Environmental Control Aspects for Fabrication, Reprocessing and Waste Disposal of Alternative LWR and LMFBF Fuels

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November 1979

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CONTENTS

LIST OF FIGURES	vi
LIST OF TABLES	vii
1.0 INTRODUCTION	1.1
2.0 SUMMARY	2.1
2.1 FUEL REPROCESSING PLANT	2.2
2.2 FUEL FABRICATION PLANTS	2.2
2.3 WASTE REPOSITORY	2.6
3.0 DESCRIPTION OF FUEL CYCLES	3.1
3.1 ALTERNATIVE LWR FUEL CYCLES	3.1
3.2 LMFBR FUEL CYCLES	3.3
4.0 CHARACTERIZATION OF FUELS	4.1
4.1 CHARACTERIZATION OF FUELS AT FABRICATION	4.1
4.1.1 Technical Basis	4.1
4.1.2 Alternative LWR Fuels	4.1
4.1.3 Alternative LMFBR Fuels	4.2
4.1.4 Fuel Characteristics	4.2
4.2 CHARACTERIZATION OF FUELS - REPROCESSING	4.4
4.2.1 LWR Analysis	4.5
4.2.2 LMFBR Analysis	4.7
4.2.3 Spent Fuel Characterization	4.7
4.3 CHARACTERIZATION OF FUEL WASTES AT DISPOSAL	4.10
4.3.1 High-Level Liquid Waste	4.10
4.3.2 Intermediate Level Liquid Waste	4.11

4.3.3	Activated Fuel Residue	4.12
4.3.4	Heavy Metal Oxide Waste	4.12
5.0	FUEL CYCLE FACILITIES: WASTE TREATMENT AND EFFLUENT CONTROL	5.1
5.1	REPROCESSING	5.1
5.1.1	FRP Process Description	5.1
5.1.2	FRP Waste Treatment and Effluent Control Systems	5.2
5.2	FUEL FABRICATION AND REFABRICATION	5.10
5.2.1	Process Description	5.10
5.2.2	Waste Treatment Systems	5.11
5.3	WASTE DISPOSAL	5.12
6.0	COMPARISON OF ENVIRONMENTAL IMPACTS OF ALTERNATIVE FUELS	6.1
6.1	BASES FOR COMPARISON	6.1
6.1.1	Comparison Method for FRP, FRFP and FFP Effluents	6.1
6.1.2	Comparisons Method for the Waste Repository	6.2
6.2	COMPARISONS OF ALTERNATIVE FUELS	6.3
6.2.1	Relative Dose Commitments for the FRP	6.3
6.2.2	Relative Dose Commitments for the FRP	6.7
6.2.3	Relative Dose Commitments for the PFRFP	6.7
6.2.4	Relative Dose Commitments for the UFRFP	6.9
6.2.5	Comparison of Hazard Indices for Waste Repository Releases	6.9
7.0	REFERENCES	7.1
	APPENDIX A - LWR FUEL CYCLE CHARACTERISTICS	A.1

APPENDIX B - FRESH FUEL COMPOSITIONS	.	.	.	.	.	.	.	B.1
APPENDIX C - SPENT FUEL COMPOSITIONS	.	.	.	.	.	.	.	C.1
APPENDIX D - WASTE COMPOSITONS	.	.	.	.	.	.	.	D.1
APPENDIX E - MISCELLANEOUS DATA	.	.	.	.	.	.	.	E.1

FIGURES

4.1	LWR Analysis	4.6
4.2	LMFBR Analysis	4.8
5.1	Flow Diagram for Emissions from FRP Waste Treatment Systems	5.3

TABLES

2.1	Relative 50-Year Total Body Dose Commitments per MTHM for a One-Year Release	2.3
2.2	Relative 50-Year Bone Dose Commitments per MTHM for a One-Year Release	2.4
2.3	Relative Hazard Indices for Waste Repository Effluents per MTHM Charged to Reactor	2.5
3.1	LWR Alternative Fuel Cycle Characterization	3.2
3.2	LMFBR Design Parameters	3.4
3.3	Annual Steady-State LMFBR Fuel Quantities	3.5
4.1	Alternative LWR Fuel Types	4.2
4.2	LMFBR Fuel Regions	4.3
4.3	Compositions and Enrichments of Alternative Fuels	4.3
4.4	Elemental Compositions of Alternative Fuels	4.4
4.5	LWR Alternative Fuel Burnups	4.5
4.6	LMFBR Fuel Burnup and Core Residence Times	4.5
4.7	Spent Fuel Heavy Element Compositions	4.9
4.8	Tritium Produced in Alternative Fuels	4.9
5.1	Fuel Reprocessing Plant Primary Waste Stream	5.4
5.2	FRP Waste Treatment System Performance	5.6
5.3	Fuel Reprocessing Plant Overall Decontamination Factors for Some Important Nuclides	5.8
5.4	Fabrication of Alternative Fuels	5.11
6.1	Relative Dose Commitments per MTHM for Fuel Reprocessing Plant Emissions	6.4
6.2	Relative Dose Commitments per MWd for Fuel Reprocessing Plant Emissions	6.6

6.3	Relative Dose Commitment per MTM for Fuel Fabrication Plant Emissions	6.8
6.4	Relative Dose Commitment per MWd for Fuel Fabrication Plant Emissions	6.9
6.5	Relative Dose Commitments per MTHM for Plutonium-Fuel Refabrication Plant Emissions	6.10
6.6	Relative Dose Commitment per MWd for Plutonium Fuel Refabrication Plant Emissions	6.11
6.7	Relative Dose Commitment per MTHM for High Activity Uranium Fuel Refabrication Plant Emissions	6.12
6.8	Relative Dose Commitments per MWd for High Activity Uranium Fuel Refabrication Plant Emissions	6.13
6.9	Waste Repository Effluents: Number of Maximum Permissible Concentrations (MPCs) in Ground Water per MTHM Charged to Reactor	6.14
A.1	LWR Alternative Fuel Cycle Characterization	A.1
A.2	Annual Fuel Quantities (MTHM) for LWR Fuel Cycles at Steady-State	A.2
A.3	Annual Steady-State Resource Requirements and Discard Quantities (MTHM) for LWR Fuel Cycles	A.3
B.1	Fresh Fuel Compositions	B.2
B.2	Assumed Fuel Impurities	B.3
B.3	Mass, Composition, and Exposure of Fuel Assembly Structural Materials	B.4
B.4	Structural Material Compositions	B.5
C.1	Radioactive Fission Product Inventory (Ci/MTHM) in Spent Fuel--1.5 Years Decay Time	C.2
C.2	Radionuclide Inventory (Ci/MTHM) of Actinides and Daughters in Spent Fuel--1.5 Year Decay Time	C.4
C.3	Radionuclide Inventory (Ci/MTHM) of Activated Fuel Impurities--1.5 Years Decay Time	C.5

C.4	Radionuclide Inventory (Ci/MTHM) in Activated Standard Materials--1.5 Years Decay Time	C.6
D.1	Radionuclide Inventory (Ci/MTHM Charged to Reactor) in Fuel Reprocessing Plant Wastes Stored in Repository	D.2
D.2	Radionuclide Inventory (Ci/MTHM Charged to Reactor) in Fuel Fabrication Plant Wastes Stored in Repository	D.5
E.1	Fifty-Year Population Dose Commitment Factors	E.2
E.2	Nuclide Sorption Coefficients for Bedded Salt Site and Maximum Permissible Concentration	E.5

1.0 INTRODUCTION

The Department of Energy's Division of Environmental Control Technology has an ongoing program at Pacific Northwest Laboratory (PNL) to analyze the environmental control aspects of developing nuclear fuel cycles. As part of this program alternative LWR and LMFBF fuel cycles have been analyzed to determine the potential for increased requirements for environmental control of radioactive emissions.

These alternative fuel cycles differ with respect to the types of fuels and the types of facilities used to fabricate, reprocess and refabricate the fuels. The fuel types chosen for this analysis are representative of steady-state alternative LWR and LMFBF fuel cycles. Recent work at PNL (Heeb et al. 1979) was used as the basis for choosing the LWR alternatives, and the LMFBF alternatives are representative of current designs (Haffner et al. 1977). Using the fuel type as a basis, the dose commitments and hazard indices (using current technologies) of fabrication, reprocessing, and waste disposal operations are estimated.

Detailed descriptions of the LWR and LMFBF fuel cycles, alternative fuels, fabrication and reprocessing facilities and their waste treatment and control systems and potential dose commitments and hazard indices are presented in the text of the report. Supporting data, including fuel and waste compositions, are contained in the appendixes.

2.0 SUMMARY

Environmental control aspects of alternative fuel cycles have been analyzed by evaluating fabrication, reprocessing, and waste disposal operations. Various indices have been used to assess potential environmental control requirements. For the fabrication and reprocessing operations, 50-year dose commitments were used. Waste disposal was evaluated by comparing projected nuclide concentrations in ground water at various time periods with maximum permissible concentrations (MPCs).

Three different fabrication plants were analyzed, which process the variety of fuel compositions: 1) a fuel fabrication plant (FFP) to produce low-activity uranium and uranium-thorium fuel rods; 2) a plutonium fuel refabrication plant (PFRFP) to produce plutonium-uranium and plutonium-thorium fuel rods; and 3) a uranium fuel refabrication plant (UFRFP) to produce fuel rods containing the high-activity isotopes ^{232}U and ^{233}U . A 50-year dose commitment resulting from annual operation of each plant is used as the figure of merit for comparison. Each plant's dose commitments are discussed separately because of the different degrees of effluent control found for each plant type.

Source terms for the analysis of effluents from the fuel reprocessing plant (FRP) were calculated using the fuel burnup codes LEOPARD, CINDER and ORIGEN. Effluent quantities are estimated for each fuel type and comparisons are made on the basis of 50-year dose commitments resulting from the annual effluent releases.

The geological medium chosen for the waste repository analysis was bedded salt. The repository site is modeled on the Waste Isolation Pilot Program (WIPP) site in New Mexico. Wastes assumed to be stored in the repository include high-level vitrified waste from the FRP, packaged fuel residue (hulls and hardware) from the FRP, and transuranic (TRU) contaminated wastes from the FFP, PFRFP, and UFRFP. The potential environmental significance was determined by estimating the ground-water concentrations of the various nuclides over a time span of a million years. The maximum permissible concentration (MPC) for

each nuclide was used along with the estimated ground-water concentration to generate a biohazard index for the comparison among fuel compositions.

Tables 2.1, 2.2, and 2.3 summarize the comparisons for fabrication, reprocessing, and waste disposal. The comparisons are presented as ratios of the dose commitment or hazard index for each fuel type to that of the appropriate reference fuel type.

2.1 FUEL REPROCESSING PLANT

As can be seen in Table 2.1 the total body dose commitments from the reprocessing of alternative fuels are all about the same or less than the commitment from reference LEU fuels. Since the control technologies are assumed to be the same, the variations result from differences in the effluent compositions for the core regions (i.e., excluding LMFBR blanket regions). All of the alternatives have approximately the same impacts for whole body doses. The LMFBR cores have slightly less impact, mainly because of the reduced amounts of tritium present in the spent fuel.

For the bone dose comparisons shown in Table 2.2, the fuels higher in ^{233}U series actinides produce significantly larger doses than other fuels. These values are calculated assuming the use of current technologies; the size of the doses implies that more stringent controls would be needed for an FRP processing these fuels.

2.2 FUEL FABRICATION PLANTS

As noted earlier three fabrication facilities were analyzed in this study. The FFP facility fabricates LEU and thorium fuels. It requires the least stringent environmental controls of all fabrication operations. For the postulated alternative fuels, those containing thorium have the longest environmental effect (see Table 2.1).

For the PFRFP the dose commitments increased in rough proportion to the concentration of plutonium in the fuel. Similarly the impact from the UFRFP

TABLE 2.1. Relative 50-Year Total Body Dose Commitments per MTHM for a One-Year Release^(a)

Fuel Cycle Facility	LWR FUEL Type ^(b)								LMFBR Fuel Type ^(b)				
	LEU	ReLEU	Du-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U-Axial Blanket	Th Radial Blanket
Reprocessing Plant	1.0 ^(c)	1.1	0.97	1.1	0.98	1.1	1.1	1.0	0.73	0.60	0.061	0.041	0.030
Fuel Fabrication Plant ^(d)	1.8	-(e)	6.6	-	7.6	-	-	-	-	-	0.18	-	6.8
Fuel Refabrication Plant-Pu fuels	-	-	-	-	-	-	1.0 ^(c)	1.8	1.0	1.4	-	2.8x10 ⁻⁸ ^(f)	-
Fuel Refabrication Plants- ²³³ U fuels	-	1.0 ^(c)	-	2.6	-	3.0	-	-	-	-	-	-	-

(a) Each entry is ratio of fuel type's dose commitment to that of reference fuel type.

(b) LWR Fuels

LEU is low enriched uranium oxide fuel containing some recycled uranium.

ReLEU is low enriched uranium oxide fuel containing high-activity recycled uranium.

DU-Th is denatured uranium-thorium oxide fuel.

ReDU-Th is denatured uranium-thorium oxide fuel containing high-activity recycled uranium.

HEU-Th is highly enriched uranium-thorium oxide fuel.

ReHEU-Th is highly enriched uranium-thorium oxide fuel containing high-activity recycle uranium.

Pu-U is plutonium-uranium oxide fuel.

Pu-Th is plutonium-thorium oxide fuel.

LMFBR Fuels

Inner Core is plutonium-uranium oxide fuel.

Outer Core is plutonium-uranium oxide fuel.

U Radial Blanket is depleted uranium oxide.

U Axial Blanket is depleted uranium oxide.

Th Radial Blanket is thorium oxide.

(c) Reference fuel type for comparison. All comparisons are for equivalent control technologies.

(d) Reference fuel type for comparison is once-through (no recycle) LEU.

(e) Dash indicates that this fuel type is not fabricated at this plant.

(f) This fuel type is fabricated as an integral part of the core fuel elements.

TABLE 2.2. Relative 50-Year Bone Dose Commitments per MTHM for a One-Year Release^(a)

Fuel Cycle Facility	LWR Fuel Type ^(b)								LMFBR Fuel Type ^(b)				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
Reprocessing Plant	1.0 ^(c)	31.0	79.0	133.0	96.0	209.0	1.8	37	1.9	2.2	0.10	0.076	16.0
Fuel Fabrication Plant ^(d)	1.9	-(e)	11.0	-	13.0	-	-	-	-	-	0.20	-	12.0
Fuel Fabrication Plant-Pu fuels	-	-	-	-	-	-	1.0 ^(c)	1.7	0.97	1.3	-	1.6x10 ^{-8(f)}	-
Fuel Refabrication Plant- ²³³ U fuels	-	1.0 ^(c)	-	2.6	-	3.1	-	-	-	-	-	-	-

(a) Each entry is ratio of fuel type's dose commitment to that of the reference fuel type.

(b) LWR Fuels

LEU is low enriched uranium oxide fuel containing some recycled uranium.

ReLEU is low enriched uranium oxide fuel containing high-activity recycled uranium.

DU-Th is denatured uranium-thorium oxide fuel.

ReDUTH is denatured uranium-thorium oxide fuel containing high-activity recycled uranium.

HEU-Th is highly enriched uranium-thorium oxide fuel

ReHEU-Th is highly enriched uranium-thorium oxide fuel containing high-activity recycled uranium.

Pu-U is plutonium-uranium oxide fuel.

Pu-Th is plutonium-thorium oxide fuel.

LMFBR Fuels

Inner Core is plutonium-uranium oxide fuel.

Outer Core is plutonium-uranium oxide fuel.

U Radial Blanket is depleted uranium oxide.

U Axial Blanket is depleted uranium oxide.

Th Radial Blanket is thorium oxide.

(c) Reference fuel type for comparison.

(d) Reference fuel type for comparison is once-through (no recycle) LEU.

(e) A dash indicates that this fuel type is not fabricated at this plant.

(f) This fuel type is fabricated as an integral part of the core fuel elements.

TABLE 2.3. Relative Hazard Indices for Waste Repository Effluents per MTHM Charged to Reactor(a)

Time After Generation of Waste (Years)	LWR Fuel Type(b)								LMFBR Fuel Type(b)				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
2,100	1.0 ^(c)	1.6	1.2	1.6	1.2	1.7	1.3	1.4	5.3	4.0	1.5	0.60	0.55
32,000	1.0 ^(c)	22.0	10.0	46.0	1.1	41.0	11.0	32.0	2.7	2.9	0.25	0.24	20.0
230,000	1.0 ^(c)	0.40	0.57	22.0	0.39	1.9	3.7	6.5	3.4	4.8	0.65	0.082	0.0051

(a) Each entry is ratio of fuel type's hazard index to that of the reference fuel type.

(b) LWR Fuels

LEU is low enriched uranium oxide fuel containing some recycled uranium

ReLEU is low enriched uranium oxide fuel containing high-activity recycled uranium.

DU-Th is denatured uranium-thorium oxide fuel.

ReDU-Th is denatured uranium-thorium oxide fuel containing high-activity recycled uranium.

HEU-Th is highly enriched uranium-thorium oxide fuel.

ReHEU-Th is highly enriched uranium-thorium oxide fuel containing high-activity recycled uranium.

Pu-U is plutonium-uranium oxide fuel.

Pu-Th is plutonium-thorium oxide fuel.

LMFBR Fuels

Inner Core is plutonium-uranium oxide fuel.

Outer Core is plutonium-uranium oxide fuel.

U Radial Blanket is depleted uranium oxide.

U Axial Blanket is depleted uranium oxide.

Th Radial Blanket is thorium oxide.

(c) Reference fuel type for comparison.

increased relative to the ^{232}U concentration. Most of the dose commitment was from ^{232}U and its daughters. For the assumed levels of environmental control the concentration of ^{232}U determined the relative impact.

2.3 WASTE REPOSITORY

Relative hazard indices for the alternative fuels are displayed in Table 2.3. The hazard indices are calculated from projected ground-water concentrations and MPCs for the nuclides present. Several points in time, representing maximum concentrations for important nuclides, are included. The relative hazard indices vary considerably for both fuel type and time. Overall, the LMFBR fuels have a higher hazard per MTHM than the reference LEU fuel. However, other fuel types exhibit significant peaks and all the fuels have a period of relatively high hazard compared to LEU. This analysis has not been exhaustive and only represents an overview of repository impacts. A more detailed comparison would use site-specific geologic and hydrologic data and more refined leach rate models.

3.0 DESCRIPTION OF FUEL CYCLES

The current LWR nuclear fuel cycle is a "once-through" cycle in which no recycling of fuel materials occurs. Fuel elements are fabricated, exposed in a reactor to the required burnup, and discharged to storage.

Alternative fuel cycles are being investigated that would better utilize natural resources by recycling nuclear fuel but reduce the potential for fissile material diversion over the previously proposed mixed uranium-plutonium recycle system. These cycles involve reprocessing spent fuel to recover unused fuel and refabricating the recovered material into fresh fuel elements. Wastes generated by this process would be retired to long-term storage at a waste repository.

The alternative fuel cycles considered in this study are those using LWRs and those involving liquid metal fast breeder reactors (LMFBRs). The LMFBR cycles are assumed to use plutonium produced by LWRs and produce as bred material either ^{233}U or more plutonium.

3.1 ALTERNATIVE LWR FUEL CYCLES

Recent work at PNL has described in some detail nine LWR fuel cycles (Heeb et al. 1979) as alternatives to the current once-through cycle that utilizes low-enriched uranium (LEU) fuel. The alternative fuel cycles were developed in response to two needs: to better utilize resources by recycling nuclear fuels, and to discourage the use of the fissile materials for weapons proliferation. These alternative cycles are used in this study as a basis for environmental release comparisons.

No recycling occurs in the once-through cycle; the fuel's principal fissile isotope is ^{235}U obtained from natural uranium. The spent fuel is retired to waste. The nine alternative fuel cycles all incorporate uranium recycle. Some of the cycles also reuse plutonium, and some use thorium to generate ^{233}U .

In the recent PNL work, fuel mass flows and detailed isotopic information were determined for the nine alternative fuel cycles for a 30-year period.

During this period, a transition was made in each cycle from the use of LEU fuel to the use of a variety of fuel types. At the end of 30 years, each cycle was at or near a steady-state condition. Steady state is reached when the annual energy contribution of each fuel type within a cycle remains constant. At steady state, natural resource requirements and effluent and waste quantities are also constant. The fuel compositions for the last time step of the 30-year analysis were chosen for the present study as representative of steady-state conditions for each cycle.

The alternative fuel cycles use several different types of fuel. Table 3.1 shows the fuel types used in each cycle and indicates whether or not plutonium was recycled in each case.

TABLE 3.1. LWR Alternative Fuel Cycle Characterization

<u>Cycle</u>	<u>Fuel Types Used(a)</u>	<u>Recycle Materials</u>
A	LEU, Pu-U	U and Pu
B	LEU, Pu-Th, Re LEU	U and Pu
C	LEU, Pu-Th, Re DU-Th	U and Pu
D	LEU, Pu-Th, Re HEU-Th	U and Pu
E	HEU-Th, Re HEU-Th	U
F	DU-Th, Re DU-Th	U
G	DU-Th, Re DU-Th, Pu-Th, Pu-U	U and Pu
H	DU-Th, Re DU-Th, Pu-U	U and Pu
I	HEU-Th, Re LEU	U
Once-through	LEU	None

- (a) LEU is low-enriched uranium oxide fuel.
 Re LEU is low-enriched uranium oxide fuel containing recycled high-activity uranium.
 DU-Th is denatured uranium-thorium oxide fuel.
 Re DU-Th is denatured uranium-thorium oxide fuel containing recycled uranium.
 HEU-Th is high-enriched uranium-thorium oxide fuel.
 Re HEU-Th is high-enriched uranium thorium oxide fuel containing recycled uranium.
 Pu-Th is plutonium-thorium oxide fuel.
 Pu-U is plutonium-uranium oxide fuel.

In all the cycles, ^{235}U from natural uranium was the only external source of fissile material. Uranium was recycled until its net reactivity worth was equal to or less than that of enrichment plant tails (0.2% ^{235}U). Thorium was not reused in any cycle.

Plutonium use varied from cycle to cycle. In cycles E, F, and I (see Table 3.1) the generated plutonium was discarded as waste. Cycles A and H recycled the plutonium into a Pu-U oxide fuel containing a near-tails concentration of fissile uranium. In cycles B, C, D, and G, plutonium was reused in a Pu-Th oxide fuel.

Annual refueling was assumed, with one-third of the core replaced each year. Two years were allowed for cooling, reprocessing, and fabrication, and thus a total of 5 years passed before recycled fuel was charged back into the reactor. Total annual energy produced by all the fuels in a given cycle was about 870 GWD. The discharge exposure varied slightly with fuel type, but was about 30 MWd/MTHM.

These fuel cycles differ considerably in resource utilization, proliferation resistance, and cost per unit power generated. Information on mass flows, resource requirements, and discard quantities is presented in Appendix A of this document. Section 4 discusses in detail the characteristics of the various fuels, and Appendix B provides detailed fuel compositions.

3.2 LMFBR FUEL CYCLES

Implementing and using LMFBRs involves the exchange of bred fissile material with LWRs. Plutonium generated in an LWR is postulated to be used to fuel an LMFBR, which in turn could produce plutonium and/or ^{233}U for use in LWRs or other LMFBRs. The characteristics of the LWR fuels that could be involved in this synergistic system are similar to some of the fuels analyzed for the LWR alternative cycles.

Two possibilities for the LMFBR cycle are considered here. Both cycles assume a mixed oxide (Pu-U) core with a depleted uranium axial blanket. The

radial blanket for one cycle, however, is composed of depleted uranium, while the blanket of the other contains thorium.

The core and axial blanket design assumed for this study has been the most prominent proposed LMFBR configuration (General Electric 1976) and represents one of the most efficient breeder designs (on a conversion ratio basis).

Table 3.2 indicates the LMFBR characteristics assumed for this study.

Table 3.3 shows the annual steady-state mass flows for each of the two LMFBR fuel cycles.

TABLE 3.2. LMFBR Design Parameters^(a)

Electric Power (MWe)	1,200
Thermal Power (MWt)	3,736
Core Height (in.)	46.8
Core Radius (in.)	65.3
Axial Blanket Thickness (in.)	13.0
Radial Blanket Thickness (in.)	13.6
Number of Fuel Assemblies	380
Number of Control Assemblies	20
Number of Blanket Assemblies	184
Fuel Pin Diameter (in.)	0.286
Pins per Fuel Assembly	271
Cladding Thickness (in.)	0.012
Fuel Smear Density (% TD)	88
Core Composition (%)	
Fuel	41.43
Structure	14.96
Sodium	43.61
Number of Fuel Pins per Year ^(b)	51,490
Heavy Metal in Core per year ^(b) kg	18,222
Fraction of Core Replaced per Year	0.5
Fraction of Radial Blanket Replaced per year	0.2

(a) From the HEDL study on alternative FBR core designs (Haffner et al. 1977).

(b) 315 full power days assumed.

TABLE 3.3. Annual Steady-State LMFBR Fuel Quantities (MTHM)

	<u>Inner Core</u>	<u>Outer Core</u>	<u>Axial Blanket</u>	<u>Radial Blanket^(b)</u>
<u>Uranium Radial</u>				
<u>Blanket Design</u>				
Charge	9.11	9.10	10.27	7.40
Discharge ^(a)	8.49	8.58	10.24	7.37
<u>Thorium Radial</u>				
<u>Blanket Design</u>				
Charge	9.11	9.10	10.27	6.64
Discharge ^(a)	8.49	8.58	10.24	6.62

(a) Sum of uranium, plutonium, and thorium quantities only.

(b) For comparative purposes the core and axial blanket parameters for the Th radial blanket case are assumed to be the same as in the uranium case. The core performance would vary slightly with different blankets but the difference is negligible for the purposes of this study.

4.0 CHARACTERIZATION OF FUELS

In order to compare the environmental effects of fuel cycle operations for the various alternative fuels, effluents from facility operations must be estimated. The characteristics of the input material to each facility and the process characterization will determine the nature of the waste streams and effluents. This section characterizes the alternative fuel inputs to fabrication, reprocessing, and waste disposal operations. The rationale for the choice of fuels for analysis and the methods used to generate the source terms is included in this discussion.

4.1 CHARACTERIZATION OF FUELS AT FABRICATION

The important properties of the alternative fuels at fabrication are the fresh fuel composition, fuel form, and cladding composition. The selection of fuel types chosen for analysis is discussed here, followed by a brief description of the fuel characteristics.

4.1.1 Technical Basis

The basis for the alternative LWR fuel types is work done for the Fuels Refabrication and Development (FRAD) program as discussed in Section 3.1. For LMFBR fuels, data was obtained from the study on alternative FBR core designs (Haffner et al. 1977).

4.1.2 Alternative LWR Fuels

As discussed in Section 3.1, the nine alternative fuel cycles reach or nearly reach a steady-state condition by the end of 30 years of fuel recycling. There are 34 steady-state fuels in the nine fuel cycles; however, several of these are identical and many others are similar. There are eight basic fuel types among the steady-state fuels. These are listed in Table 4.1.

Composition variations occur within fuel types. The steady-state plutonium-thorium (Pu-Th) fuel from fuel cycle B is somewhat different in composition from that of fuel cycle D. Most of the fuel types have only slight variations from cycle to cycle and the choice for analysis was arbitrary.

TABLE 4.1. Alternative LWR Fuel Types

	<u>Fuel(a)</u>	<u>Abbreviation</u>
1	Low-enriched (3.2%) uranium containing recycled uranium	LEU
2(b)	Low-enriched (2.5%) uranium containing recycled uranium	Re LEU
3	Denatured (20% enriched) uranium in thorium	DU-Th
4(b)	Denatured (4.3% enriched) recycled uranium in thorium	Re DU-Th
5	High-enriched (93%) uranium in thorium	HEU-Th
6(b)	High-enriched (56%) recycled uranium in thorium	Re HEU-Th
7	Plutonium in depleted uranium	Pu-U
8	Plutonium in thorium	Pu-Th
9	Low-enriched (3.0%) uranium (Fresh UO ₂)	Once-through LEU

(a) All fuels are oxides.

(b) These fuels contain ²³³U as well as ²³⁵U.

In the case of recycle denatured uranium-thorium (DU-Th) fuel, however, there is a wide range of composition to be found even within one fuel cycle. Rather than analyze all variations of recycle DU-Th, the fuel whose uranium had been recycled the greatest number of times was used. Thus, the representatives of recycle DU-Th and the nonrecycle DU-Th bracket the entire range of DU-Th compositions to be found in the fuel cycles.

Included for comparison in this analysis is the low-enriched uranium (LEU) fuel of the once-through cycle. This fuel is not reprocessed and contains no recycled materials. Spent fuel of this type is disposed of as waste.

4.1.3 Alternative LMFBR Fuels

The alternative FBR core design study examined a variety of core and blanket compositions. For this study a two-zone core, an axial blanket composition and two radial blanket compositions were chosen (five different regions in all). These regions are described in Table 4.2.

4.1.4 Fuel Characteristics

All nine LWR and five LMFBR fuel types are fabricated as heavy-metal oxide pellets encased in cladding. The material used for cladding and associated

TABLE 4.2. LMFBR Fuel Regions

Region	Fuel Material
Core	
Inner zone	Pu-U oxide
Outer zone	Pu-U oxide
Axial Blanket	Depleted U oxide
Radial Blanket	Th oxide or depleted U oxide

fuel assembly hardware for LMFBR fuels is stainless steel. For LWR fuels, Zircaloy cladding is used, and the hardware is composed of stainless steel, Inconel and Microbrazed. Detailed information on the composition of the structural materials can be found in Appendix B.

Heavy element compositions of each fuel type are presented in Tables 4.3 and 4.4. Table 4.3 shows the fissile isotopes and diluents present in each

TABLE 4.3. Compositions and Enrichments of Alternative Fuels

	Fuel Type	Primary Fissile Isotopes	Primary Diluents	Enrichment(a)
<u>LWR</u>				
1	LEU	^{235}U	^{238}U	3.2%
2	Re LEU	^{235}U , ^{233}U	^{238}U	2.5%
3	DU-Th	^{235}U	Th, ^{238}U	20.0%
4	Re DU-Th	^{235}U , ^{233}U	Th, ^{238}U	5.3%
5	HEU-Th	^{235}U	Th	93.1%
6	Re HEU-Th	^{235}U , ^{233}U	Th	5.45%
7	Pu-U	^{239}Pu , ^{241}Pu	^{238}U	61.5%
8	Pu-Th	^{239}Pu , ^{241}Pu	Th	57.8%
9	Once-through LEU	^{235}U	^{238}U	3.0%
<u>LMFBR</u>				
A	Inner Core	^{239}Pu , ^{241}Pu	^{238}U	71.1%
B	Outer Core	^{239}Pu , ^{241}Pu	^{238}U	71.1%
C	Radial Blanket	^{235}U	^{238}U	0.3%
D	Axial Blanket	^{235}U	^{238}U	0.3%
E	Radial Blanket	None	Th	--

(a) Enrichment is defined as the quantity of primary fissile isotopes divided by the quantity of all isotopes of that element, multiplied by 100.

TABLE 4.4. Elemental Compositions of Alternative Fuels

		Wt% in Fuel			
		Fuel	Th	U	Pu
<u>LWR</u>					
1	LEU		-	100.0	-
2	Re LEU		-	100.0	-
3	DU-Th		76.8	23.2	-
4	Re DU-Th		18.2	81.8	-
5	HEU-Th		95.7	4.3	-
6	Re HEU-Th		93.0	7.0	-
7	Pu-U		-	92.3	7.7
8	Pu-Th		86.5	-	13.5
9	Once-through LEU		-	100.0	-
<u>LMFBR</u>					
A	Inner Core		-	87.7	12.3
B	Outer Core		-	83.9	16.1
C	U Radial Blanket		-	100.0	-
D	U Axial Blanket		-	100.0	-
E	Th Radial Blanket	100.0		-	-

fuel type, along with the fissile isotope enrichment. Table 4.4 shows the breakdown by heavy element for each fuel. Reprocessing takes place 1.5 years after discharge, and recycled material is refabricated 0.5 year after reprocessing. Detailed heavy metal compositions by nuclide are given in Appendix B.

4.2 CHARACTERIZATION OF FUELS - REPROCESSING

Fuel exposures were calculated on the basis of time and power density. The LWR fuels averaged 30 GWd/MTHM. Actual burnups are listed in Table 4.5. Burnup data for the LMFBR fuel regions are shown in Table 4.6.

Radionuclide source terms for this analysis were generated with the aid of three computer codes: LEOPARD, CINDER, and ORIGEN (Barry 1963, England 1964, and Bell 1973, respectively.) The CINDER code, which uses recent ENDF/B-V cross-section data, was used to calculate fission product quantities

TABLE 4.5. LWR Alternative Fuel Burnups

	<u>Fuel</u>	<u>Burnup, GWd/MTHM(a)</u>
1	LEU	30.0
2	Re LEU	30.0
3	DU-Th	32.1
4	Re DU-Th	30.4
5	HEU-Th	32.8
6	Re HEU-Th	32.7
7	Pu-U	29.9
8	Pu-Th	32.3
9	Once-through LEU	30.0

(a) Residence time in reactor is 26,280 hours or 3 years in all cases.

TABLE 4.6. LMFBR Fuel Burnup and Core Residence Times

	<u>Fuel</u>	<u>Burnup, GWd/MTHM</u>	<u>Residence Time, Yr</u>
A	Inner Core	63	2
B	Outer Core	50	2
C	U Radial Blanket	6	5
D	U Axial Blanket	4	2
E	Th Radial Blanket	3	5

for each fuel type. ORIGEN calculated heavy element quantities, structural activation, fuel impurity activation, and the quantity of tritium produced as a fission product. LEOPARD was used for the LWR fuels to generate spectral information for input to ORIGEN, and to provide a reference spent fuel heavy element composition to which ORIGEN results were matched.

4.2.1 LWR Analysis

The flow of information in the LWR analysis is shown in Figure 4.1. Fresh fuel compositions for the alternative LWR fuel types were input to the LEOPARD code and exposure was accumulated over a 3-year period. Neutron flux and

FIGURE 4.1. LWR Analysis

spectral data calculated by LEOPARD were input to ORIGEN; these data plus cross-section data from LEOPARD were input to CINDER. As indicated in Figure 4.1, these two codes generated the required radionuclide inventory information. Isotopic weight fractions calculated by ORIGEN and CINDER for U and Pu were matched (requiring a minor normalization) to those calculated by LEOPARD to ensure consistent and comparable results.

4.2.2 LMFBR Analysis

Figure 4.2 shows the flow of information in the LMFBR analysis. In this case, only CINDER and ORIGEN were used; neutron fluxes calculated by ORIGEN were input to the CINDER code. Each code produced radionuclide inventory information as indicated in Figure 4.2.

4.2.3 Spent Fuel Characterization

After discharge from the reactor, the fuel is assumed to be held for a period of 1.5 years prior to reprocessing. Appendix C presents a radionuclide inventory for all spent fuel types at reprocessing. Table 4.7 summarizes the heavy metal elemental composition of the spent fuels, listing the most important actinides. A complete inventory of radionuclides can be found in Appendix C, including fission products, activation products, and an isotopic breakdown of the actinides.

One of the more important radionuclides, from the standpoint of biological hazard, is tritium (^3H). Tritium is produced in two ways in the fuel; as a fission product (1 to 2 ^3H atoms produced per 10^4 fissions), and by an (n,α) with ^6Li . A concentration of 0.5 ppm of natural Li in the fuel was assumed for this study. This value probably overestimates the Li concentration in most fuels. The choice of 0.5 ppm was arbitrary, since many fuel analyses report Li concentrations simply as less than an upper limit. The various alternative LWR and LMFBR fuels produce differing amounts of tritium for a given Li charge, mainly due to neutron flux spectral differences. The amounts of tritium produced in each fuel by the main production processes are shown in Table 4.8.

FIGURE 4.2. LMFBR Analysis

TABLE 4.7. Spent Fuel Heavy Element Compositions(a)

Fuel Type	Weight Percent						
	Th	Pa	U	Np	Pu	Am	Cm
<u>LWR</u>							
1 LEU	-(b)	-	98.89	0.10	1.00	0.02	-
2 Re LEU	-	-	98.97	0.02	0.99	0.01	-
3 DU-Th	77.35	0.01	22.20	0.06	0.37	0.01	-
4 Re DU-Th	18.07	-	80.54	0.41	0.96	0.01	-
5 HEU-Th	96.28	0.01	3.66	0.05	0.01	-	-
6 Re HEU-Th	93.79	-	5.92	0.25	0.04	-	-
7 Pu-U	-	-	93.49	0.01	5.87	0.43	0.20
8 Pu-Th	87.40	0.01	1.48	-	10.20	0.74	0.18
9 Once-through LEU	-	-	98.97	0.04	0.97	0.02	-
<u>LMFBR</u>							
A Inner Core	-	-	84.93	0.04	14.83	0.19	-
B Outer Core	-	-	82.64	0.03	17.08	0.26	-
C U Radial Blanket	-	-	97.44	0.01	2.55	-	-
D U Axial Blanket	-	-	98.14	0.01	1.85	-	-
E Th Radial Blanket	97.35	0.01	2.64	-	-	-	-

(a) Compositions at reprocessing, 1.5 years after discharge from reactor.

(b) Blank entry indicates that the weight percent of the element is less than 0.005.

TABLE 4.8. Tritium Produced in Alternative Fuels

Fuel	Tritium Produced (Ci/MTHM)(a)		
	By Fissions	By ${}^6\text{Li}(n,\alpha)3\text{H}$	Total
<u>LWR</u>			
1 LEU	450	125	575
2 Re LEU	500	120	620
3 DU-Th	400	115	515
4 Re DU-Th	460	90	550
5 HEU-Th	390	125	415
6 Re HEU-Th	420	115	535
7 Pu-U	540	60	600
8 Pu-Th	520	35	555
9 One-through LEU	460	125	585
<u>LMFBR</u>			
A Inner Core	1700	2	1702
B Outer Core	1400	1	1401
C U Radial Blanket	140	1	141
D U Axial Blanket	96	0.5	96.5
E Th Radial Blanket	37	0.5	37.5

(a) Assumed 0.5 ppm Li (7.5% ${}^6\text{Li}$) charged to reactor.

In this analysis, the LWR fuels were clad with Zircaloy while the LMFBR fuels were clad with stainless steel. The type of cladding surrounding the fuel is important for determining the amount of ^3H reaching the reprocessing plant. Tritium diffuses rather easily through stainless steel cladding; approximately 80% of the tritium produced will diffuse through to the reactor coolant. In Zircaloy cladding, only about 1% of the tritium will escape from the cladding. Hydrogen forms a chemical bond with the Zircaloy and tends to be retained with the cladding material. About 15% of the tritium produced becomes bonded to the Zircaloy cladding material and stays with the hulls during reprocessing. Thus, only about 85% of the tritium produced in Zircaloy-clad fuel enters the effluent streams of the reprocessing plant.

4.3 CHARACTERIZATION OF FUEL WASTES AT DISPOSAL

The reprocessing, fabrication, and refabrication of nuclear fuels generate a variety of wastes. Some of these wastes are of low activity and have historically been disposed of in a shallow land burial site. Current proposals for disposing of high-level and transuranic-contaminated waste require placement in a relatively more secure site. In this study these wastes are assumed to be disposed of in a deep geological repository. The following paragraphs describe the assumptions associated with modeling the waste repository.

The wastes stored at the repository come from several sources. The four most important wastes, classified by source and activity are:

- High-Level Liquid Waste (HLLW)
- Intermediate-Level Liquid Waste (ILLW)
- Activated Fuel Residue
- Heavy Metal Oxide Waste.

The first three wastes are generated at the fuel reprocessing plant (FRP), while the last is produced by the fabrication process.

4.3.1 High-Level Liquid Waste

High-level liquid waste (HLLW) is defined by Federal regulations as the aqueous waste resulting from the operation of the first cycle solvent

extraction system and the concentrated wastes from subsequent extraction cycles in a facility for reprocessing irradiated reactor fuels (10 CFR 50). Over 99% of the fission products and transplutonium elements extracted from spent fuel at a fuel reprocessing plant are contained in HLLW. Regulations stipulate that HLLW must be solidified within 5 years after reprocessing. The solidified waste is placed in sealed containers prior to shipment offsite. The sealed containers, called canisters, must be transferred to a Federal repository no later than 10 years after reprocessing.

For this study the reference HLLW solidification process is vitrification. HLLW vitrification processes convert high-level liquid waste to an inert glass by the application of heat and the addition of glass forming agents. A volume reduction factor of about 8 is achieved.

The reference vitrification process, spray calcination/in-can melting, was developed at the Pacific Northwest Laboratory (PNL). This study assumes that the HLLW undergoes solidification as it is generated by the reprocessing of fuel 1.5 years after discharge from a nuclear power plant. The flowsheet consists of four steps: 1) calcination, 2) vitrification, 3) canister handling, and 4) effluent treatment.

4.3.2. Intermediate Level Liquid Waste

Intermediate level liquid waste (ILLW) is a composite waste of variable composition. It may contain many, or all, of the following:

- laboratory liquid wastes, including samples and reagents
- solvent wash wastes, including entrained and degraded solvent
- off-gas scrubber residues
- iodine scrubber residues
- fuel receiving and storage liquid wastes, including ion exchange, regeneration wastes, cask decontaminations wastes, and cask cooldown system wastes
- solvent burner quench pot scrubber solutions

- building and equipment decontamination wastes
- building floor drainage
- liquid waste from personnel decontamination facilities
- cold chemical wastes from the cold chemical makeup area.

Combining the ILLW with the HLLW before solidification is a feasible operating mode for the reprocessing plant. The total volume of glass produced would be increased by approximately 34%. This study assumed that HLLW and ILLW would be combined before vitrification.

4.3.3 Activated Fuel Residue

Fuel residue is generated in the course of reprocessing when fuel assemblies are sheared into short lengths and leached with hot nitric acid. The residue includes short lengths of fuel cladding (Zircaloy for LWRs and stainless steel for LMFBRS) and fuel assembly hardware (support rods, poison rods, massive end fittings, fuel support grids, assorted springs and spacer elements). The type and amounts of materials found in the residue depend on the fuel cycle and the fuel fabricator. The acid-leached fuel cladding sections are expected to contain about 0.05% of the original fuel.

Obvious incentives exist for compacting the fuel residue prior to packaging. However, the simplicity of packaging the residue without compaction strongly recommends this procedure for remote hot cell work. The method considered in this report would monitor the residue for undissolved fuel, dry it, blend it with sand, and package it with the hardware.

4.3.4 Heavy Metal Oxide Waste

Operations at the fuel fabrication and refabrication plants unavoidably generate some transuranic-contaminated wastes. For this study, it was assumed that 0.5% of the plant input was lost to this waste, which is eventually stored in the repository.

5.0 FUEL CYCLE FACILITIES: WASTE TREATMENT AND EFFLUENT CONTROL

Emissions and waste streams from fuel cycle facilities can be estimated using process flowsheets, waste treatment systems information, and material inputs. This section describes the physical processes and postulated radio-nuclide release pathways of the fuel reprocessing plant, fuel fabrication plant and waste repository.

5.1 REPROCESSING

The function of the fuel reprocessing plant (FRP) is to recover and purify the uranium, thorium, and plutonium present in spent fuels so that they can be recycled to a nuclear power plant. The reprocessing plant also provides facilities for converting uranium, thorium, and plutonium nitrate solutions to UF_6 , ThO_2 and PuO_2 , respectively.

5.1.1 FRP Process Description

The reference FRP incorporates the following process steps:

- underwater storage of the spent fuel awaiting processing
- recovery and purification of the uranium, thorium, and plutonium by solvent extraction (using the Purex or Thorex process) to produce nitrate solutions of these metals
- conversion of the plutonium from nitrate solution to solid plutonium dioxide (PuO_2)
- conversion of the thorium from nitrate solution to solid thorium dioxide (ThO_2)
- conversion of the uranium from nitrate solution to solid uranium hexafluoride (UF_6).

The assumptions for operation of the reference FRP include:

- The processing rate is 2000 MTHM/yr.
- Spent fuel is aged 6 months (after discharge from reactor) before being received at the FRP.

- Spent fuel is stored at the FRP for 1 year and thus has been out of the reactor for 1.5 years before it is processed.
- Average burnup of the spent fuel is 30,000 MWd/MTHM.

Each of the above processes generates radioactive wastes. Most of these wastes undergo treatment processes in order to limit environmental releases. The wastes occur in solid, liquid and gaseous forms; the waste treatment processes used reflect the form and level of radionuclide content of the wastes.

5.1.2 FRP Waste Treatment and Effluent Control Systems

Figure 5.1 illustrates the assumed waste treatment system arrangement at the FRP. For the purposes of this study only gaseous effluents leave the FRP. The basis for the system descriptions are as detailed in the Commercial Waste Technology Study (DOE 1979).

Table 5.1 outlines the sources of radionuclides, quantities, and destinations of the various waste streams. The table is organized into functional components consistent with Figure 5.1. Table 5.2 describes the waste treatment system performance and effluent pathways. Table 5.3 lists the overall plant decontamination factors for some of the more important nuclides. The following paragraphs briefly describe each waste treatment system at the FRP.

5.1.2.1 Waste Vitrification System

High-level liquid waste (HLLW), containing over 99% of the fission products and transplutonium elements, results from operation of the solvent extraction system at the FRP. The intermediate-level liquid waste (ILLW) results from the final solvent extraction purification cycles at the FRP. In this study, the two waste streams were assumed to be combined for waste treatment.

The HLLW and ILLW are solidified by the waste vitrification process, which produces various secondary gaseous waste streams. These gaseous wastes are subjected to filtration and ruthenium and iodine sorption processes before being routed to the atmospheric protection system (APS).

FIGURE 5.1. Flow Diagram for Emissions from FRP Waste Treatment Systems

5.1.2.2 Fuel Residue Packaging System

The fuel residue is produced during the shearing and leaching of fuel assemblies and includes short lengths of fuel cladding (hulls) and associated

TABLE 5.1. Fuel Reprocessing Plant Primary Waste Stream

Waste Treatment System	Sources of Primary Waste	Radionuclide Content in Primary Waste					
		Fission Products ^(a)		Actinides ^(b)		Activation Products ^(c)	
Waste vitrification system	Main plant HLLW, ILLW, and silica gel	³ H	0.081	U	0.006	All	0
		⁸⁵ Kr	0	Th	0.006		
		¹²⁹ I	0.008	Pu	0.006		
		All others	1.0	All others	1.0		
Fuel residue packaging system	Fuel residue (hardware and hulls)	³ H	0 (LMFBR)	All	5 x 10 ⁻⁴	All	1.0
		³ H	0.15 (LWR)				
		⁸⁵ Kr	0				
		¹²⁹ I	0				
		All others	5 x 10 ⁻⁴				
ILW incineration system	Combustible trash and ventilation filters from main plant, PuO ₂ and ThO ₂ conversion, and UF ₆ plant; IX bead resins	¹²⁹ I	2 x 10 ⁻³	U	5.3 x 10 ⁻⁴	All	0
		Zr	2 x 10 ⁻⁵	Th	2.5 x 10 ⁻³		
		Nb	2 x 10 ⁻⁵	Pu	2.5 x 10 ⁻³		
		Ru	2 x 10 ⁻⁵	All others	1.1 x 10 ⁻⁵		
		All others	1.1 x 10 ⁻⁵				
LLW incineration system	Main plant combustible trash	³ H	0	All	1 x 10 ⁻⁹	All	0
		¹²⁹ I	0				
		⁸⁵ Kr	0				
		All other	1 x 10 ⁻⁹				
Degraded solvent system	Degraded solvent	¹²⁹ I	1 x 10 ⁻⁴	Th	1 x 10 ⁻⁴	All	0
		Zr	1 x 10 ⁻⁶	Pu	1 x 10 ⁻⁴		
		Nb	1 x 10 ⁻⁶	All others	1 x 10 ⁻⁶		
		Ru	1 x 10 ⁻⁶				
		All others	1 x 10 ⁻¹⁰				
Vessel off-gas system	Process vessel off-gas	³ H	0.001	All	1 x 10 ⁻⁷	All	0
		⁸⁵ Kr	1 x 10 ⁻⁶				
		¹²⁹ I	0.005				
		All others	1 x 10 ⁻⁷				

TABLE 5.1. (contd)

Waste Treatment System	Sources of Primary Waste	Radionuclide Content in Primary Waste					
		Fission Products ^(a)		Actinides ^(b)		Activation Products ^(c)	
Dissolver off-gas system	Fuel shear and dissolver off-gas	³ H	0.05	All	1 x 10 ⁻⁷	All	0
		¹⁴ C	1.0				
		⁸⁵ Kr	1.0				
		¹²⁹ I	0.99				
		Ru	2 x 10 ⁻⁴				
		All others	1 x 10 ⁻⁷				
Failed equipment and noncombustible waste system	Noncombustible trash and failed equipment from main plant, ThO ₂ and PuO ₂ conversion, and UF ₆ facility	All	2 x 10 ⁻⁶	U	1.1 x 10 ⁻⁴	All	0
				Th	2 x 10 ⁻⁴		
				Pu	2 x 10 ⁻⁴		
				All others	2 x 10 ⁻⁶		
Cement immobilization system	UO ₆ plant fluorinator bed residues and fines and K ₂ UO ₄ mud	Zr	1.1 x 10 ⁻⁷		U		2.8 x 10 ⁻³
		Nb	1.1 x 10 ⁻⁷		Th		1.2 x 10 ⁻⁶
		Ru	1.1 x 10 ⁻⁶		Pu		1.2 x 10 ⁻⁶
		All others	0	All others	0		
Atmospheric protection system	Building ventilation air	All	1 x 10 ⁻¹¹	All	1 x 10 ⁻¹¹	All	0
Stack (no treatment) 0	Vaporized excess water and UF ₆ process off-gas	³ H	0.87 (LMFBR)		U	1 x 10 ⁻⁵	All
		³ H	0.72 (LWR)	Th	2 x 10 ⁻¹¹		
		¹²⁹ I	1 x 10 ⁻⁵	Pu	2 x 10 ⁻¹¹		
		Zr	1.3 x 10 ⁻⁴		All others	1 x 10 ⁻¹⁶	
		Nb	4 x 10 ⁻¹²				
		Ru	1 x 10 ⁻¹⁰				
		All others	1 x 10 ⁻¹⁶				

(a) Expressed as fraction of inventories in Table C.1 (fission products) and Table C.3 (activated impurities)

(b) Expressed as fraction of inventory in Table C.2.

(c) Expressed as fraction of inventory in Table C.4.

TABLE 5.2. FRP Waste Treatment System Performance

Waste Treatment System	Source of Emission	Nuclide	Decontamination Factor	Emission Released to:
Vitrification system	Process off gas, ventilation air, and excess water	^3H	1.0	APS
		^{129}I	1×10^3	
		$^{106}\text{Ru-Rh}$	1×10^{10}	
		All others	5×10^{10}	
Fuel residue packaging system		All	1×10^9	APS
ILW incineration system	Incinerator off gas and water evaporated from spent scrubber solutions	^3H	1.25	APS
		^{129}I	1.05	
		$^{95}\text{Zr-Nb}$	1×10^7	
		$^{106}\text{Ru-Rh}$	1×10^{11}	
		Pu	1×10^7	
		U	1×10^7	
		Th	1×10^7	
		Other fission products		5×10^7
		Other actinides	2×10^6	
		Activation products		1×10^8
LLW incinerator system	Incinerator off gas	All	1.25×10^{11}	APS
	Water evaporated from spent scrubber solution	All	1×10^8	APS
Degraded solvent system	Incinerator off gas	^{129}I	1.0	Vessel off-gas system
		All others	1×10^6	
Vessel off-gas system	Process off gas	^3H	1.0	APS
		^{85}Kr	1.0	
		^{14}C	1.0	
		$^{106}\text{Ru-Rh}$	20	
		^{129}I	1×10^3	
		All others	1×10^4	

TABLE 5.2. (contd)

Waste Treatment System	Source of Emission	Nuclide	Decontamination Factor	Emission Released to:
Dissolver off-gas system	Process off gas and vaporized excess water	^3H	1.0	APS
		^{14}C	1×10^2	
		^{85}Kr	10	
		$^{106}\text{Ru-Rh}$	2×10^4	
		^{129}I	1×10^3	
		All others	1×10^5	
Failed equipment and noncombustible waste system	Ventilation air	All	1×10^{11}	APS
	Vaporized excess water	All	5×10^8	Stack
Cement immobilization system	Process off gas	^3H	1×10^9	APS
		^{129}I	1×10^4	
		All others	1×10^9	
	Vaporized excess water	^3H	1×10^3	Stack
		^{129}I	1×10^6	
		All others	3×10^{11}	
Atmospheric protection system	--	^3H	1.0	Stack
		^{85}Kr	1.0	
		^{129}I	1.0	
		^{14}C	1.0	
		$^{106}\text{Ru-Rh}$	1.1	
		All others	1×10^4	

(a) Assumes that volatile ruthenium is not converted to particulate ruthenium before it reaches the APS.

TABLE 5.3. Fuel Reprocessing Plant Overall
Decontamination Factors for Some
Important Nuclides

<u>Nuclide</u>	<u>Decontamination Factor</u>
^3H	1.2 (LWR) ^(a)
	1.0 (LMFBR)
^{14}C	10^2
^{85}Kr	10^1
^{129}I	3.3×10^2
^{106}Ru	7.1×10^7
All U	10^5
All Pu	5×10^{12}
All Th	5×10^{10}

(a) 15% of the tritium remains with the
LWR Zircaloy clad

hardware. This fuel residue is blended with sand and packaged into stainless steel canisters, releasing small amounts of airborne radioactivity. The air is routed to the APS before being discharged.

5.1.2.3 Low- and Intermediate-Level Waste Incineration

Low-level and intermediate-level wastes (LLW and ILW) at the FRP consist mostly of general trash and ventilation filters. These wastes are separated on the basis of activity level and burned in two separate incinerators. The combustion gases are scrubbed and cooled before being routed to the APS. Water evaporated from the scrubber solution is also routed to the APS.

5.1.2.4 Degraded Solvent System

The organic solvent used in the extraction process is slowly degraded by chemical and radiation damage, decreasing its effectiveness. Small amounts of degraded solvent are routinely discarded and incinerated. The off gas from this process is routed to the vessel off-gas system for treatment.

5.1.2.5 Vessel Off-Gas System

Input to the vessel off-gas system includes gases emitted from process vessels, storage tanks, and HLLW concentrators as well as the degraded solvent incineration off gas. This system removes particles, iodine, and oxide of nitrogen from the off gas. The emissions from the vessel off-gas system are routed to the APS.

5.1.2.6 Dissolver Off-Gas System

The off gas produced during the chopping and leaching of fuel assemblies contains essentially all of the ^{129}I , ^{14}C , and ^{85}Kr present in the spent fuel, as well as some of the tritium and ruthenium. The dissolver off-gas system combines recovery of ^{129}I , ^{14}C , and ^{85}Kr into an integrated system. No tritium recovery system is included in the reference FRP, and all the tritium available for release is released via either the dissolver off-gas system and APS or the excess water vaporizer.

The iodine recovery system uses silver-nitrate-impregnated amorphous silicic acid to remove iodine as well as a ruthenium adsorber and filter bank. The ^{14}C is captured as CO_2 on a zeolite bed, and the ^{85}Kr is cryogenically removed. The off gas from this system is routed to the APS.

5.1.2.7 Failed Equipment and Noncombustible Waste System.

These wastes include large pieces of failed equipment as well as noncombustible scrap. All of these materials are contaminated to varying degrees. The failed equipment is cut into pieces, if necessary, and the waste is decontaminated and packaged for disposal. Small amounts of radioactivity are released to the air and eventually routed to the APS. Decontamination solutions are evaporated in the excess water vaporizer.

5.1.2.8 Cement Immobilization System

Low-level wet and solid wastes are immobilized by mixing with cement and packaging for shipment. Small quantities of airborne radioactivity generated by this process are routed to the APS for treatment. Small amounts of radioactivity are also released via the excess water vaporizer.

5.1.2.9 Atmospheric Protection System (APS)

The APS is the final stage of filtration for building ventilation air and process off gas at the FRP, and consists of a prefilter followed by a bank of high efficiency particulate air (HEPA) filters. Emissions from the APS are discharged to the stack.

5.2 FUEL FABRICATION AND REFABRICATION

The function of the fuel fabrication plant (FFP) and the fuel refabrication plants (FRFP) is to produce fuel elements for LWRs and LMFBRs. The FFP and FRFPs utilize uranium, plutonium, and thorium oxides in producing uranium and binary mixture fuels.

5.2.1 Process Description

The appropriate quantities of UO_2 , ThO_2 , and PuO_2 powders are mixed, formed into dense fuel pellets, and sealed within cladding tubes. The clad material is Zircaloy in the case of LWRs and stainless steel for LMFBRs. The assumed plant capacity is 1500 MTHM of oxide fuel per year for the FFP, and 400 MTHM/yr for the FRFPs.

The fuel rods are produced through several process steps:

- PuO_2 , ThO_2 , and UO_2 powders are mechanically blended in the proper proportions.
- The mixed oxides are pressed into cakes.
- The oxide cake is granulated and blended.
- The granules are pressed into "green" pellets.
- The green pellets are sintered in high-temperature furnaces.
- The sintered pellets are finish ground.
- Finished pellets are washed and inspected.
- Finished pellets are loaded into cladding (tubing cut to finished length and with bottom end-plug welded into place).

- The cladding is decontaminated, a pellet hold-down device is inserted, and the top end plug is welded into place.

The completed fuel rod is then withdrawn from the restricted access area and additional operations are performed. Such operations include:

- clad degreasing
- clad etching and rinse
- x-ray inspection of fuel pellet arrangement
- autoclaving of fuel elements
- fuel-clad leak check and dimensional inspection.

Since the activity levels of the fuels vary so widely, we have considered three types of fuel fabrication plants have been considered in this study. The first plant type produces fuels containing thorium and low-activity uranium only, and shall be referred to as the fuel fabrication plant (FFP). Fuels that contain plutonium require remote fabrication operations and stricter effluent control; these fuels are produced at the plutonium refabrication plant (PFRFP). Fuels containing high-activity uranium (^{232}U) require a plant using remote maintenance as well as remote operations. These fuels are made at the high-activity uranium fuel refabrication plant (UFRFP). Table 5.4 shows which fuel types are fabricated at each plant.

5.2.2. Waste Treatment Systems

Waste treatment at a fuel fabrication or refabrication plant is much less complex than that at the FRP, since the radionuclide content of the waste is

TABLE 5.4. Fabrication of Alternative Fuels

<u>Plant</u>	<u>LWR Fuel Types</u>	<u>LMFBR Fuel Types</u>	<u>Overall Plant Decontamination Factor</u>
Fuel Fabrication Plant (FFP)	LEU, DU-Th, HEU-Th, once-through LEU	U Radial Blanket, Th Radial Blanket	All Nuclides: 3.2×10^4
Plutonium Fuel Refabrication Plant (PFRFP)	Pu-U, Pu-Th	Inner Core, Outer Core, U Axial Blanket(a)	Am: 10^{10} All others: 8×10^{11}
High-Activity Uranium Fuel Refabrication Plant (UFRFP)	Re LEU, Re DU-Th, Re HEU-Th	None	All nuclides same(b)

(a) This is an integral part of the core fuel element, and so must be fabricated at the PFRFP.

(b) No value has been determined.

limited to actinides and their daughters. The pertinent waste treatment systems at the three plants are similar to those at the FRP, but are of different efficiencies. For this reason, the overall decontamination factor varies among the three plants.

5.2.2.1 Waste Treatment at the FFP

The reference FFP incorporates the direct conversion process whereby UF_6 is converted to UO_2 . This process produces only gaseous wastes, but other processes in the FFP do produce some liquid waste. For the reference plant considered, the dose commitment from liquid waste releases is less than 5% of that from gaseous waste releases. Thus, for this comparative study, liquid releases will be ignored. A more detailed description of the reference fuel fabrication facility can be found in the ORNL report (Pechin et al. 1975) on UO_2 LWR fuels.

The gaseous waste treatment system at the FFP employs scrubbers and filters for decontamination of the waste streams. The overall plant decontamination factor for all nuclides is 3.2×10^4 .

5.2.2.2 Waste Treatment at the PFRFP

The reference PFRFP is essentially that described in the Commercial Waste Technology report (DOE 1979). All effluents are gaseous, and secondary wastes from the various waste treatment process are routed through a bank of HEPA filters before release. The overall decontamination factor for the PFRFP is 1×10^{10} for americium and 8×10^{11} for all other nuclides.

5.2.2.3 Waste Treatment at the UFRFP

No data are currently available for decontamination factors for the UFRFP, but the waste treatment system was assumed to be about the same as for the PFRFP. The decontamination factor was assumed to be the same for all nuclides.

5.3 WASTE DISPOSAL

The geologic repository system is assumed to provide protection from radioactive waste in two ways. The first is to contain the waste for a sufficient length of time so that the hazardous nuclides decay to innocuous levels

and unrestricted release to the environment presents no radiological hazard. The second way is to limit the rate of release of the nuclides to the biosphere such that their concentration in the constantly renewed local surface water and air never exceeds acceptable limits. The available system components that provide the means for protection can be reduced to 1) the waste form, 2) the containment package, and 3) the geological media.

This report compares wastes from various nuclear fuel types on the basis of the hazard index associated with a postulated release scenario. The scenario for nuclide releases from a geologic repository involves the following steps:

- penetration of the repository by ground water after 1000 years
- failure of waste canister also at 1000 years
- leaching of radionuclides from the waste at a rate of 0.1% per year for 1000 years
- transport of the nuclides through the surrounding geology by ground water traveling at 100 m/year over a migration path that is 10 km long
- transport of the nuclides through the biosphere to man.

The transportation of the radionuclides through the geosphere is modeled by the latest version of the GETOUT code (DeMier et al. 1979), which simulates the release of radionuclides from the repository through contact and leaching by flowing groundwater, subsequent transport of the nuclides in the groundwater (including the effects of convection, sorption, and radioactive decay), and ultimate release of the nuclides to the surface water.

The geological site chosen for this analysis is a layer of bedded salt deposited between dolomites and sandstones, with the dolomites lying between the repository and the biosphere. The disposal site is modeled after the Waste Isolation Pilot Program (WIPP) site at Los Medanos, New Mexico (Dosch and Lynch 1978, Serne et al. 1977). The nuclide sorption constants (k_d) that describe the rate of transport by ground water were biased in a conservative direction. This allows the most rapid credible nuclide transport to be assumed.

6.0 COMPARISON OF ENVIRONMENTAL IMPACTS OF ALTERNATIVE FUELS

This section presents the comparison of environmental impacts of reprocessing and fabricating alternative fuels and of storing the resulting wastes. The bases for comparison are outlined, and the results of the comparison for each facility are discussed. The nuclides responsible for substantial contributions to and variations among the dose commitments and biohazards of the various fuel types are identified.

6.1 BASES FOR COMPARISON

Using the information presented in Section 5, it is possible to compare the various LWR fuel types and LMFBR fuel regions. For the fuel fabrication plant (FFP), fuel refabrication plants (FRFPs), and the fuel reprocessing plant (FRP), comparisons were made on the basis of a 50-year population dose commitment resulting from one year of plant operation. For the waste repository, a hazard index based on maximum permissible concentrations (MPCs) was used for the comparison.

6.1.1 Comparison Method for FRP, FRFP and FFP Effluents

Emissions from the FRP were calculated using the spent fuel radionuclide inventory of Appendix C and the waste stream and decontamination factor information of Section 5.1. Similar calculations were performed for the FFP and FRFPs, using the radionuclide inventory of Appendix B and decontamination factors from Section 5.2. These emissions are in units of curies/MTHM per year of operation.

Dose commitment factors for each nuclide of importance were estimated. These factors represent the 50-year population dose commitment per curie of each nuclide for a 1-year airborne release at a generic midwestern site. Dose factors for total body, thyroid, bone, lung, and gastrointestinal (GI) tract are listed in Appendix E.

For comparison purposes, the dose commitments for each organ were scaled to a reference fuel type. For the FFP, the reference type is the once-through

LEU fuel. The reference fuel type for the PFRFP is the Pu-U fuel (LWR fuel type 7), and for the UFRFP, it is Re LEU (LWR fuel type 2). The dose commitments from each fabrication plant are compared. The difference in the degree of effluent control among the plant types prevents comparing dose commitments among the three fabrication plants.

The reference fuel for the FRP is LEU fuel (fuel type 1 in Table 4.5). The once-through LEU is by definition not reprocessed.

6.1.2 Comparisons Method for the Waste Repository

As part of the analysis of the potential environmental effects of operating a nuclear waste repository, the consequences of ground-water intrusion were studied. If ground water came into contact with the stored wastes, the radioactive material would be leached out and transported to the biosphere. The leaching process, if it occurred, would take place very slowly and it would take a substantial length of time for most radionuclides to reach the biosphere.

Using the radionuclide inventories in the waste of each fuel type, ground-water concentrations at various times were calculated for each nuclide of importance. The radionuclide transport by ground water was modeled using the GETOUT code (DeMier et al. 1979). A leach rate of 0.1% per year was assumed with a ground-water velocity of 100 m/year. The distance to the biosphere was assumed to be 10 km. The distribution constant, k_d , was estimated for each nuclide of interest, based on available data and present understanding of nuclide-rock interactions.

The GETOUT code calculated ground-water concentrations for each nuclide at various times. Comparisons were made at times when important nuclides reached their peak concentration in the ground water. For each of these comparisons, each nuclide concentration was divided by its maximum permissible concentration (MPC). A sum over all nuclides was calculated for each fuel type and used as a hazard index to compare the fuel types. This method for calculating the hazard index is taken from a recent overview of safety indices (Voss 1979).

6.2 COMPARISONS OF ALTERNATIVE FUELS

Using the method described in Section 6.1, comparisons have been made among the various LWR fuel types and LMFBR fuel regions. For the reprocessing and fabrication plants, the relative dose commitments were calculated both on the basis of dose commitment per MTHM and on the basis of dose commitment per MWD. The latter comparison is an attempt to weigh the environmental impact of reprocessing, fabricating, or refabricating various fuels against the benefit of the energy they produce.

Tables 6.1 through 6.8 present the results of the relative dose commitment calculations. The various fuel types are compared on a per MTHM basis in Tables 6.1, 6.3, 6.5, and 6.7. The first row of entries for each affected organ shows the relative dose commitment to that organ from each fuel type. Listed under this are the percent contributions to the dose commitment by each nuclide of importance. Tables 6.2, 6.4, 6.6, and 6.8 present only the relative dose commitments on per MWD basis.

6.2.1 Relative Dose Commitments for the FRP

As can be seen in Table 6.1, there is a substantial range in the relative dose commitments for the fuel reprocessing plant (FRP). It is apparent from the table that the reprocessing of LMFBR fuels produces, in most cases, a substantially smaller dose commitment than the reprocessing of LWR fuels. In the case of the axial and radial blankets, this effect is largely due to the lower neutron fluxes (and, therefore, lower fission densities) to which these fuel regions are exposed. Table 6.2, however, shows that even when this is taken into account, the LMFBR fuels still provide a lower dose commitment. This difference primarily results from the lower tritium production and retention in LMFBR fuel elements.

Tritium is a significant and often dominant contributor to the dose commitment for all organs except bone. Since 80% of the tritium produced in LMFBRs diffuses away through the stainless steel cladding during the core life, tritium tends to play a lesser role in the dose commitment from the reprocessing of these fuels. The tritium in the spent fuel comes partly from neutron

TABLE 6.1. Relative Dose Commitments per MTHM for Fuel Reprocessing Plant Emissions^(a)

Organ	LWR Fuel Type								LMFBR Fuel Type				
	1	2	3	4	5	6	7	8	A	B	C	D	E
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
Total Body	1.0	1.1	0.97	1.1	0.98	1.1	1.1	1.0	0.73	0.60	0.061	0.041	0.030
³ H (fission)	80% ^(b)	80%	75%	77%	72%	68%	91%	92%	99%	98%	98%	98%	54%
³ H (activation) ^(c)	20	17	19	13	20	17	9	95	0	1	1	1	1
²³² U	0	3	6	9	7	14	0	2	0	0	0	0	21
²³³ U	0	0	0	0	0	0	0	0	0	0	0	0	21
Others	0	0	0	1	1	1	0	1	1	1	1	1	3
GI Tract	1.0	1.1	0.91	0.97	0.91	0.95	1.1	0.99	0.74	0.61	0.060	0.042	0.019
³ H (fission)	79	82	79	84	77	79	91	94	96	97	97	97	83
³ H (activation)	20	17	20	15	22	19	5	18	1	0	1	1	1
⁸⁵ Kr	1	1	1	1	1	1	0	0	1	1	1	1	5
²³³ U	0	0	0	0	0	0	0	0	0	0	0	0	9
Others	0	0	0	0	0	1	0	1	2	2	1	1	0
Thyroid	1.0	1.1	0.93	1.04	0.93	1.0	1.1	1.0	0.99	0.82	0.073	0.049	0.030
³ H(fission)	70	69	68	69	66	65	79	79	63	63	71	74	47
³ H (activation)	17	14	17	12	19	16	8	4	0	0	0	0	1
¹²⁹ I	12	16	14	18	14	18	13	17	36	35	23	25	49
Others	1	1	1	1	1	1	0	0	1	2	1	1	3
Bone	1.0	1.1	0.93	1.04	0.93	1.0	1.1	1.0	0.99	0.82	0.073	0.049	0.030
⁸⁵ Kr	56	2	1	1	1	1	18	1	39	29	48	46	1
¹⁰⁶ Ru	4	0	0	0	0	0	3	0	9	7	6	7	0
¹²⁹ I	4	0	0	0	0	0	2	0	6	4	6	5	0
¹⁴ C	15	0	0	0	0	0	3	0	0	0	0	0	0
²³² U	0	91	94	92	94	95	0	80	0	0	0	0	46

TABLE 6.1. (contd)

Organ	LWR Fuel Type								LMFBR Fuel Type				
	1	2	3	4	5	6	7	8	A	B	C	D	E
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
Bone (contd)	<u>1.0</u>	<u>31</u>	<u>79</u>	<u>133</u>	<u>96</u>	<u>209</u>	<u>1.8</u>	<u>37</u>	<u>1.9</u>	<u>2.2</u>	<u>0.10</u>	<u>0.076</u>	<u>16</u>
^{233}U	0	4	5	5	5	3	0	12	0	0	0	0	53
^{234}U	6	2	0	2	0	1	1	0	0	0	0	0	0
^{238}U	0	0	0	0	0	0	1	0	0	0	10	13	0
^{238}Pu	7	0	0	0	0	0	48	4	19	27	2	1	0
^{239}Pu	1	0	0	0	0	0	1	0	5	5	26	25	0
^{240}Pu	1	0	0	0	0	0	4	0	7	7	2	2	0
^{241}Pu	4	0	0	0	0	0	19	2	14	19	0	0	0
Others	2	1	0	0	0	0	0	1	1	2	0	1	0
Lung	<u>1.0</u>	<u>2.7</u>	<u>4.0</u>	<u>6.0</u>	<u>4.5</u>	<u>9.1</u>	<u>1.1</u>	<u>2.4</u>	<u>0.76</u>	<u>0.63</u>	<u>0.062</u>	<u>0.044</u>	<u>0.71</u>
^3H (fission)	79	33	18	13	15	8	90	39	92	92	96	95	2
^3H (activation)	19	7	5	2	4	2	9	2	0	0	0	0	0
^{106}Ru	0	0	0	0	0	0	0	0	5	5	2	3	0
^{232}U	0	55	73	78	75	85	0	50	0	0	0	0	43
^{233}U	0	3	4	4	5	3	0	8	0	0	0	0	56
^{234}U	0	1	0	2	0	1	0	0	0	0	0	0	1
Others	2	1	0	0	1	1	1	1	3	3	2	2	0

(a) Low Enriched Uranium (LEU), LWR Fuel Type 1, used as reference case. Underlined entries are ratios of each fuel type's dose commitment to that of LEU.

(b) Percent contribution by each nuclide to dose commitment.

(c) Assumes 0.5 ppm natural Li as a fuel impurity.

TABLE 6.2. Relative Dose Commitments per MWd for Fuel Reprocessing Plant Emissions^(a)

Organ	LWR Fuel Type								LMFBR Fuel Type				
	1	2	3	4	5	6	7	8	A	B	C	D	E
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
Total Body	1.0	1.1	0.91	1.0	0.90	1.0	1.1	0.93	0.35	0.36	0.31	0.31	0.30
GI Tract	1.0	1.1	0.85	0.92	0.83	0.87	1.1	0.92	0.35	0.37	0.30	0.32	0.19
Thyroid	1.0	1.1	0.87	0.95	0.85	0.92	1.1	0.93	0.47	0.49	0.37	0.37	0.30
Bone	1.0	31	74	131	88	192	1.8	34	0.76	0.96	0.50	0.57	160
Lung	1.0	2.7	3.7	5.7	4.1	8.3	1.1	2.2	0.36	0.38	0.31	0.33	7.1

(a) Low Enriched Uranium (LEU), LWR Fuel Type 1, used as reference case. Entries are ratios of each fuel type's dose commitment of that of LEU.

activation of ^6Li ; natural Li is assumed to be present in all the fuels at a concentration of 0.5 ppm by weight. However, the dose commitment from tritium would still be important even if no Li had been present in the fuel.

The lighter uranium isotopes, ^{232}U and ^{233}U , are present only in high-activity U and spent Th fuels. These isotopes are significant contributors, when present, to the bone and lung dose commitments, and are of somewhat smaller importance to the total body dose commitments.

Other isotopes contributing significantly to the total dose commitment are ^{129}I for the thyroid, and ^{85}Kr , ^{14}C , and plutonium isotopes for bone (when not overshadowed by ^{232}U).

As Table 6.1 shows, the total body, GI tract, and thyroid dose commitments per MTHM do not vary much among the LWR fuel types, and are considerably lower for the LMFBR fuel types. For bone and lung dose commitments, the presence of ^{232}U and Pu isotopes in some of the spent fuels results in a large variation among the fuel types. The thorium-based fuel types produced the highest dose commitments for bone and lung.

6.2.2 Relative Dose Commitments for the FRP

Table 6.3 shows the dose commitment comparisons per MTHM for the FRP, while Table 6.4 shows those per MWD produced by the fuel. There is again substantial variation among the fuel types fabricated at this plant. The most important contributors to the dose commitment tend to be ^{232}Th and ^{234}U ; the variation in the concentrations of these two account for most of the differences in dose commitment among fuel types.

Thorium-based fuels tend to produce the highest dose commitments, especially for bone and total body. The LMFBR uranium radial blanket generally produced the lowest dose commitments, since it contains so little of the higher-activity uranium isotopes.

6.2.3 Relative Dose Commitments for the PFRFP

The relative dose commitments for the PFRFP are shown in Tables 6.5 and 6.6. The major contributors to the dose commitment from the Pu-based fuels

TABLE 6.3. Relative Dose Commitment per MTM for Fuel Fabrication Plant Emissions^(a)

Organ	LWR Fuel Type				LMFBR Fuel Type	
	1 LEU	3 DU-Th	5 HEU-Th	9 Once-Through LEU	C U Radial Blanket	E Th Radial Blanket
<u>Total Body</u>	<u>1.8</u>	<u>6.6</u>	<u>7.6</u>	<u>1.0</u>	<u>0.18</u>	<u>6.8</u>
²³⁰ Th	0% ^(b)	0%	0%	0%	0%	3%
²³² Th	0	79	84	0	0	97
²³⁴ U	83	20	15	84	22	0
²³⁵ U	2	1	1	3	2	0
²³⁶ U	8	0	0	0	0	0
²³⁸ U	7	0	0	13	76	0
<u>GI Tract</u>	<u>1.7</u>	<u>1.3</u>	<u>1.0</u>	<u>1.0</u>	<u>0.28</u>	<u>0.0071</u>
²³⁰ Th	0	0	0	0	0	4
²³² Th	0	0	0	0	0	96
²³⁴ Th	8	2	0	13	45	0
²³⁴ U	81	92	96	72	13	0
²³⁵ U	2	4	4	3	1	0
²³⁶ U	9	0	0	0	0	0
²³⁸ U	0	2	0	12	42	0
<u>Thyroid</u>	<u>1.8</u>	<u>1.6</u>	<u>1.4</u>	<u>1.0</u>	<u>0.003</u>	<u>1.1x10⁻⁴</u>
²³⁰ Th	0	0	0	0	0	100
²³⁴ Th	1	0	0	10	19	0
²³⁴ U	99	100	100	100	76	0
²³⁸ U	0	0	0	0	5	0
<u>Bone</u>	<u>1.9</u>	<u>11</u>	<u>13</u>	<u>1.0</u>	<u>0.20</u>	<u>12</u>
²³⁰ Th	0	0	0	0	0	3
²³² Th	0	87	91	0	0	97
²³⁴ Th	0	0	0	0	1	0
²³⁴ U	81	13	9	83	22	0
²³⁵ U	2	0	0	3	2	0
²³⁶ U	9	0	0	0	0	0
²³⁸ U	8	0	0	14	76	0
<u>Lung</u>	<u>1.8</u>	<u>2.0</u>	<u>1.9</u>	<u>2.0</u>	<u>0.19</u>	<u>0.76</u>
²³⁰ Th	0	0	0	0	0	2
²³² Th	0	29	37	0	0	98
²³⁴ U	82	68	61	83	22	0
²³⁵ U	2	2	2	3	2	0
²³⁶ U	8	0	0	0	0	0
²³⁸ U	7	1	0	14	76	0

(a) LWR Fuel Type 9 (once-through LEU) is used as the reference case.
Underlined entries are ratios of each fuel type's dose commitment to that of once-through LEU.

(b) Percent contribution by each nuclide to the dose commitment.

TABLE 6.4. Relative Dose Commitment per MWd for Fuel Fabrication Plant Emissions^(a)

Organ	LWR Fuel Type				LMFBR Fuel Type	
	1 LEU	3 DU-Th	5 HEU-Th	9 Once-Through LEU	C U Radial Blanket	E Th Radial Blanket
Total Body	1.8	6.2	7.0	1.0	0.90	68
GI Tract	1.7	1.2	0.91	1.0	1.4	0.071
Thyroid	1.8	1.5	1.3	1.0	0.32	0.0011
Bone	1.9	10	12	1.0	1.0	120
Lung	1.8	1.9	1.7	1.0	0.95	7.6

(a) LWR fuel type 9 (once-through LEU) used as reference case. Entries are ratios of each fuel type's dose commitment to that of once-through LEU.

are ^{238}Pu , ^{241}Pu , and ^{241}Am . The dose commitment variation among fuel types is directly related to amount of Pu in each fuel. The LMFBR axial blanket is included only because it is an integral part of the core and thus must be fabricated at the PFRFP.

6.2.4 Relative Dose Commitments for the UFRFP

Tables 6.7 and 6.8 show the comparisons of dose commitments from the UFRFP per MTHM and per MWd, respectively. The variation in dose commitment is a direct function of the amount of ^{232}U and its daughter, ^{228}Th , in each fuel. If the same release factors used for the PFRFP are assumed for the UFRFP, the dose commitments for the fuels containing ^{232}U are generally about two orders of magnitude lower than those for the fuels fabricated at the PFRFP.

6.2.5 Comparison of Hazard Indices for Waste Repository Releases

Hazard indices were calculated for releases from the bedded salt repository, following the method of a recent overview of safety indices (Voss 1979). Indices for wastes from each of the alternative LWR and LMFBR fuel types were calculated at various decay times. The times chosen for analysis were those

TABLE 6.5. Relative Dose Commitments per MTHM for Plutonium-Fuel Refabrication Plant Emissions(a)

Organ	LWR Fuel Type		LMFBR Fuel Type		
	7 Pu-U	8 Pu-Th	A Inner Core	B Outer Core	D U-Axial Blanket (b)
<u>Total Body</u>	<u>1.0</u>	<u>1.8</u>	<u>1.0</u>	<u>1.4</u>	<u>2.8×10^{-8}</u>
^{234}U	0% ^(c)	0%	0%	0%	24%
^{235}U	0	0	0	0	2
^{238}U	0	0	0	0	74
^{238}Pu	33	30	17	17	0
^{239}Pu	2	1	3	3	0
^{240}Pu	3	4	5	5	0
^{241}Pu	14	15	18	18	0
^{241}Am	46	50	57	57	0
<u>GI Tract</u>	<u>1.0</u>	<u>1.8</u>	<u>1.1</u>	<u>1.4</u>	<u>2.0×10^{-6}</u>
^{234}U	0	0	0	0	24
^{235}U	0	0	0	0	2
^{238}U	0	0	0	0	74
^{238}Pu	29	27	14	14	0
^{239}Pu	1	1	2	2	0
^{240}Pu	3	3	3	3	0
^{241}Pu	15	16	19	19	0
^{241}Am	52	53	62	62	0
<u>Thyroid</u>	<u>1.0</u>	<u>1.9</u>	<u>1.3</u>	<u>1.8</u>	<u>4.1×10^{-8}</u>
^{234}U	0	0	0	0	94
^{238}U	0	0	0	0	6
^{241}Pu	23	23	23	23	0
^{241}Am	77	77	77	77	0
<u>Bone</u>	<u>1.0</u>	<u>1.7</u>	<u>0.97</u>	<u>1.3</u>	<u>1.6×10^{-8}</u>
^{234}U	0	0	0	0	22
^{235}U	0	0	0	0	2
^{238}U	0	0	0	0	76
^{238}Pu	45	43	24	24	0
^{239}Pu	2	2	5	5	0
^{240}Pu	5	5	7	7	0
^{241}Pu	24	25	32	32	0
^{241}Am	25	25	32	32	0
<u>Lung</u>	<u>1.0</u>	<u>1.7</u>	<u>0.85</u>	<u>1.1</u>	<u>1.3×10^{-6}</u>
^{234}U	0	0	0	0	22
^{235}U	0	0	0	0	2
^{236}U	0	0	0	0	76
^{238}Pu	62	59	38	38	0
^{239}Pu	2	2	6	6	0
^{240}Pu	5	6	9	9	0
^{241}Pu	1	1	2	2	0
^{241}Am	29	31	45	45	0

(a) Pu-U fuel (LWR fuel type 7) used as reference case. Underlined entries are each fuel type's dose commitment to that of Pu-U.

(b) This is produced at the Pu Refabrication Plant because it is an integral part of the core fuel elements.

(c) Percent contribution by each nuclide to the dose commitment.

TABLE 6.6. Relative Dose Commitment per MWd for Plutonium Fuel Refabrication Plant Emissions(a)

Organ	LWR Fuel Type		LMFBR Fuel Type		
	7 Pu-U	8 Pu-Th	A Inner Core	B Outer Core	D U Axial Blanket
Total Body	1.0	1.7	0.47	0.84	2.1×10^{-7}
GI Tract	1.0	1.7	0.52	0.84	1.5×10^{-5}
Thyroid	1.0	1.8	0.62	1.1	3.1×10^{-7}
Bone	1.0	1.6	0.46	0.78	1.2×10^{-7}
Lung	1.0	1.6	0.40	0.66	9.7×10^{-6}

(a) Pu-U fuel (LWR fuel type 7) used as reference case. Entries are each fuel type's dose commitment to that of Pu-U.

at which the hazard index reached a relative maximum due to one or more nuclides attaining its peak ground-water concentration.

The results of the calculations are presented in Table 6.9. The ratio of the ground-water concentration (per MTHM changed to the reactor) to the maximum permissible concentration (MPC) is listed for each nuclide if the ratio is greater than 10^{-3} . These ratios, when summed over the nuclides, provide a hazard index for comparing the potential impacts of wastes from reprocessing and fabrication of the various fuel types.

Several fission products migrate rather quickly (compared to the heavy elements) through the bedded salt and neighboring rocks, and so tend to peak in ground-water concentration comparatively early. The hazard index at 2090 years is dominated by fission products and is highest for the LMFBR inner and outer core wastes. Of the other fuel types, those that contain significant quantities of ^{233}U have somewhat higher hazard indices than the others.

At 8300 years decay time, only ^{135}Cs is present in the biosphere at concentrations comparable to the MPC. The Pu-based fuels tend to produce more of this fission product, which results in a higher biohazard index for wastes from these fuels. Other nuclides have either decayed away or have not yet reached the biosphere in significant concentrations.

TABLE 6.7. Relative Dose Commitment per MTHM for High Activity Uranium Fuel Refabrication Plant Emissions^(a)

<u>Organ</u>	<u>LWR Fuel Type</u>		
	<u>2</u> <u>ReLEU</u>	<u>4</u> <u>ReDU-Th</u>	<u>6</u> <u>ReHEU-Th</u>
<u>Total Body</u>	<u>1.0</u>	<u>2.6</u>	<u>3.0</u>
²²⁸ Th	20% ^(b)	20%	20%
²³² U	84	84	85
²³³ U	6	3	2
Others	1	3	3
<u>GI Tract</u>	<u>1.0</u>	<u>2.4</u>	<u>3.0</u>
²²⁸ Th	50	53	56
²³² U	32	34	35
²³³ U	14	8	6
Others	4	5	3
<u>Thyroid</u>	<u>1.0</u>	<u>2.6</u>	<u>3.2</u>
²³⁸ Th	98	99	99
Others	2	1	1
<u>Bone</u>	<u>1.0</u>	<u>2.6</u>	<u>3.1</u>
²³⁸ Th	32	33	33
²³² U	61	62	64
²³³ U	6	3	2
Others	1	2	1
<u>Lung</u>	<u>1.0</u>	<u>2.4</u>	<u>3.1</u>
²²⁸ Th	40	41	41
²³² U	52	54	55
²³³ U	6	3	2
Others	2	2	2

(a) ReLEU (LWR fuel type 2) used as the reference case. Underlined entries are ratio of each fuel type's dose commitment to that of ReLEU.

(b) Percent contribution by each nuclide to the dose commitment.

TABLE 6.8. Relative Dose Commitments per MWd for High Activity Uranium Fuel Refabrication Plant Emissions^(a)

Organ	LWR Fuel Type		
	2 ReLEU	4 ReDU-Th	6 ReHEU-Th
Total Body	1.0	2.6	2.8
GI Tract	1.0	2.4	2.8
Thyroid	1.0	2.6	2.9
Bone	1.0	2.6	2.8
Lung	1.0	2.4	2.8

(a) ReLEU (LWR fuel type 2) used as reference case. Entries are ratios of each fuel type's dose commitment to that of ReLEU.

The concentrations of uranium nuclides peak at approximately 37,000 years. Wastes from fabrication and reprocessing of fuels that contain ^{233}U tend to result in higher hazard indices than the other fuels.

The biohazards at 230,000 years are dominated by ^{237}Np . This nuclide is produced from neutron captures on ^{236}U as well as by decay of ^{241}Am , which is present in quantity in spent Pu-based fuels. Thus, the recycle U fuels (which contain relatively more ^{236}U) and the Pu-based fuels produce higher concentrations of ^{237}Np .

The only substantial peak at very long times occurs at 1,153,000 years, when plutonium concentration peaks. The only significant contributor here is ^{242}Pu , resulting, of course, in higher hazard indices for the plutonium-based fuels.

It should be recognized that the ground-water concentration values reflected in Table 6.9 are the result of a calculation with an arbitrary set of assumptions for leach rates, ground-water flow, path length, and porosity of the geological media. Changing any of these parameters, as well as changing the nuclide/rock sorption interactions, can change the calculated

TABLE 6.9. Waste Repository Effluents: Number of Maximum Permissible Concentrations (MPCs) in Ground Water per MTHM Charged to Reactor

Time, yrs ^(a)	Nuclide	LWR Fuel Type								LMFBR Fuel Type				
		1	2	3	4	5	6	7	8	A	B	C	D	E
		LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
2090	¹⁴ C	4.2-1 ^(b)	4.2-1	2.7-1	2.8-1	4.3-1	3.8-1	2.0-1	1.4-1	7.1-2	3.6-2	1.0-2	6.4-3	6.3-3
	⁷⁹ Se	1.1+1	2.2+1	1.8+1	2.5+1	2.1+1	3.2+1	8.3+0	1.3+1	1.8+1	1.5+1	1.0+0	6.4-1	2.4+0
	⁹³ Mo	3.1+0	3.1+0	2.2+0	1.6+0	3.0+0	2.4+0	8.1-1	5.2-1	1.0+3	6.8+2	6.4+2	2.3+2	2.0+2
	⁹⁹ Tc	4.1+0	3.7+1	4.2+1	3.7+1	4.1+1	3.8+1	4.0+1	4.2+1	7.3+1	6.2+1	4.1-1	3.3+0	2.6+0
	¹²⁹ I	4.5+2	6.8+2	4.9+2	7.0+2	5.0+2	7.4+2	5.7+2	6.3+2	1.4+3	1.1+3	7.5+1	4.6+1	5.4+1
	Hazard Index	4.7+2	7.5+2	5.5+2	7.0+2	5.7+2	8.1+2	6.2+2	6.8+2	2.5+3	1.9+3	7.2+2	2.8+2	2.6+2
8300	¹³⁵ Cs	4.3+0	4.0+0	2.8+0	6.2+0	4.4+0	4.7+0	9.3+0	1.2+1	3.0+1	2.4+1	1.7+0	1.0+0	9.0-1
	Hazard Index	4.3+0	4.0+0	2.8+0	6.2+0	4.4+0	4.7+0	9.3+0	1.2+1	3.0+1	2.4+1	1.7+0	1.0+0	9.0-1
37,000	²³⁵ Ra	<10 ⁻³	1.1-1	5.5-2	2.0-1	<10 ⁻³	1.9-1	<10 ⁻³	5.9-2	<10 ⁻³	<10 ⁻³	<10 ⁻³	<10 ⁻³	1.1-1
	²²⁵ Ac	<10 ⁻³	1.1-1	5.5-2	2.0-1	<10 ⁻³	1.9-1	<10 ⁻³	5.9-2	<10 ⁻³	<10 ⁻³	<10 ⁻³	<10 ⁻³	1.1-1
	²²⁹ Th	<10 ⁻³	1.6-1	8.3-2	2.9-1	<10 ⁻³	2.9-1	<10 ⁻³	8.8-2	<10 ⁻³	<10 ⁻³	<10 ⁻³	<10 ⁻³	1.7-1
	²³⁴ Th	6.6-2	6.6-2	1.3-2	5.0-2	<10 ⁻³	<10 ⁻³	6.3-2	<10 ⁻³	5.8-2	5.6-2	6.8-2	6.8-2	<10 ⁻³
	²³³ U	1.2-1	1.5+1	7.8+0	2.8+1	<10 ⁻³	2.7+1	1.5-1	8.5+0	1.7-1	2.2-1	1.7-2	1.2-2	1.6+1
	²³⁴ U	5.6-1	3.6+0	7.1-1	1.0+1	8.7-1	6.8+0	8.4+0	1.8+1	1.9+0	2.1+0	5.6-2	5.5-2	1.6-1
	²³⁶ U	7.0-2	1.2-2	2.8-1	2.2-1	2.2-2	1.0+0	3.2-1	5.0-1	9.4-2	1.3-1	3.3-2	3.3-2	<10 ⁻³
	²³⁸ U	3.3-2	3.3-2	6.3-3	2.5-2	<10 ⁻³	<10 ⁻³	3.2-2	10 ⁻⁸	2.9-2	2.8-2	3.4-2	3.4-2	<10 ⁻³
	Hazard Index	8.5-1	1.9+1	8.8+0	3.9+1	8.9-1	3.5+1	9.0+0	2.7+1	2.3+0	2.5+0	2.1-1	2.0-1	1.7+

TABLE 6.9. (contd)

Time, yrs	Nuclide	LWR Fuel Type								LMFBR Fuel Type				
		1	2	3	4	5	6	7	8	A	B	C	D	E
		LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
230,000	²²⁵ Ra	<10 ⁻³	2.3-2	1.2-2	4.2-2	<10 ⁻³	4.1-2	<10 ⁻³	1.3-2	<10 ⁻³	<10 ⁻³	<10 ⁻³	<10 ⁻³	2.4-2
	²²⁵ Ac	<10 ⁻³	2.3-2	1.2-2	4.2-2	<10 ⁻³	4.1-2	<10 ⁻³	1.3-2	<10 ⁻³	<10 ⁻³	<10 ⁻³	<10 ⁻³	2.4-2
	²²⁹ Th	<10 ⁻³	3.5-2	1.8-2	6.3-2	<10 ⁻³	6.2-2	<10 ⁻³	1.9-2	<10 ⁻³	<10 ⁻³	<10 ⁻³	<10 ⁻³	3.7-2
	²³³ Pa	4.9-1	1.9-1	2.8-1	1.6+0	1.9-1	9.5-1	1.8+0	3.3+0	1.6+0	2.4+0	3.2-1	3.9-2	<10 ⁻³
	²³³ U	9.3-2	2.6-2	5.6-2	3.6-1	10 ⁻³	2.2-1	1.2-1	2.3-1	1.4-1	1.8-1	1.2-2	1.3-2	<10 ⁻³
	²³⁷ U	<u>1.6+1</u>	<u>6.5+0</u>	<u>9.3+0</u>	<u>3.5+1</u>	<u>6.4+0</u>	<u>3.2+1</u>	<u>6.1+1</u>	<u>1.1+2</u>	<u>5.5+1</u>	<u>7.9+1</u>	<u>1.1+1</u>	<u>1.3+0</u>	<u>1.0-3</u>
	Hazard Index	1.7+1	6.8+0	9.7+0	3.8+1	6.6+0	3.3+1	6.3+1	1.1+2	5.7+1	8.2+1	1.1+2	1.4+0	8.6-2
1,153,000	²⁴² Pu	<u>2.9-3</u>	<u>2.8-3</u>	<u>1.0-3</u>	<u>1.2-2</u>	<u><10⁻³</u>	<u><10⁻³</u>	<u>1.2-1</u>	<u>2.1-1</u>	<u>3.7-1</u>	<u>7.6-2</u>	<u>2.3-2</u>	<u><10⁻³</u>	<u><10⁻³</u>
	Hazard Index	2.9-3	2.8-3	1.0-3	1.2-2	<10 ⁻³	<10 ⁻³	1.2-1	2.1-1	3.7-1	7.6-2	2.3-2	<10 ⁻³	<10 ⁻³

(a) Time since discharge of fuel from reactor; leaching starts 1000 years after discharge from the reactor.

(b) 4.2-1 = 4.2 x 10⁻¹. Each entry is ratio of ground-water concentration to MPC for nuclide. Hazard index is sum of entries.

concentrations of the nuclides in the ground water as well as the timing of the hazard index peaks. Furthermore, a large dilution factor would apply at the point at which the groundwater enters a river. These values were calculated for the purpose of a general comparison among fuel wastes from varying types, and are useful only for this purpose.

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APPENDIX A

LWR FUEL CYCLE CHARACTERISTICS

LWR FUEL CYCLE CHARACTERISTICS

In a recent PNL study (Heeb et al. 1979), nine alternative LWR fuel cycles were analyzed to determine economic characteristics, natural resource requirements, and energy generation. Eight representative steady-state fuel types have been selected from these cycles as a basis for the present study.

The alternative fuel cycle study traced the material requirements of a generic pressurized water reactor (PWR) over a 30-year period at a constant annual energy output. During the 30 years, a transition was made from uranium-only fuel to the use of a variety of fuel types. Each cycle is characterized by the fuel types used within the cycle. Table A.1 shows the fuel types used in each cycle.

TABLE A.1. LWR Alternative Fuel Cycle Characterization

<u>Cycle</u>	<u>Fuel Types Used(a)</u>	<u>Recycle Materials</u>
A	LEU, Pu-U	U and Pu
B	LEU, Pu-Th, Re LEU	U and Pu
C	LEU, Pu-Th, Re DU-Th	U and Pu
D	LEU, Pu-Th, Re HEU-Th	U and Pu
E	HEU-Th, Re HEU-Th	U
F	DU-Th, Re DU-Th	U
G	DU-Th, Re DU-Th, Pu-Th, Pu-U	U and Pu
H	DU-Th, Re DU-Th, Pu-U	U and Pu
I	HEU-Th, Re LEU	U
Once-through	LEU	None

(a) LEU	is	Low-enriched uranium oxide fuel
ReLEU	is	Low-enriched uranium oxide fuel containing recycled high-activity uranium
DU-Th	is	Denatured uranium-thorium oxide fuel
ReDU-Th	is	Denatured uranium-thorium oxide fuel containing recycled uranium
HEU-Th	is	High-enriched uranium-thorium oxide fuel
ReHEU-Th	is	High-enriched uranium-thorium oxide fuel containing recycled uranium
Pu-Th	is	Plutonium-thorium oxide fuel
Pu-U	is	Plutonium-uranium oxide fuel

At the end of the 30-year period, each alternative cycle was approximately in a steady-state condition, in which the energy contributed by each fuel type within the cycle was nearly constant from year to year. With this condition fulfilled, annual natural resource requirements, quantities of fuel, and quantities of waste generated were also constant.

The nine alternative fuel cycles were compared with the once-through fuel cycle; this latter cycle employs low-enriched uranium fuel only. There is no reprocessing and therefore no recycling of material in this cycle.

Table A.2 shows the quantities of each fuel type used at steady state for the alternative fuel cycles and the throwaway cycle. Annual resource requirements and quantities of discarded material are shown in Table A.3. No recycling of thorium was assumed to occur in the alternative fuel cycle study. Plutonium was recycled in some cycles and discarded in others. Other discarded materials include uranium tails (containing less than 0.2% ^{235}U) from the enrichment process, and depleted uranium, with a net reactivity less than or equal to that of tails.

TABLE A.2. Annual Fuel Quantities (MTHM) for LWR Fuel Cycles at Steady-State

Cycle	Fuel Type							
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th
A	20.3						8.7	
B	21.0	3.2						4.4
C	20.0			4.0				4.4
D	20.0					4.2		4.1
E					8.5	18.1		
F			12.5	14.3				
G			8.0	14.1			4.6	0.8
H			7.2	11.6			7.7	
I		6.3			12.7			
Once-through	28.9							

TABLE A.3. Annual Steady-State Resource Requirements and Discard Quantities (MTHM) for LWR Fuel Cycles

	Fuel Cycle									Once-Through
	A	B	C	D	E	F	G	H	I	
<u>Resource</u>										
Natural U	94.7	97.6	87.6	87.6	67.3	112.4	72.4	66.2	101.0	158.7
Thorium	0	3.9	6.7	7.6	24.9	17.4	15.3	11.6	12.2	0
<u>Discard Material</u>										
Plutonium	0	0	0	0	0	0.14	0	0	0.15	0.28
Depleted Uranium	7.9	0	0	0	0	1.4	0	7.1	0	27.8
Enrichment Tails	75.9	0	91.2	123.4	66.9	109.5	70.5	61.6	100.5	129.7
Thorium	0	3.8	6.5	7.4	24.3	16.9	14.9	11.2	11.9	0

APPENDIX B

FRESH FUEL COMPOSITIONS

FRESH FUEL COMPOSITIONS

Detailed compositions (as assumed for this study) for fabricated fresh fuel are presented in this appendix. Data are provided for the LWR fuel types and LMFBR fuel regions, separated into heavy metal compositions, fuel impurities, and structural material compositions.

Table B.1 shows the heavy metal compositions by nuclide for each alternative fuel and the once-through LEU. Actinides present only in trace quantities are included. Most of the LWR fuel composition information was taken from the 30-year study (Heeb et al., 1979). Similarly, much of the information for LMFBR fuel regions came from a HEDL study on alternative FBR core designs (Haffner et al. 1979).

The concentrations of some isotopes of biological import were not calculated in the above-mentioned studies, and were determined specifically for this study. The quantities of ^{232}U in Re LEU, Re DU-Th, and Re HEU-Th at steady state were calculated by hand using results from ORIGEN.

The ^{238}Pu concentration in the LMFBR core regions was found by assuming that the Pu was obtained from reprocessing spent LWR LEU fuel. This gave a weight fraction of ^{238}Pu in Pu of 0.013.

The uranium composition of the core and blankets of the LMFBR is the same as enrichment cascade tails containing 0.3% ^{235}U . Assuming that this would indeed be the source of the uranium, the ^{234}U concentration was calculated.

The concentrations of all the actinides present in trace amounts in the fuel were found by calculating the decay products of the primary actinides over 6 months, which is the time that elapses between reprocessing and fabrication or refabrication.

In Table B.2, assumed impurity concentrations for thorium and uranium oxide fuels are listed. The trace quantities of these elements do not affect the neutronic performance of these fuels but could give rise to long-lived or

TABLE B.1. Fresh Fuel Compositions (kg/MTHM)

Nuclide	LWR Fuel Type									LMFBR Fuel Type				
	1	2	3	4	5	6	7	8	9	A	B	C	D	E
	LEU	ReLEU	Du-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Once-Through LEU	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
²³² Th	0.0	0.0	767.774	181.942	956.959	930.094	0.0	869.482	0.0	0.0	0.0	0.0	0.0	1000.0
²³² U	0.0	0.0	0.0	0.057	0.0	0.070	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²³³ U	0.0	23.262	0.0	32.067	0.0	27.354	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²³⁴ U	0.500	4.595	0.441	13.170	0.379	8.850	0.028	0.019	0.275	0.012	0.012	0.014	0.014	0.0
²³⁵ U	32.061	1.594	46.444	11.044	40.075	10.773	4.630	0.0	29.999	2.63	2.52	3.00	3.00	0.0
²³⁶ U	0.638	0.733	0.0	25.605	0.0	14.736	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²³⁸ U	961.802	969.816	185.341	736.172	2.587	8.193	921.429	0.0	969.726	874.00	836.00	997.00	997.00	0.0
²³⁸ Pu	0.0	0.0	0.0	0.0	0.0	0.0	3.07	5.05	0.0	1.6	2.1	0.0	0.0	0.0
²³⁹ Pu	0.0	0.0	0.0	0.0	0.0	0.0	34.111	54.138	0.0	71.6	93.7	0.0	0.0	0.0
²⁴⁰ Pu	0.0	0.0	0.0	0.0	0.0	0.0	21.176	42.470	0.0	32.0	40.6	0.0	0.0	0.0
²⁴¹ Pu	0.0	0.0	0.0	0.0	0.0	0.0	11.375	21.235	0.0	14.8	19.4	0.0	0.0	0.0
²⁴² Pu	0.0	0.0	0.0	0.0	0.0	0.0	7.251	12.673	0.0	3.89	5.10	0.0	0.0	0.0
²⁰⁸ Tl	0.0	9.7-11	0.0	2.5-10	0.0	3.1-10	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²¹² Pb	0.0	5.6-8	0.0	1.4-7	0.0	1.8-7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²¹² Bi	0.0	5.3-9	0.0	1.4-8	0.0	1.7-8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²²⁰ Rn	0.0	8.5-11	0.0	2.2-10	0.0	2.7-10	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²²⁴ Ru	0.0	4.9-7	0.0	1.3-6	0.0	1.6-6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²²⁵ Ra	0.0	2.7-10	0.0	3.7-10	0.0	3.2-10	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²²⁶ Ra	9.4-11	4.7-12	1.4-10	3.2-11	1.2-10	3.2-11	1.4-11	0.0	8.8-11	7.7-12	7.4-12	8.8-12	8.8-12	0.0
²²⁵ Ac	0.0	1.8-10	0.0	2.5-10	0.0	2.1-10	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²²⁸ Th	0.0	9.5-5	0.0	2.5-4	0.0	3.0-4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²²⁹ Th	0.0	4.9-5	0.0	6.8-5	0.0	5.8-5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²³⁰ Th	6.9-7	6.3-6	6.1-7	1.8-5	5.2-7	1.2-5	3.9-8	1.3-8	3.8-7	1.7-8	1.7-8	1.9-8	1.9-8	1.98-4
²³¹ Th	1.3-10	6.4-12	1.9-10	4.5-11	1.6-10	4.4-11	1.9-11	0.0	1.2-10	1.1-11	1.0-11	1.2-11	1.2-11	0.0
²³⁴ Th	1.4-8	1.4-8	2.7-9	1.0-8	3.7-11	1.2-10	1.3-8	0.0	1.4-8	1.2-8	1.2-8	1.4-8	1.4-8	0.0
²³⁷ U	0.0	0.0	0.0	0.0	0.0	0.0	4.1-8	6.4-7	0.0	4.5-7	5.9-7	0.0	0.0	0.0
²³⁷ Np	0.0	0.0	0.0	0.0	0.0	0.0	1.1-4	2.1-4	0.0	1.5-4	1.9-4	0.0	0.0	0.0
²⁴¹ Am	0.0	0.0	0.0	0.0	0.0	0.0	2.6-1	4.9-1	0.0	3.5-1	4.6-1	0.0	0.0	0.0

TABLE B.2. Assumed Fuel Impurities

<u>Impurity</u>	<u>g/MT UO₂</u>	<u>g/MT ThO₂</u>
Al	75.0	5.0
Ba	10.0	10.0
Be	0.5	0.5
B	0.1	1.6
Ca	30.0	30.0
Cd	0.2	0.5
C	50.0	50.0
Cl	6.0	6.0
Cr	100.0	100.0
Co	8.0	8.0
Cu	20.0	20.0
F	2.0	2.0
Fe	200.0	100.0
Pb	10.0	10.0
Mg	10.0	20.0
Mn	8.0	8.0
Mo	30.0	30.0
Ni	60.0	60.0
N	10.0	10.0
P	7.0	7.0
Si	60.0	60.0
Ag	0.5	0.5
Sn	3.0	3.0
W	10.0	10.0
V	3.0	10.0
Sm	0.3	3.0
Eu	0.2	3.0
Gd	0.6	1.0
Dy	0.7	1.0
H	2.0	2.0
Li	0.5	0.5
Na	15.0	100.0
Ti	0	10.0
Zn	40.0	40.0

otherwise significant radioactive activation products. For a particular fuel type the impurity levels are estimated by mixing the uranium and thorium impurities. The estimated impurity concentrations were derived from historical fabrication experience on similar fuels (Farwell and Eggert 1971, Crain and Hutchinson, 1966).

Table B.3 summarizes the structural material makeup associated with the LWR and LMFBR fuel type. Table B.4 shows the structural material compositions used in the analysis.

TABLE B.3. Mass, Composition, and Exposure of Fuel Assembly Structural Materials

Structural Material	LWR		LMFBR	
	Mass (kg/MTHM)	Percentage ^(a) Exposure	Mass (kg/MTHM)	Percentage Exposure
Zircaloy-4	235.0	100	0	-
302 SS	4.2	60	0	-
304 SS	37.1	40	0	-
316 SS	0	-	335.6 ^(b)	100
Inconel	12.8	100	0	-
Microbrazed 50	2.6	100	0	-

(a) This is the percentage of the average core exposure a particular type of structural material receives.

(b) This value is for the core and axial blanket regions. For the uranium and thorium radial blankets the appropriate values are 108.3 and 299.3 respectively.

TABLE B.4. Structural Material Compositions

Element	Weight %					
	Zircaloy-4	302 SS	304 SS	316 SS	Inconel	Nicrobraz 50
C	0.012	0.151	0.078	0.050	0.04	0.010
Mn	0.0020	2.018	1.957	1.60	0.20	0.010
Si	-	0.101	0.98	0.56	0.20	-
P	-	0.0454	0.0438	0.005	-	10.31
S	0.0035	0.0303	0.0292	0.007	0.007	0.01
Cr	0.125	18.16	18.54	17.60	19.0	12.95
Ni	0.0020	9.08	8.783	13.70	52.5	76.60
Mo	-	-	-	2.30	3.0	-
Cu	0.0020	-	-	0.01	0.10	-
N	0.0039	-	-	0.010	-	0.0068
Co	0.0010	-	-	0.01	trace	-
V	0.0020	-	-	0.02	-	-
Al	0.0024	-	-	0.01	0.60	0.01
B	Trace	-	-	0.001	-	0.005
Cd	Trace	-	-	0.01	-	-
Ta	-	-	-	0.02	-	-
As	-	-	-	0.01	-	-
Fe	0.225	70.41	69.75	64.08	18.0	0.047
Zr	97.911	-	-	-	-	-
Sn	1.600	-	-	-	-	-
H	0.001	-	-	-	-	-
O	0.0950	-	-	-	-	0.043
Hf	0.0078	-	-	-	-	-
Ti	0.0020	-	-	-	0.800	-
W	0.002	-	-	-	-	-
Nb	-	-	-	-	5.55	-

APPENDIX C

SPENT FUEL COMPOSITIONS

SPENT FUEL COMPOSITIONS

In order to determine the quantity of radioactive effluents and the compositions of waste streams from a fuel reprocessing plant, the radionuclide inventory in the spent fuel must be determined. Tables C.1 through C.4 present radioactive inventory at the time of reprocessing, which serves as a source term both for dose commitment calculations and the waste repository inventory determination.

The contents of the tables are as follows:

Table C.1: Radioactive Fission Product Inventory

Table C.2: Radionuclide Inventory of Actinides and Their Daughters

Table C.3: Radionuclide Inventory of Activated Fuel Impurities

Table C.4: Radionuclide Inventory in Activated Structural Material

In all cases reprocessing occurs 1.5 years after discharge from the reactor.

TABLE C.1. Radioactive Fission Product Inventory (Ci/MTHM) in Spent Fuel--1.5 Years Decay Time

Nuclide	LWR Fuel Type									LMFBR Fuel Type				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Once-Through LEU	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
³ H	4.5+02	5.0+02	4.0+02	4.6+02	3.9+02	4.2+02	5.4+02	5.2+02	4.6+02	3.4+02	2.8+02	2.8+01	1.9+01	7.4+00
⁷⁹ Se	3.3-01	6.9-01	5.4-01	7.8-01	6.3-01	9.9-01	2.5-01	3.5-01	3.3-01	5.6-01	4.6-01	3.2-02	2.0-02	7.4-02
⁸⁵ Kr	7.9+03	1.2+04	1.2+04	1.4+04	1.4+04	1.8+04	4.3+03	6.0+03	7.7+03	1.1+04	9.0+03	6.9+02	4.8+02	1.3+03
⁸⁹ Sr	4.5+02	5.1+02	8.1+02	7.2+02	9.5+02	1.0+03	2.6+02	3.8+02	4.4+02	9.9+02	7.9+02	3.7+01	5.6+01	7.6+01
⁹⁰ Sr	6.9+04	7.5+04	9.7+04	9.1+04	1.1+05	1.1+05	3.3+04	4.0+04	6.8+04	8.0+04	6.3+04	5.5+03	3.7+03	7.8+03
⁹⁰ Y	6.9+04	7.5+04	9.7+04	9.1+04	1.1+05	1.1+05	3.3+04	4.0+04	6.8+04	8.0+04	6.3+04	5.3+03	3.7+03	7.8+03
⁹¹ Y	1.4+03	1.4+03	2.1+03	1.9+03	2.4+03	2.5+03	8.5+02	1.1+03	1.3+03	3.0+03	2.4+03	1.1+02	1.7+02	1.8+02
⁹³ Zr	2.6+01	2.8+00	3.3+00	3.2+00	3.6+00	3.7+00	1.8+00	2.0+00	2.6+00	4.1+00	3.3+00	2.7-01	1.7-01	2.5-01
⁹⁵ Zr	3.7+03	3.7+03	4.6+03	4.1+03	4.8+03	4.8+03	3.1+03	3.5+03	3.7+03	1.1+04	8.5+03	3.6+02	5.1+02	3.3+02
⁹⁵ Nb	8.0+03	7.9+03	9.8+03	8.9+03	1.0+04	1.0+04	6.7+03	7.4+03	8.0+03	2.3+04	1.8+04	7.7+02	1.1+03	7.1+02
⁹⁹ Tc	1.2+02	1.1+01	1.3+01	1.1+01	1.3+01	1.1+01	1.2+01	1.3+01	1.2+01	2.2+01	1.9+01	1.4+00	8.1-01	7.2-01
¹⁰³ Ru	8.3+01	7.8+01	5.5+01	5.8+01	4.0+01	3.4+01	9.8+01	9.2+01	8.4+01	3.4+02	2.7+02	1.0+01	1.4+01	2.0+00
¹⁰⁶ Ru	1.6+05	1.6+05	6.9+04	9.6+04	2.6+04	2.2+04	3.2+05	3.3+05	1.7+05	8.0+05	6.5+05	2.6+04	2.5+04	1.2+03
¹⁰⁷ Pd	9.6-02	9.8-02	3.9-02	5.5-02	1.2-02	9.9-03	2.5-01	2.7-01	1.0-01	3.8-01	3.2-01	1.8-02	1.0-02	5.7-04
¹¹⁰ Ag	8.4+01	8.8+01	3.2+01	4.8+01	6.4+00	7.4+00	2.0+02	1.5+02	9.1+01	4.7+02	2.8+02	4.5+00	3.0+00	5.3-02
^{110m} Ag	6.5+02	6.8+02	2.4+02	3.6+02	5.0+01	5.7+01	1.6+03	1.1+03	7.0+02	6.1+01	3.6+01	5.8-01	3.9-01	6.8-03
¹²³ Sn	1.8+02	2.2+02	1.7+02	2.0+02	1.7+02	2.0+02	2.1+02	2.2+02	1.9+02	1.4+03	1.2+03	4.3+01	5.3+01	1.5+01
¹²⁶ Sn	4.6-01	8.9-01	5.3-01	8.8-01	5.6-01	9.8-01	6.5-01	7.5-01	4.8-01	2.3+00	1.9+00	1.1-01	6.3-02	7.3-02
¹²⁴ Sb	5.9-01	8.4-01	4.9-01	6.3-01	5.2-01	7.1-01	6.7-01	6.4-01	6.2-01	1.3+01	7.5+00	1.6-01	1.4-01	3.9-02

TABLE C.1. (contd)

Nuclide	LWR Fuel Type									LMFBR Fuel Type				
	LEU	ReLEU	OU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Once-Through LEU	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
¹²⁵ Sb	5.2+03	8.6+03	5.2+03	8.2+03	5.2+03	8.5+03	7.2+03	7.9+03	5.3+03	3.2+04	2.6+04	1.3+03	9.0+02	6.2+02
¹²⁶ Sb	4.6-01	8.9-01	5.3-01	8.8-01	5.6-01	9.8-01	6.5-01	7.5-01	4.8-01	2.3+00	1.9+00	1.1-01	6.3-02	7.3-02
^{125m} Te	1.3+03	2.1+03	1.3+03	2.0+03	1.3+03	2.1+03	1.8+03	1.9+03	1.3+03	7.7+03	6.4+03	3.1+02	2.2+02	1.5+02
¹²⁷ Te	3.3+02	5.5+02	4.5+02	6.0+02	5.1+02	7.2+02	4.0+02	5.0+02	3.4+02	1.7+03	1.4+03	5.2+01	6.3+01	5.8+01
^{127m} Te	3.3+02	5.5+02	4.5+02	6.0+02	5.1+02	7.2+02	4.0+02	5.0+02	3.4+02	1.7+03	1.4+03	5.2+01	6.3+01	5.8+01
^{129m} Te	6.9-01	9.3-01	8.3-01	9.9-01	9.0-01	1.0+00	7.5-01	8.5-01	7.0-01	2.8+00	2.3+00	9.1-02	1.3-01	9.0-02
¹²⁹ I	2.7-02	4.2-02	2.9-02	4.2-02	3.0-02	4.4-02	3.4-02	3.8-02	2.7-02	8.1-02	6.7-02	4.5-03	2.7-03	3.3-03
¹³⁴ Cs	6.6+04	6.2+04	6.3+04	5.2+04	6.8+04	6.2+04	5.4+04	5.3+04	6.7+04	2.9+05	1.8+05	7.7+03	4.5+03	2.8+03
¹³⁵ Cs	4.4-01	4.0-01	5.5-01	6.2-01	4.4-01	4.7-01	9.4-01	1.2+00	4.2-01	3.1+00	2.5+00	1.7-01	1.1-01	9.1-02
¹³⁷ Cs	9.4+04	9.8+04	1.0+05	1.0+05	1.0+05	1.1+05	9.4+04	1.0+05	9.4+04	2.0+05	1.6+05	1.1+04	7.0+03	7.2+03
¹⁴¹ Ce	1.1+01	1.1+01	1.3+01	1.2+01	1.3+01	1.4+01	9.7+00	1.1+01	1.0+01	3.4+01	2.7+01	1.1+00	1.6+00	1.0+00
¹⁴⁴ Ce	2.6+05	2.4+05	3.1+05	2.6+05	3.2+05	3.0+05	2.2+05	2.4+05	2.6+05	6.0+05	4.8+05	2.2+04	2.6+04	1.9+04
¹⁴⁷ Pm	1.1+05	9.6+04	1.2+05	1.1+05	1.1+05	1.0+05	1.1+05	1.2+05	1.1+05	1.6+05	1.6+05	1.2+04	1.1+04	7.8+03
¹⁴⁸ Pm	2.4-01	2.1-01	2.9-01	3.1-01	2.3-01	2.3-01	4.5-01	5.3-01	2.3-01	2.9+00	2.3+00	7.7-02	1.0-01	3.6-02
¹⁵¹ Sm	3.1+02	2.8+02	2.4+02	3.9+02	2.2+02	2.1+02	9.2+02	1.3+03	3.0+02	2.5+03	2.4+03	2.0+02	1.5+02	6.3+01
¹⁵⁴ Eu	7.9+03	7.2+03	6.0+03	5.0+03	5.8+03	4.7+03	9.4+03	8.9+03	8.2+03	3.1+04	2.3+04	1.2+03	5.8+02	1.5+02
¹⁵⁵ Eu	4.9+03	4.5+03	3.6+03	3.5+03	3.0+03	2.5+03	9.5+03	1.0+04	5.0+03	5.2+04	3.4+04	1.7+03	9.4+02	1.4+02
¹⁶⁰ Tb	3.1+00	3.2+00	1.1+00	1.5+00	3.9-01	3.6-01	7.6+00	7.9+00	3.3+00	6.9+01	4.4+01	8.6-01	7.1-01	1.4-02

(a) Assumes that 80% of the ³H diffused through the stainless steel clad (LMFBRs only).

TABLE C.2. Radionuclide Inventory (Ci/MTHM) of Actinides and Daughters in Spent Fuel --1.5
Year Decay Time

Nuclide	LWR Fuel Type								Once-Through LEU	LMFBR Fuel Type				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th		Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
²⁰⁸ Tl	3.5-03	7.6+01	9.3+01	2.3+02	1.1+02	3.4+02	1.2-03	3.8+01	1.4-03	2.4-03	1.2-03	6.2-04	1.8-04	1.1+01
²¹² Pb	9.7-03	2.1+02	2.6+02	6.4+02	3.1+02	9.5+02	3.4-03	1.1+02	4.0-03	6.8-03	3.3-03	1.7-03	5.0-04	3.1+01
²¹² Bi	9.7-03	2.1+02	2.6+02	6.4+02	3.1+02	9.5+02	3.4-03	1.1+02	4.0-03	6.8-03	3.3-03	1.7-03	5.0-04	3.1+01
²¹² Po	6.2-03	1.4+02	1.6+02	4.1+02	2.0+02	6.1+02	2.2-03	6.8+01	2.6-03	4.4-03	2.1-03	1.1-03	3.2-04	2.0+01
²¹⁶ Po	9.7-03	2.1+02	2.6+02	6.4+02	3.1+02	9.5+02	3.4-03	1.1+02	4.0-03	6.8-03	3.3-03	1.7-03	5.0-04	3.1+01
²²⁰ Rn	9.7-03	2.1+02	2.6+02	6.4+02	3.1+02	9.5+02	3.4-03	1.1+02	4.0-03	6.8-03	3.3-03	1.7-03	5.0-04	3.1+01
²²⁴ Ra	9.7-03	2.1+02	2.6+02	6.4+02	3.1+02	9.5+02	3.4-03	1.1+02	4.0-03	6.8-03	3.3-03	1.7-03	5.0-04	3.1+01
²²⁸ Th	9.7-03	2.1+02	2.6+02	6.4+02	3.1+02	9.5+02	3.4-03	1.1+02	4.0-03	6.8-03	3.3-03	1.7-03	5.0-04	3.1+01
²³¹ Pa	5.0-05	5.2-04	3.3+00	7.3-01	3.8+00	3.7+00	5.7-06	2.8+00	2.8-05	2.3-07	2.6-07	7.4-07	4.2-07	3.6+00
²³³ Pa	6.6-01	1.4-01	1.4+01	7.9+00	1.9+01	1.8+01	9.1-02	1.3+01	2.5-01	2.4-01	1.7-01	9.8-02	7.0-01	1.2+01
²³² U	2.2-02	2.3+02	4.4+02	7.3+02	5.3+02	1.2+03	7.9-03	1.8+02	8.7-03	2.0-02	9.8-03	3.7-03	1.5-03	4.6+01
²³³ U	1.2-04	4.9+01	1.2+02	1.7+02	1.4+02	2.0+02	2.4-05	1.3+02	7.3-05	1.3-05	1.1-05	8.7-06	6.2-06	2.5+02
²³⁴ U	1.9+00	2.7+01	6.4+00	7.3+01	1.0+01	5.3+01	6.5-01	4.9+00	9.9-01	1.3-02	1.1-02	2.4-03	1.5-03	2.3+00
²³⁵ U	2.0-02	3.6-03	3.8-02	1.8-02	2.6-02	1.2-02	5.7-03	1.8-04	1.8-02	2.7-03	3.2-03	5.0-03	5.4-03	8.1-06
²³⁶ U	5.4-01	6.3-02	3.3-01	1.4+00	3.0-01	8.6-01	2.7-02	1.2-03	2.2-01	1.8-02	1.4-02	8.8-03	6.6-03	2.4-06
²³⁷ U	2.8+00	2.8+00	1.2+00	2.1+00	2.4-03	1.3-02	2.5+01	4.5+01	3.0+00	2.0+01	3.0+01	1.7-02	6.5-03	4.2-10
²³⁸ U	3.1-01	3.2-01	5.9-02	2.4-01	8.6-04	2.7-03	3.0-01	3.3-08	3.2-01	2.6-01	2.6-01	3.2-01	3.2-01	2.3-08
²³⁷ Np	6.6-01	1.4-01	4.1-01	2.8+00	3.5-01	1.7+00	9.2-02	9.0-03	2.5-01	2.4-01	1.7-01	9.8-01	7.0-02	2.5-07
²³⁹ Np	1.1+01	3.9+00	3.8+00	3.6+00	4.6-03	2.6-02	5.4+02	7.5+02	1.2+01	7.5+01	6.7+01	1.0-04	1.9-05	1.2-12
²³⁸ Pu	4.6+03	1.2+03	2.2+03	1.7+04	1.8+03	5.4+03	5.4+04	9.5+04	1.7+03	2.5+04	3.9+04	8.4+01	4.3+01	5.2-05
²³⁹ Pu	3.6+02	3.6+02	1.4+02	3.7+02	7.5-01	3.2+00	1.3+03	1.7+03	3.5+02	5.4+03	6.0+03	1.5+03	1.1+03	7.7-05
²⁴⁰ Pu	4.6+02	5.0+02	1.3+02	3.1+02	5.5-01	2.5+00	3.9+03	7.9+03	4.6+02	8.0+03	9.8+03	1.3+02	6.9+01	4.7-06
²⁴¹ Pu	1.1+05	1.1+05	4.9+04	8.4+04	9.5+01	5.0+02	9.8+05	1.8+06	1.2+05	8.0+05	1.2+06	6.8+02	2.6+02	1.7-05
²⁴¹ Am	3.9+02	3.8+02	1.7+02	3.0+02	3.0-01	1.6+00	4.8+03	1.0+04	4.1+02	4.8+03	7.3+03	3.1+00	8.7-01	7.8-08
²⁴² Am	1.3+01	1.2+01	6.0+00	1.2+01	6.2-03	4.3-02	8.1+02	2.0+03	1.3+01	1.8+02	1.9+02	1.3-02	1.4-03	2.2-10
^{242m} Am	1.3+01	1.2+01	6.0+00	1.2+01	6.2-03	4.3-02	6.3+02	1.4+03	1.3+01	1.8-02	1.9+02	1.3-02	1.4-03	2.2-10
²⁴³ Am	1.1+01	3.9+00	3.8+00	3.6+00	4.6-03	2.6-02	5.4+02	7.5+02	1.2+01	7.5+01	6.7+01	1.0-04	1.9-05	1.2-12
²⁴² Cm	3.3+03	3.1+03	1.4+03	1.9+03	1.6+00	9.8+00	7.4+04	1.2+05	3.7+03	1.3+04	1.3+04	7.6-01	1.3-01	1.3-08
²⁴⁴ Cm	1.3+03	4.2+02	4.5+02	3.5+02	4.6-01	2.5+00	1.3+05	1.3+05	1.5+03	2.2+03	1.4+03	3.5-04	4.7-05	3.0-12

TABLE C.3. Radionuclide Inventory (Ci/MTHM) of Activated Fuel Impurities--1.5 Years Decay Time

Nuclide	LWR Fuel Type									LMFBR Fuel Type				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Once-Through LEU	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
³ H ^(a)	1.1+02	1.0+02	1.0+02	8.0+01	1.1+02	1.0+02	5.3+01	3.0+01	1.1+02	4.3-01	2.8-01	1.5-01	1.0-01	1.0-01
¹⁴ C	2.5-01	2.5-01	2.3-01	1.6-01	2.7-01	2.3-01	1.0-01	6.0-02	2.7-01	3.5-03	2.3-03	1.3-03	9.1-04	9.0-04
⁵⁴ Mn	5.3+00	5.0+00	3.6+00	4.5+00	2.9+00	2.9+00	4.7+00	1.6+00	5.4+00	4.0+01	2.6+01	7.4+00	1.1+01	2.6+00
⁵⁵ Fe	5.7+01	5.7+01	2.9+01	2.8+01	3.2+01	2.7+01	1.3+01	2.9+01	6.1+01	5.6+01	3.7+01	1.5+01	1.5+01	5.7+00
⁵⁸ Co	4.3-01	4.1-01	5.0-01	4.5-01	4.5-01	4.5-01	4.7-01	3.2-01	4.3-01	4.2+00	2.8+00	6.4-01	1.1+00	4.5-01
⁶⁰ Co	7.6+02	7.3+02	7.3+02	5.8+02	7.8+02	7.1+02	4.6+02	2.8+02	7.9+02	2.4+01	1.6+01	7.6+00	6.4+00	5.3+00
⁶³ Ni	3.1+00	3.1+00	2.6+00	1.7+00	3.3+00	2.8+00	7.7-01	3.5-01	3.4+00	5.3-01	3.4-01	1.9-01	1.4-01	1.4-01
⁶⁵ Zn	1.8+01	1.7+01	1.7+01	1.3+01	1.9+01	1.7+01	1.0+01	6.3+00	1.8+01	0.0 ^(b)	0.0	0.0	0.0	0.0
⁹⁹ Tc	8.5-04	8.0-04	8.9-04	7.8-04	8.6-04	8.2-04	7.1-04	4.7-04	8.7-04	1.6-03	1.0-03	5.9-04	4.2-04	4.1-04
¹⁸⁵ W	2.2-01	2.1-01	2.2-01	1.7-01	2.3-01	2.1-01	1.6-01	9.5-02	2.3-01	0.0	0.0	0.0	0.0	0.0

(a) This result assumes 0.5 ppm Li as impurity in fresh fuel.

(b) An entry of 0.0 indicates that the nuclide is not included in the ORIGEN library for the reactor type.

(c) Assumes that 80% of the ³H diffused through the stainless steel clad (LMFBRs only).

TABLE C.4. Radionuclide Inventory (Ci/MTHM) in Activated Standard Materials --1.5 Years Decay Time

Nuclide	LWR Fuel Type								Once-Through LEU	LMFBR Fuel Type				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th		Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
³ H	3.2-02	3.2-02	3.4-02	3.6-02	3.2-02	3.3-02	4.4-02	4.9-02	3.2-02	8.2-01	6.2-01	2.2-01	2.6-01	1.6-01
¹⁴ C	1.7-01	1.8-01	1.5-01	1.2-01	1.8-01	1.6-01	9.9-02	8.7-02	1.8-01	7.3-02	3.6-02	9.4-03	5.9-03	5.8-03
²² Na	0.0(a)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.0+01	2.0+01	3.3+00	6.8+00	2.6+00
⁵⁴ Mn	1.6+02	1.6+02	1.5+02	1.5+02	1.6+02	1.6+02	1.7+02	9.2+01	1.6+02	4.4+04	3.0+04	5.6+03	1.0+04	4.4+03
⁵⁵ Fe	4.2+03	4.5+03	3.2+03	2.3+03	4.4+03	3.8+03	1.2+03	2.5+02	4.5+03	6.0+04	4.1+04	1.1+04	1.4+04	8.7+03
⁵⁸ Co	6.3+01	6.3+01	6.6+01	6.6+01	6.5+01	6.8+01	8.4+01	8.2+01	6.2+01	3.3+03	2.2+03	3.4+02	7.6+02	2.7+02
⁶⁰ Co	3.9+03	4.0+03	3.4+03	2.9+03	4.0+03	3.7+03	2.7+03	2.3+03	4.0+03	9.4+02	6.4+02	2.0+02	2.1+02	1.6+02
⁵⁹ Ni	3.3+00	3.6+00	2.6+00	1.8+00	3.5+00	3.0+00	9.4-01	5.4-01	3.6+00	4.9+00	3.4+00	1.3+00	1.1+00	1.0+00
⁶³ Ni	5.3+02	5.7+02	4.1+02	3.0+02	5.6+02	4.9+02	1.6+02	1.0+02	5.7+02	1.6+02	1.1+02	4.0+01	3.6+01	3.1+01
⁶⁵ Zn	9.4+00	1.0+01	6.9+00	5.0+00	9.9+00	8.5+00	4.1+00	3.3+00	1.0+01	0.0	0.0	0.0	0.0	0.0
⁹¹ Y	2.0-02	2.0-01	1.9-01	1.8-01	2.1-01	2.1-01	2.1-01	2.1-01	2.0-01	0.0	0.0	0.0	0.0	0.0
⁹⁵ Zr	7.3+01	7.6+01	6.3+01	5.5+01	7.7+01	7.3+01	5.7+01	5.2+01	7.6+01	0.0	0.0	0.0	0.0	0.0
^{93m} Nb	1.4-02	1.5-02	1.3-02	1.1-02	1.4-02	1.3-02	9.9-03	9.1-03	1.5-02	4.2-01	2.8-01	1.7-01	9.6-02	1.3-01
⁹⁵ Nb	1.6+02	1.6+02	1.3+02	1.2+02	1.6+02	1.6+02	1.2+02	1.1+02	1.6+02	1.1-03	7.2-04	1.1-04	2.3-04	8.1-05
⁹⁹ Tc	1.8-02	1.1-02	1.0-02	9.8-03	1.1-02	1.0-02	1.1-02	1.0-02	1.1-02	4.0-01	2.7-01	1.1-01	9.4-02	8.2-02
¹²⁵ Sb	2.6+01	2.8+01	2.0+01	1.4+01	2.7+01	2.4+01	7.3+00	4.8+00	2.8+01	0.0	0.0	0.0	0.0	0.0
^{125m} Te	1.1+01	1.2+01	8.1+00	5.8+00	1.1+01	9.8+00	3.0+00	2.0+00	1.2+01	0.0	0.0	0.0	0.0	0.0

(a) An entry of 0.0 indicates that the nuclide is not included in the ORIGEN library for the reactor.

(b) Assumes that 80% of the ³H diffused through the stainless-steel clad (LMFBRs only).

APPENDIX D

WASTE COMPOSITIONS

WASTE COMPOSITIONS

Material stored in the waste repository has two possible origins: the fuel reprocessing plant (FRP) and the fuel fabrication plants. Wastes from the FRP include vitrified high-level and intermediate-level wastes and packaged fuel residue (hardware and leached hulls). The radionuclide inventory in FRP wastes is shown in Table D.1.

Also stored in the repository are wastes generated at the fuel fabrication plants. Losses to waste were assumed to be 0.5% of plant input. Table D.2 presents the radionuclide waste inventory from the fabrication plants.

In all cases, the wastes are assumed to have decayed for 1000 years, at which time the repository breach by groundwater occurs and leaching begins.

TABLE D.1. Radionuclide Inventory (Ci/MTIM Charged to Reactor) in Fuel Reprocessing Plant Wastes Stored in Repository^(a)

Nuclide	LWR Fuel Type								LMFBR Fuel Type				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
¹⁴ C	3.85-1	3.81-1	3.32-1	2.53-1	3.94-1	3.48-1	1.79-1	1.30-1	6.52-2	3.31-2	9.16-3	5.83-3	5.78-3
⁵⁹ Ni	3.57+0	3.58+0	2.55+0	1.83+0	3.49+0	3.03+0	9.40-1	5.37-1	4.89+0	3.33+0	1.28+0	1.13+0	1.00+0
⁶³ Ni	3.10-1	3.11-1	2.24-1	1.62-1	3.03-1	2.64-1	8.82-2	5.41-2	9.09-3	6.18-3	2.37-3	2.10-3	1.84-3
⁷⁹ Se	3.31-1	6.79-1	5.38-1	7.71-1	6.27-1	9.75-1	2.51-1	3.46-1	5.56-1	4.52-1	3.20-2	1.95-2	7.28-2
⁸⁷ Rb	1.94-5	2.70-5	2.98-5	3.23-5	3.37-5	4.14-5	9.74-6	1.33-5	2.55-5	2.02-5	1.73-6	1.12-6	3.10-6
⁹⁰ Sr	1.39-6	1.51-6	1.95-6	1.83-6	2.15-6	2.29-6	6.66-7	8.07-7	1.61-6	1.27-6	1.11-7	7.53-8	1.58-7
⁹³ Zr	2.74+0	2.89+0	3.42+0	3.23+0	3.66+0	3.81+0	1.87+0	2.01+0	4.08+0	3.27+0	2.63-1	1.68-1	2.53-1
⁹³ Mo	1.16-2	1.17-2	8.37-3	5.96-3	1.13-2	9.14-3	3.02-3	1.94-3	3.73+0	2.55+0	2.39+0	8.66-1	7.62-1
⁹⁹ Tc	1.23+1	1.10+1	1.28+1	1.13+1	1.25+1	1.14+1	1.20+1	1.28+1	2.21+1	1.86+1	1.25-1	1.00+0	8.00-1
¹⁰⁷ Pd	9.64-2	9.80-2	3.86-2	5.50-2	1.15-2	9.90-3	2.47-1	2.73-1	3.78-1	3.18-1	1.48-5	1.05-2	5.67-4
¹²⁶ Sn	4.62-1	8.86-1	5.25-1	8.77-1	5.55-1	9.72-1	6.48-1	7.40-1	2.30+0	1.89+0	1.09-1	6.23-2	7.28-2
¹²⁹ I	2.71-2	4.17-2	2.93-2	4.18-2	2.99-2	4.44-2	3.43-2	3.76-2	8.11-2	6.65-2	4.53-3	2.75-3	3.27-3
¹³⁵ Cs	4.35-1	4.01-1	5.51-1	6.24-1	4.43-1	4.73-1	9.40-1	1.17+0	3.07+0	2.47+0	1.72-1	1.05-1	9.14-2
¹³⁷ Cs	9.68-6	1.01-5	1.06-5	1.04-5	1.07-5	1.11-5	9.70-6	1.06-5	2.06-5	1.65-5	1.15-6	7.27-7	7.49-7
¹⁵¹ Sm	1.83-1	1.63-1	1.85-1	2.28-1	1.28-1	1.23-1	5.36-1	7.32-1	1.45+0	1.41+0	1.18-1	8.58-2	3.70-2
²²⁶ Ra	4.82-5	2.87-4	1.27-4	8.02-4	1.65-4	6.01-4	7.87-4	1.72-3	1.33-4	1.37-4	2.97-2	1.52-7	1.04-4
²²⁹ Th	1.37-4	2.71-2	6.64-2	9.40-2	7.47-2	1.12-1	9.26-5	7.04-2	1.23-4	1.45-4	1.93-5	1.37-5	1.36-1
²³⁰ Th	2.69-4	1.52-3	5.13-4	4.22-3	6.84-4	2.99-3	4.73-3	1.02-2	8.13-4	8.43-4	1.68-6	8.62-7	2.66-4
²³² Th	0.0	0.0	4.99-4	1.17-4	6.22-4	6.06-4	0.0	5.64-4	4.27-11	4.78-11	3.21-12	2.26-12	6.47-4
²³¹ Pa	5.11-5	5.10-4	3.24+0	7.16-1	3.73+0	3.60+0	6.45-6	2.71+0	9.14-7	1.06-6	1.46-6	1.18-6	3.50+0
²³² U	1.77-8	2.96-6	1.75-4	3.91-5	2.12-4	1.94-4	6.84-9	7.16-5	2.10-8	1.01-8	2.98-9	1.52-9	1.86-5

TABLE D.1. (contd)

Nuclide	LWR Fuel Type								LMFBR Fuel Type				
	LEU	ReLEU	DJ-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
²³³ U	3.01-3	2.96-1	7.30-1	1.04+0	8.21-1	1.23+0	2.56-3	7.76-1	3.20-3	3.96-3	4.19-4	2.98-4	1.49+0
²³⁴ U	3.37-2	1.77-1	4.91-2	4.81-1	6.59-2	3.32-1	6.52-1	1.41+0	1.16-1	1.20-1	2.10-4	1.09-4	1.40-2
²³⁵ U	1.26-4	2.44-5	2.32-4	1.10-4	1.60-4	7.61-5	5.01-5	2.10-5	4.92-5	5.59-5	3.97-5	3.97-5	4.94-8
²³⁶ U	3.43-3	4.95-4	2.09-3	8.95-3	1.80-3	5.24-3	1.02-2	1.05-2	1.60-3	1.84-3	7.62-5	5.15-5	1.44-8
²³⁸ U	1.91-3	1.92-3	3.62-4	1.46-3	5.24-6	1.66-5	1.83-3	6.16-8	1.61-3	1.59-3	1.96-3	1.98-3	1.38-10
²³⁷ Np	7.29-1	2.07-1	4.41-1	2.85+0	3.49-1	1.72+0	9.03-1	1.75+0	1.05+0	1.38+0	9.82-2	6.99-2	2.48-7
²³⁸ Pu	1.31-1	1.16-1	6.19-2	1.48-1	4.63-3	1.40-2	7.45+0	1.80+1	1.63+0	1.71+0	3.24-4	1.20-4	1.34-10
²³⁹ Pu	2.41+0	2.22+0	9.29-1	2.28+0	4.54-3	1.97-1	2.25+1	3.01+1	3.42+1	3.74+1	8.99+0	6.59+0	4.55-7
²⁴⁰ Pu	5.66+0	3.73+0	1.80+0	2.55+0	4.16-3	2.00-2	3.39+2	3.52+2	4.92+1	5.74+1	7.33-1	3.80-1	2.58-8
²⁴¹ Pu	1.87-1	5.54-2	6.95-2	4.99-2	6.06-5	3.35-4	3.87+1	2.95+1	1.43-1	6.41-2	3.74-9	3.55-10	2.24-17
²⁴² Pu	9.55-3	9.35-3	3.31-3	4.02-3	4.90-6	2.69-5	2.41-1	4.35-1	1.22-1	1.52-1	2.74-6	5.91-7	4.78-14
²⁴⁴ Pu	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.91-9	1.77-9	4.92-16	1.64-16	0.0
²⁴¹ Am	8.37+1	8.21+1	3.61+1	6.34+1	6.44-2	3.55-1	1.04+3	2.20+3	1.01+3	1.52+3	6.63-1	1.87-1	1.66-8
²⁴³ Am	1.02+1	3.54+0	3.46+0	3.29+0	4.19-3	2.32-2	4.96+2	6.82+2	6.83+1	6.16+1	9.18-5	1.71-5	1.12-12
²⁴² Cm	1.09-1	1.03-1	5.17-2	9.92-2	5.33-5	3.66-4	6.91+0	1.69+1	1.54+0	1.62+0	1.08-4	1.17-5	1.89-12
²⁴³ Cm	1.06-9	1.01-9	3.61-10	3.28-10	4.47-13	2.52-12	1.30-8	1.12-8	3.20-8	2.17-8	2.68-13	3.10-14	3.26-21
²⁴⁴ Cm	3.32-14	1.04-14	1.14-14	8.43-15	1.16-17	6.22-17	3.89-12	3.13-12	3.83-12	2.34-12	6.48-19	2.13-19	0.0
²⁴⁵ Cm	1.86-1	5.53-2	6.94-2	4.98-2	6.05-5	3.34-4	3.86+1	2.94+1	1.43-1	6.40-2	3.74-9	3.55-10	2.43-2
²⁴⁶ Cm	2.59-2	7.46-3	8.37-3	4.04-3	3.58+1	3.91-5	3.85+1	1.63+0	6.97-3	2.11-3	2.95-11	1.98-12	1.03-19
²⁴⁷ Cm	1.07-7	7.73-7	3.57-8	3.50-8	0.0	0.0	2.30-5	7.03-6	3.77-8	7.96-9	4.99-18	1.48-18	0.0

(a) This inventory includes vitrified HLLW and ILLW and packaged fuel residue (hulls and hardware). A decay time of 1000 years is assumed.

TABLE D.2. Radionuclide Inventory (Ci/MTHM Charged to Reactor) in Fuel Fabrication Plant Wastes Stored in Repository^(a)

Nuclide	LWR Fuel Type								LMFBR Fuel Type				
	LEU	ReLEU	DU-Th	ReDU-Th	HEU-Th	ReHEU-Th	Pu-U	Pu-Th	Inner Core	Outer Core	U Radial Blanket	U Axial Blanket	Th Radial Blanket
²²⁹ Th	0.0	9.97-2	0.0	1.38-1	0.0	1.18-1	3.03-6	5.66-6	3.94-6	5.16-6	0.0	0.0	0.0
²³⁰ Th	1.34-4	1.23-3	1.18-4	3.51-3	1.02-4	2.36-3	8-13-4	1.32-3	4.23-4	5.54-4	3.74-6	3.74-6	1.92-5
²³² Th	0.0	0.0	4.19-4	9.94-5	5.23-4	5.05-4	0.0	4.74-4	0.0	0.0	0.0	0.0	5.45-4
²³² U	0.0	1.53-4	0.0	3.98-4	0.0	4.88-4	0.0	0.0	0.0	0.0	0.0	0.0	0.0
²³³ U	0.0	1.10+0	0.0	1.52+0	0.0	1.30+0	8.76-5	1.64-4	1.14-4	1.49-4	0.0	0.0	0.0
²³⁴ U	1.55-2	1.42-1	1.37-2	4.08-1	1.18-2	2.74-1	9.40-2	1.53-1	4.89-2	6.40-2	4.36-3	4.36-3	0.0
²³⁵ U	3.43-4	1.71-5	4.97-4	1.18-4	4.29-4	1.16-4	5.97-5	1.60-5	4.94-5	5.47-5	3.21-5	3.21-5	0.0
²³⁶ U	1.79-3	2.33-4	0.0	8.10-3	0.0	4.67-3	6.44-4	1.29-3	9.74-4	1.24-3	0.0	0.0	0.0
²³⁸ U	1.60-3	1.62-3	3.10-4	1.23-3	4.31-6	1.37-5	1.54-3	3.80-8	1.46-3	1.40-3	1.66-3	1.66-3	0.0
²³⁷ Np	0.0	0.0	0.0	0.0	0.0	0.0	3.26-2	6.10-2	4.24-2	5.55-2	0.0	0.0	0.0
²³⁸ Pu	0.0	0.0	0.0	0.0	0.0	0.0	1.06-1	1.75-1	5.54-2	7.27-2	0.0	0.0	0.0
²³⁹ Pu	0.0	0.0	0.0	0.0	0.0	0.0	1.02+1	1.61+1	2.14+1	2.79+1	0.0	0.0	0.0
²⁴⁰ Pu	0.0	0.0	0.0	0.0	0.0	0.0	2.11+1	4.23+1	3.18+1	4.04+1	0.0	.0	0.0
²⁴² Pu	0.0	0.0	0.0	0.0	0.0	0.0	1.41-1	2.47-1	7.57-2	9.89-2	0.0	0.0	0.0
²⁴¹ Am	0.0	0.0	0.0	0.0	0.0	0.0	4.06+1	7.60+1	5.28+1	6.92+1	0.0	0.0	0.0

(a) Assuming 0.5% heavy element losses to waste, and 1000 years decay time.

APPENDIX E

MISCELLANEOUS DATA

MISCELLANEOUS DATA

The dose commitment factors that were used to calculate the relative dose commitments reported in this study are shown in Table E.1. These are 50-year population dose commitment factors for atmospheric releases at a generic mid-western site. The factors are in units of person-rem/50 years per curie of nuclide released, and are given for four organs and the total body.

The generic midwestern site is characterized as follows:

- There is fenced boundary, including 4.7 km^2 in a rectangular form 2.35 km by 2.0 km. A moderately large river runs through one corner of the site.
- The average population density out to an 80 km radius is 174 persons per square kilometer, making a total population of 3.5×10^6 persons.
- The nearest dwelling is a farm house located 1.3 km from the site. A milk cow eats from the pasture for seven months of the year and the garden is irrigated by water from the river.
- Drinking water and boating and swimming are the main uses of the river. Fish taken from the river are consumed at an average rate of 6.9 kg/yr.
- The site receives 61 cm (24 in.) of precipitation every year.
- The average annual wind speed is 14.8 km/hr (9.2 mph). Air releases are from a 10 m stack.
- The river average annual flow rate is $1420 \text{ m}^3/\text{sec}$ (50,000 cfs).

The models used to calculate radiation dose factor have been developed by Brenchley et al. 1977.

Table E.2 presents nuclide sorption coefficients (k_d) for the dolomites and sandstones, which are the rock strata surrounding the bedded salt site

TABLE E.1. Fifty-Year Population Dose Commitment Factors
(Person-rem/50 years per Ci/year released)

Nuclide	Total Body	GI Tract	Thyroid	Bone	Lung
³ H	1.28E-02	1.28E-02	1.28E-02	0.00E+00	1.28E-02
¹⁴ C	8.08E-01	8.08E-01	8.08E-01	4.04E+00	8.08E-01
²² Na	2.77E+01	2.77E+01	2.77E+01	2.77E+01	2.77E+01
⁵⁴ Mn	2.84E+00	3.61E+00	2.78E+00	2.78E+00	5.93E+00
⁵⁵ Fe	3.21E-02	7.07E-01	7.55E-07	1.99E-01	2.17E-01
⁵⁸ Co	8.42E-01	1.53E+00	7.80E-01	7.80E-01	2.86E+00
⁶⁰ Co	4.34E+01	4.58E+01	4.31E+01	4.31E+01	5.65E+01
⁵⁹ Ni	1.52E-01	7.01E-02	0.00E+00	9.09E-01	1.47E-01
⁶³ Ni	4.00E-01	1.88E-01	0.00E+00	1.19E+01	4.01E-01
⁶⁵ Zn	2.60E+00	3.01E+00	1.49E+00	2.26E+00	3.43E+00
⁷⁹ Se	1.10E+01	1.35E+01	5.98E-02	5.98E-02	8.63E-01
⁸⁵ Kr	4.75E-05	4.75E-05	4.75E-05	4.75E-05	9.08E-05
⁸⁹ Sr	3.42E-01	2.59E+00	8.85E-05	1.19E+01	3.14E+00
⁹⁰ Sr	5.52E+02	6.50E+01	5.18E-06	2.42E+03	2.16E+01
⁹⁰ Y	2.66E-04	1.15E+00	1.40E-04	4.83E-03	3.81E-01
⁹¹ Y	3.02E-02	3.28E+00	2.21E-03	1.04E+00	3.83E+00
⁹³ Zr	5.18E-02	1.50E-01	2.72E-02	9.67E-01	4.10E-01
⁹⁵ Zr	5.67E-01	1.81E+00	5.15E-01	7.57E-01	4.49E+00
^{93m} Nb	4.80E-02	2.63E-01	3.20E-03	5.62E-01	5.62E-01
⁹⁵ Nb	2.97E-01	1.10E+00	2.87E-01	3.19E-01	1.42E+00
⁹⁹ Tc	2.70E-02	3.38E+00	1.25E-06	6.73E-02	1.82E+00
¹⁰³ Ru	2.30E-01	1.03E+00	2.27E-01	2.35E-01	1.36E+00
¹⁰⁶ Ru	8.63E-01	1.04E+01	8.49E-01	1.12E+00	2.19E+01
¹⁰⁷ Pd	7.35E-02	7.12E+00	0.00E+00	0.00E+00	1.70E-01
^{110m} Ag	6.98E+00	1.72E+01	6.95E+00	7.00E+00	1.74E+01
¹¹⁰ Ag	3.11E-03	2.08E+00	2.53E-03	5.35E-03	4.21E-01
¹²³ Sn	6.44E-02	3.34E+00	4.37E-02	1.85E+00	5.19E+00
¹²⁶ Sn	2.60E+00	5.40E+00	2.30E+00	8.40E+00	1.30E+00
¹²⁴ Sb	1.29E+00	4.82E+00	1.23E+00	1.39E+00	6.87E+00
¹²⁵ Sb	4.67E+00	5.84E+00	4.62E+00	4.83E+00	8.61E+00
¹²⁶ Sb	1.26E+00	3.37E+00	1.25E+00	1.27E+00	2.98E+00
^{125m} Te	3.30E-02	1.02E+00	7.02E-02	2.26E-01	7.07E-01
¹²⁷ Te	3.22E-05	1.29E-01	3.39E-05	3.46E-05	1.47E-02
^{127m} Te	1.08E-01	3.22E+00	2.27E-01	8.87E-01	2.16E+00
^{129m} Te	1.39E-02	3.91E+00	2.54E-01	6.61E-01	2.65E+00
¹²⁹ I	3.96E+01	2.74E+01	1.01E+04	3.14E+01	2.68E+01
¹³⁴ Cs	2.53E+01	1.40E+01	1.37E+01	1.97E+01	1.53E+01
¹³⁵ Cs	8.11E-01	4.09E-02	6.04E-07	1.98E+01	2.08E-01
¹³⁷ Cs	2.78E+01	2.08E+01	2.06E+01	2.86E+01	2.19E+01
¹⁴¹ Ce	3.20E-02	8.51E-01	2.86E-02	7.35E-02	8.41E-01
¹⁴⁴ Ce	5.54E-01	8.61E+00	1.39E-01	7.87E+00	1.76E+01

TABLE E.1. (contd)

Nuclide	Total Body	GI Tract	Thyroid	Bone	Lung
¹⁴⁷ Pm	5.74E-02	5.01E-01	7.12E-07	1.51E+00	1.19E+00
¹⁴⁸ Pm	4.95E-02	1.28E+00	4.89E-02	5.58E-02	7.51E-01
¹⁵¹ Sm	3.87E-01	6.36E-01	3.23E-01	1.87E+00	1.12E+00
¹⁵⁴ Eu	2.84E+01	3.04E+01	2.72E+01	4.06E+01	3.77E+01
¹⁵⁵ Eu	1.18E+00	1.56E+00	1.01E+00	2.83E+00	2.71E+00
¹⁶⁰ Tb	1.02E+00	2.89E+00	9.71E-01	1.37E+00	4.42E+00
¹⁸⁵ W	6.04E-04	7.18E-01	3.02E-06	1.71E-02	1.00E+00
²⁰⁸ Tl	0.0	0.0	0.0	0.0	0.0
²¹² Pb	6.70E-02	5.40E-02	0.0	9.20E-01	2.00E+01
²¹² Bi	4.30E-03	2.60E-05	0.0	6.40E-03	2.20E+00
²¹² Po	0.0	0.0	0.0	0.0	0.0
²¹⁶ Po	0.0	0.0	0.0	0.0	0.0
²²⁰ Rn	0.0	0.0	0.0	0.0	0.0
²²⁴ Ra	6.70E+00	1.30E+00	0.0	2.10E+01	4.40E+02
²²⁸ Th	1.32E+02	4.10E+01	9.47E+00	3.62E+03	1.82E+04
²²⁹ Th	1.00E+04	6.50E+00	5.70E-01	2.00E+05	5.10E+04
²³⁰ Th	1.40E+03	6.90E-01	2.00E-03	4.60E+04	9.10E+03
²³¹ Th	2.90E-04	5.60E-02	0.00E+00	6.40E-03	3.40E-02
²³² Th	1.80E+03	5.90E-01	0.00E+00	5.20E+04	1.30E+04
²³¹ Pa	3.57E+03	5.59E+00	1.29E+00	9.15E+04	1.03E+03
²³³ Pa	2.40E-01	7.63E-01	2.37E-01	2.58E-01	8.69E-01
²³² U	8.08E+01	4.22E+00	1.40E-02	1.13E+03	3.99E+03
²³³ U	1.46E+01	3.94E+00	3.31E-02	2.40E+02	9.56E+02
²³⁴ U	1.43E+01	3.87E+00	4.25E-02	2.30E+02	9.38E+02
²³⁵ U	1.32E+01	4.59E+00	2.93E-03	2.18E+02	8.80E+02
²³⁶ U	1.37E+01	3.60E+00	1.24E-03	2.21E+02	8.98E+02
²³⁷ U	1.58E-02	3.12E-01	1.56E-02	1.65E-02	1.99E-01
²³⁸ U	1.23E+01	3.23E+00	7.29E-04	2.09E+02	8.23E+02
²³⁷ Np	1.25E+03	1.29E+01	8.14E+00	3.04E+04	9.46E+02
²³⁹ Np	5.80E-03	2.76E-01	5.78E-03	6.29E-03	9.02E-02
²³⁸ Pu	1.24E+03	4.14E+00	7.11E-03	4.93E+04	3.27E+03
²³⁹ Pu	1.39E+03	3.78E+00	4.58E-03	5.73E+04	3.09E+03
²⁴⁰ Pu	1.39E+03	3.85E+00	7.54E-03	5.72E+04	3.09E+03
²⁴¹ Pu	2.32E+01	9.82E-02	1.93E-02	1.15E+03	2.75E+00
²⁴² Pu	1.60E+03	3.90E+00	2.80E-04	3.10E+04	1.00E+04
²⁴¹ Am	1.21E+03	5.24E+00	1.03E+00	1.82E+04	1.09E+03
²⁴² Am	0.0	0.0	0.0	0.0	0.0
^{242m} Am	1.21E+03	6.48E+00	1.19E+00	1.84E+04	4.40E+02
²⁴⁵ Am	1.19E+03	1.25E+01	7.54E+00	1.82E+04	1.04E+03
²⁴² Cm	1.77E+01	4.01E+00	1.37E-03	2.67E+02	7.04E+02
²⁴⁴ Cm	6.32E+02	4.46E+00	1.28E-02	1.06E+04	1.09E+03

assumed for the waste repository. The k_d value relates the transport velocity of each nuclide to the groundwater velocity:

$$\frac{V_{\text{nuclide}}}{V_{\text{ground water}}} = \frac{1}{1 + \beta k_d}$$

where β is a parameter relating to the bulk density porosity of the medium. For this study a β representative of sedimentary rock was used.

Also included in Table E.2 are the maximum permissible concentrations (MPC) of each nuclide in water intended for public release (10 CFR 20). These values were used in determining the relative hazard indices in the waste repository analysis.

TABLE E.2. Nuclide Sorption Coefficients for Bedded Salt Site and Maximum Permissible Concentration

<u>Nuclide</u>	<u>Sorption Coefficient (ml/gm)</u>	<u>Maximum Permissible Concentration (MPC), $\mu\text{Ci/ml}$</u>
^{14}C	0	8×10^{-4}
^{59}Ni	30	2×10^{-4}
^{63}Ni	30	3×10^{-5}
^{79}Se	0	3×10^{-5}
^{87}Rb	0.5	1×10^{-4}
^{90}Sr	0.5	3×10^{-7}
^{93}Zr	500	8×10^{-4}
^{93}Mo	0	3×10^{-6}
^{99}Tc	0	3×10^{-4}
^{107}Pd	200	3×10^{-6}
^{126}Sn	30	3×10^{-6}
^{129}I	0	6×10^{-8}
^{135}Cs	0.5	1×10^{-4}
^{138}Cs	0.5	2×10^{-5}
^{151}Sm	50	4×10^{-4}
^{210}Pb	30	1×10^{-7}
^{225}Ra	3	3×10^{-6}
^{226}Ra	3	3×10^{-8}
^{225}Ac	500	2×10^{-6}
^{229}Th	500	3×10^{-6}
^{230}Th	500	2×10^{-6}
^{232}Th	500	2×10^{-6}
^{234}Th	500	2×10^{-5}
^{231}Pa	300	9×10^{-7}
^{233}Pa	300	1×10^{-4}
^{232}U	3	3×10^{-5}
^{233}U	3	3×10^{-5}
^{234}U	3	3×10^{-5}
^{235}U	3	3×10^{-5}
^{236}U	3	3×10^{-5}
^{238}U	3	4×10^{-5}
^{237}Np	20	3×10^{-6}
^{238}Pu	100	5×10^{-6}
^{239}Pu	100	5×10^{-6}
^{240}Pu	100	5×10^{-6}
^{241}Pu	100	2×10^{-4}
^{242}Pu	100	5×10^{-6}
^{244}Pu	100	4×10^{-6}
^{241}Am	500	4×10^{-6}
^{243}Am	500	4×10^{-6}
^{243}Cm	500	5×10^{-6}
^{244}Cm	500	7×10^{-6}
^{245}Cm	500	4×10^{-6}
^{246}Cm	500	4×10^{-6}
^{247}Cm	500	4×10^{-6}

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