CASK HANDLING OPERATIONS

D.1 Operations at Storage Basins, Ports, and the Disposition Facilities

Casks must conform to handling requirements at the reactor and independent spent fuel storage (ISFS) basins, shipping docks, and the disposition facility. Spent fuel casks are usually shipped horizontally, and the casks must have lifting features for the cask handling cranes to move them to a vertical position at the unloading basin, at docks, and at the disposition facilities. Also, the cask design should reduce decontamination efforts before the loaded cask is shipped and before the empty cask is returned.

D.2 Transportation Routing

Truck transportation of spent fuel is similar to any non-nuclear transportation of heavy loads, with the exception that some states have extra restrictions, especially for overweight loads. DOT regulations require transport of radioactive materials with no unnecessary delays (49 CFR 177.853). If the truck must be parked for any length of time, warning devices must be placed as specified in 49 CFR 392 in addition to surveillance by the motor vehicle operator or another qualified representative of the motor carrier.

Rail transportation of spent fuel is similar to heavy loads routinely transported, including large loads of hazardous non-radioactive materials. Routes are fixed by rail locations, and urban areas cannot be readily bypassed by alternative routes. Certain routine restrictions may also be imposed by the states or dictated by poor track conditions in some areas.

Ports normally selected for shipping or receiving a transport ship with foreign spent fuel casks will be equipped with necessary cranes and rail or truck facilities to accommodate the spent fuel casks. If crane facilities are not available at the port selected for pickup of casks, the transporting ship must be equipped with the necessary lifting gear to load the cask(s). If foreign fuel or waste from any future reprocessing is returned to its original port by ship, the same requirements would apply. The cranes and lifting gear should be capable of retrieving a cask dropped into the water adjacent to the ship or dock.

Container ships having a port of call near the foreign reactors will be selected to transport foreign fuel to a U. S. port of call. The casks will be unloaded from the deck on rail cars or rolled off and then connected to truck equipment for transport to the receiving facility in the United States.

E. PACKAGES AND CASKS FOR RADIOACTIVE WASTES

Some packages and casks are available for radioactive wastes generated by water basin, transportation, and disposition facility activities; but other casks will also be required to meet this function as the amount of waste increases. If a decision is made to reprocess, additional casks will be required for reprocessing wastes. These packages and casks are described in this section.

E.1 Wastes from Storage Basins

Wastes generated during operation of storage basins (see Section III., D.1) are normally packaged in 210-liter (55-gallon) drums meeting DOT Specification 17C (49 CFR 178.115). Degenerated resins, evaporator bottoms, and other liquid or slurry wastes are mixed in a concrete matrix or equivalent solidifying agent before shipment. Obsolete equipment and other large objects will be placed in DOT Specification 7A (49 CFR 178.350) steel boxes having dimensions of approximately $1.2 \times 1.8 \times 1.8$ m ($4 \times 6 \times 6$ ft.)

Wastes that are classified as non-TRU wastes will be shipped by truck to a land burial site. Commercial trucks and vans with shielding (for example, the Atcor and Hittman vans) are currently in use.

Wastes from storage basins that are classified as LLW-TRU wastes will be transported by truck or rail to a geological repository. The waste will be placed in a Type B overpack if required by DOT regulation 49 CFR 173.

The two largest^o unshielded Type B overpacks available are

- The largest Type B overpack is the ATMX-600 railcar presently in use at DOE's Rocky Flats facility. These cars have a $2.7 \times 2.7 \times 15.2$ -m cargo bay and a useful load of 45,800 kg. Up to two hundred and sixteen 210-liter (55-gallon) drums can be loaded directly into the car. The car can also carry 140 drums which can be preloaded into two standard $2.4 \times 2.4 \times 6.1$ m ($8 \times 8 \times 20$ ft) cargo containers.¹⁹ Steel boxes may also be used as waste containers and loaded on the railcar.
- The "Super Tiger" overpack manufactured by Protective Packaging, Inc., Louisville, KY, is a double steel walled container with 25 cm (10 in.) of polyurethane thermal and shock insulation separating the walls. Interior dimensions are $1.9 \times 1.9 \times 4.4$ m ($6.3 \times 6.3 \times 14.3$ ft), and the outside dimensions conform to standards for $2.4 \times 2.4 \times 6.1$ m ($8 \times 8 \times 20$ ft) cargo containers.

The inner cavity will hold forty-two 210-liter (55-gallon) steel drums. Maximum payload is 13,600 kg; and the containers can be transported by truck, rail, or cargo vessel. This overpack is certified by IAEA.²⁰

Shielded Type B overpacks are available in a variety of sizes and shield thicknesses.

- The NECO B3-1 provides 15 cm (6 in.) of lead shielding and holds a single 210-liter steel drum.
- The Brookhaven Waste Shipment Overpack²¹ is a combination concrete vault and steel overpack. The shielding is comprised of 30 cm (12 in.) of 1800-kg (4000-lb) test reinforced concrete on the inner vault and a minimum of 1.3 cm (0.5 in.) of steel on the steel overpack constructed of steel plate and tubular steel vertical reinforcing. Two sizes for the overpack include
 - a) cavity of $1.2 \times 2.4 \times 1.2$ m ($4 \times 8 \times 4$ ft) and outer dimensions of $1.5 \times 2.7 \times 1.5$ m ($5 \times 9 \times 5$ ft)
 - b) cavity of $1.2 \times 2.4 \times 1.5$ m ($4 \times 8 \times 5$ ft) and outer dimensions of $1.5 \times 2.7 \times 1.8$ m ($5 \times 9 \times 6$ ft)
- Reference 22 includes a partial listing of other available shipping casks that provide varying amounts of shielding.

E.2 Wastes from Reprocessing of Spent Fuel

If a decision is made to reprocess spent fuel, large volumes of solid wastes will require transportation to a domestic or foreign disposition facility. The casks and packages required are briefly summarized from ERDA-77-75.²³

High-Level Waste will be transported in casks weighing less than 100 MT. These casks are expected to hold 1.6 m^3 of solidified waste and are assumed to be transported by rail; no such casks exist but conceptual designs exist. The cask must satisfy Type B packaging criteria and are expected to resemble rail casks for spent fuel.

Cladding Waste will be transported in a rail cask weighing about 65 MT. This rail cask also will resemble a spent fuel cask.

Radioactive Gases that are collected during reprocessing will be packaged in 50-liter cylinders as a compressed gas. No decision on disposition of these pressurized cylinders has been reached. Several options are being considered. Two of these are 1) transport these radioactive cylinders to a geologic

repository for disposal or 2) store the cylinders in an engineered surface storage facility while most of the radionuclides decay. Then release the residual gas to the environment. The first option was selected for analysis in the GEISs on the Spent Fuel Policy. The following assumptions are made. Krypton will be packaged in pressurized 50-liter cylinders, and six will be shipped in a rail cask. Other gases will be chemically fixed in a solid that will be packaged in 210-liter (55-gallon) steel drums and shipped as dry solid waste.

Other Wastes include dry solid wastes and salts of intermediate-level wastes that are incompatible with the high-level liquid waste solidification. Although no regulations specify the waste form of these wastes before transport, the following assumptions were made to determine the environmental effects in the GEISs on Spent Fuel Policy. This analysis assumes that these wastes, often classified according to TRU element content, will be solidified into some suitable solid matrix such as concrete and transported to land burial sites or a disposition facility in packages described in E.1 of this section or in the cladding waste casks.

Plutonium Shipments are assumed to be made only within the FRP-MOX facility because of its collocation. If these facilities are not collocated, plutonium would be transported in a shipping container that is expected to consist of a pressure vessel containing as many as four annular canisters in a special overpack. The overpack would provide radiation shielding and protection against credible accidents.²⁴

F. GENERIC CASKS

In this study, rail and truck casks are assumed to be fabricated as the need and economic justification develop. An inherent factor is that the lag time for fabrication of an additional already licensed cask is about 18 months for a legal weight truck cask and about 24 months or more for a rail cask. The lead time required to design, test, license, and build a fleet of a new generation of casks is 6 to 8 years, depending upon the complexity of the cask. The generic selection of cask types for assessments of this study is described in the following section. The generic casks were selected to develop an upper boundary on the nonradioactive environmental effects.

F.1 Land Transport

The following assumptions were used with GEISs for Spent Fuel Policy for Shipment of spent fuel by land transport.

- Thirty percent (by weight) of the domestic LWR spent fuel shipped from reactors to storage basins is in legal weight truck casks, and 70% (by weight) of the spent fuel is shipped in rail casks. Of the rail shipments, 50% is in IF-300 (or equivalent) type casks; and 50% is in NLI 10/24 (or equivalent) type casks. Overweight truck casks would reduce the miles traveled by truck; therefore, the assumption that all truck shipments are legal weight is conservative. The 30/70 split is the split forecasted between now and the year 2000 and is based upon current and projected reactors shipping capabilities.
- All CANDU fuel is assumed to be shipped in NLI 10/24 (or equivalent) type casks.
- All spent fuel is shipped from the storage basins to the disposition facility by rail in NLI 10/24 (or equivalent) type casks.
- NLI 10/24 (or equivalent) type cask with a modified basket will be used to ship any canned spent fuel. End fittings are assumed to be cut off, and the damaged assembly is canned to fit in the NLI 10/24 cavity.
- Domestic LWR spent fuel is shipped by legal weight truck casks between basins at different sites belonging to the same utility and between basins belonging to a different utility. The assumed shipping is for temporary relief. This spent fuel is assumed to be returned to the originating utility before shipment to disposition.
- Thirty percent (by weight) of the domestic LWR spent fuel is assumed to be shipped directly from reactor basins to the disposition facility in legal weight truck casks, and 70% (by weight) in rail casks. Fifty percent of the spent fuel shipped by rail is in IF-300 (or equivalent) type casks, and 50% is shipped in NLI 10/24 (or equivalent) type casks.
- Packaged non-TRU solid wastes from storage basins and any reprocessing facilities are assumed to be placed in truck vans (13.4 m³/van) or in shielded containers with a capacity of fourteen 210-liter (55-gallon) drums for transport to a land burial site.
- Packaged low-level TRU wastes are assumed to be placed in Type B overpacks for transport to a disposition facility by truck from storage basins or by rail from any future reprocessing facilities.

F.2 Maritime Transportation

NLI 10/24, TN-12, or similar casks are assumed to be used for transporting 80% of foreign LWR spent fuel (by weight) and all CANDU spent fuel to the United States. The other 20% of the foreign LWR spent fuel is assumed to be transported in TN-8, TN-9, or similar casks. Further, rail casks and TN-8 and TN-9 casks are assumed to be transported on regularly scheduled cargo ships that provide roll-on and roll-off or containerized services. For purposes of this study, a cargo ship is assumed to carry four rail casks or four TN-8 or TN-9 (or equivalent) casks per vessel.

If spent fuel is returned to the foreign originator, the above modes are assumed. If wastes are returned after reprocessing, casks for solidified high-level liquid wastes and fuel residues, including cladding hulls, are assumed to weigh about 100 MT. Other wastes are assumed to be placed in a cargo container like the "Super Tiger" or a shielded container similar to that used for land shipments to a disposition facility in the United States.

REFERENCES FOR SECTION III,C

1. *Code of Federal Regulations, Title 10, Part 71 (10 CFR 71)*, U. S. Government Printing Office, Washington, DC (1976).
2. *Code of Federal Regulations, Title 49, Parts 170-179 (49 CFR 170-179)*, U. S. Government Printing Office, Washington, DC (1975).
3. Memorandum of Understanding Between DOT and AEC for Regulation of Safety in the Transportation of Radioactive Materials Under the Jurisdiction of DOT and NRC, 38 F. R. 8466, p. 2028 in *1 CCH Nuc. Reg. Rep.*, Chapter 1009, New York, NY (March 22, 1973).
4. *Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition*, Safety Series No. 6, International Atomic Energy Agency, Vienna (April 1973).
5. T. A. Mensah. "Regulations Laid Down by the International Convention on the Safety of Life at Sea (1960) and the IMCO Code of Practice for Maritime Transport of Nuclear Material." Presented at *Symposium on Maritime Carriage of Nuclear Materials*, Stockholm, June 18-22, 1972.
6. *Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants*. USAEC Report WASH-1238 (NUREG-75/038), Washington, DC (December 1972).
7. C. E. McDonald. "Radioactive Material Package Design Reviews." Presented at *4th International Symposium on Packaging and Transportation of Radioactive Materials*, Miami Beach, FL, September 22-27, 1974.
8. Henry E. Lippek and C. Richard Schuller. *Legal, Institutional and Political Issues in Transportation of Nuclear Materials at the Back End of the Nuclear Fuel Cycle*. Research Report PNL-2457, Battelle Human Affairs Research Centers, Seattle, WA (September 1977).
9. R. E. Best, J. A. Allen, P. A. Aucoin, G. D. Ball, C. C. Hoffman, M. E. Mason, W. A. Propes, and T. A. Vizzini. *Capabilities of U. S. Domestic Transportation Systems for the Shipment of Radioactive Wastes*. USERDA Report Y/OWI/SUB-77/22330. Prepared by the Office of Waste Isolation, Oak Ridge, TN (September 1977).
10. W. A. Brobst. "The State of State Regulations." Presented at *4th International Symposium on the Packaging and Transportation of Radioactive Materials*, Miami Beach, FL, September 22-27, 1974.

11. J. A. Allen, P. A. Aucoin, G. E. Leamon, and J. A. Smith. *U. S. Light Water Reactor Spent Fuel Cask Transportation - Status of Current Capabilities and Limitations, February 1977*. Prepared for USERDA by Nuclear Assurance Corporation, Atlanta, GA (February 1977).
12. B. F. Teer. "Spent Fuel Transportation." Reprint from *Public Utilities Fortnightly*, Transnuclear, Inc., White Plains, NY (August 14, 1975).
13. B. F. Teer. "Transnuclear Spent Fuel Casks." Reprint from *Public Utilities Fortnightly*, Transnuclear, Inc., White Plains, NY (November 1975).
14. *The Nuclear Industry-1974*. USAEC Report WASH-1174-74, Washington, DC (1974).
15. "Safety Analysis Report for NLI 1/2 Truck Cask." Docket No. 11-9010, National Lead Industries, Inc., Wilmington, DE (November 1972).
16. *Safety Analysis Report for the NLI-10/24 Shipping Cask*. Docket No. 71-9023, prepared by NLI Industries, Wilmington, DE (February 1976).
17. *Design and Analysis Report IF-300 Shipping Cask*. Docket No. 701220, General Electric Company, San Jose, CA (January 1971).
18. H. W. Curtis. "UK Experience in the Marine Transport of Irradiated Fuel." Presented at *Symposium on Maritime Carriage of Nuclear Materials*, Stockholm, June 18-22, 1972.
19. F. E. Adcock and J. D. McCarthy. *ATMX-600 Rail Car Safety Analysis Report for Packaging (SARP)*, RFP-2244, Rev. 1, Rockwell International, Golden, CO (February 1977).
20. *Directory of Packagings for Transportation of Radioactive Materials*, p. 251. USAEC Report WASH-1279, USAEC Division of Waste Management and Transportation, Washington, DC (October 1973).
21. *Directory of Packagings for Transportation of Radioactive Materials*, p. 163-5. USAEC Report WASH-1279, USAEC Division of Waste Management and Transportation, Washington, DC (October 1973).

22. *Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle*, Vol. 3, Section 22.7.5. ERDA-76-43, Battelle Pacific Northwest Laboratories, Seattle, WA (May 1976).
23. *Light Water Reactor Fuel Reprocessing and Recycling*. ERDA-77-75, Washington, DC (July 1977).
24. R. E. Best and J. L. Ridihalgh. "The Development, Design and Evaluation of a Packaging for the Transportation of Plutonium." *Proceedings of the Fourth International Symposium on the Packaging and Transportation of Radioactive Materials*, CONF-740901, Miami Beach, FL, September 1974.

IV. ASSUMPTIONS AND MODELS USED TO ASSESS ENVIRONMENTAL EFFECTS FROM STORAGE FACILITIES

A. METEOROLOGY

A.1 Releases From Normal Operations

A meteorological model for calculating dispersion and deposition of radionuclides was developed from data taken from a midwest location in the United States. The meteorological data include mean windspeed and stability categories in 22.5° sectors (Appendix A, Table A-1, DOE-EIS-0015D).¹ Relative concentration (X/Q , sec/m^3) values were calculated for different release heights for each 22.5° sector at distances representing centers of population groups as shown in Reference 1. Methods used for calculating relative concentration were adapted from those described in *Meteorology and Atomic Energy - 1968*.² X/Q as a function of distance averaged over all sectors (not population weighted) is shown in Figure IV-1. The X/Q values were weighted for population distribution to obtain an annual integrated average ($\bar{X}/Q$) for the population living within 80 km of the release point. A deposition velocity of 0.01 m/sec was used to calculate deposition of radioiodine and particulates; however, an undepleted plume (no deposition) was used for calculating the direct pathways (inhalation and submersion) from atmospheric releases of radioactive materials. This conservatism was used instead of calculating dose from resuspension of deposited radioactivity. Annual average relative concentration values for an individual receiving the highest dose at the facility perimeter (at a distance of 0.8 km from the exhaust stack) and for the 80-km-radius population are shown in Table IV-1. Also shown is the fractional deposition (fraction of release) of radioiodine and particulates within an 80-km radius.

Calculations of the effects of deposited nuclides include the assumption that the fractional deposition shown in Table IV-1 occurs within the 80-km radius, and the remainder is deposited in the eastern United States.

FIGURE IV-1. $\bar{X}/Q$ as a Function of Distance Averaged Over All Sectors

TABLE IV-1

Relative Concentration and Deposition Fractions

Release Height, m	X/Q for Maximum Individual, * sec/m^3	$\bar{X}/Q$ for 80-km Population, sec/m^3	Fractional Deposition Within 80 km
0	4.6×10^{-6}	7.7×10^{-9}	0.20
10	2.9×10^{-6}	7.5×10^{-9}	0.19
45	4.9×10^{-7}	5.7×10^{-9}	0.13
110	6.9×10^{-8}	2.8×10^{-9}	0.07
200	3.4×10^{-8}	1.3×10^{-9}	0.03

* At a distance of 0.8 km.

A.2 Accidental Releases

The meteorological dispersing conditions during accidental short-term releases are taken from *U.S. Nuclear Regulatory Guide 1.3*.³ The X/Q values for 0- to 8-hour release times and the closest offsite location (800 m) are $9 \times 10^{-4} \text{ sec}/\text{m}^3$ for a ground-level release, $5 \times 10^{-5} \text{ sec}/\text{m}^3$ for a 45-m release height, and $2 \times 10^{-5} \text{ sec}/\text{m}^3$ for a 110-m release height. Doses from postulated accidents are calculated only for the maximum individual through the inhalation and submersion pathways.

B. LOCAL DOSE

B.1 Maximum Individual Dose

The maximum dose commitment to an individual for one year's releases is calculated by using the models described in Section IV., B.2, "80-Kilometer Radius Population Dose." The models include a meteorological dispersion factor that results in concentrations about twice as high as those for average conditions at the same distance. No other assumptions, such as increased local diet, etc., to optimize the maximum dose were made because of conservatism in the dose models.

B.2 80-Kilometer Radius Population Dose

The population within an 80-km radius surrounding each generic facility site is assumed to average 1,700,000 persons in 1978 (Figure IV-2). The dose commitment from one year's releases is calculated by using the models described under each nuclide or pathway.

FIGURE IV-2. Population Growth with Time

The population dose from each generic facility is also calculated for each year's releases occurring during the period of spent fuel handling and storage. Doses were summed for the years of operation for each facility. The effects of long-lived nuclides persisting in the environment are included for a 100-year period following the study period; details are given in the discussion of specific nuclides. The population surrounding the plants is assumed to increase in proportion to United States growth (Figure IV-2).

B.3 Uptake-Dose Models

B.3.1 Tritium

The local annual whole body dose rate from the release of tritium (assumed to be 100% tritiated water, HTO) is estimated by using the approach given in the model described in ORNL-4992, Chapter Two.⁴ This ORNL model assumes that the water in man's diet reaches the same specific activity as moisture in atmosphere. An average daily intake of 1300 g H₂O in food and 1200 g H₂O in drinking water is assumed; however, the drinking water is assumed to be diluted 100-fold to allow for drinking water sources (groundwater) that would not be in equilibrium with atmospheric moisture. A dose conversion factor for foodstuffs and drinking water pathways of 52 rem/yr per Ci/m³ (H₂O) was used in the calculations. This factor is based upon the same parameters as the atmospheric factor described below.

In addition to the ORNL model for dietary intake of tritium, the dose from inhalation and skin absorption which resulted from exposure to the annual average tritium atmospheric concentrations was calculated. The dose conversion factor used at SRL was modified for use in these calculations and was documented in Reference 5 for the active man breathing rate; this factor was corrected for the normal breathing rate.⁶ The SRL factor incorporates a quality factor of 1.7 and assumes a 12-day biological half-life; a quality factor of 1.0 was assumed for these calculations. The dose conversion factor used was then 1.8×10^9 mrem/yr per Ci/m³.

The dose to the local population from worldwide tritium recycling (described in Section IV., D., "Worldwide Dose") is calculated for the fuel handling and storage period and 100 years thereafter and is added to the doses described above for "first-pass" exposure.

B.3.2 Carbon-14

The local annual dose rates from the release of ^{14}C (assumed to be available to the biosphere as CO_2) is estimated by using the model described in ORNL-4992, Chapter Two.⁵ The ORNL model assumes that the specific activity ($^{14}\text{C}/\text{total C}$) in human tissues is equal to the average steady-state value in the atmosphere. About 99% of the resulting dose is caused by the ingestion (dietary) exposure mode.⁴ The ORNL dose-rate factors for the most significant organ doses are 1.28×10^{12} mrem/yr per Ci/m^3 , total body, and 2.22×10^{12} mrem/yr per Ci/m^3 , red marrow. Local dose from recycled ^{14}C is calculated for the fuel handling and storage period and 100 years thereafter and is added to the doses for "first-pass" exposure.

B.3.3 Krypton-85

NCRP Report No. 44⁷ includes the dose conversion factors for ^{85}Kr exposure used in this statement. These factors are 1.5×10^7 mrem/yr per Ci/m^3 , total body, and 3.1×10^7 mrem/yr per Ci/m^3 , lung, for the most significant organ doses. Skin dose is about 120 times the whole body dose,⁷ but the dose is not included in this assessment because of the very low frequency of severe health effects from skin doses. Local dose from recycled ^{85}Kr is calculated for the fuel handling and storage period and 100 years thereafter and is added to the dose for "first-pass" exposure.

B.3.4 Iodine-129

The uptake-dose model used for ^{129}I calculations was developed by Soldat.⁸ Soldat calculates ^{129}I doses to the thyroids of individuals at 1, 4, and 14 years of age and to adult individuals from dietary pathways. Table IV-2, taken from Reference 8, shows the dietary habits assumed and dose conversion factors for the four age groups. The sum of the adult dose factors (5.94×10^{15} mrem/yr per Ci/m^3) is used to calculate maximum individual and population doses. Buildup of ^{129}I in the soil is assumed to contribute an additional 1% per year to the dose from milk and leafy vegetable pathways (Table IV-2) and is included in the calculation of doses. The fraction of the ^{129}I released (shown in Table IV-1) is assumed to be deposited within the 80-km radii of the fuel handling and storage facilities.

Because of its long half-life (1.6×10^7 years), ^{129}I will persist in the environment and potentially expose the population long after it has been released to the environment. For this assessment, the effects of ^{129}I released through the end of fuel

handling and storage are estimated for an additional 100 years. The EPA in its analysis of the fuel cycle⁹ used this 100-year cutoff. The EPA also recognized that effects from ¹²⁹I releases could impose additional health risks for future generations if the ¹²⁹I remains in the biosphere beyond the 100 years. Although only the fraction of ¹²⁹I available through the soil-root pathway is assumed to contribute to population dose in the 100-year period following release, the man-rem thyroid dose is approximately equal to the dose in the period of continuing ¹²⁹I release.

TABLE IV-2

Thyroid Doses From Unit Concentration of ¹²⁹I in Air⁸

<i>Exposure Pathway</i>	<i>mrem/yr per pCi/m^{3a}</i>			
	<i>1 yr</i>	<i>4 yr</i>	<i>14 yr</i>	<i>Adult</i>
Inhalation ^b	21	11	18	39
Milk Consumption ^a	5800	2400	2000	3100
Leafy Vegetable Consumption ^d	0	510	730	1500
Beef Consumption ^e	0	320	500	1300

- a. Maximum dose rate after equilibrium is reached between thyroid and intake rate. Soil-root pathway would add an additional ~1.3% (per year of accumulation) to the milk pathway and ~1.0% (per year of accumulation) to the vegetable pathway.
- b. Inhalation rates are 5.6, 7.0, 13.5, and 20 m³/day for individuals 1, 4, and 14 years of age and for adult individuals, respectively.
- c. One liter per day of fresh milk from cow grazing on contaminated pasture 12 months per year.
- d. Leafy vegetable consumption was assumed to be 0, 32, 54, and 72 kg/yr for individuals 1, 4, and 14 years of age and for adult individuals, respectively.
- e. Beef consumption was assumed to be 0, 25, 45, and 80 kg/yr for individuals 1, 4, and 14 years of age and for adult individuals, respectively.

B.3.5 Other Nuclides and Pathways

In addition to the dose calculations for ^3H , ^{14}C , ^{85}Kr , and ^{129}I described previously, the contributions of other nuclides released from the fuel handling and storage facilities are calculated with the models and dose conversion factors described in ORNL-4992.⁴

B.3.5.1 External Exposure to Gamma-Emitters Deposited on the Ground

Tables in ORNL-4992, Chapter Three,⁴ give the annual dose rate and the accumulated dose to individuals from continuous deposition of radionuclides at a rate of one $\mu\text{Ci}/(\text{cm}^2\text{-hr})$. The ground is modeled as a plane surface with a uniform distribution of activity. The exposed individual is represented by a point receptor one meter above the ground. The tabulated dose rates and total doses include radioactive decay and the buildup of radioactive daughter products after deposition; no environmental removal processes are included. Conversion factors for obtaining dose rates at the end of a given period and total doses integrated over a given period for the nuclides are given in Table IV-3. These nuclides proved to be significant to man.

The $\bar{X}/Q$ for a depleted plume and a deposition velocity of 0.01 m/sec are used to calculate ground deposition. Exposure to deposited nuclides was assumed to continue for a 100-year period after the study period; radioactive decay was the only removal factor included in the calculations.

B.3.5.2 Inhalation

Exposure to the radioactive releases from fuel handling and storage by the inhalation pathway was calculated for the nuclides not previously considered in nuclide-specific models. The $\bar{X}/Q$ for an undepleted plume was used in this calculation, and resuspension was not included. When the undepleted plume concentration is used, the estimated dose is more conservative than a depleted plume concentration and a resuspension factor. The contribution of long-term resuspension should be evaluated in site-specific environmental assessments. Breathing rate is assumed to be 20 m^3/day . Dose conversion factors, taken from ORNL-4992,⁴ result in 50-year dose commitments. The ORNL models for nuclide uptake, retentions, and distribution in the body are for the most part based upon ICRP models,^{6,10} including the ICRP-2 lung model. Dose factors for the more-significant nuclides in the inhalation pathway are given in Table IV-4.

TABLE IV-3

Dose Rate and Integrated Dose Conversion Factors From Deposited Radionuclides⁴

Nuclide	Dose Rate, ^a mrem/yr per $\mu\text{Ci}/(\text{cm}^2\text{-hr})$		Total Dose ^a mrem per $\mu\text{Ci}/(\text{cm}^2\text{-hr})$	
	1 yr	20 yr	1 yr	20 yr
⁵⁵ Fe	2.67×10^8	1.17×10^9	1.39×10^8	1.90×10^{10}
⁶⁰ Co	2.97×10^{10}	2.24×10^{11}	1.52×10^{10}	3.12×10^{12}
¹²⁹ I	1.16×10^9	2.31×10^{10}	5.78×10^8	2.31×10^{11}
¹³⁴ Cs	1.79×10^{10}	6.2×10^{10}	9.47×10^9	1.07×10^{12}
¹³⁷ Cs	7.24×10^9	1.17×10^{11}	3.63×10^9	1.26×10^{12}
¹⁵⁴ Eu	1.62×10^{10}	1.66×10^{11}	8.19×10^9	2.09×10^{12}

^a. For continuous deposition for the periods shown.

TABLE IV-4

50-yr Dose Conversion Factors for Inhalation of Radionuclides,
rem/ μCi Inhaled⁴

Nuclide	Whole Body	Bone	Lung	Liver	Kidneys
⁵⁵ Fe	0.0009	0.001	0.006	0.004	-
⁶⁰ Co	0.006	-	0.74	0.006	0.001
⁹⁰ Sr	0.222	11.1	1.2	-	-
¹³⁴ Cs	0.056	0.037	0.55	0.1	0.034
¹³⁷ Cs	0.032	0.061	0.45	0.079	0.028
¹⁴⁴ Ce	0.065	1.2	1.0	0.49	0.30
¹⁵⁴ Eu	0.19	3.03	0.93	0.28	1.64
²³⁸ Pu	143.5	5705	188.5	816.4	608.7
²³⁹ Pu	159.2	6559	177.2	896.2	676.4
²⁴⁰ Pu	159.0	6551	177.2	896.2	676.4
²⁴¹ Pu	2.55	124	0.165	6.41	12.1
²⁴¹ Am	135.6	2069	62.6	717.4	1016
²⁴³ Am	132.5	2064	59.4	697.7	999.3
²⁴² Cm	3.38	50.9	40.5	51.9	15.4
²⁴⁴ Cm	74.5	1256	64.7	541.9	347.6

Doses are calculated by the inhalation pathway for the TRU actinides. Soluble forms are assumed in calculating doses to all organs except lung; insoluble forms are assumed in calculating doses to lung (introducing an over-estimation of dose, depending upon the soluble-insoluble fractions).

B.3.5.3 Foodstuffs

The ORNL-4992 models for uptake of radionuclides in man by the dietary pathway are used in this assessment. The models use the TERMOD computer code which estimates uptakes through consumption of milk, beef, and plants contaminated directly by deposited radionuclides and by uptake from the soil. The code is described in ORNL-4992, Chapter Two.⁴

A deposition velocity of 0.01 m/sec was used to calculate deposition. The ORNL model assumes the following foodstuff consumption rate:

Above Surface Food	0.25 kg/day
Milk	1.0 L/day
Beef	0.3 kg/day

Direct deposition on vegetation is assumed to be the more-significant route to man than the soil-root pathway. The TERMOD model also assumes a 4% per year movement from the root zone to an unavailable soil sink. For these reasons, the foodstuff pathway is not included in dose estimates for 100 years after fuel handling and storage, when direct deposition of releases during the study period has ceased. The significance of resuspension as a means of continuing direct deposition, the movement of radionuclides in various soils, and the effect of agricultural practices on this movement could be evaluated in site-specific assessments.

Dose conversion factors taken from ORNL-4992 for nuclides that are significant in the foodstuffs pathway are listed in Table IV-5.

B.3.5.4 Direct External Exposure

Calculations with the ORNL-4992 model and dose conversion factors show that doses from external exposure to the radionuclides dispersed in the atmosphere surrounding the fuel handling and storage sites can be neglected when compared with other dose pathways.

TABLE IV-5

50-yr Dose Commitment Factors for Ingestion of Radionuclides,
rem/ μ Ci Ingested*

<i>Nuclide</i>	<i>Total Body</i>	<i>Bone</i>	<i>Muscle</i>	<i>Liver</i>	<i>Kidney</i>	<i>Spleen</i>
^{55}Fe	0.0003	0.0005	-	0.001	-	0.002
^{60}Co	0.004	-	-	0.002	0.0008	0.001
^{90}Sr	0.166	8.31	-	-	-	-
^{134}Cs	0.075	0.05	0.13	0.14	0.04	0.10
^{137}Cs	0.043	0.08	0.08	0.11	0.04	0.09
^{238}Pu	0.017	0.68	-	0.10	0.07	-
^{239}Pu	0.019	0.79	-	0.11	0.08	-
^{240}Pu	0.019	0.79	-	0.11	0.08	-
^{241}Pu	-	0.015	-	-	-	-
^{241}Am	0.05	0.82	-	0.29	0.41	-
^{242}Cm	-	0.02	-	0.02	-	-
^{244}Cm	0.03	0.50	-	0.02	0.14	-

C. UNITED STATES DOSE

C.1 Atmospheric Concentrations and Population Distribution

Releases from fuel recycle plants are assumed to expose the population east of the generic facility location in the United States in addition to the local populations included in the 80-km radius population. The United States exposure was calculated for the nuclides described in subsequent sections for the releases from the base case during the fuel handling and storage period (1978-2000). Effects of long-lived nuclides were assessed for an additional 100 years (ending in the year 2100). The National Oceanic and Atmospheric Administration (NOAA)¹¹ population-weighted concentration estimated for ^{85}Kr dispersal over the United States (and part of Canada) is 2.5×10^{-10} (person-Ci)/m³ (1970 population for 1 curie annual release from the midwestern United States). This value is used in subsequent sections, together with annual releases, dose conversion factors, and population corrected to the appropriate year (Figure IV-2), to estimate doses to the eastern United States population (assumed to be 80% of total United States population).

C.2 Tritium

The NOAA population-concentration estimate was combined with the ORNL-4992 tritium uptake models and dose conversion factors described for local dose estimates. The ORNL models include the assumption that tritium in foodstuff water and surface drinking water supplies is in equilibrium with tritium in atmospheric water. Global recycling is included in proportion to the fraction of the world population in the United States.

C.3 Carbon-14

The dose to the eastern United States population is calculated as previously described for tritium, by using the NOAA estimates and ORNL-4992 uptake and dose calculational models. The uptake model includes the assumption that the $^{14}\text{C}/\text{C}$ ratio in the body reaches rapid equilibrium with the atmospheric ratio. The United States dose from global recycling is also included.

C.4 Krypton-85

Krypton-85 doses to the eastern United States population are calculated by using the NOAA estimates for "first-pass" exposures and NCRP-44 dose conversion factors. Global recycling is also included.

C.5 Iodine-129

The fraction of the radioiodine not deposited in the 80-km radius is assumed to be deposited outside the 80-km radius zone in the eastern United States. Doses are calculated by combining the NOAA projection with the uptake-dose models proposed by Soldat⁸ and described for the local population dose estimates.

C.6 Other Nuclides

The NOAA projections are combined with the ORNL models and dose conversion factors described in Section IV., B., "Local Dose." The fraction of the particulates not deposited in the 80-km radius is assumed to be distributed over the eastern United States.

D. WORLDWIDE DOSE

D.1 Population Growth

The world population is assumed to grow from a 1970 value of 3.56×10^9 at a rate of 1.9%/yr⁹ (Figure IV-2). Distribution of specific nuclides on a global basis is discussed in the following sections. Dose calculations include the direct effects from the study period and persistent effects for 100 years thereafter.

D.2 Tritium

The worldwide dose due to tritium exposure is estimated by diluting the tritium oxide released into the atmosphere (2×10^{18} m³ air) and the circulating waters¹⁰ (1.4×10^3 H₂O) of the Northern Hemisphere (the circulating waters include the oceans to a 75-m depth). The tritium is assumed to reach equilibrium between atmospheric water and the circulating waters. Dose calculations include inhalation and skin absorption of tritium in the atmospheric moisture. A dose conversion factor of 1.8×10^6 rem/yr per Ci/m³ is used. Also included is the dose estimated by assuming that man's body water reaches the same tritium concentrations as the circulating waters. A dose conversion factor of 52 rem/yr per Ci/m³ (H₂O) is used (see Section IV., B., "Local Dose," for discussion of these factors). The buildup of ³H is calculated by using the input values from DOE-EIS-0015D¹ corrected for radioactive decay. After the fuel handling and storage period, the available tritium is reduced by radioactive decay. The doses are summed over the study period and 100 years thereafter.

The Northern Hemisphere's population (80% of the world population) is assumed to be exposed to the resulting concentrations. The United States population contribution is subtracted and added to the United States dose values.

D.3 Carbon-14

The worldwide dose from ^{14}C is estimated according to the technique used by Hayes and MacMurdo,¹² which was based upon earlier UNSCEAR estimations.¹³ The stratosphere, troposphere, and ocean surface are considered a single reservoir (I) that exchanges with the deep ocean. The removal rate of ^{14}C to the deep ocean is given by the expression¹³

$$0.96 e^{-0.0209 t} + 0.04 e^{-0.000125 t}$$

The buildup of ^{14}C is calculated with the input values from DOE-EIS-0015D¹ and the removal rate given above. Doses were calculated by using the curies available in Reservoir I and the dose conversion factor 1.15×10^{-10} (rem/Ci in Reservoir I) derived from Hayes and MacMurdo.¹² For the 100-year period after the study period, the curies available in Reservoir I was depleted by using the exponential expression given previously, and doses to the steadily increasing population were estimated. The doses for the entire period are then summed to represent the total effects of fuel handling and storage. The Suess effect (dilution of ^{14}C by CO_2 released from fuel-burning operations) is neglected because it is not of sufficient magnitude relative to the uncertainties in the ^{14}C assumptions to warrant inclusion.

The United States population contribution of ^{14}C is subtracted and added to the United States dose values.

D.4 Krypton-85

The worldwide dose from ^{85}Kr is estimated by diluting the releases into the world's atmosphere ($4 \times 10^{18} \text{ m}^3$).⁹ The concentrations are calculated by using release data from DOE-EIS-0015D,¹ corrected for radioactive decay. Doses for the study period (buildup) and 100 years thereafter (decreasing ^{85}Kr dose as a result of radioactive decay) are summed to estimate the effects of the fuel handling and storage facilities.

The fraction of the worldwide dose of ^{85}Kr occurring in the United States is subtracted and added to the United States dose.

E. LOCAL AND REGIONAL DOSE — FOREIGN COUNTRIES

Calculation of local (80-km radius) and regional dose for populations of foreign countries presupposes knowledge of foreign countries involved, population density and distribution, and meteorological dispersion conditions. Also needed is a knowledge of agricultural practices and eating habits of these people. The location of a nuclear facility in a country is also important. For example, a nuclear facility located on the coastline, where prevailing winds carry atmospheric effluents out to sea, causes a much smaller regional dose, whereas, a nuclear facility located where prevailing winds carry atmospheric effluents over large land masses, causes regional population doses varying with area and population density. For example, in the United States, the regional dose involves the population in the eastern half of the country and parts of Canada.

Since foreign countries are not identified in the generic environmental impact statement on storage of spent foreign fuels, the generic environment used to calculate foreign effects is assumed to be the same as that for the United States doses. In this United States generic environment, a population density of 85 people/km² for the 80-km radius population and 50 people/km² (average for Eastern United States) was used for the regional population. This generic environment is used with the knowledge that meteorological conditions, food habits, and population densities vary widely in foreign countries. Table IV-6 shows the wide range of population densities in foreign countries compared with the United States. Specific environmental effects in these foreign countries can be determined, if desired, in a later site specific environmental impact statement.

TABLE IV-6

United States and Foreign Country Population Densities

<i>Location</i>	<i>Population,¹⁴ Thousands</i>	<i>Area, km²</i>	<i>Population Density, People/km²</i>
80-km Radius	1,700	20,106	85
Eastern U.S. ¹⁵	149,639	2,990,183	50
United States(all) ¹⁵	203,235	9,161,161	22
Argentina	25,383	2,766,889	9
Australia	13,502	7,686,848	2
Bangladesh	76,815	143,998	533
Brazil	107,145	8,511,965	13
Burma	31,240	676,552	46
Cambodia	8,110	181,035	45
Canada	22,831	9,976,139	2
China	838,803	9,596,961	87
Denmark	5,059	43,069	117
Egypt	37,233	1,001,449	37
France	52,786	547,026	97
Germany - West	61,832	248,577	249
Germany - East	16,850	108,178	156
Great Britain	55,962	244,046	229
Iceland	218	103,000	2
Ireland	3,127	70,283	44
India	598,097	3,287,590	182
Iraq	11,124	434,924	26
Israel	3,371	20,770	162
Italy	55,810	301,225	185
Japan	110,953	372,313	298
Korea	50,515	220,277	229
Laos	3,303	236,800	14
Norway	4,007	324,219	12
Spain	35,472	504,782	70
Sweden	8,193	449,964	18
Thailand	41,869	514,000	82
USSR	254,382	22,402,200	11
Vietnam	45,211	329,556	137
World	3,967,000	135,830,000	29

REFERENCES FOR SECTION IV

1. *Storage of U.S. Spent Power Reactor Fuel*. USDOE Report DOE-EIS-0015D, U.S. Department of Energy, Washington, DC (May 1978).
2. *Meteorology and Atomic Energy — 1968*. Ed. D. H. Slade. USAEC Report TID-24190, p. 190, Technical Information Center, Oak Ridge, TN (1968).
3. *U. S. Nuclear Regulatory Guide 1.3, Rev. 2*, U.S. Nuclear Regulatory Commission, Division of Document Control, Washington, DC (1974).
4. *A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment*. USERDA Report ORNL-4992, Oak Ridge National Laboratory, Oak Ridge, TN (1976).
5. W. L. Marter. *Environmental Effects of a Tritium Gas Release from the Savannah River Plant on May 2, 1974*. USAEC Report DP-1369, Savannah River Laboratory, E. I. du Pont de Nemours and Co., Aiken, SC (1974).
6. *Report of the International Commission on Radiological Protection*. ICRP Publication 2, International Commission on Radiological Protection, London (1959).
7. *Krypton-85 in the Atmosphere — Accumulation, Biological Significance and Control Technology*. NCRP Report No. 44, National Council on Radiation Protection and Measurements, Washington, DC (1975).
8. J. K. Soldat. "Radiation Doses from ^{129}I in the Environment." *Health Physics* 30, 61 (1976).
9. *Environmental Analysis of the Uranium Fuel Cycle, Part III — Nuclear Fuel Reprocessing*. Report EPA-520/9-73-003D, Environmental Protection Agency, Washington, DC (1973).
10. *Report of the International Commission on Radiological Protection*. ICRP Publication 10, International Commission on Radiological Protection, London (1968).

11. L. Machta, G. J. Ferber, and J. L. Heffter. "Regional and Global Scale Dispersion of Krypton-85 for Population Dose Calculations." In *Proc. Symp. on Physical Behavior of Radioactive Contaminants in the Atmosphere*. Report IAEA/WMO, p. 411, Vienna (1975).
12. D. W. Hayes and K. W. MacMurdo. "Carbon-14 Production by the Nuclear Industry." *Health Physics*, 32, 215 (1977).
13. *Report of the United Nations Scientific Committee on the Effects of Atomic Radiation, New York, 1961*. Official Records of the General Assembly, 17th Session, Supplement No. 16 (A/5216).
14. Estimated 1975 population data for foreign countries from *Statistical Yearbook 1976*, United Nations, New York, NY (1977).
15. United States 1970 population data from *Historical Statistics of the United States, Part I*, U.S. Department of Commerce, Bureau of Census, Washington, DC (1975).

V. ASSUMPTIONS AND METHODOLOGY USED TO ASSESS ENVIRONMENTAL EFFECTS OF TRANSPORTATION

This section describes the methods used to assess the environmental effects of shipping spent fuel and wastes. These methods are used to determine the environmental effects given in the generic environmental impact statement on the Spent Fuel Policy. Methods are also included for calculating radiation doses to the transportation workers and the public along shipping routes; injuries and fatalities resulting from transportation accidents; nonradiological, thermal, and other transportation related environmental releases; and consumption of key materials (fossil fuels, etc.) during transportation of these spent fuels or wastes.

In most existing analyses of environmental effects from transportation, radiation doses to the public and transportation workers assume spent fuel is shipped about 150 to 180 days after discharge from the reactor. Current shipping casks are designed to provide shielding for gamma and neutron radiations and heat dissipation for spent fuel cooled for about 0.5 year or slightly less.

The casks must meet Federal regulatory standards and requirements for package design, including quality assurance and hypothetical accident tests to protect the contents in transportation accidents. Accidents reported for transportation of hazardous materials have been classified as minor, moderate, severe, extra severe, and extreme. Worldwide experience with spent fuel casks and full-scale testing of obsolete spent fuel casks indicate that these shipping casks will not be breached in minor, moderate, or severe accidents. In a test series simulating an extreme accident (collision of a rail car traveling at 131 kph (81 mph) into an unyielding concrete wall and then exposing the cask to an enveloping fire for ~100 minutes), the cask was not breached; and none of the contents was lost to the environment. However, to conservatively estimate environmental effects, this analysis assumes that an extra severe or extreme accident may damage a cask sufficiently to cause a loss of some of the contents of the cask to the environment.

If a cask filled with 0.5-year cooled fuel is involved in an extreme accident with no remedial action for a day or more, the self-heating of the fuel elements in casks may cause cask venting with some dispersal of gases from within the cask.

Volatile fission products and some aerosols may also be dispersed from the cask cavity, through any breaches in the damaged cask assembly, to the surrounding environs.

The environmental effects of transportation of spent fuel and waste in the two generic environmental impact statements that support the Spent Fuel Policy were determined by using the methods presented in this section. Spent fuel directly addressed by the Spent Fuel Policy is cooled 4 to 10 years or more before shipment. However, some spent fuel transshipped between utility water basins to relieve spent fuel storage problems of individual reactors may be cooled as little as 2 years. In this section, methodology and assumptions used for spent fuel that is cooled for 2, 4, 5, 10, or 25 years before shipment are discussed. Also discussed are assumptions for shipping wastes from storage basins and any future reprocessing facilities.

A. DEMOGRAPHIC MODEL

The analyses of the risks of either normal transportation or accidents during transportation depend upon the assumed population density, i.e., the average number of people per unit area. Because population densities vary greatly, three different population density zones suggested in NUREG 0170¹ were used in the GEISs on the Spent Fuel Policy. These zones were divided into urban, suburban, and rural areas. The average population densities were forecast for 1984 based upon values obtained from 1970-census data² and the demographic areas used in NUREG 0170.¹ In the forecast 1984 values, the average population densities in urban and suburban areas were assumed to remain constant; and the population increases were assumed to cause expansion of the land area occupied by urban and suburban populations.

The United States Census Bureau has delineated so-called "urbanized areas" to provide a better separation of urban and rural population in the vicinities of the larger cities. An urbanized area consists of a central city with 50,000 or more inhabitants and surrounding closely settled territory. Areas of large nonresidential tracts devoted to such urban land used as railroad yards, airports, factories, parks, golf courses, and cemeteries are excluded in computing the population density. The 1970 Census showed the average population density in urbanized areas was 1,303 people/km² (3,375 people/mi²); 31.5% of the total population lived within the central cities of urbanized areas; and 26.8% lived in the urban fringe (or suburban area) — for a total of 58.3% living inside urbanized areas.

The 1970 Census also showed the average urban housing area consisted of 4 to 5 housing units per acre or about 3,861 people/km²

(10,000 people/mi²). In NUREG 0170¹ (this value was assumed for urban population density), 54% of the urbanized area population was assumed to live in the central city; and 18.2% of the urbanized area was assumed to be occupied by the central city. About 15.2% of the total population lived in areas classified as urban, but these areas were outside the urbanized areas in and around the larger cities. NUREG 0170 classified this area as a suburban area. The average population density in these areas was 719 people/km² (860 people/mi²). The 1970 Census also showed that about 26.5% of the total population lived in rural areas, which contain about 98.5% of the land area of the United States and its possessions. The average population density was 6 people/km² (15 people/mi²).

For purposes of this assessment, the nationwide demographic model,¹ based upon 1970-census data, was adjusted for population projections for 1984.³ Population densities in urban and suburban areas were assumed to remain constant, but the areas were assumed to expand proportionally to the population growth predictions. Also, the model was adjusted for only the land area of the conterminous United States.

The fraction of total land area, fraction of total population, and associated population densities for each of the population zones for the year 1984 are shown in Table V-1. The year 1984 was chosen because large numbers of shipments begin in that year. The rate of change of the population for the period 1985 through 2000 is estimated to vary from 0.9% to 0.6%/yr. The assessment of shipments after 1984 includes adjustments for projected population growth (see Section IV of this report for growth projections).

TABLE V-1

Demographic Model for 1984

<i>Population Zone</i>	<i>Fraction of Land Area^a</i>	<i>Fraction of Population</i>	<i>Assumed Population</i>	<i>Population Density, persons/km²</i>
Urban	0.0024	0.317	71,552,000	3,861
Suburban	0.0182	0.446	100,654,000	719
Rural	0.979	0.237	53,568,000	7.1

a. Conterminous United States only.

B. GENERAL ASSUMPTIONS

General assumptions regarding casks and waste packages are summarized in this section. The effects of radioactive decay are shown in Table V-2 for radioactive and thermal properties of spent fuel. Categories of shipments and assumed distances per shipment are listed. Also, assumptions are shown for accident resistance of shipping packages, frequency of accidents during transportation, and the consequences of subjecting a spent fuel cask to accidents of different severities.

TABLE V-2

Radioactive Properties of Spent Fuel Normalized to 0.5-Year Cooled Fuel

Cooling Times of Spent Fuel →	<i>Ratio of Nuclides and Decay Heat to 0.5-Year Cooled Fuel</i>					
	0.5 yr	2 yrs	4 yrs	5 yrs	10 yrs	25 yrs
<i>Radioactivity Property</i>						
Total Fission Products	1.0	0.32	0.16	0.13	0.084	0.030
Gamma-Emitting Fission Products ^a	1.0	0.27	0.10	0.067	0.035	0.022
Neutrons Emitted	1.0	0.65 ^b	0.57	0.55	0.46	0.29
<i>Decay Heat</i>						
LWR	1.0	0.32	0.15	0.12	0.069	0.047
CANDU	1.0	0.23	0.08	0.06	0.03	0.02

a. With gammas >100 keV.

b. After the ²⁴²Cm decays, the majority of the neutrons are from ²⁴⁴Cm decay.

B.1 Thermal and Radiological Characteristics

In the GEISS on Spent Fuel Policy, existing and new casks designed for shipping LWR spent fuel cooled from 120 to 150 days are assumed to be used to ship spent fuel. These casks are designed to limit combined neutron and gamma radiations to no more than 10 mrem/hr at 2 m (6 ft) from the carrier for the 120- to 150-day cooled fuel. Therefore, the radiation rates will be much less than 10 mrem/hr at 2 m (6 ft) from the cask shipments of aged spent fuel considered in this study.

Table V-2 shows the radionuclide properties and decay heat for 2-, 4-, 5-, 10-, and 25-year cooled fuel normalized to 0.5-year cooled LWR and CANDU fuel. These values are used to estimate radiation exposures to transportation workers and the population exposed during transportation of spent fuel.

The thermal load of existing casks when filled with 2-, 4-, 5-, and 10-year cooled BWR and PWR elements and CANDU fuel is shown in Table V-3. Table V-3 also shows the licensed thermal loading of the casks.

TABLE V-3

Thermal Loading for Shipping Casks

Cask Type	Type of Shipment,		Thermal Loading, kW								
	LWR - No. of		2-Yr Cooled								
	Fuel Assemblies		4-Yr Cooled		5-Yr Cooled		10-Yr Cooled		Licensed Value		
	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR			
NFS, NAC-1	1	2	2.73	2.24	1.24	1.02	1.01	0.832	0.586	0.480	11.5
NLI-1/2	1	2	2.73	2.24	1.24	1.02	1.01	0.832	0.586	0.480	10.6
TN-8	3	-	a	-	3.73	-	3.03	-	1.76	-	35.5
TN-9	-	7	-	a	-	3.57	-	2.91	-	1.68	24.5
IF-300	7	18	a	a	8.71	9.19	7.10	7.48	4.10	4.31	11.8 dry ^b 62 water-filled
NLI 10/24	10	24	a	a	12.4	12.2	10.1	9.98	5.86	5.76	70
CANDU Fuel, MTU											
NLI 10/24	7.6 MTU ^c		-		5.6		4.2		2.2		70

a. Fuel cooled 2 years will be transported only on legal weight trucks.

b. The IF-300 cask is authorized for maximum thermal loads of 62 kW with the cask cavity filled with water and for maximum thermal loads of 11.8 kW with the cask cavity filled with air.

c. Four baskets each containing 1.9 MTU.

In addition to spent fuel shipments, wastes are shipped from the ISFS basin. Low-level waste (LLW) from the operation of the ISFS basin is assumed to contain small quantities of radionuclides. Therefore, the radiation levels are low, and the dose rates are assumed to be less than 0.2 mrem/hr to occupants of the truck cab and less than 2 mrem/hr at 2 m (6 ft) from the side of the truck. Little if any of the wastes from the ISFS basins will be classified as TRU wastes, and transportation of such wastes to a Federal disposition facility would not make a significant contribution to dose commitments from transportation activities.

Casks designed for shipping wastes from any future re-processing facilities are assumed to limit radiations to less than DOT maximum radiation limits (10 mrem/hr at 2 m (6 ft) from the transport vehicle).⁴ The thermal loading of a cask containing solidified high-level wastes may be as much as 30 kW. The packages containing other reprocessed wastes have a much lower thermal loading.

B.2 Modes of Spent Fuel Shipments

In the GEIS supporting the Spent Fuel Policy, spent fuel is assumed to be shipped from

- Domestic reactor basins to other reactor basins
- Domestic reactor basins to ISFS basin
- Domestic reactor basins to a disposition facility
- ISFS basins to a disposition facility
- Disposition facility or ISFS basin to reprocessing plant
- Reactor basins to reprocessing plant
- Foreign ports to U.S. port or return
- U.S. ports to ISFS basin or return
- U.S. ports to disposition facility or return

The assumptions used for each of these categories of shipment are discussed in the following sections. The term disposition facility is used in this report and in the GEIS on *Storage of U.S. Spent Power Reactor Fuels*. It denotes some undefined facility that receives the spent fuel from reactors and ISFS facilities. This disposition facility may be for disposal of spent fuel as waste; or, if recycling is approved, wastes from a reprocessing plant.

B.2.1 Transshipment from Domestic Reactor Basin to Other Reactor Basins

All shipments are assumed to be light water reactor (LWR) fuel shipped in legal weight truck casks (1 PWR or 2 BWR elements per load). The average shipping distance between reactor basins is assumed to be 160 km (100 mi).

Some of these contracts for the transshipping will be assumed to require return of this transshipped fuel to its point of origin before shipment to the United States Government. Other contracts may allow shipment to the United States Government from the second reactor. To be conservative, all transshipped fuel will be assumed to be returned to the original reactor basin before shipment to the Government. Some of this fuel may be cooled a minimum of 2 years; and, to conservatively estimate the effect of this shipping category, the fuel will be assumed to be cooled 2 years.

B.2.2 Domestic Reactor Basin to ISFS Basin

The assumptions used for this shipment category are the following:

- All spent fuel will be LWR fuel.
- Thirty percent of the spent fuel will be shipped in truck casks, and 70% will be shipped in rail casks.
- The shipping distance is assumed to be 1600 km (1000 mi).
- Cooling time of the spent fuel shipped in this category is 4 years.

B.2.3 Domestic Reactor Basins to Disposition Facility

The assumptions used for this category of shipment includes the following:

- All spent fuel will be LWR fuel.
- Thirty percent (by weight) of the spent fuel will be shipped in truck casks, and the remainder will be shipped in rail casks. A larger percentage of future reactors will be equipped with rail facilities than for existing reactors, and the percentage of truck shipments will decrease. Because the capacity of truck casks is small compared with rail casks, a shift to transport by rail will decrease the number of

shipments. Therefore, in this assumption, radiation doses to transportation workers and the population along transportation routes are conservatively assessed.

- The shipping distance is assumed to average 2400 km (1500 mi).
- In GEISs on the Spent Fuel Policy, all spent fuel transferred to the disposition facility must be cooled a minimum of 5 years. Much of the fuel transferred to the disposition facility will be cooled more than 5 years; but to be conservative, the environmental effects will be determined by assuming that all fuel has been cooled 5 years.

B.2.4 ISFS Basin to Disposition Facility

The assumptions for this shipping category include the following:

- Spent fuel will be LWR or CANDU fuel as dictated by the specific case considered.
- All spent fuel will be shipped in rail casks.
- The shipping distance is assumed to be 2400 km (1500 mi).
- The fuel schedule assumptions in the GEISs on Spent Fuel Policy would result in spent fuel cooled 12 years or more after reactor discharge, if the fuel is shipped in this category. To conservatively estimate the environmental effects of this shipping category, the cooling time of 12 years will be assumed for this fuel.

B.2.5 Disposition Facility or ISFS Basin to Fuel Reprocessing Plant (FRP)

The assumptions for the shipping category are the following:

- Spent fuel will be LWR or CANDU fuels as dictated by the specific case considered.
- All spent fuel will be shipped in a rail cask.
- The shipping distance is assumed to be 2400 km (1500 mi). Actual distance will probably be <2400 km (<1500 mi). If fuel reprocessing plants are built to include reprocessing of fuel retrieved from the disposition facility (such as a geologic reprocessing), one or more FRPs would probably be located closer to the disposition facility.

- An average age of 25 years since reactor discharge is assumed for the spent fuel shipped from the disposition facility or ISFS basins to the reprocessing plant(s).

B.2.6 Reactor Basins to Spent Fuel Reprocessing Plant (FRP)

The assumptions for this shipping category are the following:

- U.S. spent fuel will be shipped in rail and truck casks as previously described in B.2.3 of this section. For foreign fuels reprocessed in foreign countries, 20% (by weight) of the fuel will be shipped in overweight truck casks and the rest will be in rail casks.
- The shipping distance is assumed to be 2400 km (1500 mi). Actual distance will probably be less than 2400 km (1500 mi).
- Cooling time for the spent fuel will be determined by the delay of a decision to reprocess. If foreign spent fuel is reprocessed in foreign countries, transport from the reactors to a fuel reprocessing plant is expected after a cooling time of a few years. Cooling times are dependent upon the option selected, and the effects of cooling time are analyzed in the GEISs for the Spent Fuel Policy for each option assessed.

B.2.7 Foreign Ports to U.S. Ports

The assumptions for this shipping category include the following:

- Twenty percent of the LWR spent fuel will be shipped in overweight truck casks (TN-8, TN-9, etc.); and 80% will be shipped in rail casks. All CANDU spent fuel will be shipped in rail casks.
- The average shipping distance is assumed to be 7300 nautical miles for all ships delivering casks to eastern sea ports (assuming use of the Suez and Panama Canals). The average shipping distance is assumed to be 6300 nautical miles when shipments from eastern Asia are made to western ports (spent fuel goes directly to a disposition site) and all other shipments are made to eastern ports.
- The spent fuel is assumed to be cooled 5 years.

B.2.8 U.S. Ports to ISFS Basins

The assumptions for this shipping category are the following:

- Twenty percent of the LWR spent fuel will be shipped by overweight truck, and the rest will be shipped in rail casks. All CANDU spent fuel will be shipped by rail.
- The shipping distance is assumed to be 800 km (500 mi).
- The fuel is assumed to be cooled 5 years.

B.2.9 U.S. Ports to Disposition Facility

The assumptions for this shipping category include the following:

- Twenty percent of the LWR spent fuel will be shipped by overweight truck, and the rest will be shipped in rail casks. All CANDU spent fuel will be shipped in rail casks.
- The shipping distance is assumed to be 800 km (1500 mi).
- The fuel is assumed to be cooled 5 years.

B.3 Categories of Waste Shipments

Wastes are assumed to be shipped from

- ISFS basins to a shallow land burial site or a disposition facility. Assumptions for shipment of wastes generated at reactor basins are not included in this assessment nor in the GEISs on the Spent Fuel Policy.
- The spent fuel reprocessing-mixed oxide facility (FRP-MOX) to shallow land burial or to a disposition facility.

The assumptions used for each of these categories of shipment are as follows.

B.3.1 ISFS Basins to Land Burial Site and Disposition Facility

All shipments are assumed to be made by truck. The shipping distance from the ISFS basin to the land burial site is assumed to be 800 km (500 mi). Small quantities of TRU waste may be shipped from the ISFS basin to the disposition facility. This distance is assumed to be 2400 km (1500 mi).

B.3.2 FRP-MOX to Land Burial Sites or Disposition Facilities

The assumptions used for these categories of shipment include:

- Low-level waste (LLW) will be shipped to a land burial site by truck for a distance of not more than 800 km (500 mi).
- Solidified HLW, cladding (fuel residue) wastes, intermediate level waste (ILW), and low-level TRU wastes will be shipped in casks or other Type B packages on rail cars. The average shipping distance between the FRP and disposition facility is assumed to be 2400 km (1500 mi) as in B.2.5 of this section "Disposition Facility or ISFS Basin to Fuel Re-processing Plant (FRP)."

B.4 Abnormal Events During Transportation

Transportation accidents in routine commerce are relatively common. Because radioactive materials move in this routine commerce, they will sometimes be expected to be involved in accidents.

Because of the radiological characteristics of spent fuel and reprocessing wastes, the casks for this transport are designed to withstand accidents. Of primary concern is the remote possibility of a release of radionuclide or high-radiation levels from a cask damaged in an accident environment. Transportation statistics, accident scenarios, and knowledge of cask design used to estimate the probability and consequences of such abnormal events are described in this section.

Spent fuel considered in this assessment will be transported in Type B casks designed to withstand severe accident environments. An estimated 50,000 shipments per year⁵ are now made in the United States in Type B packaging. During the 25 years through 1972, an estimated 3600 casks⁶ containing irradiated fuel have been transported in routine commerce. In 1975, spent fuel shipments totaled 254 by truck and 17 by rail.¹ The Type B packagings, including casks for spent fuel, have an excellent record of package integrity in the United States and internationally.^{5,7,8}

No reported accident in the United States or internationally has subjected a Type B package to the most severe design impact; but, in the United States one or two fires which occurred after an accident have approached the design hypothetical fire criteria. Of eight severe vehicle accidents during transportation of Type B packages, three involved irradiated fuel casks where the truck overturned.

The casks remained intact, and no release of radionuclides or increase in radiation levels was detectable. In the worst event, the cask received only superficial damage and remained attached to its trailer, but the driver was killed.

B.4.1 Transportation Accident Statistics⁹

B.4.1.1 Land Transportation

Transportation industry vehicle accidents during routine commerce on land have a wide range of severities. Most accidents occur at low vehicle speeds and have relatively minor consequences. In general, as speed increases, accident severity also increases. However, an accident severity is not a function of vehicle speed only. Other factors such as the type of accident, the kind of equipment involved, and the location of the accident can have an important bearing on accident severity.

Furthermore, damage to any package in a transportation accident in routine commerce is not directly related to accident severity. In a series of accidents of the same severity or in a single accident involving a number of similar packages, damage to packages may vary from none to extensive. In relatively minor accidents, serious damage to some packages can occur from impacts on sharp objects or from being struck or crushed by other cargo. Conversely, damage to well-constructed packages from impacting a flat yielding surface may be minimal, even in very severe accidents.

Spent fuel and wastes are assumed to be shipped by commercial carrier on conventional transportation equipment. These shipments would be subject to the same accident environment as commercial shipments of nonradioactive commodities. Accident data for truck shipments are compiled by the Bureau of Motor Carrier Safety of the Federal Highway Administration.¹⁰ Accident data for rail shipments are compiled by the Office of Safety of the Federal Railroad Administration.¹¹ Summaries of hazardous materials accidents are compiled for each calendar year by the Office of Hazardous Materials of the United States Department of Transportation and are included in annual reports required by the Hazardous Materials Transportation Act, Title 1, Public Law 93-633.¹²

A recent study⁶ has analyzed the probability of rail and truck accidents based upon accident data supplied by the United States Department of Transportation. Accidents are classified by severity into five categories as functions of vehicle speed and fire duration. The five categories and their associated

probabilities for both rail and truck accidents are shown in Table V-4. (International rail and truck statistics are expected to be similar.)

Of particular concern are the transportation accident statistics that apply to spent fuel shipments. Consideration of the regulatory standards and requirements for package design and quality assurance, results of tests, and past experience support a conclusion that Type B packages will withstand all except extra severe or extreme accidents. The extra severe or extreme accidents can be postulated with very low frequencies $<1.6 \times 10^{-11}$ accidents per vehicle kilometer or $<1 \times 10^{-11}$ accidents per vehicle mile. Even if an extra severe or extreme accident occurs, package failure with subsequent release of radioactive contents, is unlikely. Estimates of the release probability for particular package designs have been discussed in References 13, 14, and 15.

TABLE V-4
Rail and Truck Accident Probabilities^a

Severity	Vehicle Speed, mph	Fire Duration, hr	Probability per Vehicle Mile ^a	
			Rail	Truck
Minor	0-30	<1/2	6×10^{-9}	6×10^{-9}
	0-30	0	4.7×10^{-7}	4×10^{-7}
	30-50	0	2.6×10^{-7}	9×10^{-7}
		Total	7.4×10^{-7}	1.3×10^{-6}
Moderate	0-30	1/2-1	9.3×10^{-10}	5×10^{-11}
	30-50	<1/2	3.3×10^{-9}	1×10^{-8}
	50-70	<1/2	9.9×10^{-10}	5×10^{-9}
	50-70	0	7.5×10^{-8}	3×10^{-7}
		Total	8.0×10^{-8}	3.1×10^{-7}
Severe	0-30	>1	7.0×10^{-11}	5×10^{-12}
	30-50	>1	3.9×10^{-11}	1×10^{-11}
	30-50	1/2-1	5.1×10^{-10}	1×10^{-10}
	50-70	1/2-1	1.5×10^{-10}	6×10^{-12}
	>70	<1/2	1×10^{-11}	1×10^{-10}
	>70	0	8×10^{-10}	8×10^{-9}
		Total	1.5×10^{-9}	8.2×10^{-9}
Extra Severe	50-70	>1	1.1×10^{-11}	6×10^{-13}
	>70	1/2-1	1.6×10^{-12}	2×10^{-13}
		Total	1.3×10^{-11}	8×10^{-13}
Extreme	>70	>1	1.2×10^{-13}	2×10^{-14}
		Total	1.2×10^{-13}	2×10^{-14}

a. Vehicle mile for rail is cask or car mile.

B.4.1.2 Maritime Transportation

Relatively few statistical analyses exist for maritime shipping. Statistics summarized below were obtained from safety studies for nuclear-powered cargo vessels. The average frequency of ship collisions for the period 1964 to 1974 was $4 \times 10^{-3}/\text{yr}$ per ship for all ships over 1800 gross MT (2000 gross tons).¹⁶ The rate of collisions decreased after 1970 even though the number of ships in operation steadily increased. Most of the collisions occurred near and in busy harbors or in congested shipping lanes. For example, during 1964-1974, of 59 collisions occurring on the New York - Rotterdam route, five occurred in New York Harbor or its vicinity; 54 occurred in the English Channel, North Sea, or Rotterdam approach; and none occurred in the open sea.

Containerships that would be used to transport spent fuel casks are equipped with high quality navigational and electronic gear. Also, the crew members are highly qualified and well trained. The frequency of collision of these containerships is estimated to be about 10^{-6} per trip.¹⁷ The frequency of collision with ensuing fire is estimated to be $<10^{-7}$ per trip based upon the statistic that only one of the 13 containerships involved in collisions during the period 1970 through 1975 caught fire and exploded. That containership struck a tanker.¹⁸

Casks stored on the deck are not expected to be impacted by a ship striking the containership and are expected to remain in place if the containership strikes another vessel. A second ship striking the containership is not expected to severely damage casks tied down in the rollon-rolloff hold.

B.4.2 Mitigating the Consequences of Transportation Accidents⁹

In a U.S. transportation accident, carriers of radioactive material are required to follow DOT-prescribed procedures¹⁹ designed to mitigate the consequences. IAEA regulations specify similar procedures and actions by emergency response personnel. DOT regulations require prompt reporting of any transportation incident involving shipment of radioactive material in which fire, breakage, spillage, or suspected radioactive contamination occurs. The regulations also specify guidelines for remedial actions in situations involving actual or suspected release of radioactivity from a shipping container. Vehicles used for transporting radioactive material must be monitored after each shipment; they may not be returned to service until the dose rate on accessible surfaces is below prescribed levels, and no significant removable radioactive surface contamination is evident.

An inter-governmental radiological assistance program^{20,21} provides personnel equipped to monitor radiation and trained to act as advisors to aid in radiological incidents such as a transportation accident involving nuclear material. The Federal radiological assistance program is coordinated by the Division of Operational Safety of DOE. The program provides a mechanism whereby 13 Federal agencies coordinate their radiological emergency activities with the activities of state and local health departments, and police, fire, and civil defense agencies.

In the event of a transportation accident, trained personnel from the radiological assistance program are available to help

- Evaluate the radiological health hazard
- Minimize personnel exposure to radiation and/or radioactive materials
- Minimize the spread of radioactive contamination
- Minimize damaging effects on property
- Assist in carrying out emergency rescue and first-aid procedures necessary to save life and minimize injury
- Provide technical information to appropriate authorities and medical advice on the treatment of injuries complicated by radioactive contamination
- Provide information to the public as quickly as possible to minimize undue public alarm and to assist in the orderly conduct of emergency activities

In accidents at sea, emergency assistance may not be available. If a collision or other ship accident occurs in or near a U.S. harbor, emergency assistance is provided by the U.S. Coast Guard and fire-fighting crew.

B.4.3 Fuel Element Failure During Transportation

A cladding failure rate of 0.25% during normal transport was assumed in WASH-1238⁶ and subsequent studies^{1,9} to assess the amount of radionuclides that may be accumulated in the coolant or atmosphere in the cavity of a spent fuel cask. The 0.25% was arbitrarily selected and was believed to be conservative (high) in the absence of experience data. Gaseous radionuclides from these "failed" fuel rods were then assumed to be released to the environs, if the cavity of the cask were overpressured (relief valve operated) or if a vehicle accident somehow breached the cask containment system.

No damage to spent fuel due to normal transportation has been observed either by visual inspection or radioactive gas monitoring at the Receiving Basin for Offsite Fuel (RBOF) at DOE's Savannah River Plant. RBOF has received over 1200 casks of domestic and foreign spent fuel, and the average cask loading was 25 assemblies (primarily metallic fuel). Satisfactory experience is expected in spent fuel shipments because fuels are shipped in rugged casks and are fixed in place inside the cask with spacers or impactor limiters. Therefore, the spent fuel is expected to withstand vibrations, shocks, stresses, etc., during normal transport without damage to cladding of fuel rod. LWR and CANDU spent fuel is similarly packaged for shipment.

Since no cladding damage has been observed in these shipments received at RBOF, this experience can be used to estimate the frequency for cladding failure during normal transportation based upon statistical treatment of rare events. When rare events are distributed in time, the usual and reasonable assumption is that the number of events in a fixed time interval has a Poisson distribution and that the intervals between events have an exponential distribution.²² With the experience of no spent fuel damage for transport of over 1200 casks (about 30,000 assemblies), the statement may be made with 95% confidence that the failure rate is between zero and 1×10^{-4} per assembly shipped.

For purposes of the GEISs on the Spent Fuel Policy, assumptions made for failure rate of cladding on fuel rods are

- During normal transportation, the failure rate is about 0.01% (or 1×10^{-4} failure per fuel rod).
- If the spent fuel shipment is involved in an accident (probability of about 10^{-6} /vehicle kilometer or mile), the cladding failure rate is 0.25% or greater, depending upon the severity of the accident environment.

B.4.4 Postulated Accidents for Shipments of Spent Fuel⁹

The key design goal of spent fuel casks is to provide safe transport of spent fuel (i.e., no leakage of radionuclide and shielding of nuclear radiations to DOT limits). The casks are rugged, thick-walled containers designed to retain shielding and containment integrity in virtually all transport accident situations. For licensing these casks, a set of hypothetical accident conditions (Type B package tests) have been specified which simulate severe accident environments. Type B test criteria include impact, puncture, fire, and immersion tests.

Recently, the protection provided by Type B package requirements has been investigated by intensive analytical studies,²³⁻²⁵ small-scale studies,²⁶⁻²⁹ and full-scale tests of actual shipping containers.^{30,31}

The testing program for spent fuel casks (truck and rail) at Sandia Laboratories includes:

- Analytical investigation of a railroad-grade crossing accident³² and a truck hitting a concrete wall
- Small-scale model testing of computer analyses
- Full-scale testing of analyzed accidents by using excess spent fuel casks

For the full-scale test of the railroad crossing "accident," a locomotive struck a truck loaded with a spent fuel cask at an impact velocity of 131 kph (82 mph). The cask was superficially damaged, and no damage occurred to the unirradiated fuel assembly carried in the cask.³³

The first truck test was made at an impact velocity of 98 kph (61 mph). Damage to the cask was slight, and no damage occurred to the fuel element carried by the cask.³³ The truck test was repeated with the same cask at an impact velocity of 135 kph (84 mph). The impact limiter on the cask was destroyed, and the head end of the cask was deformed but did not leak. The unirradiated fuel element (N. S. Savannah) carried in the cask was buckled at the impact end, but no cladding failure occurred.³⁴

In the fourth test conducted by Sandia Laboratories, a spent fuel cask on a rail car was propelled into a reinforced concrete wall at a velocity of 131 kph (81 mph). The rail car and cask were then subjected to an enveloping fire of JP-4 fuel for ~100 minutes. Fire temperature ranged between 1000 to 1150°C (1800 to 2100°F). This test simulates an extreme accident. The cask maintained its integrity (no loss of contents) during the collision and ensuing fire.

A spent fuel cask provides double containment for irradiated fuel during transport. The first barrier to an accidental release of radioactivity is the fuel cladding. The second barrier is the combination of inner shell, gamma shield, and outer shell (the cask body). For a release to occur during transport, radioactive material must first escape to the cask cavity from perforated fuel rods. The material must then be released from the cask atmosphere as a result of a penetration of cask containment. The cask may leak by one of three paths: actuation of pressure relief mechanisms, loss of closure head seal, or breach of the cask. Because of the cask design, barrier failure from accidental breach of the cask is considered extremely unlikely.

Some cask designs include a pressure-relief system, expected to vent when the internal pressure exceeds a preset level. Most systems are designed to reseal after excess pressure is relieved.

As considered in the GEISs on the Spent Fuel Policy, the spent fuel is carried in casks filled with helium or air. The most probable consequence of an accident that breaches one of these casks is a release of some of this helium or air from the cavity. Extreme accidents may cause the rupture of some fuel rod cladding and release fission gases from the fuel rod into the cask cavity and, if the cask is breached, then to the environs.

The Design and Analysis Report for the IF 300³⁵ cask indicates that, in a very severe accident (with no corrective action for two days or more), 120-day cooled PWR fuel may self-heat to temperatures that may rupture the cladding. However, as considered in the GEISs on the Spent Fuel Policy, spent fuel will be cooled 2 to 25 years when shipped. The decay heat of spent fuel cooled 2 years is 30% of that for fuel cooled 0.5 year. Calculations show that even if the cask, containing fuel cooled 2 years or more, is subjected to a lengthy fire in an accident environment, self-heating of the fuel rods will not cause cladding failures with subsequent release of fission products and actinides. The maximum fuel rod temperature is calculated to be 350°C (660°F) during normal transport or 390°C (730°F) if the cask is subjected to fire in an accident environment. Temperatures of fuel rods in other rail casks and truck casks are expected to be lower.

A fraction of the volatile fission products in a spent fuel rod is released from the fuel matrix and accumulated in rod void spaces. Table V-5 provides an estimate³⁶ of fission gas fractions which accumulate in fuel rod void spaces. If a fuel rod fails in shipment, fission gases which have accumulated in rod void spaces will escape to the cask cavity. The failure rate of fuel rods during normal transportation is assumed to be 0.01% and in transportation accidents 0.25% to 1% depending upon accident severity. Overpressurization of the cavity is not expected during normal transportation of long-cooled fuel.

Postulated accidents, response of safety systems, expected actions of emergency assistance personnel, and expected consequences are summarized for truck and rail shipments in Table V-6 through V-11. The consequences for casks on ships, involved in collisions, groundings, and fires are estimated to be no greater than for rail shipments.

TABLE V-5

Estimated³⁶ Activity Available for Release
in Penetration of Fuel Rod Cladding

<i>Radioactive Material</i>	<i>Percent in Fuel Rod Void Spaces</i>
⁸⁵ Kr	30
¹²⁹ I	10
³ H	1

TABLE V-6

Spent Fuel Rail Cask — Minor and Moderate Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Consequence</i>
Train derailment involves spent fuel cask. Estimated frequency $\sim 7 \times 10^{-7}$ /car mile.	<ol style="list-style-type: none"> 1. Derailment accident occurs at low to moderate speed. 2. Rail car leaves track and may overturn. 3. Confinement barriers of cask remain intact. 	<ol style="list-style-type: none"> 1. Radiation warning signs on cask caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to assist local public safety and to transport carrier officials to control site to recover cask. 4. Railroad crews restore car to track. 5. Confinement barriers of cask contain all radioactive materials. 	No release of radio-nuclides and no increase of radiation levels.
Train derailment involves spent fuel cask with 1/2-hr or less fire. Estimated frequency $\sim 3 \times 10^{-9}$ /car mile.	<ol style="list-style-type: none"> 1. Derailment accident occurs. 2. Rail car leaves track and may overturn. 3. Spent fuel cask is involved in 1/2-hr (or less) fire. 4. Confinement barriers of cask remain intact. 	<ol style="list-style-type: none"> 1. Radiation warning signs on cask caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to assist local public safety and to transport carrier officials to control site to recover cask. 4. Railroad crews restore car to track. 5. Confinement barriers of cask contain all radioactive materials. 	No release of radio-nuclides and no increase of radiation levels.

TABLE V-7

Spent Fuel Rail Cask - Severe Accidents

Accident Description	Sequence of Events	Safety System	Consequences								
Collision or derailment results in loss of neutron shielding from spent fuel rail cask. Estimated frequency $\sim 8 \times 10^{-8}$ /car mile.	<ol style="list-style-type: none">1. Collision and/or derailment accident occurs at moderate to high speed.2. Rail car overturns.3. Cask leaves rail car.4. Water jacket is ruptured and neutron shield water is lost.	<ol style="list-style-type: none">1. Radiation warning signs on cask caution onlookers to keep distance.2. Accident is reported to local and Federal officials.3. Interagency radiological assistance personnel available to assist local public safety, to transport carrier officials to control site, to minimize personnel exposure, and to recover cask.4. Remaining confinement barriers of cask remain intact and contain all radioactive materials.	No radioactive material release. Design basis - the radiation level at 3 ft from the damaged cask is less than 1000 mrem/hr for casks containing fuel cooled ~ 0.5 year.								
Fire accompanying accident causes a rail cask cavity to overpressurize; relief valve operates. Estimated frequency $\sim 5 \times 10^{-10}$ /car mile.	<ol style="list-style-type: none">1. Collision and/or derailment accident occurs at moderate speed.2. Spent fuel cask is involved in 1/2-hr (or longer) fire.3. Cask cavity overpressurizes, and pressure relief valve operates to relieve pressure.4. 0.1% of cavity atmosphere is released from cask due to operation of pressure relief valve.5. Cavity atmosphere contains some radioactivity from a small percentage of fuel rods with failed cladding.	<ol style="list-style-type: none">1. Radiation warning signs on cask caution onlookers to keep distance.2. Accident is reported to local and Federal officials.3. Interagency radiological assistance personnel available to assist local public safety and to transport carrier officials to control site, to minimize personnel exposure, and to recover cask.4. Confinement barriers of cask remain intact and limit release to a small percentage of cavity atmosphere.	Venting of cask occurs at ground level. If 0.25% of fuel rods exhibit failed cladding, the following release fractions apply: <table><tr><th>Isotope</th><th>Release Fraction</th></tr><tr><td>^{85}Kr</td><td>7.5×10^{-7}</td></tr><tr><td>^{129}I</td><td>2.5×10^{-7}</td></tr><tr><td>^3H</td><td>2.5×10^{-8}</td></tr></table>	Isotope	Release Fraction	^{85}Kr	7.5×10^{-7}	^{129}I	2.5×10^{-7}	^3H	2.5×10^{-8}
Isotope	Release Fraction										
^{85}Kr	7.5×10^{-7}										
^{129}I	2.5×10^{-7}										
^3H	2.5×10^{-8}										

TABLE V-8

Spent Fuel Rail Cask - Extra Severe or Extreme Accidents

Accident Description	Sequence of Events	Safety System	Consequences								
Train collision or derailment involves spent fuel rail cask that is subjected to severe impact and fire. Estimates $<1 \times 10^{-11}$ /car mile.	<ol style="list-style-type: none">1. Collision and/or derailment accident occurs at high speed.2. Spent fuel cask strikes massive object and decelerates almost instantaneously.3. Cask is involved in a fire which lasts longer than 1 hour.4. Closure head seal breached by gasket failure or by cask lid bolts damaged.5. 100% of cavity atmosphere is released, and cladding on 1% of the fuel rods is ruptured.	<ol style="list-style-type: none">1. Accident is reported to local and Federal officials.2. Interagency radiological assistance personnel available to assist local public safety and carrier officials to control site, to minimize personnel exposure, and to recover cask.3. Only a small opening exists in cask. Accident does not result in gross breach of cask containment.	<p>Venting of cask occurs at ground level. If 1% of fuel rods exhibit failed cladding, the following release fractions apply:</p> <table><tr><th>Isotope</th><th>Release Fraction</th></tr><tr><td>^{85}Kr</td><td>3×10^{-3}</td></tr><tr><td>^{129}I</td><td>1×10^{-3}</td></tr><tr><td>^3H</td><td>1×10^{-4}</td></tr></table> <p>Particulates are $<10^{-8}$ (other F. P. and, possibly, actinides).</p> <p>Possibly some increase in radiation levels around the closure head - with gasket failure only little increase in radiation levels. Any closure head displacement from the cask body would cause radiation levels that are proportional to the width of the resultant gap. Full-scale tests of spent fuel shipping casks suggest the probability that a significant gap width is very small.</p>	Isotope	Release Fraction	^{85}Kr	3×10^{-3}	^{129}I	1×10^{-3}	^3H	1×10^{-4}
Isotope	Release Fraction										
^{85}Kr	3×10^{-3}										
^{129}I	1×10^{-3}										
^3H	1×10^{-4}										

TABLE V-9

Spent Fuel Truck Cask -- Minor and Moderate Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Consequences</i>
Truck collision or overturn accident involves fuel cask. Estimated frequency $\sim 1 \times 10^{-6}$ /vehicle mile.	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Confinement barriers of cask remain intact. 	<ol style="list-style-type: none"> 1. Radiation warning signs on cask caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to assist local public safety and to transport carrier officials to control site and recover cask. 4. Confinement barriers of cask contain all radioactive materials. 	No release of radio-nuclides and no increase in radiation levels.
Truck collision or overturn accident involves spent fuel cask with 1/2-hour (or less) fire. Estimated frequency $< 6 \times 10^{-9}$ /vehicle mile.	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs at low to moderate speed. 2. Truck leaves roadway and may overturn. 3. Spent fuel cask is involved in 1/2-hour (or less) fire. 4. Confinement barriers of cask remain intact. 	<ol style="list-style-type: none"> 1. Radiation warning signs on cask caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site to recover cask. 4. Confinement barriers of cask contain all radioactive materials. 	No release of radio-nuclides and no increase in radiation levels.

TABLE V-10

Spent Fuel Truck Cask - Severe Accidents

Accident Description	Sequence of Events	Safety System	Consequences								
Truck collision or overturn results in loss of neutron shielding from spent fuel truck cask. Estimated frequency 3×10^{-7} /vehicle mile.	<ol style="list-style-type: none">1. Collision or overturn accident occurs.2. Truck leaves roadway and overturns.3. Water jacket is ruptured, and neutron shield water is lost.	<ol style="list-style-type: none">1. Radiation warning signs on cask caution onlookers to keep distance.2. Accident is reported to local and Federal officials.3. Interagency radiological assistance personnel available to assist local public safety and to transport carrier officials to control site, to minimize personnel exposure, and to recover cask.4. Remaining confinement barriers of cask remain intact and contain all radioactive materials.	No radioactive material release. Design basis - the radiation levels at 3 feet from the damaged cask is less than 1000 mrem/hr for casks containing fuel cooled about 0.5 yr.								
Fire accompanying accident causes truck cask cavity to overpressurize; relief valve operates. Estimated frequency 1×10^{-11} /vehicle mile.	<ol style="list-style-type: none">1. Collision or overturn accident occurs at moderate to high speed.2. Spent fuel cask is involved in 1/2-hour (or longer) fire.3. Cask cavity overpressurizes, and pressure relief valve operates to relieve pressure.4. 0.1% of cavity atmosphere is released from cask due to operation of pressure relief valve.5. Cavity atmosphere contains some radioactivity from a small percentage of fuel rods with failed cladding.	<ol style="list-style-type: none">1. Radiation warning signs on cask caution onlookers to keep distance.2. Accident is reported to local and Federal officials.3. Interagency radiological assistance personnel available to assist local public safety and to transport carrier officials to control site, to minimize personnel exposure, and to recover cask.4. Confinement barriers of cask remain intact and limit release to a small percentage of cavity atmosphere.	Venting of cask occurs at ground level. If 0.25% of fuel rods exhibit failed cladding, the following release fractions apply. <table><tr><th>Isotope</th><th>Release Fraction</th></tr><tr><td>^{85}Kr</td><td>7.5×10^{-7}</td></tr><tr><td>^{129}I</td><td>2.5×10^{-7}</td></tr><tr><td>^3H</td><td>2.5×10^{-8}</td></tr></table>	Isotope	Release Fraction	^{85}Kr	7.5×10^{-7}	^{129}I	2.5×10^{-7}	^3H	2.5×10^{-8}
Isotope	Release Fraction										
^{85}Kr	7.5×10^{-7}										
^{129}I	2.5×10^{-7}										
^3H	2.5×10^{-8}										

TABLE V-11

Spent Fuel Truck Cask -- Extra Severe and Extreme Accidents

Accident Description	Sequence of Events	Safety System	Consequences								
Collision or overturn causes spent fuel truck cask to be subjected to severe impact and fire. Estimated frequency $<1 \times 10^{-12}$ /vehicle mile.	<ol style="list-style-type: none">1. Collision or overturn accident occurs at high speed.2. Spent fuel cask strikes massive object and decelerates almost instantaneously.3. Cask is involved in a fire which lasts longer than 1 hour.4. Closure head seal breached by gasket failure or by cask lid bolts being damaged.5. 100% of cavity atmosphere is released, and cladding on 1% of the fuel rods is ruptured.	<ol style="list-style-type: none">1. Accident is reported to local and Federal officials.2. Interagency radiological assistance personnel available to assist local public safety and carrier officials to control site, to minimize personnel exposure, and to recover cask.3. Only a small opening exists in cask. Accident does not result in gross breach of cask containment.	<p>Venting of cask occurs at ground level. If 1% of fuel rods exhibit failed cladding, the following release fractions apply.</p> <table><tr><th>Isotope</th><th>Release Fraction</th></tr><tr><td>^{85}Kr</td><td>3×10^{-3}</td></tr><tr><td>^{129}I</td><td>1×10^{-3}</td></tr><tr><td>^3H</td><td>1×10^{-4}</td></tr></table> <p>Particulates are $<10^{-8}$ (other F. P. and, possibly, actinides).</p> <p>Possibly some increase in radiation levels around the closure head - with gasket failure, only a little increase in radiation levels. Any closure head displacement from the cask body would cause radiation levels that are proportional to the width of the resultant gap. Full-scale tests of spent fuel shipping casks suggest the probability that a significant gap width is very small.</p>	Isotope	Release Fraction	^{85}Kr	3×10^{-3}	^{129}I	1×10^{-3}	^3H	1×10^{-4}
Isotope	Release Fraction										
^{85}Kr	3×10^{-3}										
^{129}I	1×10^{-3}										
^3H	1×10^{-4}										

B.4.5 Postulated Accidents for Shipments of Reprocessing Wastes

Casks and other packages used for shipment of wastes from any reprocessing plant will be designed to provide protection from high-radiation dose rates, containment of radionuclides and, for HLW glass wastes, dissipation of decay heat. The casks or steel drums and boxes are placed in Type B overpacks for shipment and are designed to retain shielding and containment integrity in virtually all transport accident situations. Licensing requirements for these packages, including tests specified for hypothetical accident conditions, are similar to those for spent fuel casks.

Brief discussions follow for casks and packages used for different waste forms.

B.4.5.1 High-Level Wastes

Shipping casks for high-level wastes are expected to resemble spent fuel shipping casks weighing about 100 MT (110-tons), including fins to dissipate heat from radioactive decay of radionuclides within the cask. The cask, designed for rail transport, provides double containment for vitrified high-level wastes.

The first barrier to a release in a transportation accident is the canister containing the vitrified mass. The second barrier is the shielded shipping cask. For a release to occur during transport, the canister must be damaged; the vitrified mass must be fractured into small particles during the canister damage; the cask closure system must be breached; and then some driving force must expel the respirable particles from within the damaged canister into the environs.

A transportation accident that results in breach of a cask may also result in failure of the high-level waste canister and breakup of the glass into small, potentially respirable particles. A recent study³⁷ has measured the production of respirable (i.e., sub 10-micron) fines from high-speed impacts of stainless steel canisters containing borosilicate glass like that proposed for high-level waste solidification. In the study, canister damage was found to be generally limited to small cracks, and the fraction of sub 10-micron fines produced was very small and varied with the speed of impact. At a canister impact speed of 97 kph (60 mph) the fraction of sub 10-micron fines produced was about 10^{-3} wt % of glass inventory. Only a small fraction of the respirable particles produced by severe impact would be expected to escape to the cask cavity from cracks in a canister

and, in the unlikely event that the cask is breached, from the confines of the cask cavity to the outside environment.

The waste (assumed to be borosilicate glass) is heated to at least 900°C (1650°F) during processing. It is unlikely that the vitrified waste temperature would be exceeded in any credible transportation accident involving a fire. Thermal decomposition or a release of volatile material will not be a problem if the material is later heated to a lower temperature. Also, pressure buildup in these waste forms appears insignificant up to 900°C except for the volatilization of water accidentally added to a filled, baked out canister before closure welding.

Postulated accidents, response of safety systems, expected actions of emergency assistance personnel, and expected consequences are summarized in Tables V-12, V-13, and V-14.

B.4.5.2 Cladding Wastes

Fuel residue wastes are assumed to be shipped in a conceptual cask that is lead-filled, double-walled, stainless steel, and weighs about 65 MT. Three canisters containing cladding hulls and other fuel residues in a grout matrix are positioned in an insert inside the cask cavity.

The first barrier to a release in a transportation accident is the canister containing the cemented mass. The second barrier is the shielded shipping cask. For a release to occur during transport, the canister must be damaged; the cemented mass must be fractured into small particles during the canister damage; and the cask closure system must be breached before respirable particles can be expelled from the cask cavity into the environs.

A transportation accident that causes a breach of the cask may also cause the failure of the canister(s) and crush the cemented mass into small, potentially respirable particles. Recent impact tests of two types on concretes (bare masses containing simulated waste sludges and high-level defense wastes) converted 0.016% to 0.4%³⁸ to particles of less than 4 microns. Only a small fraction of the respirable particles produced by an extra severe or extreme transportation accident would be expected to escape to the cask cavity from cracks in a canister and, in the unlikely event that the cask is breached, from the cask cavity to the outside environment.

Postulated accidents, expected response of safety systems, and emergency personnel and expected consequences are summarized in Tables V-15 and V-16.

TABLE V-12

Solidified High-Level-Waste Rail Cask -- Minor and Moderate Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Release</i>
Train derailment involves high-level-waste cask	1. Derailment accident occurs at low to moderate speed.	1. Radiation warning signs on cask caution onlookers to keep distance.	None
Expected frequency $\sim 7 \times 10^{-7}$ /car mile	2. Railcar leaves track and may overturn.	2. Accident is reported to local and Federal officials.	
	3. Confinement barriers of cask remain intact.	3. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover cask.	
		4. Railroad crews restore car to track.	
		5. Confinement barriers of cask contain all radioactive materials.	
Train derailment involves high-level-waste cask in $\frac{1}{2}$ -hr or less fire.	1. Derailment accident occurs.	1. Radiation warning signs on cask caution onlookers to keep distance.	None
Expected frequency $\sim 3 \times 10^{-9}$ /car mile.	2. Railcar leaves track and may overturn.	2. Accident is reported to local and Federal officials.	
	3. Spent fuel cask is involved in $\frac{1}{2}$ -hr (or less) fire.	3. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover cask.	
	4. Confinement barriers of cask remain intact.	4. Railroad crews restore car to track.	
		5. Confinement carriers of cask contain all radioactive materials.	
Unusual transport condition erodes cask surface.	1. Cask is shipped with surface contamination which is within limits prescribed by Federal regulations.	1. Radiation warning signs on cask caution onlookers to keep distance.	Less than 2×10^{-6} Ci of mixed fission products and actinides for 25 yr aged spent fuel.
Expected frequency $\sim 10^{-3}$ /shipment.	2. Shipment encounters heavy rain storm or violent wind and dust storm.	2. Casks are decontaminated and monitored before shipment to ensure that removable surface contamination levels are within limits prescribed by Federal regulations.	
	3. Exceptional weather erodes cask surface.		
	4. 10% of residual surface contamination is removed by weathering.		

TABLE V-13

Solidified High-Level-Waste Rail Cask — Severe Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Release</i>
Collision or derailment results in loss of neutron shielding from high-level-waste cask.	<ol style="list-style-type: none"> 1. Collision and/or derailment Accident occurs at moderate to high speed. 2. Railcar overturns. 3. Cask leaves railcar. 4. Water jacket is ruptured and neutron shield water is lost. 	<ol style="list-style-type: none"> 1. Radiation warning signs on cask caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site, to minimize personnel exposure, and to recover cask. 4. Remaining confinement barriers of cask remain intact and contain all radioactive materials. 	<p>No radioactive material release.</p> <p>Design basis - the radiation level at 3 ft from the damaged cask is less than 1000 mrem/hr for casks containing HLW with 10-yr cooling.</p>
Expected frequency $\sim 8 \times 10^{-8}$ /car mile.			

TABLE V-14

Solidified High-Level-Waste Rail Cask — Extra Severe to Extreme Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Release</i>
Collision or derailment subjects high-level-waste cask to severe impact and fire.	<ol style="list-style-type: none"> 1. Collision and/or derailment accident occurs at high speed. 2. High-level-waste cask strikes massive object and decelerates almost instantaneously. 3. Cask is involved in a fire lasting more than 1 hr. 4. Closure head seal breached by gasket failure or by cask lid bolts damaged. 5. Mechanical shock causes small cracks in half of high-level-waste canisters. 6. Breakup of waste glass results in production of small percentage of respirable fines. 	<ol style="list-style-type: none"> 1. Accident is reported to local and Federal officials. 2. Interagency radiological assistance personnel available to assist local public safety and carrier officials to control site, to minimize personnel exposure, and to recover cask. 3. Only a small opening exists in cask. Accident does not result in gross breach of containment. 	<p>For glass waste a mixed fission product and actinide release in the form of sub-10 μ fines is postulated with a release fraction of $<10^{-8}$.</p>
Expected frequency $<1 \times 10^{-11}$ /car mile.			

TABLE V-15

Cladding Waste Rail Cask — Minor, Moderate, or Severe Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Release</i>
Derailment involves fuel residues cask. Expected frequency $\sim 7 \times 10^{-7}$ /car mile.	1. Derailment occurs. 2. Railcar leaves track and may overturn. 3. Confinement barriers of cask remain intact.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to aid local public safety and transport carrier officials to control site and recover cask. 4. Railroad crews restore car to track. 5. Confinement barriers of cask contain all radioactive materials.	None
Derailment involves fuel residues cask and $\frac{1}{2}$ -hr or less fire. Expected frequency $\sim 3 \times 10^{-9}$ /car mile.	1. Derailment occurs. 2. Railcar leaves track and may overturn. 3. Cask may leave car. 4. Confinement barriers of cask remain intact.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to aid local public safety and transport carrier officials to control site and recover cask. 4. Railroad crews restore car to track. 5. Confinement barriers of cask contain all radioactive materials.	None
Unusual transport condition erodes cask surface. Expected frequency $\sim 10^{-9}$ /shipment.	1. Cask is shipped with surface contamination which is within limits prescribed by Federal regulations. 2. Shipment encounters heavy rain-storm or violent wind and dust storm. 3. Exceptional weather erodes cask surface. 4. Weathering releases 10% of residual surface contamination.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Casks are decontaminated and monitored before shipment to ensure that removable surface contamination levels are within limits prescribed by Federal regulations.	2×10^{-6} Ci of mixed fission products and actinides for 25-yr cooled spent fuel.

TABLE V-16

Cladding Waste Rail Cask — Extra Severe or Extreme Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Release</i>
Collision or derailment subjects fuel residues cask to severe impact and fire.	<ol style="list-style-type: none"> 1. Collision and/or derailment speed. 2. Fuel residues cask strikes massive object and decelerates almost instantaneously. 3. Cask is involved in fire more than 1 hr. O 4. Closure head seal breached by gasket failure or by cask lid bolts damaged. 5. Small perforations occur in all waste canisters. 	<ol style="list-style-type: none"> 1. Accident is reported to local and Federal officials. 2. Interagency radiological assistance personnel available to aid local public safety and transport carrier officials to control site and recover cask. 3. Only a small opening exists in cask. Accident does not result in gross breach of cask containment. 	Respirable particle release occurs with a release fraction of $<1 \times 10^{-6}$.
Expected frequency $<1 \times 10^{-11}$ /car mile.			

B.4.5.3 Intermediate-Level Wastes

Intermediate-level wastes are assumed to be thoroughly mixed in concrete cast in canisters for shipment in cladding waste casks. The discussion and tables in B.4.5.2 of this section also apply to shipment of the solidified intermediate-level wastes.

B.4.5.4 Low-Level Transuranic (LLW-TRU) Wastes

Incinerator ashes or chopped-up noncombustible wastes are assumed to be thoroughly mixed with grout and cast in 210-L (55-gal) drums. These waste containers are shipped in a Type B overpack that provides impact and thermal protection for transportation accidents.

The first barrier to a release in a transportation accident is the steel drum containing the solid concrete mass. The second barrier is the walls of the protective overpack. For a release to occur during transport, the drum must be damaged; the cement mass must be fractured into small particles; and concurrently, the protective overpack must be breached before respirable particles can reach the environment.

In an extra severe or extreme transportation accident, there is a small possibility that the overpack may be breached. Some of the drum contents, perhaps 10%, may be fractured and crushed and produce some small particles. From data in DP-1400,³⁸ about 0.1% of the crushed concrete may be assumed to be respirable size particles. Only a small fraction of the crushed concrete mass resulting from such an accident would be expected to escape from the damaged drums to the protective overpack and then through breaches in the overpack to the surrounding environment.

Releases of the cemented mass to the environment are not expected in moderate and severe accidents. The consequences of postulated accidents and the response of safety systems and emergency personnel are shown in Tables V-17 and V-18.

B.4.5.5 Low-Level Wastes

Low-level wastes are assumed to be transported in truck vans or lightly shielded casks that provide some protection in transportation accidents. From Table V-4, the probability of fire for truck accidents is about 3.2×10^{-8} per truck kilometer (2×10^{-8} per truck mile), considering all accident types. A fire in an accident environment is assumed to convert no more than 10% of the waste in drums to small particles. Also, 10% of the small particles released from the drums or lightly shielded cask is assumed to

TABLE V-17

Low-Level TRU Waste Containers — Minor, Moderate or Severe Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Release</i>
Derailment involves low-level TRU waste container. Expected frequency $\sim 7 \times 10^{-7}$ /car mile.	<ol style="list-style-type: none"> 1. Derailment occurs. 2. Railcar leaves track and may overturn. 3. Confinement barriers of Type B overpack remain intact. 	<ol style="list-style-type: none"> 1. Radiation warning signs on overpack caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover overpack. 4. Confinement barriers of Type B overpack contain all radioactive material. 	None
Derailment involves low-level TRU waste container and $\frac{1}{2}$ -hr or less fire. Expected frequency $\sim 3 \times 10^{-9}$ /car mile.	<ol style="list-style-type: none"> 1. Derailment occurs. 2. Railcar leaves track and may overturn. 3. Low-level TRU waste container is involved in $\frac{1}{2}$-hour (or less) fire. 4. Confinement barriers of Type B overpack remain intact. 	<ol style="list-style-type: none"> 1. Radiation warning signs on overpack caution onlookers to keep distance. 2. Accident is reported to local and Federal officials. 3. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover overpack. 4. Confinement barriers of Type B overpack contain all radioactive material. 	None
Low-level TRU waste shipment made in improperly closed packages and improperly closed overpack. Expected frequency 10^{-5} /shipment.	<ol style="list-style-type: none"> 1. A shipment of low-level TRU waste is made with improperly closed packages. 2. Some radioactive material is released from the packages to the interior of the overpack. 3. A small fraction of the material released from the waste packages escapes from the confinement of the overpack during the trip if the overpack is also improperly closed. 	<ol style="list-style-type: none"> 1. Radiation warning signs on overpack caution onlookers to keep distance. 2. Quality assurance and package test requirements reduce probability of shipments with improperly closed packages. 3. Sealed overpack prevents release to the environment of material spilled from improperly closed packages. 	Release fraction is 10^{-8} . Assume release of respirable particles at ground level.

TABLE V-18

Low-Level TRU Waste Containers - Extra Severe or Extreme Accidents

<i>Accident Description</i>	<i>Sequence of Events</i>	<i>Safety System</i>	<i>Release</i>
Low-level TRU waste container is subjected to severe impact and fire. Expected frequency $<1 \times 10^{-11}$ /car mile.	<ol style="list-style-type: none">1. Collision and/or derailment occurs at high speed.2. Overpack strikes massive object and decelerates almost instantaneously.3. Overpack is involved in fire which lasts longer than 1 hour.4. Waste packages are breached by force of accident.5. Only rips or holes are created in overpack; waste packages are retained within the confines of the overpack.	<ol style="list-style-type: none">1. Accident is reported to local and Federal officials.2. Interagency radiological assistance to personnel available to assist local public safety and transport carrier officials to control site and recover overpack.3. Only small openings exist in overpack. Accident does not result in gross breach of containment.	Release fraction is 10^{-5} .

escape from the truck or cask. The release fraction of respirable fines is assumed to be 0.01. The hazard from this waste release is very small because of the low concentration of radionuclides in the waste.

B.5 Transportation Workers

Assumptions concerning transportation workers are discussed below.

B.5.1 Personnel Requirements

Truck shipments are assumed to be made on exclusive-use trucks and therefore two drivers are assumed to travel directly from the shipping facility to the receiving facility. The overall average speed, including driving time and periodic stops, is assumed to be about 60 km/hr (40 mph/hr) for 800-, 1600-, or 2400-km (500-, 1000-, or 1500-mi) trips.

The total manpower required for trucking operations is arbitrarily assumed to be double the manpower requirements for drivers. This total includes manpower for contingencies, truck breakdowns and servicing, supporting personnel and supervisory overheads. With this assumption, manpower requirements for round trips by truck, including stops, are

- 160-km (100-mi) shipments, 30 man-hours
- 800-km (500-mi) shipments, 100 man-hours
- 1600-km (1000-mi) shipments, 200 man-hours
- 2400-km (1500-mi) shipments, 300 man-hours

Manpower requirements for rail operations are based upon the time spent by the train crew on the train while it is in motion. The average velocity of the train, while it is in motion, is assumed to be ~60 kph (~40 mph) for trips of 800-, 1600-, 2400-km (500-, 1000-, or 1500-mi). A crew of five and an average train of 70 cars are assumed. On a prorated basis, the man-hours expended by the crew per car transported round trip is assumed to be 4 man-hours for a shipping distance of 1600 km (1000 mi).

The total manpower required for rail operations is arbitrarily assumed to be triple the manpower requirements for crewmen to include contingencies; supporting personnel, including those for switchings; rail yard controls, etc; and supervisory overhead. Therefore, the manpower requirements assumed for round trips by rail (filled cask and return of empty cask) are

- 800-km (500-mi) shipments, 6 man-hours/rail cask
- 1600-km (1000-mi) shipments, 12 man-hours/rail cask
- 2400-km (1500-mi) shipments, 18 man-hours/rail cask

Cargo ships selected for transport of spent fuel casks are assumed to travel at an average speed of 19 knots,* including port and refueling stops. The cruising speed of the container-ship is 24 knots. A crew of 20 is assumed. The manpower requirements for ship personnel assumed for transport of spent fuel casks are prorated by the ratio of cask weight to total net cargo. Four NLI 10/24 casks represent 5.7% of the cargo and four TN-8 or TN-9 casks represent 2.3% of the cargo.

The total manpower required for sea and harbor operations involving transport of spent fuel, including loading and unloading of casks at U.S. ports, is assumed to be double the manpower requirements of the crew. Therefore, the manpower requirements assumed for round trips by ship (filled and returned empty cask) are

- Average shipping distance of 7260 nautical miles, 850 man-hours per four NLI 10/24 casks, or 340 man-hours per four TN-8, TN-9 casks
- Average shipping distance of 6310 nautical miles, 740 man-hours per four NLI 10/24 casks, or 300 man-hours per four TN-8, TN-9 casks

B.5.2 Injuries and Fatalities in Transportation Accidents

Injuries caused by impact, crushing, etc., are fairly frequent in land transportation accidents. Fatalities resulting from these injuries occur less frequently. Both transportation crews and members of the population may be involved in these accidents. For instance, in truck accidents, nonfatal injuries occur in about 1/3 of all truck accidents; and fatal injuries occur in about 3 out of 100 accidents.

Probability rates for nonradiological injuries and fatalities for U.S. traffic accidents for 1969⁶ are shown in Table V-19 for truck and rail shipments. These rates, based upon 1969 accident statistics, are assumed for calculating injuries and fatalities attributable to shipments considered in the GEISS on Spent Fuel Policy.

* Knots = nautical miles per hour.

TABLE V-19

Risks of Injury or Fatality from Accidents
During Transportation

<i>Mode and Probability Description</i>	<i>Injuries</i>	<i>Fatalities</i>
Truck ^a		
Probability/accident	0.51	0.03
Probability/vehicle mile	9×10^{-7}	5×10^{-8}
Rail ^a		
Probability/accident	2.7	0.2
Probability/car mile	4×10^{-7}	3×10^{-8}
Ship		
Probability/trip	b	6×10^{-8}

a. Probabilities from WASH-1238, Section V.⁶

b. No estimate of injuries.

For the period 1965 to 1974, 43 deaths were tabulated in U.S. Coast Guard reports for 3076 cargo ship collisions, for a frequency of 1.4×10^{-2} fatalities per collision. The cumulative total of cargo ship accidents of all classifications was 7450 with 183 fatalities for an overall frequency of 2.5×10^{-2} fatalities per accident. As discussed in B.4.1.2 of this section the estimated frequency of collision of containerships assumed for shipping spent fuel is about 10^{-6} per trip. From the above statistics, the frequency of fatalities from all types of cargo ship accidents is no more than 6×10^{-8} fatalities per trip.

C. METHODS USED TO ASSESS ENVIRONMENTAL EFFECTS
OF TRANSPORTATION

This section describes the methods used to calculate population dose and the occupational dose to truck drivers and rail or ship crewmen during normal transportation and the dose to the maximum individual that might result from a transport accident.

C.1 Summary

The methods for determining population and transportation worker dose from normal transportation given in this section were adapted from NUREG 0170.¹ These methods determine the doses to the following population groups:

- Persons who live or work near the land transport route or ports
- Persons traveling on the railroad or highway near the radioactive shipments
- Persons exposed during carrier stops
- Truck drivers and train and ship crew (referred to as crewmen)

Table V-20 shows the results of spent fuel cooled 0.5 year and shipped 2400 km (1500 mi) by rail or truck. This table shows the relative comparison between rail and truck transport with the normal fuel load and shows the magnitude of the dose per trip to different sectors of the population per trip. The dose values given in Table V-20 are assumed for the radiation level from the rail car or truck at the DOT limit of 10 mrem/hr at 2 m (6 ft) from the side of the vehicle. The dose per trip to the population is determined by the mode of shipment rather than the quantity of spent fuel carried. As can be seen, the crew transporting the fuel receives the largest dose during shipment by either rail or truck.

Table V-21 shows the results of dose calculations in terms of dose per mile to transport one MTU of spent fuel cooled 0.5 year. These results show that doses per truck mile are much larger than per rail mile. Also, as the capacity of the cask is increased, the doses to transportation workers and the general population decrease for shipping a given quantity of spent fuel.

The method used to determine the maximum individual dose during an accident is given in C.3.2 of this section. This method, used in conjunction with the release fractions given in Tables V-6 through V-18, allows estimates of the dose commitment to the maximum individual.

During normal transportation, the maximum individual dose to any transportation worker is determined by the number of trips and total miles traveled. The maximum individual dose to anyone living close to the route is determined by the distance he lives from the road or railroad and the number of shipments passing.

TABLE V-20

Population and Transportation Workers Dose^a - Normal Transport of a Spent Fuel Cask

Population Exposed	Dose per 1500-Mile Shipment ^b		Dose per 7300-Mile Trip
	Rail ^c man-rem/trip	Truck man-rem/trip	Ship ^d man-rem/trip
Persons along the route (for ship-entry and exit from the port)	1.5×10^{-3}	1.5×10^{-3}	1.7×10^{-3}
Persons traveling on the highway or railroad	1.4×10^{-5}	1.5×10^{-3}	-
Persons exposed during stops of the carrier vehicle	1.4×10^{-3}	1.9×10^{-2}	3.3×10^{-2}
General population total	2.9×10^{-3}	2.2×10^{-2}	3.5×10^{-2}
Crewmen or drivers (transportation workers)	1.8×10^{-2}	6.5×10^{-2}	2.4×10^{-1}
Totals	2.1×10^{-2}	8.7×10^{-2}	2.8×10^{-1}

a. Calculated based on 0.5-year cooled spent fuel.

b. Data provided from "RADTRAN" program³⁹ by John M. Taylor, Sandia Lab, if the dose rate is assumed to be 10 mrem/hr at 6 ft from the vehicle carrying the cask.

c. For carriage by regular train. Transport by special train increases the dose for crewmen to 2.1×10^{-2} man-rem, but population dose to other sectors of the population changes very little when compared with regular train.

d. While the ship is at sea, only the crew is exposed to radiations.

TABLE V-21

Comparative Population and Transportation Worker Dose for Different Spent Fuel Casks

Type of Cask	Average ^a Loading, MTU	Dose, man-rem/mi - MTU ^b	
		Transportation Workers	General Population
Truck (legal weight)	0.424	1.0×10^{-4}	3.5×10^{-5}
Truck (overweight)	1.36	2.0×10^{-4}	3.7×10^{-5}
Rail			
IF-300	3.30	3.6×10^{-6}	5.8×10^{-7}
NLI-10/24	4.58	2.6×10^{-6}	4.1×10^{-7}

a. Based on 38% by weight BWR elements and 62% by weight PWR elements.

b. Based on 1500-mile trips for 0.5-year cooled fuel.

The maximum dose to a truck driver is estimated to be about 0.5 rem/yr and is less for other individuals. If an extra severe or extreme transportation accident occurred and fission gases escaped from the damaged spent fuel cask (a very low probability), the maximum dose to an individual of the population downwind of the accident involving aged fuel is in the order of 10 mrem.

C.2 Dosimetry Methods for Normal Transport

The methods used to calculate doses from transportation of LWR and CANDU spent fuel and wastes by truck, rail, and ship are developed in this section based upon the RADTRAN program.³⁹ The methods assess dose to the transportation workers and to the general population exposed along the route of travel and during stops of the shipments. Methods were developed to calculate doses to persons in highway vehicles traveling in the same and opposite direction from truck shipments. This type calculation is not made for rail or ship. Because the number of persons in a train passing another train carrying rail cask(s) is small, and the exposure time is short, the resultant dose is very small and does not contribute significantly to the total population dose from rail shipments. Likewise, a cargo ship carrying spent fuel casks does not get close to other ships on the high seas and, therefore, no exposure occurs to persons other than crew members. In the harbor, few ships approach closer than one-half mile to the cargo ship carrying casks, and the accumulated dose to personnel on those other ships is negligible compared with the population dose while the ship is at berth.

The method for assessing population and transportation worker doses involves dose rate as a function of distance from a point source of radiation. The point source approximation is acceptable for distances between the receptor and the source of more than twice the length of the source. At smaller distances, the point source approximation overpredicts exposure and, therefore, will provide a conservative estimate of dose. The dose rate relationship is given by the following:

$$D(d) = \frac{K e^{-\mu d} B(d)}{\left(\frac{d}{d_0}\right)^2} \quad (1)$$

where:

$D(d)$ = distance from point source (ft)

d_0 = distance from point source where the dose rate = K

K = the dose rate (mrem/hr) at a distance equal to d_0

μ = absorption coefficient for air (0.00118 ft^{-1})

$B(d)$ = Berger buildup factor in air $B(d) = 0.0006d + 1$
(dimensionless)⁶

C.2.1 Dose to Persons Surrounding the Transport Route as Shipment Passes

An expression for the total integrated dose absorbed by an individual at a distance $\underline{x}$ from the path of a radioactive shipment with dose rate factor $\underline{K}$ passing at velocity $\underline{V}$ has been derived⁶ from Equation 1 and is given by

$$D(x) = 2 \frac{K}{V} I(x) \quad (2)$$

where:

V = shipment speed (ft/hr)

x = perpendicular distance of individual from shipment path (ft)

$$I(x) = \int_x^{\infty} \frac{e^{-\mu r} B(r) dr}{r(r^2 - x^2)^{\frac{1}{2}}}$$

For a K of 1 mrem/hr and a V of 1 mile/hr, the absorbed dose as a function of $\underline{x}$ is as shown in Figure V-1.

FIGURE V-1. Dose Received by an Individual as a Shipment Passes¹

In order to obtain integrated population dose in sectors of length L and width d on both sides of the shipment route (Figure V-2), Equation 2 is multiplied by the average population density and L and integrated over the width of the strip on both sides of the highway or railroad. For a ship entering a harbor, exposure of a resident population on only one side of the ship is assumed and Equations 3 through 7 are multiplied by 0.5 to conservatively estimate the population dose to residents and workers along the path of the ship.

$$\text{Dose (Truck or Rail)} = 2 \cdot \text{PD} \cdot L \cdot \int_{\min}^d D(x) dx \quad (3)$$

where:

Dose = integrated population dose in strip (person-mrem)
on both sides of truck or rail routes

PD = average population density (persons/ft²)

L = length of strip (ft)

$\min$ = minimum distance from population to shipment
centerline (ft)

d = maximum distance over which exposure is
evaluated (ft)

$D(x)dx$ = incremental dose function from Equation 2 (mrem/ft)

Equation 3 predicts an infinite dose as $\min$ approaches 0; thus, a limit on this value must be set. Values for $\min$ were selected based upon actual roadway dimensions. A value of 2600 feet was selected for d based upon a previous assessment.⁶

For a radioactive package with dose rate factor K , a single trip is considered to involve three population density zones: rural, suburban, and urban. The total population dose resulting from the trip of length L (feet) is made up of the sum of the doses received in each of the three zones:

$$\text{Dose} = \text{Dose}_r + \text{Dose}_s + \text{Dose}_u$$

where the subscripts r , s , and u refer to rural, suburban, and urban, respectively. The use of the integrated dose expression of Equation 3 results in the following expression:

$$\begin{aligned} \text{Dose} = 4 \cdot K \cdot L \cdot & \left[\frac{f_r \cdot \text{PD}_r}{V_r} \cdot I_r + \frac{f_s \cdot \text{PD}_s}{V_s} \cdot I_s \right. \\ & \left. + \frac{f_u \cdot \text{PD}_u}{V_u} \cdot I_u \right] \end{aligned} \quad (4)$$

- - Populated Zone with uniform population density PD
- L - Length of populated strip
- d - Maximum distance over which exposure is evaluated
- min - Smallest distance between exposable population and shipment centerline

FIGURE V-2. Dose to Person Living Along the Transport Route¹

where:

f_r = fraction of distance traveled in rural population density zone

f_s = fraction of distance traveled in suburban population density zone

f_u = fraction of distance traveled in urban population density zone

PD_r = population density (rural) (people/ft²)

PD_s = population density (suburban) (people/ft²)

PD_u = population density (urban) (people/ft²)

$$I_r = \int_{\min_r}^d I(x) dx$$

$$I_s = \int_{\min_s}^d I(x) dx$$

$$I_u = \int_{\min_u}^d I(x) dx$$

$\min_r$ = minimum distance from exposable population to shipment centerline (ft) (rural)

$\min_s$ = minimum distance from exposable population to shipment centerline (ft) (suburban)

$\min_u$ = minimum distance from exposable population to shipment centerline (ft) (urban)

V_r = average speed in rural area (ft/hr)

V_s = average speed in suburban area (ft/hr)

V_u = average speed in urban area (ft/hr)

Spent fuel shipments by truck are assumed to be routed on freeways and 4-lane roads in low- and medium-population density zones. Therefore, the urban term is not modified for transportation in city streets. Trucks leaving freeways or 4-lane roads to make stops are assumed to travel short distances. The population dose during stops, as calculated in C.2.2 of this section, is assumed to include travel between the truck stops and the 4-lane roads and freeways.

The upper integration limit $\underline{d}$ was taken to be 2600 ft, and the lower limits $\underline{\min_r} = \underline{\min_s} = \underline{\min_u} = 100$ ft in all three population density zones. With these limits, the dimensionless integral $\underline{I_r} = \underline{I_s} = \underline{I_u}$ was evaluated numerically and found to be equal to 2.42.

When the value of the dimensionless integral $\underline{I_r} = \underline{I_s} = \underline{I_u}$ is substituted for Equation 4, the following expression results:

$$\text{Dose} = 4 \cdot K \cdot L \cdot 2.42 \cdot \left[\frac{f_r \cdot PD_r}{V_r} + \frac{f_s \cdot PD_s}{V_s} + \frac{f_u \cdot PD_u}{V_u} \right] \quad (5)$$

If the population densities $\underline{PD}$ are expressed as persons/mi² and the velocities $\underline{V}$ are expressed in miles per hour (mph), the dose received per mile traveled is

$$\begin{aligned} \text{Dose (person-rem per mile)} &= 3.47 \times 10^{-10} \cdot K \cdot \left[\frac{f_r \cdot PD_r}{V_r} \right. \\ &\quad \left. + \frac{f_s \cdot PD_s}{V_s} + \frac{f_u \cdot PD_u}{V_u} \right] \quad (6) \end{aligned}$$

The annual normal population dose for this shipment scenario is obtained by multiplying the above equation by the total number of miles per year for the type of shipment, or $\underline{SPY} \times \underline{FMPS}$,

where:

$\underline{SPY}$ = number of shipments per year

$\underline{FMPS}$ = average distance traveled (miles) per shipment

In this assessment, the final expression for calculating the normal population dose to surrounding population while a given type of shipment is moving becomes

$$\left. \begin{array}{l} \text{Dose} \\ \text{(person-rem} \\ \text{per year)} \\ \text{Truck or Rail} \end{array} \right\} = 3.47 \times 10^{-10} \cdot K \cdot \text{SPY} \cdot \text{FMPS} \cdot \left[\frac{f_r \cdot \text{PD}_r}{V_r} + \frac{f_s \cdot \text{PD}_s}{V_s} + \frac{f_u \cdot \text{PD}_u}{V_u} \right] \quad (7)$$

where:

K = 1000 mrem/hr for a spent fuel cask containing fuel cooled about 0.5 yr. This value is equivalent to 10 mrem/hr at 10 ft from the centerline of the carrier vehicle.

SPY = number of shipments per year

FMPS = average distance (miles) per shipment

f_r, f_s, f_u = fraction of distance traveled in rural, suburban, and urban areas, respectively

$\text{PD}_r, \text{PD}_s, \text{PD}_u$ = population density (person/mi²) in rural, suburban, and urban areas, respectively

V_r, V_s, V_u = average speed (mph) in rural, suburban, and urban areas, respectively

C.2.2 Dose to Population During Shipment Stops

If the land shipment stops for crew change, meals, refueling, etc., or the ship is at berth, people in an area around the stop point are exposed. The population dose is again obtained by integrating a form of Equation 1 which includes an annular differential element, $2\pi r dr$:

$$\text{Dose} = K \cdot \Delta T \cdot \text{PD} \int_x^d (2\pi r) \left(\frac{e^{-\mu r} B(r)}{r^2} \right) dr \quad (8)$$

where:

Dose = integrated population dose per shipment (person/mrem)

ΔT = total stop time per shipment (hr)

PD = Population density

Numerical evaluation of the integral when $x = 10$ ft and $d = 2600$ ft is 27.275.¹

Stops are assumed to be made in different population density zones. By selecting the length of time of stops in each population zone per trip the integrated population dose is man-rem per year. The man-rem per year resulting from stops for a given shipment is given by:

$$\text{Dose} = Q_1 \cdot K \cdot \text{SPY} \cdot \left[(\Delta T_r)(PD_r) + (\Delta T_s)(PD_s) + (\Delta T_u)(PD_u) \right] \quad (9)$$

where:

ΔT_r = total stop time in rural population density zones (hours)

ΔT_s = total stop time in suburban population density zones (hours)

ΔT_u = total stop time in urban population density zones (hours)

$Q_1 = 2.54 \times 10^{-9}$ (rem-km²/mrem-ft²) (for $x = 10$ feet and $d = 2600$ feet)

$K = 1000$ mrem/hr (for shipments radiating 10/mrem/hr at 6 feet from the carrier vehicle)

C.2.3 Dose to Crewmen

The annual dose to crewmen is obtained directly from Equation 1 by using an average source-to-crew characteristic distance $\underline{d}$ for each transport mode

$$(\text{Dose})_{\text{crew}} = Q_3 \cdot K \cdot \text{SPY} \cdot N_{\text{crew}} \cdot \frac{e^{-\mu d} B(d)}{d^2} \cdot \Delta T_{\text{ship}} \quad (10)$$

where:

N_{crew} = number of crewmen aboard

d = average distance to crew compartment (ft)

$Q_3 = 10^{-3}$ (rem/mrem)

ΔT_{ship} = average time required for a shipment = $\frac{f_r}{V_r} + \frac{f_s}{V_s} + \frac{f_u}{V_u}$

$K = 1000$ mrem/hr

$\text{SPY} = \text{shipments/yr}$

Mode

C.2.4

Figure V-3 shows a vehicle carrying radioactive material.

FIGURE V-3. Dose to Persons in Vehicles Sharing the Highway with the Shipment¹

The vehicle is traveling at a speed V_T along with other vehicles in the same lane. Occasionally, vehicles traveling in the opposite direction pass the vehicle in the other lane. Two separate doses are computed: a) the dose to persons traveling in the opposite direction from the shipment, and b) the dose to persons traveling in the same direction as the shipment.

C.2.4.1 Dose to Persons Traveling in the Opposite Direction

Both the shipment and the oncoming traffic are assumed to be moving at speed $\underline{V_T}$. The dose received by an individual in an oncoming vehicle may be computed by assuming that his vehicle is at rest, and he is passed by the shipment at a speed of $2\underline{V_T}$. An expression for the integrated dose from a moving source was given in Equation 2.

Thus, the average integrated dose received by a person in an oncoming vehicle passing the truck at a distance $\underline{x}$ is

$$D = \frac{2K}{(2V_T)} I(x) \quad (11)$$

The average number of oncoming vehicles $\underline{N_c}$ per mile is

$$N_c = \frac{N'}{V_T} \quad (12)$$

where:

N' = the traffic count (average number of cars per hour traveling in one direction)

Let $\underline{P}$ be the average number of persons per vehicle. Thus the average number of persons ($\underline{N_p}$) who travel in the opposite direction to the shipment and who are exposed per mile traveled by the truck is

$$N_p = N_c \cdot P = \frac{N' \cdot P}{V_T} \quad (13)$$

The population dose to persons traveling in the opposite direction to a shipment is given by $\underline{D} \times \underline{N_p} \times \underline{FMPS}$, where $\underline{FMPS}$ is the average distance per shipment. Multiplication of this number by $\underline{SPY}$, the annual number of shipments of that type, results in the annual population dose for the given shipment:

$$\begin{aligned} \text{Dose} &= \frac{K}{V_T} \cdot I(x) \cdot \frac{N'}{V_T} \cdot P \cdot FMPS \cdot SPY \\ &= K \cdot I(x) \cdot \frac{N'}{V_T^2} \cdot P \cdot FMPS \cdot SPY \end{aligned} \quad (14)$$

The traffic count $\underline{N'}$ and the average velocity $\underline{V_T}$ depend upon the population density zone and the time of day (i.e., rush hour or normal traffic). The value of the integral $\underline{I(x)}$ depends upon the distance $\underline{x}$ of closest approach, which in turn depends upon the type of road. The assumptions made for the different values for $\underline{x}$ and the corresponding values for $\underline{I(x)}$ are tabulated below.

Type of Road	$x(ft)$	$I(x)(ft^{-1})$
Freeway	50	2.9×10^{-2}
4-Lane	30	4.8×10^{-2}

The following additional assumptions are made:

1. All rural and suburban truck travel is on freeways.
2. Urban travel may be on freeways or 4-lane roads.
3. The traffic count doubles during the commuter rush periods (applicable in urban and suburban population zones).
4. The average speeds decrease by a factor of 2 during commuter rush periods (applicable in urban and suburban population zones).
5. Urban travel on freeways and 4-lane roads during rush hour is half the average suburban velocity.
6. Urban travel on freeways during nonrush hours is the average rural velocity. Urban travel on 4-lane roads during nonrush hours is the average suburban velocity.

Under these assumptions, the following expression is obtained for the annual population dose in man-rem per year to persons traveling in a direction opposite to the shipment for a given shipment type:

$$(\text{Dose})_{\text{opp}} = Q \cdot K \cdot \text{SPY} \cdot \text{FMPS} \cdot P \cdot F \quad (15)$$

where:

$$F = f_r \frac{N'_r \cdot I_{\text{fwy}}}{(V_{Tr})^2} + f_s \left(\frac{f_{rh} \cdot 2N'_s \cdot I_{\text{fwy}}}{(V_{Ts}/2)^2} + \frac{f_n \cdot N'_s \cdot I_{\text{fwy}}}{(V_{Ts})^2} \right) \\ + f_u \left[f_{\text{fwy}} \left(\frac{f_{rh} \cdot 2N'_u \cdot I_{\text{fwy}}}{(V_{Ta}/2)^2} + \frac{f_n \cdot N'_u \cdot I_{\text{fwy}}}{(V_{Tr})^2} \right) \right. \\ \left. + f_{4l} \left(\frac{f_{rh} \cdot 2N'_u \cdot I_{4l}}{(V_{Ts}/2)^2} + \frac{f_n \cdot N'_u \cdot I_{4l}}{(V_{Ts})^2} \right) \right]$$

Symbols in the equations are as follows:

f_r, f_s, f_u = fractions of distance traveled in rural, suburban, and urban zones, respectively

f_{rh} = fraction of distance traveled in rush hour traffic

f_n = fraction of distance traveled in normal traffic

f_{fwy} = fraction of travel on freeways or interstates

f_{4l} = fraction of travel on 4-lane roads

V_{Tr} = average velocity on freeways (mph)

V_{Ts} = average velocity on freeways in suburban population density zones and on all 4-lane roads (mph)

$I_{fwy} = I(50 \text{ ft}) = 2.9 \times 10^{-2} \text{ ft}^{-1}$

$I_{4l} = I(30 \text{ ft}) = 4.8 \times 10^{-2} \text{ ft}^{-1}$

$Q = 10^{-3} \frac{\text{rem}}{\text{mrem}} \times \frac{1 \text{ mile}}{5280 \text{ ft}} = 1.89 \times 10^{-7}$

$K = 1000 \text{ mrem/hour}$ for shipments where the dose rate at 6 feet from the vehicle is 10 mrem/hr

The total annual dose to persons traveling in direction opposite to that of the shipment is computed using Equation 15.

C.2.4.2 Dose to Persons Traveling in the Same Direction as the Shipment

On the average, vehicles carrying radioactive material move at the same speed as the rest of the traffic. Thus, vehicles traveling in the same direction as the shipment can be modeled as a static set of vehicles at fixed distances from the shipment. The dose in millirem received by a person located at distance x from the radioactive material may be computed by multiplying the dose rate from Equation 2 by the duration ΔT of the exposure

$$D = \frac{Ke^{-\mu x} B(x)}{x^2} \cdot \Delta T \quad (16)$$

For a given scenario, the total annual exposure time is given by the quotient of total miles per year (miles per shipment $\times$ shipments per year) and average velocity:

$$\Delta T_{\text{ann}} = \frac{\text{FMPS} \cdot \text{SPY}}{V_T} \quad (17)$$

People are assumed to be distributed uniformly along the shipment path with a linear density given by

$$\text{Linear Density (persons/mile)} = \frac{N' \cdot P}{V_T} \quad (18)$$

The annual dose to persons traveling in the same direction as the shipment, for a given scenario, is determined by multiplying the expression for the dose given in Equation 16 by the linear density given in Equation 18 by using Equation 17 for ΔT_{ann} and by integrating over x from some minimum distance d out to a maximum distance "max".

$$(\text{Dose})_{\text{same dir.}} = 2 \cdot \frac{N' \cdot P}{V_T} \cdot \frac{\text{FMPS} \cdot \text{SPY}}{V_T} \cdot K \cdot \int_d^{\text{max}} \frac{e^{-\mu x} B(x)}{x^2} dx \quad (19)$$

The factor of 2 takes into account vehicles ahead of and behind the shipment.

As in the case of persons traveling in the opposite direction, N' and V_T depend upon the population density zone and the time of day (rush hour or normal traffic). Also, the distance d of closest approach depends upon the type of road. The average values selected for d are 100 ft for freeways and interstates and 30 ft for 4-lane roads. Using the same traffic assumptions as made for the calculation of the dose to persons traveling in the direction opposite to that of the shipment, the following expression is obtained for the annual dose (for a given shipment scenario) received by persons traveling in the same direction as the shipment.

$$(\text{Dose})_{\text{same dir.}} = Q' \cdot K \cdot \text{FMPS} \cdot \text{SPY} \cdot P \cdot F \quad (20)$$

where the traffic factor F is the same as that given in Equation 15, except that:

$$I_{\text{fwy}} = I_1 (100 \text{ ft}) = 0.008$$

$$I_4 = I_1 (30 \text{ ft}) = 0.031 \text{ and}$$

$$I_1(d) = \int_d^{2600 \text{ ft}} \frac{e^{-\mu x} B(x)}{x^2} dx$$

The constant Q' is

$$Q' = 2 \times 10^{-3} \frac{\text{rem}}{\text{mrem}} \times \frac{1 \text{ mile}}{5280 \text{ ft}} = 3.79 \times 10^{-7}$$

$K = 1000 \text{ mrem/hr}$ for shipments where the dose rate at 6 feet from the vehicle is 10 mrem/hr

The annual dose is computed for each shipment type by using Equation 20 to obtain the total annual dose to persons traveling along the route in the same direction as the shipment.

C.2.5 Parameters for Normal Transportation Calculations

The parameters of vehicle speed, fraction of travel in each population zone, length of stops, etc., assumed for calculating population doses are shown in Table V-22.

TABLE V-22

Parameters for Calculating Radiation Doses During Normal Transportation

	<i>Truck</i>	<i>Rail</i>	<i>Ship</i>
Number of crewmen (or drivers)	2	5	20
Mean velocity while crew is aboard	51 mph	38 mph	22 ^a mph
Distance from cask to crew	30 ft	500 ft	200
Total time of high-population stops (per 1500-mile trip)	1 hr	0	-
Total time of medium-population stops (per 1500-mile trip)	5 hr	0	-
Total time of low-population stops (per 1500-mile trip)	2 hr	48 hr	-
Fraction of travel in high-population zones	0.05	0.05	-
Velocity in high-population zones	15 mph	15 mph	-
Fraction of travel in medium-population zones	0.05	0.05	-
Velocity in medium-population zones	25 mph	25 mph	-
Fraction of travel in low-population zones	0.90	0.90	0.18 ^b
Velocity in low-population zones	55 mph	40 mph	2-5 mph
Fraction of travel in high-population zones during rush-hour traffic	0.08	-	-
Fraction of travel in high-population zones during normal traffic	0.92	-	-
Fraction of travel on 4-lane roads and freeways	1.0	-	-
Traffic count - vehicles passing a specific point			
- low population	2×10^3	1	-
- medium population	1×10^4	5	-
- high population	2×10^4	5	-
Number of people per vehicle	2	3	-

^a Based on a ship speed of 24 knots at sea.

^b Based on estimate of time for entering and leaving harbors. At sea, other ships seldom are closer than 1/2 mile to the ship carrying casks.

C.3 Assumptions for Calculation of Maximum Individual Dose and Consequences of Transportation

During normal transportation, the maximum dose is received by the transportation crew from direct radiation. For example, during truck shipment, the driver transporting the shipment becomes the maximum individual. The ship and train crewmen transporting radioactive materials receive lower doses. Individuals of the general public who receive the maximum radiation dose are the individuals who live near the rail or truck route. No inhalation or ingestion of radionuclides occurs during normal transportation because no release of radionuclides is encountered.

If the carrier vehicle is involved in a transportation accident, physical injuries, some of which may cause fatalities, frequently occur to drivers, crewmen, or members of the public involved in the accident. In severe to extreme accidents, the neutron shield on spent fuel may be punctured or damaged with a resultant decrease of neutron shielding and moderate radiation exposures. In extreme accidents causing severe damage to the cask, possibly some radionuclides could be released from the cask cavity; but the probability of this occurring is very low. Radiation injury as a result of extreme accidents is highly improbable, but death or serious injury of the crewmen is expected from the impact, crushing, and other forces generated by the accident.

C.3.1 Normal Transportation

The transportation workers who incur the maximum individual doses are truck drivers hauling casks full time and making fifty 2400-km (1500-mi) trips each year. This dose is calculated by using Equation 10 (in C.1 of this section). Train crewmen normally do not stay with the shipment for the whole trip. These crewmen stay at a greater distance from the load during the trip; thus, their dose should be <0.3 times that received by truck drivers. Ship crewmen normally stay away from the load during transport, but inspection during watches requires a crewman to walk by the load. The maximum individual dose per trip by ship is estimated to be about 3 times that of a truck driver transporting spent fuel for 2400 km (1500 mi). However, an individual ship is not likely to transport spent fuel more than 4 or 5 times in one year.

The individual of the general population who incurs the maximum dose is the person who lives beside the highway or an ISFS basin or a disposition facility. The dose per shipment is

calculated by using Equation 2 and by assuming the person lives 31 m (100 ft) from the highway. Using the $I(x)$ value from Figure V-1, the equation becomes

$$\text{Dose/Trip (mrem)} = 8 \times 10^{-6} \frac{K}{V} \quad (21)$$

where:

K = dose rate factor (mrem/hr)

V = average velocity (mph) of the shipment passing that point

For 0.5-yr spent fuel, K is 1000 mrem/hr, and the annual dose to the individual becomes

$$\text{Dose/yr} = \frac{8 \times 10^{-3}}{V} \times \text{total number casks passing/yr} \quad (22)$$

For example, if 1000 truck shipments annually travel at 13 kph (8 mph) on the highway past the residence of an individual located 31 m (100 ft) from the highway, his dose is 1 mrem/yr.

C.3.2 Transportation Accidents

Experience in transportation of spent fuel casks⁵ and full-scale testing of such casks^{3,4} shows that truck drivers will sustain injuries, usually fatal, in severe to extreme accidents necessary to damage spent fuel casks. Expected injuries and fatalities for rail and truck shipments are calculated for miles traveled by loaded casks and emptied casks being returned. Statistics shown in Table V-19 are used. The frequency of fatalities for maritime transport is less than rail and is estimated to be $<6 \times 10^{-8}$ fatalities per trip by containership but statistics for injuries are not available. These statistics include the transportation workers and the general public.

Maximum shielding damage will be loss of neutron shielding. If an accident damages the neutron water shields of a spent fuel cask and the water leaks out, the dose rate in areas that may be occupied is expected to be about 600 mrem/hr or less for aged fuel considered in the GEISs for the Spent Fuel Policy. If a transportation worker or spectator stays near the cask until local or state police or an emergency or radiological assistance team arrives (less than one hour), the dose to the individual would be <600 mrem.

Recent studies and actual tests show that loss of gamma shielding is not credible for the spent fuel casks considered in the GEISs for the Spent Fuel Policy. These casks are designed with a closure consisting of an inner lid providing primary shielding and an outer lid serving as an impact limiter. Recent studies³⁰⁻³⁴ suggest that with this arrangement, the inner lid will remain in place even in extreme accidents such as a truck hitting an unyielding concrete barrier when traveling at a speed of 135 kph (84 mph). Any future casks designed for reprocessing wastes will be designed to meet the appropriate standards and regulations which apply to spent fuel casks.

Any release of radionuclides from a cask subjected to an extra severe or extreme accident is assumed to occur at ground level over a period of about one hour. The individual who incurs the maximum dose is assumed to be 100 m (330 ft) downwind and remains there throughout the release. Meteorological conditions at the time of the accident are assumed to be Pasquill weather Type F. The relative concentration (X/Q) of 3×10^{-2} sec/m³ for the short-term release of radionuclides during these weather conditions is consistent with assumptions described in Regulatory Guide 1.3.⁴⁰

Formulas used to calculate maximum individual dose downwind from such a release are

$$\text{Inhalation Dose (mrem)} = 3.5 \times 10^5 Q \cdot D_c \cdot \frac{X}{Q} \quad (23)$$

where:

Q = curies of the radionuclide released from the cask

D_c = dose conversion factor 50-year dose commitment -
rem/ μ Ci for the radionuclide inhaled - from
ORNL 4992,⁴¹ Table 4-2

$\frac{X}{Q}$ = relative concentration factor (sec/m³)

$$\text{Submersion Dose} = 3.17 \times 10^{-8} Q \cdot R \cdot \frac{X}{Q} \quad (24)$$

where:

Q = curies of the gaseous radionuclide released from
the cask

R = submersion dose factor (mrem/hr Ci of the radionuclide/m³)
from ORNL-4992,⁴¹ Table 3-8

$\frac{X}{Q}$ = relative concentration factor (sec/m³)

C.4 Occupational Exposure - Transportation Workers

The exposures to crewmen on trains or ships and drivers of trucks during transportation operations are classified as occupational exposures. The loading and unloading of casks and waste packages at the shipping or receiving facilities are assumed to be performed by the operating personnel; therefore, the incurred exposure is included in occupational exposure of the operating facilities.

At the U.S. port of entry, casks containing foreign fuel are assumed to be unloaded to truck or rail car by two handlers who attach and detach rigging equipment to transfer the cask from the ship to the land carrier. The handlers are assumed to spend about 15 minutes per cask in an average radiation field of about 20 times the radiation dose rate at 3.1 m (10 ft) from the centerline of the cask. The dose to these handlers and the dose to the ship crew is considered the occupational dose for transport by ship.

C.5 Nonradiological Releases

Combustion of fuel in engines of trucks, locomotives, and ships releases nonradioactive gases and particulates to the environment. Emission factors used to calculate the quantities of pollutants are shown in Table V-23. The emission rates shown are based upon a cask-mile basis for truck and rail. The emission rates shown for the containership are shown for a 6400-MT (7000-ton) net cargo. Four 100-MT (110-ton) spent fuel casks on a ship represent 5.7% of the cargo or four overweight truck casks represent 2.3% of the cargo. These factors are applied to the values in Table V-23 to calculate prorated emissions for carrying rail casks or overweight truck casks.

C.6 Thermal Effects

Thermal discharges during transportation operations are the heat produced during combustion of fuel to drive the carrier vehicles and the radioactivity decay heat radiated from the surfaces of casks. The amount of fuel consumed for cumulative mileages for the transportation modes is based upon fuel consumption rates of $4.7 \times 10^{-4} \text{ m}^3/\text{km}$ (0.20 gal/mi) for trucks, $5.9 \times 10^{-4} \text{ m}^3/\text{cask km}$ (0.25 gal/cask mi) for rail and for ship, 0.019 m^3 (5.0 gal) per nautical mile for a cargo of four 100-MT (110-ton) rail casks or 0.0076 m^3 (2.0 gal) per nautical mile for a cargo of four 40-MT (44-ton) casks. The heat of combustion of diesel fuel is $10,730 \text{ kW-hr/m}^3$ (138,700 Btu/gal) of diesel fuel consumed.

TABLE V-23

Emission Factors for Operation of Transportation Equipment

	<i>Truck</i> <i>lb/mt^a</i>	<i>Rail</i> <i>lb/cask car mile^b</i>	<i>Ship</i> <i>lb/nautical mile^c</i>
Particulates	2.6×10^{-3}	6.3×10^{-3}	9.3
Sulfur Oxides (SO _x as SO ₂)	5.4×10^{-3}	1.4×10^{-2}	7.0
Carbon Monoxide	4.5×10^{-2}	3.3×10^{-2}	5.6
Hydrocarbons	7.4×10^{-3}	2.4×10^{-2}	4.2
Nitrogen Oxides (NO _x as NO ₂)	7.4×10^{-2}	9.3×10^{-2}	6.5
Aldehydes (as HCHO)	6.0×10^{-4}	1.4×10^{-3}	0.32
Organic Acids	6.0×10^{-4}	1.8×10^{-3}	-

a. Fuel consumption rate of 0.20 gallons per mile⁴² and emission rates for diesel trucks in Reference 43.

b. Fuel consumption for the average train of 70 cars is 6.7 gallons per mile.⁴² The weight of a spent fuel cask and its car is 2.6 times the average car weight and on a prorated weight basis the diesel fuel consumption per cask car mile is 0.2 gallons. Emission rates for the average train are from Reference 43.

c. Fuel consumption rate of 88 gal/nautical mile is assumed for the 20,000-ton (dead weight) containership which consumes 190 tons of fuel/day at a speed of 24 knots. The emission factors shown are prorated from EPA values for a ship consuming 19 gallons of fuel per nautical mile.⁴³

The decay heat released for casks containing spent fuel are small in comparison to the heat released from combustion of fuel during the transport of the casks. The decay heat released to the environment from spent fuel casks was calculated from portal to portal by assuming the following transit times between the shipping and receiving facilities.

- 160 km (100 mi) (truck) 3 hours
- 1600 km (1000 mi) (truck) 26 hours
- 1600 km (1000 mi) (rail) 160 hours
- 2400 km (1500 mi) (truck) 38 hours
- 2400 km (1500 mi) (rail) 240 hours
- 6300 nautical miles (ship) 13.5 days
- 7300 nautical miles (ship) 15.5 days

REFERENCES FOR SECTION V

1. *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes*, Vol. 1. NUREG 0170, Docket No. PR-71, 73 (40 FR 23768), United States Nuclear Regulatory Commission, Washington, DC (December 1977).
2. *Statistical Abstracts of the United States 1974*. 95th Annual Edition, United States Department of Commerce Social and Economic Statistics Division, United States Bureau of the Census, Washington, DC (1974).
3. "Projections of the Population of the United States: 1977-2050." United States Bureau of the Census, *Current Population Reports*, Series II, United States Government Printing Office, Washington, DC (1977).
4. *Light Water Reactor Fuel Reprocessing and Recycling*. USERDA Report ERDA 77-75 (July 1977).
5. J. W. Langhaar. "Transport Experience with Radioactive Materials." In *Proceedings of the International Symposium on the Management of Wastes from the LWR Fuel Cycle*, p 469, CONF-76-0701, Denver, CO (July 1976).
6. *Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants*. USAEC Report WASH-1238, Washington, DC (December 1972).
7. A. W. Grella. "A Review of Five Years Accident Experience in the U. S. A. Involving Nuclear Transportation." Presented at the *International Seminar on the Design, Construction, and Testing of Packaging for the Safe Transport of Radioactive Materials*, IAEA-SR-10/5, Vienna, August 23-27, 1976.
8. W. C. McCluggage. "The AEC Accident Record and Recent Changes in AEC Manual Chapter 0529." In *Proceedings of the Third International Symposium on Packaging and Transportation of Radioactive Materials*, p 108, BNWL-SA-3906, Richland, WA (August 1971).
9. *Technology for Commercial Waste Management*. USDOE Report ERDA 77-85 (Draft), Battelle Pacific Northwest Laboratories, Richland, WA (December 1977).

10. *Accidents of Large Motor Carriers of Property.* Bureau of Motor Carrier Safety, Federal Highway Administration, United States Department of Transportation, Washington, DC (Published annually).
11. *Summary and Analysis of Accidents on Railroads in the United States.* Office of Safety, Federal Railroad Administration, United States Department of Transportation, Washington, DC (Published annually).
12. *Sixth Annual Report of the Secretary of Transportation on Hazardous Materials Control, Calendar Year 1975.* Washington, DC (1975).
13. T. A. McSweeney, et al. *An Assessment of the Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Truck.* BNWL-1846, Battelle, Pacific Northwest Laboratories, Richland, WA (August 1975).
14. *Transportation Accident Risks in the Nuclear Power Industry, 1975-2020.* EPA-520/3-75-023, United States Environmental Protection Agency, Washington, DC (1975).
15. W. A. Brobst. "Transportation Accidents: How Probable?" *Nuclear News*, 16 (7), 48 (May 1973).
16. V. U. Minorsky. *Probability of Collisions on Selected Routes.* MA-RD-920-77021, George G. Sharp, Inc., New York, NY (September 30, 1976).
17. V. U. Minorsky, C. W. Parker, and J. C. Gotimer. "Ship Accident Studies." *Symposium on Safety of Nuclear Ships* (proceedings to be published by OECD Nuclear Energy Agency). Hamburg, West Germany (December 1977).
18. D. T. Mallett. *Report on Fire/Explosion Probability Study.* MA-RD-920-77042, George G. Sharp, Inc., New York, NY (November 1976).
19. *Code of Federal Regulations, Title 49, Parts 171.15, 174.566, and 177.861.* United States Government Printing Office, Washington, DC (1975).
20. *Interagency Radiological Assistance Plan.* ERDA-10, Energy Research and Development Administration, Washington, DC (April 1975).
21. *Energy Research and Development Administration Radiological Assistance Plan.* USERDA Report ERDA-60, Division of Operational Safety, Washington, DC (July 1975).

22. L. M. Arnett. *Confidence Limits for Parameters of Poisson and Binomical Distribution*. USERDA Report DP-1416, Savannah River Laboratory, E. I. du Pont de Nemours and Company, Aiken, SC (April 1976).
23. O. A. Kelly and W. C. Stoddart. *Highway Vehicle Impact Studies: Tests and Mathematical Analyses of Vehicle, Package and Tiedown Systems Capable of Carrying Radioactive Material*. USAEC Report ORNL-NSIC-61, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN (February 1970).
24. L. B. Shappert. *A Guide for the Design and Operation of Shipping Casks for Nuclear Application*. Cask Designers Guide, USAEC Report ORNL-NSIC-68, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN (February 1970).
25. A. E. Spaller. *Structural Analysis of Shipping Casks, Resistance to Puncture*, Vol. 2. USAEC Report ORNL-TM-1312, Oak Ridge National Laboratory, Oak Ridge, TN (September 1966).
26. L. B. Shappert and J. H. Evans. *The Obsolete Cask Program: Initial Tests*, Vol. 14. USAEC Report ORNL-TM-1312, Oak Ridge National Laboratory, Oak Ridge, TN (November 1974).
27. G. P. Wachtell. "Drop and Fire Tests of Half-Scale Model of TN-9 Shipping Container." In *Proceedings of the Fourth International Symposium on Packaging and Transportation of Radioactive Materials*, CONF-740901, Miami Beach, FL (September 1974).
28. S. Aoki, et al. "An Experimental Study on the Drop Impact of a Spent Fuel Shipping Cask." In *Proceedings of the Third International Symposium on Packaging and Transportation of Radioactive Materials*, CONF-710801, Richland, WA (August 1971).
29. L. B. Shappert, et al. *The Obsolete Cask Program: Test Number 2*, Vol. 16. USAEC Report ORNL-TM-1312, Oak Ridge National Laboratory, Oak Ridge, TN (April 1975).
30. L. B. Shappert, et al. "The Full-Scale Cask Program at ORNL." *International Seminar on the Design, Construction and Testing of Packaging for the Safe Transport of Radioactive Materials*, IAEA-SR-10, Vienna, Austria (August 1976).
31. H. R. Yoshimura and M. Huerta. *Full-Scale Tests of Spent Nuclear Fuel Shipping Systems*. USERDA Report SAND-76-5707, Sandia Laboratories, Albuquerque, NM (July 1976).

32. A. W. Dennis. *Analytical Investigation of a Grade-Crossing Accident Between a Railroad Train and a Spent Reactor Fuel Cask*. USAEC Report SAND-74-0317, Sandia Laboratories, Albuquerque, NM (January 1975).
33. Robert M. Jefferson. "Cask Demolishes Train Superstructure - Sustains Minor Damage." *Nuclear Engineering International* 22 (259) (July 1977).
34. J. M. Jefferson and H. R. Yoshimura. *Crash Testing of Nuclear Fuel Shipping Containers*. USERDA Report SAND-77-1462C, Sandia Laboratories, Albuquerque, NM (August 1977).
35. *Design and Analysis Report - IF-300 Shipping Cask*. Docket No. 71-9001, General Electric Company, San Jose, CA (January 1971).
36. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." *Safety Guide 25 (Regulatory Guide 1.25)*. United States Nuclear Regulatory Commission, Washington, DC (March 23, 1972).
37. T. H. Smith and W. A. Ross. *Impact Testing of Vitreous Simulated High-Level Waste in Canisters*. BNWL-1903, Battelle Pacific Northwest Laboratories, Richland, WA (May 1975).
38. R. M. Wallace and J. A. Kelley. *An Impact Test for Solid Waste Forms*. USERDA Report DP-1400, Savannah River Laboratory, E. I. du Pont de Nemours and Company, Aiken, SC (March 1976).
39. J. M. Taylor and S. L. Daniel. *RADTRAN: A Computer Code to Analyze Transportation of Radioactive Material*. USERDA Report SAND 76-0243, Sandia Laboratories, Albuquerque, NM (April 1977).
40. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." *Regulatory Guide 1.3, Rev. 2*. United States Nuclear Regulatory Commission, Washington, DC (1974).
41. G. G. Killough and L. R. McKay. *A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment*. USAEC Report ORNL-4992, Oak Ridge National Laboratory, Oak Ridge, TN (March 1976).
42. D. B. Shonka, et al. *Transportation Energy Conservation Data Book (Second Edition)*. USDOE Report ORNL-5320, Oak Ridge National Laboratory, Oak Ridge, TN (October 1977).
43. *Compilation of Air Pollution Emission Factors (Second Edition)*. U.S. Environmental Protection Agency, Washington, DC (April 1973).

VI. RADIOLOGICAL HEALTH EFFECTS

Present radiation standards are based upon the assumptions that any human exposure to ionizing radiation results in some detriment to health. Some of the detriment occurs to the individual exposed (somatic effects) and some to the descendants of the individual exposed (genetic or hereditary effects). Where exposures to radiation are small (in comparison to natural background), the detriment to health is assumed to be proportional to the collective dose equivalent commitment. The collective dose commitment is usually referred to as population dose commitment and is expressed in man-rems or person-rems. Proportionality of dose to effect is based upon an assumption of linearity, with no threshold, between risk of a health effect and dose equivalent. This assumption was used by the National Academy of Sciences - National Research Council in the BEIR Report¹ in assessing the radiological effects of ionizing radiation. The Environmental Protection Agency (EPA), in a policy statement² issued in 1975, also used the assumption of linearity - no threshold in assessing the effects of radiation on populations. The following excerpts from the EPA policy statement concern the uncertainties and limitations that must be considered when a linear dose-effect relationship is used.

Although much is known about radiation dose-effect relationships at high levels of dose, a great deal of uncertainty exists when high-level dose-effect relationships are extrapolated to lower levels of dose, particularly when given at low dose rates. These uncertainties in the relationships between dose received and effect produced are recognized to relate, among many factors, to differences in quality and type of radiation, total dose, dose distribution, dose rate, and radiosensitivity, including repair mechanisms, sex, variations in age, organ, and states of health. These factors involve complex mechanisms of interaction among biological chemical, and physical systems, the study of which is part of the continuing endeavor to acquire new scientific knowledge.

Because of these many uncertainties, it is necessary to rely upon the considered judgments of experts on the biological effects of ionizing radiation. These findings are well-documented in publications by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), the National Academy of Sciences (NAS), the International Commission on Radiological Protection (ICRP), and the National

Council on Radiation Protection and Measurements (NCRP), and have been used by the Agency in formulating a policy on relationship between radiation dose and effect.

It is the present policy of the Environmental Protection Agency to assume a linear, nonthreshold relationship between the magnitude of the radiation dose received at environmental levels of exposure and ill health produced as a means to estimate the potential health impact of actions it takes in developing radiation protection as expressed in criteria, guides, or standards. This policy is adopted in conformity with the generally accepted assumption that there is some potential ill health attributable to any exposure to ionizing radiation and that the magnitude of this potential ill health is directly proportional to the magnitude of the dose received.

In adopting this general policy, the Agency recognizes the inherent uncertainties that exist in estimating health impact at the low levels of exposure and exposure rates expected to be present in the environment due to human activities, and that at these levels, the actual health impact will not be distinguishable from natural occurrences of ill health, either statistically or in the forms of ill health present. Also, at these very low levels, meaningful epidemiological studies to prove or disprove this relationship are difficult, if not practically impossible, to conduct.

The position of the National Council on Radiation Protection³ is the following:

The linear dose-effect hypothesis has been coming into frequent use in analyses in which population exposures are expressed in the form of person-rem, including doses of 1 mrem/yr or less to population groups and doses to individual organs, with linear extrapolation to damage estimates through the use of the NAS-BEIR Committee report¹ values. The indications of a significant dose rate influence on radiation effects would make completely inappropriate the current practice of summing of doses at all levels of dose and dose rate in the form of total person-rem for purposes of calculating risks to the population on the basis of extrapolation of risk estimates derived from data at high doses and dose rates.

The NCRP wishes to caution governmental policy-making agencies of the unreasonableness of interpreting or assuming 'upper limit' estimates of carcinogenic risks at a low radiation level, derived from linear extrapolation from data obtained at high doses and dose rates as actual risks, and of basing

unduly restrictive policies on such an interpretation or assumption. The NCRP has always endeavored to ensure public awareness of the hazards of ionizing radiation, but it has equally determined to ensure that such hazards are not greatly overestimated. Undue concern, as well as carelessness with regard to radiation hazards, is considered detrimental to the public interest.

The NCRP position regarding the use of the linear dose-effect extrapolations stems from the NCRP concern that government policy-making agencies may use the assumption as a basis for developing unduly restrictive policies. Despite the admonitions of the NCRP, dose-effect factors are a useful method for comparing the effects of actions involving radiation exposure to humans. To this end, the EPA dose-effect factors⁴ are used to calculate health effects in the generic environmental statement on the spent fuel policy, justified on pragmatic rather than strictly scientific grounds. The probability that the effects are lower than the calculated values must be recognized. The major value of this approach is that it provides a means of comparing relative effects of different actions; calculated health effects of specific actions are in no way to be construed as accurate or absolute effects of that action.

Latent somatic effects from ionizing radiation were estimated in the *Reactor Safety Study*.⁵ The possibility that both dose and dose rate are important considerations influencing the selection of dose-rate factors was recognized. This study presents three different estimates of radiation risk: the upper-bound estimate, the central estimate, and the lower-bound estimate. The upper-bound estimate essentially uses the BEIR Report absolute risk* values with minor modifications relating to the G.I. tract, bone, and thyroid gland. The central estimate modifies the upper-bound estimate by correction for risk reduction due to drawing out of dose in space and time and the lesser effectiveness of very small doses. A frequently quoted numerical relationship regarding radiogenic carcinogenesis is that low-dose-rate exposures are about one-fifth as effective as high-dose-rate exposures for the same total exposure. A dose effectiveness factor of 0.2 (for most organs) is used in the *Reactor Safety Study* to obtain the central estimate. The lower-bound estimate recognizes the possibility that radiation effects may be nonlinear at very low doses and that a threshold dose of 10 or 25 rem may occur. Thus, at doses and dose rates encountered in normal operation of nuclear facilities, the dose-effect factors become vanishingly small; the possibility of zero effects is not excluded.

* Absolute risk estimates are based upon the reported number of excess cancer deaths per rem that have been observed in exposed population groups, e.g., Hiroshima, Nagasaki, etc.

The International Commission on Radiological Protection published early reports^{6,7} (predating the BEIR Report) on the evaluation of risks from radiation, to be used in developing radiation protection standards. A recent ICRP report⁸ updates the data published in the earlier reports and recommends radiation dose-risk values to be used in developing protection standards. These ICRP values generally lie between the *Reactor Safety Study* and the EPA values.

The factors used by the EPA (and adopted for use in this report), in the *Reactor Safety Study* (central estimate), and by the ICRP to estimate health effects from man-rem population doses and their estimates of the percent mortality of the various effects are given in Table VI-1.

The EPA factors for fatal cancer given in Table VI-1 were derived by using an average of the BEIR Committee relative risk* estimates and absolute risk estimates; the factors used in the *Reactor Safety Study*⁵ were adapted from the BEIR Committee's absolute risk estimates. The upper-bound (no dose-rate dependence) results in dose-effect factors for fatal cancers, about 50% of the EPA values given in Table VI-1 for lung, bone, and red marrow (leukemia). The central estimate includes (for most organs) a dose-effectiveness factor of 0.2 for dose rates <1 rem/day, thus reducing lung, bone, and red marrow values to those shown in Table VI-1. The thyroid dose conversion factor used in the study is 13.4 fatal cancers/10⁶ man-rem compared with the EPA value of 15 given in Table VI-1. The comparative total cancer deaths per 10⁶ man-rem are 200 (EPA), 135 (upper bound), 58 (central estimate), and 100 (ICRP). References 4, 5, and 8 provide further discussion on relationship of somatic health effects (cancers) to dose, and the interpretive variations should be kept in mind when the health effect estimates are used.

The EPA factor of 200 genetic effects per year for 10⁶ person-rem annual exposure is based upon a relative risk model originally propounded by the NCRP⁹ and adopted in the BEIR Report.¹ This model defines the risk as the dose required to produce a mutation rate equal to that which occurs spontaneously and known as the doubling dose. The BEIR Report concludes that the doubling dose almost surely lies between 20 and 200 rem, and all of the BEIR relative risk estimates are based upon this range of exposure. The BEIR Report provides estimates of genetic effects resulting from 5 rem per generation on a population of one million. The value of 5 rem was chosen because it is equivalent to 170 mrem/yr for a 30-yr generation time. This is also the annual limit for

* Relative risk estimates are based upon the percentage increase per rem in the ambient cancer mortality.

TABLE VI-1

Radiological Dose - Effect Conversion Factors

	<i>Environmental Protection Agency^{4, 10}</i>			<i>Reactor Safety Study Central Estimates⁶</i>			<i>ICRP⁵</i>
	<i>Health Effects /10⁶ man-rem, Cancers</i>	<i>Fatal Health Effects /10⁶ man-rem, Cancers</i>	<i>% Mortality</i>	<i>Health Effects /10⁶ man-rem, Cancers</i>	<i>Fatal Health Effects /10⁶ man-rem, Cancers</i>	<i>% Mortality</i>	<i>Fatal Health Effects^a /10⁶ man-rem, Cancers</i>
Whole Body	400	200	50	116	58	50	100
Thyroid	60	12	20	134	13.4	10	5
Bone	32	16	50	1.8	1.4	80	5
Lung	40	40	100	4.4	4.4	100	20
Red Marrow (Leukemia)	54	54	100	5.7	5.7	100	20
	(Genetic Effects)			(Genetic Effects)			(Genetic Effects)
Gonads	200			258			200

a. Morbidity factors are not given in the ICRP report.

average population exposure proposed by the Federal Radiation Council.¹¹ The BEIR Report concluded that 300 to 7500 genetic effects could be expected for 5×10^6 person-rem (exposure of 5 rem in one generation for one million people). In an effort to express the genetic effects for an annual population exposure of a million man-rem, EPA adjusted the BEIR Report values to 1800 to 45,000 genetic effects per 30×10^6 person-rem (or 10^6 person-rem/yr for a 30-yr generation time). EPA then assumed a 50-yr generation time to estimate genetic effects. These genetic effects ranged from 36 to 900 genetic effects/ 10^6 person-rem of annual exposure. The EPA apparently took the geometric mean of this range to get 180 genetic effects/ 10^6 person-rem of annual exposure and rounded it to 200 genetic effects/ 10^6 person-rem of annual exposure.

The genetic effects considered in deriving genetic dose-effect estimates in the BEIR Report include congenital anomalies, constitutional and degenerative diseases, etc. Because of the seriousness of the genetic effects, e.g., mongolism, the emotional and financial stress would be somewhat similar to death impact. The generic environmental impact statements on the spent fuel policy sums genetic and cancer risk as total health effects.

REFERENCES FOR SECTION VI

1. *The Effects on Populations of Exposure to Low Levels of Ionizing Radiation*, BEIR Report. National Academy of Sciences and National Research Council, Washington, DC (November 1972).
2. *Review of the Current State of Radiation Protection Philosophy*. NCRP Report 43, National Council on Radiation Protection and Measurements, Washington, DC (1975).
3. "EPA Policy Statement on Relationship Between Radiation Dose and Effects." *Federal Register*, 41 FR 28409, Washington, DC (July 9, 1976).
4. *Radiological Quality of the Environment in the United States, 1977*. Report EPA-520/1-77-009, U. S. Environmental Protection Agency, Washington, DC (1977).
5. *An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Reactor Safety Study*, Appendix VI, Report WASH-1400 (NUREG-75/014), U.S. Nuclear Regulatory Commission, Washington, DC (1975).
6. *The Evaluation of Risks from Radiation*. ICRP Publication 8, Pergamon Press, London (1966).
7. *Radiosensitivity and Spatial Distribution of Dose*. ICRP Publication 14, Pergamon Press, London (1969).
8. *Recommendations of the International Commission on Radiological Protection*. ICRP Publication 26, Annals of the ICRP, Vol. 1, No. 3, Pergamon Press, London (1977).
9. "Somatic Radiation Dose for the General Population." Report of the Ad Hoc Committee of the National Committee on Radiation Protection and Measurements, NCRP 1960. *Science* 131: 482-486 (May 6, 1959).
10. *Environmental Analyses of the Uranium Fuel Cycle, Part 1, Fuel Supply*. Report EPA 520/9-73-003-B, U.S. Environmental Protection Agency, Washington, DC (1973).
11. *Background Material for the Development of Radiation Protection Standards*. Report No. 1, Federal Radiation Council, Washington, DC (1960).

APPENDIX -- GLOSSARY OF TERMS AND ABBREVIATIONS

actinide

The series of elements beginning with actinium, atomic number 89, and continuing through lawrencium, atomic number 103.

activation

The process of making a material radioactive by absorption of neutrons, protons, or other nuclear particles.

activation product

A nuclide formed by activation.

activity

Radioactivity or radioactive materials. A measure of the rate at which a material is emitting radiations; usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time. The standard unit of activity is the curie (Ci).

AEC

Atomic Energy Commission (discontinued with formation of ERDA and NRC on January 19, 1975).

aging

Holding radioactive fuel and wastes while short-lived radio-nuclides decay.

alpha emitter

A nuclide which undergoes radioactive decay by emitting an alpha particle, a positively charged particle.

aquifer

A water-bearing layer of permeable rock or soil.

background dose

The levels of ionizing radiation received in man's natural environment, including cosmic rays and radiation from naturally occurring radioactive elements. Background doses within the U.S. vary by a factor of about two, depending upon location.

biosphere

The part of the world in which life can exist, including the lithosphere, hydrosphere, and atmosphere; living beings together with their environment.

biota

The animal and plant life of a region.

burialground

A land area specifically designated for storage or disposal of containers of low-level radioactive solid wastes and obsolete or worn out equipment in shallow land burial.

BWR

Boiling water reactor is a nuclear reactor in which boiling water is used as the coolant.

CANDU

CANada Deuterium Uranium reactor is a nuclear reactor in which natural uranium is used as fuel. It is heavy-water moderated, reflected, and cooled.

canister

A metal container for radioactive solid waste.

cask

A container that provides shielding and containment for the shipment or storage of radioactive material.

cc

Cubic centimeters.

cfm

Cubic feet per minute.

CFR

Code of Federal Regulations, subdivided by Titles and Parts, available from U.S. Government Printing Office, Washington, DC.

10 CFR 100

U.S. Code of Federal Regulations, Title 10, Part 100, Reactor Site Criteria.

cfs

Cubic feet per second.

Ci

Curie(s).

cladding

The outer jacket of a nuclear fuel or target element.

cladding waste

Cladding waste consists of hulls, other hardware components, and residual fines which remain after spent fuel is dissolved in a reprocessing plant. Cladding waste is mostly Zircaloy, Inconel, and stainless steel. The components contain induced activity and some residual radionuclides.

compaction

Reduction in the spacing of racks that hold spent fuel in a water storage basin so that the basin can hold more fuel and still remain subcritical.

Concentration Guide (CG)

The average concentration of a radionuclide in air or water to which a worker or member of the general population may be continuously exposed without exceeding radiation dose standards.

contamination

The deposition of radioactive material on a surface or the presence of fission products in a process stream.

criticality

State of being critical: a self-sustaining neutron chain reaction in which the production of neutrons equals or exceeds the loss of neutrons.

curie

The basic unit used to describe the intensity of radioactivity in sample of material. One curie (Ci) equals 37 billion disintegrations per second.

CW

Cooling water.

dba

"decibels Audio." A unit for measuring noise levels upon which occupational standards are based.

DBE

Design Basis Earthquake. An earthquake that is postulated to be the most severe near any site. The DBE is based upon historical records and is used as a basis for facility and system design.

decay (radioactive)

The spontaneous transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide.

decommissioning

The management or disposition of worn out or obsolete nuclear facilities or contaminated sites. Decommissioning operations remove facilities such as reprocessing plants and ISFS Basins from service and reduce or stabilize radioactive contamination.

decontamination

The selective removal of radioactive material from a surface or from within another material.

deionizer

A metal vessel containing ion exchange resins, used for removing positively or negatively charged ions from water.

depleted uranium

Uranium having a percentage of uranium-235 smaller than the 0.7% found in natural uranium.

DF

An acronym for Decontamination Factor. The ratio of the concentration of a fission product in the feed stream to that of the treated effluent.

diffusion climatology

Use of average local meteorological parameters to predict atmospheric concentrations of releases of material from a specified source.

discharge capability

Describes a schedule for shipping spent fuel from the reactor to leave room in the reactor basin to accommodate the scheduled annual discharge of fuel (from 1/4 to 1/3 of the core load).

disintegration

(Radioactive decay) — the spontaneous transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. The process results in emission of energy and/or mass from the nucleus.

disposal

The planned isolation of radioactive and other waste in a manner that precludes recovery or its placement in a manner which is considered permanent because of no provision for recovery.

disposition facility

An undefined generic facility assumed, in the GEISs on the Spent Fuel Policy, to receive spent fuel from reactor and ISFS basins at some point in the schedule.

disposition mode

The mode of disposing of spent fuel in the GEISs. It could consist of either permanent disposal in a geological repository or fuel reprocessing.

DOE

Department of Energy (created October 1, 1977). Includes former Energy Research and Development Administration which was created January 19, 1975, when the Atomic Energy Commission was abolished.

dose

The amount of absorbed energy imparted to matter, when ionizing radiation passes through that matter, per unit mass of the irradiated material.

dose commitment

The amount of radiation to an individual or population over a stipulated period of time resulting from exposure to a given source.

DOT

Department of Transportation.

enriched uranium

Uranium in which the percentage of the fissionable isotope uranium-235 has been increased above the 0.7% normally found in natural uranium.

EPA

Environmental Protection Agency.

ERDA

Energy Research and Development Administration (includes part of the former AEC). Discontinued with formation of the Department of Energy (DOE) in 1977.

ERDAM

ERDA Manual (for ERDA operations and ERDA contractors).

Federal repository

A U.S. Government-controlled facility to be used for the disposal of nuclear waste.

fertile material

A material, for example uranium-238, not itself a readily fissionable material, which can be converted into a fissionable material (plutonium-239) by irradiation in a reactor.

fission (nuclear)

The spontaneous or neutron-induced splitting of a heavy nucleus into two nuclei of different masses, with the emission of 2 or more neutrons and substantial energy.

fission product

A nuclide produced by fission or from radioactive decay of the nuclide thus formed.

fissionable material

Any material that fissions from neutron absorption.

fly ash

Airborne ash from fuel burning operations.

food chain

A linear sequence of successive utilizations of nutrient energy by a series of plant and animal species.

FRC

Federal Radiation Council (now part of EPA).

frequency

The number of times an event can be postulated to occur, or actually occurs, per unit of time.

FRP

Abbreviation for fuel reprocessing plant.

fuel

(Nuclear reactor) — fissionable material used as the source of energy when placed in a nuclear reactor.

fuel assembly

A grouping of fuel elements which is not taken apart during the charging and discharging of a reactor core.

fuel cycle

The complete series of steps involved in supplying fuel for nuclear reactors. The cycle includes uranium mining and refining, uranium enrichment, fuel element fabrication, irradiation, chemical reprocessing (to recover the fissionable material remaining in the spent fuel), and disposal of radioactive waste. Later steps in the fuel cycle are re-enrichment of the enriched fuel material and refabrication into new fuel elements. In a "stowaway" fuel cycle, spent fuel is not reprocessed to recover usable fuel; spent fuel is treated as waste.

fuel element

The smallest structurally discrete part of a reactor assembly which has nuclear fuel as its principal constituent.

full core reserve

Describes a schedule for shipping spent fuel from the reactor to leave room in the reactor basin to accommodate all the spent fuel contained in the reactor.

full-cost recovery

Includes charges to the user that compensate the government for budgetary spending for capital and operating costs, return on invested capital, and costs to cover unusual hazards, e.g., insurance premiums, premium pay for hazardous work, workmen's compensation, etc.

g
grams

gal
gallons

gamma rays (γ)
High-energy, short-wavelength electromagnetic radiation emitted by a nucleus. Gamma radiation accompanies radioactive decay, neutron capture, and fission.

GAO
General Accounting Office (under the Comptroller General of the United States).

geologic storage
Storage in a repository constructed in a geologic formation.

g/L
grams per liter

gpm
gallons per minute

groundwater
Water that exists or flows in the zone of saturation beneath the land surfaces.

GWe
Gigawatts electric, i.e., one billion (10^9) watts or one thousand megawatts.

half-life
The time in which half the atoms in a radioactive substance spontaneously disintegrate to another nuclear form.

health physics
The science and profession devoted to the protection of man and his environment from unnecessary exposure to ionizing radiation.

heavy metals
All the chemical elements with atomic numbers of 81 or greater, beginning with thallium.

heavy water
Deuterium oxide, D_2O . Water in which hydrogen atoms have been replaced with deuterium atoms.

heavy water reactor
Uses heavy water (D_2O) as moderator for slowing fast neutrons. May use light or heavy water for coolant.

high-efficiency particulate air (HEPA) filter

An air filter designed to achieve 99.97% minimum efficiency in the containment of airborne radioactive particulates of greater than 0.3 micron size.

high-level liquid waste

The aqueous waste resulting from the operation of head end and first-cycle extraction system or equivalent wastes from a process not using solvent extraction in a facility for processing irradiated reactor fuels.

high-level waste (HLW)

High-level liquid waste, or products from solidification of high-level liquid waste obtained from chemical processing of irradiated fuel, and/or irradiated fuel elements, if discarded without processing.

ICRP

International Commission on Radiological Protection.

interim storage

Storage operations for which (a) surveillance and human control are provided and (b) subsequent action involving treatment, transportation, or fuel disposition is expected.

intermediate-level liquid waste

The aqueous waste, other than high-level liquid and cladding waste, from reprocessing plants that require shielding or other protective actions during handling. This includes aqueous wastes from extraction cycle evaporator overhead and miscellaneous waste solutions.

intermediate-level waste (ILW)

Intermediate-level liquid waste or products from solidification of intermediate-level liquid waste obtained from chemical processing of irradiated fuel.

ion

An atom with an electrical charge from either the loss or gain of an electron.

ion exchange

A reversible transfer between ions in solution and different ions contained in or on a crystal or resin without destruction of the crystal.

ISFS

Independent Spent Fuel Storage.

isotope

Any of the two or more forms of the same element, containing the same number of protons but different number of neutrons. The isotopes are chemically similar but have different atomic weights.

kilo

A prefix indicating one thousand (10^3) times the affixed unit, abbreviated "k."

km

kilometers (1 kilometer = 1000 meters or 0.621 mile).

kph

kilometer per hour.

kW-hr

kilowatt-hour. A unit of energy generation or consumption in a given hour.

kW-yr

kilowatt-year. A unit of energy generation or consumption in a given year.

lattice

The geometric arrangement of fuel assemblies.

light water

Normal water (H_2O), as distinguished from heavy water (D_2O).

light water reactor (LWR)

Uses light water (H_2O) as coolant and as the moderator for slowing fast neutrons. Most common types are Pressurized Water Reactors (PWR) or Boiling Water Reactors (BWR).

long-term storage

The status of radioactive waste under control and surveillance and readily retrievable. It is in such a form and location that no further processing or manipulation is considered necessary for a period of time which is very long compared with other periods of time in the nuclear fuel cycle. An example would be storage in a high-quality near-surface storage vault with an expected durability of many decades.

low-level liquid waste

The aqueous waste generated at plants handling radionuclides that are low in radionuclide content and do not require significant shielding. The concentrated low-level waste is recycled into intermediate-level waste.

low-level waste (LLW)

Low-level solid waste is miscellaneous solid waste materials that contain beta-gamma emitters and traces of TRU alpha emitters in concentration <10 nCi/g. This waste is normally made up of ash from incinerated combustible wastes, decontaminated equipment, etc. This waste is normally sent to a burial ground for disposal.

low-level transuranic waste (LLW-TRU)

LLW-TRU is miscellaneous solid waste materials that contain beta gamma emitters and transuranic radionuclides with TRU alpha emitters in concentrations >10 nCi/g. This waste requires long term storage or disposal in a geologic repository.

LWT

Abbreviation for legal weight truck.

m

(1) meter; (2) as prefix, milli. See "milli."

man-rem

The total radiation dose commitment to a given population group; the sum of the individual doses received by a population segment.

maximum permissible concentration (MPC)

The quantity of radioactive material in air, water, etc., per unit volume or weight, from which a human, if continuously exposed, should not sustain appreciable body damage.

meteorology

The science concerned with the atmosphere and its phenomena, especially as related to the weather.

metric ton (MT)

Unit of weight; 1 MT = 1000 kilograms.

mg

milligrams.

micro (μ)

Prefix indicating one millionth (1 microgram = $1/1,000,000$ of a gram or 10^{-6} gram).

milli

Prefix indicating one-thousandth (1 milli = $1/1000$ of a rem or 10^{-3} rem).

millirem

One-thousandth of a rem.

minimum critical mass

The minimum mass of fissionable material that, with a specified geometrical arrangement and material composition, will self-sustain a fission chain reaction.

mL

milliliters.

MM

Modified Mercalli (scale of earthquake intensities).

MW

Megawatt (1 MW = 1 million watts). A unit of the rate of energy production or consumption.

MW-yr

Megawatt-year. A unit of energy generation or consumption in a given year.

moderator

A material, such as water or graphite, used in a reactor to slow down high-velocity fission neutrons.

mph

miles per hour.

mr^{em}

millirems.

MTU

Metric tons of uranium (2200 pounds, or 1000 kilograms).

nano

A prefix indicating 10^{-9} times the affixed unit, abbreviated "n."

natural uranium

Uranium as found in nature. It is a mixture of the fertile uranium-238 isotope (99.3%), the fissionable uranium-235 isotope (0.7%), and a minute percentage of uranium-234.

nautical mile

International unit of linear measure (1852 meters or 6076 feet).

nCi

nanocurie(s).

NCRP

National Council on Radiation Protection and Measurements.

neutron

An uncharged elementary particle with a mass nearly equal to that of the proton. Neutrons sustain the fission chain reaction in a nuclear reactor.

noble gas

A chemically inert gas, e.g., xenon, argon, and krypton.

nonproliferation

Limits the number of nations capable of producing nuclear weapons without limiting worldwide use of nuclear power.

NO_x

An acronym for a mixture of nitrogen oxides in no particular ratio or quantity.

NRC

Nuclear Regulatory Commission (includes the regulatory branch of the former AEC).

nuclear reaction

Neutron reactions with materials that cause fission or transmutation with the simultaneous release of energy and radiation.

nuclear safety

The application of technical knowledge and administrative control to prevent an unplanned, uncontrolled nuclear chain reaction.

nucleus

The positively charged center of an atom.

nuclide

A species of atom characterized by its mass number, atomic number, and nuclear energy state, provided that the mean life in that state is long enough to be observable.

NWTS

National Waste Terminal Storage.

off-gas

Gas released by any process in fuel reprocessing or nuclear waste management.

order of magnitude

A factor of 10.

OSHA

Occupational Safety and Health Act of 1970.

overpack

Secondary (or additional) external containment for packaged nuclear waste.

parapet level

The railing and walkway area directly over the water surface of the spent fuel storage basin.

parent

A precursor radionuclide that upon disintegration yields a specified nuclide, the daughter, either directly or as a later member of a radioactive series.

pCi

picocurie(s)

perched water

Groundwater separated from the main body of groundwater by unsaturated rock or soil.

pico

Prefix indicating one-millionth of a micro unit (1 picocurie = 1/1,000,000 of a microcurie or 10^{-12} curie).

plenum

An enclosure in which a fluid is at a pressure higher than that outside the enclosure.

plutonium

A radioactive element with an atomic number of 94. Its most important isotope is fissionable plutonium-239, produced by neutron irradiation of uranium-238.

pool or pool cell

A concrete chamber filled with water to provide shielding for irradiated fuel elements.

population dose

The summation of radiation exposures received by the member of a population group over a given time period.

ppb

parts per billion

ppm

parts per million

probability

The chance of an event occurring in a unit time, usually expressed as events per year.

rad

Radiation absorbed dose. The basic unit of absorbed dose of ionizing radiation. One rad is equal to the absorption of 100 ergs of radiation energy per gram of matter.

radioactive

Unstable in a manner shown by spontaneous nuclear disintegration with accompanying emission of radiation and particles.

radioactive decay

The spontaneous decrease of a radioactive substance due to disintegration by the emission of particles and radiation.

radioactivity

The spontaneous decay or disintegration of unstable nuclei accompanied by the emission of radiation and particles.

radioisotope

An isotope of an element which decays radioactively.

radionuclide

An unstable nuclide that decays radioactively.

radwaste

Waste containing radioactive contamination.

RBOF

Receiving Basin for Offsite Fuels, a facility at the DOE's Savannah River Plant.

reactor

A device in which a fission chain reaction can be initiated, maintained, and controlled.

rem

A term used in radiation protection to express the effective dose equivalent for all forms of ionizing radiation. It is the product of the absorbed dose in rads and factors related to relative biological effectiveness.

repository

A facility or designated site for storage or disposal of high-level and TRU radioactive wastes.

reprocessing

Dissolving spent reactor fuel to recover useful materials such as thorium, uranium, and plutonium. Other radioactive materials are usually separated and treated as waste.

resin

A synthetic organic polymer that can act as an ion exchanger.

retrievability

Capability to recover waste from interim storage.

risk

The product of an event's frequency and its consequence yielding an estimate of the expected damage rate (e.g., population dose per year) from a specified event.

risk assessment

The evaluation and comparison of several independent or associated risks.

roentgen

A unit of exposure dose of ionizing radiation. It is that amount of gamma or x-rays required to produce ions carrying one electrostatic unit of electrical charge in one cubic centimeter of dry air under standard conditions.

seismicity

The tendency for the occurrence of earthquakes.

separations

Chemical processes used to separate nuclear products from each other.

shielding

The material interposed between a source of radiation and the environment for protection against the danger of radiation. Common shielding materials are concrete, water, and lead.

shipping cask

A specially designed container used for shipping radioactive materials (see cask).

smear

(Swipe) — to wipe a surface with cloth or paper to determine if any loose (smearable) contamination is present.

special nuclear materials

Materials that could be used to make nuclear weapons if they were available in sufficient quantity and purity.

spent fuel

Irradiated nuclear reactor fuel at the end of its useful life.

SSC

Sealed storage cask.

storage

Retention of waste in some type of man-made device.

storage basin

A water-filled, stainless-steel lined pool for the interim storage of spent fuel.

sump

Any low area that receives and contains drainage.

tectonic

Pertaining to structural deformation of the earth's crust.

ton

Unit of weight. 1 ton = 2000 pounds.

tonne

Unit of weight, 1 tonne = 1000 kg = 1 metric ton.

transmutation

A change of one element into another.

transuranic elements

Elements above uranium in the periodic table, that is, with an atomic number greater than 92. All 13 known transuranic elements are radioactive. Examples: neptunium, plutonium, and curium.

transuranic waste (TRU)

Any solid waste material measured or assumed to contain transuranic elements in excess of 10 nCi/g.

tritium

A radioactive isotope of hydrogen containing two neutrons and one proton in the nucleus with atomic weight of 3. It is heavier than deuterium (heavy hydrogen with atomic weight of 2). Tritium (T or ^3H) is used in industrial thickness gages, as a label in tracer experiments, in controlled nuclear fusion experiments, and in thermonuclear weapons.

TRU

See transuranic waste.

unstable

Chemical: compounds which readily decompose or change into other compounds.

Radioactive: nuclides which decay to form other nuclides and emit radiation in the process.

uranium

A natural radioactive element with the atomic number of 92 and an atomic weight of approximately 238. The two principal naturally occurring isotopes are the fissionable uranium-235 (0.7% of natural uranium) and the fertile uranium-238 (99.3% of natural uranium).

USAEC

United States Atomic Energy Commission (see AEC).

USGS

United States Geological Survey.

vittrification

The process of forming an amorphous as opposed to crystalline structure — a glassing process.

waste immobilization

Process of converting waste to a stable, solid form which ties up the radionuclides thereby preventing (or slowing) their migration to the biosphere.

waste management

The planning, execution, and surveillance of essential functions related to the control of radioactive (and nonradioactive) waste, including treatment, solidification, initial or long-term storage, surveillance, and disposal.

waste, radioactive

Equipment and materials (from nuclear operations) that are radioactive or have radioactive contamination and for which there is no recognized use or for which recovery is impractical.

water table

Upper boundary of an unconfined aquifer below which saturated groundwater occurs; defined by the levels at which water stands in wells that barely penetrate the aquifer; the water surface in an unconfined aquifer at which the pressure is atmospheric.

zeolite

Any of various hydrous silicates that can act as ion exchangers.

μ

Prefix indicating one millionth. Same as "micro."