DOE/NE-0003/6
UC-2

PRELIMINARY SAFETY AND ENVIRONMENTAL INFORMATION DOCUMENT

VOLUME VI

LIQUID-METAL FAST-BREEDER REACTORS

January 1980

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM

U.S. DEPARTMENT OF ENERGY
ASSISTANT SECRETARY FOR NUCLEAR ENERGY
WASHINGTON, D.C. 20545

Available from:

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

Price: Printed Copy: $ 8.00
Microfiche: $ 3.00

This report has been prepared jointly by Argonne National Laboratory, Oak Ridge National Laboratory, and NUS Corporation for the Office of Fuel Cycle Evaluation, U.S. Department of Energy.

FOREWORD

The Department of Energy (DOE) Nonproliferation Alternative Systems Assessment Program (NASAP) is a planned program of studies of nuclear power systems, with particular emphasis on identifying and then evaluating alternative nuclear reactor/fuel-cycle systems that have acceptable proliferation-resistance characteristics and that offer practical deployment possibilities domestically and internationally. The NASAP was initiated in 1977, in response to President Carter's April 1977 Nuclear Power Policy Statement.

The NASAP objectives are to (1) identify nuclear systems with high proliferation resistance and commercial potential, (2) identify institutional arrangements to increase proliferation resistance, (3) develop strategies to implement the most promising alternatives, and (4) provide technical support for U.S. participation in the International Nuclear Fuel Cycle Evaluation (INFCE) Program.

The NASAP is not an assessment of all future energy-producing alternatives. Rather, it is an attempt to examine comprehensively existing and potentially available nuclear power systems, thus providing a broader basis for selecting among alternative systems. The assessment and evaluation of the most promising reactor/fuel-cycle systems will consider the following factors: (1) proliferation resistance, (2) resource utilization, (3) economics, (4) technical status and development needs, (5) commercial feasibility and deployment, and (6) environmental impacts, safety, and licensing.

The DOE is coordinating the NASAP activities with the U.S. Nuclear Regulatory Commission (NRC) to ensure that their views are adequately considered at an early stage of the planning. In particular, the NRC is being asked to review and identify licensing issues on systems under serious consideration for future research, development, and demonstration. The Preliminary Safety and Environmental Information Document (PSEID) is the vehicle by which NASAP will provide information to the NRC for its independent assessment. The PSEID contains the safety and environmental assessments of the principal systems. Special safeguards measures will be considered for fuel cycles that use uranium enriched in U-235 to 20% or more, uranium containing U-233 in concentrations of 12% or more, or plutonium. These measures will include the addition of radioactivity to the fuel materials (i.e., spiking), the use of radioactive sleeves in the fresh fuel shipping casks, and other measures. The basis for the safeguards review by NRC is contained in Appendix A.

The information contained in this PSEID is an overlay of the present safety, environmental, and licensing efforts currently being prepared as part of the NASAP. It is based on new material generated within the NASAP and other reference material to the extent that it exists. The intent of this assessment is to discern and highlight on a consistent basis any safety or environmental issues of the alternative systems that are different from a reference LWR once-through case and may affect their licensing. When issues exist, this document briefly describes research, development, and demonstration requirements that would help resolve them within the normal engineering development of a reactor/fuel-cycle system.

The preparation of this document takes into consideration the NRC responses to the DOE preliminary safety and environmental submittal of August 1978. Responses to these initial comments have been, to the extent possible, incorporated into the text. Comments by the NRC on this PSEID were received in mid-August 1979 and, as a result of these comments, some changes were made to this document. Additional comments were incorporated as Appendix B. Comments that are beyond the scope and resources of the NASAP may be addressed in research, development, and demonstration programs on systems selected for additional study. The intent of this document (and the referenced material) is to provide sufficient information on each system so that the NRC can independently ascertain whether the concept is fundamentally licensable.

This PSEID was prepared for the DOE through the cooperative efforts of the Argonne National Laboratory, the Oak Ridge National Laboratory, and NUS Corporation.

TABLE OF CONTENTS

	Page
FOREWORD	i
CHAPTER 1--GENERAL DESCRIPTION	1-1
1.1 DESIGN GROUND RULES	1-1
1.2 DESIGN DESCRIPTION	1-4
1.2.1 Homogeneous Cores (General Electric Designs)	1-4
1.2.2 Heterogeneous Cores (Westinghouse Designs)	1-4
1.3 NUCLEAR ANALYSIS AND PERFORMANCE	1-5
1.3.1 Homogeneous Cores (General Electric Designs)	1-5
1.3.2 Heterogeneous Cores (Westinghouse Designs)	1-5
1.4 THERMAL-HYDRAULIC ANALYSIS AND PERFORMANCE	1-6
1.4.1 Homogeneous Cores (General Electric Designs)	1-6
1.4.2 Heterogeneous Cores (Westinghouse Designs)	1-6
1.5 MECHANICAL DESIGN AND PERFORMANCE	1-7
1.5.1 Homogeneous Cores (General Electric Designs)	1-7
1.5.2 Heterogeneous Cores (Westinghouse Designs)	1-7
REFERENCES FOR CHAPTER 1	1-8
CHAPTER 2--URANIUM-PLUTONIUM/URANIUM RECYCLE: HOMOGENEOUS LMFBR CORE (LMFBR U-Pu/U/U RECYCLE)	2-1
2.1 DESCRIPTION	2-1
2.2 FUEL MANAGEMENT	2-1
REFERENCES FOR CHAPTER 2	2-6
CHAPTER 3--URANIUM-PLUTONIUM/URANIUM SPIKED RECYCLE: HETEROGENEOUS LMFBR CORE (LMFBR U-Pu/U RECYCLE)	3-1
3.1 DESCRIPTION	3-1
3.2 FUEL MANAGEMENT	3-1
REFERENCES FOR CHAPTER 3	3-6
CHAPTER 4--URANIUM-PLUTONIUM/URANIUM SPIKED RECYCLE: HOMOGENEOUS LMFBR CORE (LMFBR U-Pu/U RECYCLE)	4-1
4.1 DESCRIPTION	4-1
4.2 FUEL MANAGEMENT	4-1
CHAPTER 5--URANIUM-PLUTONIUM/THORIUM SPIKED RECYCLE: HETEROGENEOUS LMFBR CORE (LMFBR U-Pu/U/Th/Th RECYCLE)	5-1
5.1 DESCRIPTION	5-1
5.2 FUEL MANAGEMENT	5-1
REFERENCES FOR CHAPTER 5	5-6
CHAPTER 6--URANIUM-PLUTONIUM/THORIUM SPIKED RECYCLE: HOMOGENEOUS LMFBR CORE (LMFBR U-Pu/Th RECYCLE)	6-1
6.1 DESCRIPTION	6-1
6.2 FUEL MANAGEMENT	6-1
REFERENCES FOR CHAPTER 6	6-5

TABLE OF CONTENTS (continued)

	Page
CHAPTER 7--THORIUM-PLUTONIUM/THORIUM SPIKED RECYCLE: HOMOGENEOUS LMFBR CORE	7-1
7.1 DESCRIPTION	7-1
7.2 FUEL MANAGEMENT	7-1
REFERENCES FOR CHAPTER 7	7-7
CHAPTER 8--DENATURED URANIUM-233/THORIUM CYCLE: HOMOGENEOUS LMFBR CORE	8-1
8.1 DESCRIPTION	8-1
8.2 FUEL MANAGEMENT	8-1
REFERENCES FOR CHAPTER 8	8-7
CHAPTER 9--SAFETY CONSIDERATIONS FOR THE LMFBR	9-1
9.1 INTRODUCTION	9-1
9.2 KEY LMFBR SAFETY ISSUES	9-2
9.3 STATUS OF KEY ISSUES	9-3
9.3.1 Scram-System Reliability	9-3
9.3.2 Reliability of the Residual-Heat-Removal System	9-3
9.3.3 Accommodation of Subassembly Faults	9-3
9.3.4 Integrity of the Heat-Transport System	9-3
9.3.5 Containment Design to Cope with Sodium Hazards	9-4
9.3.6 Containment Design to Mitigate Radioactivity Releases from Events Within the Design Basis	9-4
9.3.7 Accommodation of Core-Melt and Core-Disruptive Accidents	9-4
9.4 SAFETY CONSIDERATIONS ASSOCIATED WITH ALTERNATIVE CORE DESIGNS AND FUEL CYCLES	9-7
9.4.1 Low-Void-Worth Core Designs	9-7
9.4.2 Alternative Fuel Systems	9-7
REFERENCES FOR CHAPTER 9	9-8
CHAPTER 10--ENVIRONMENTAL CONSIDERATIONS	10-1
10.1 SUMMARY ASSESSMENT	10-1
10.2 REACTOR AND STEAM-ELECTRIC SYSTEM (R.G. 4.2/3.2)	10-1
10.3 STATION LAND USE	10-1
10.4 STATION WATER USE (R.G. 4.2/3.3)	10-1
10.5 HEAT-DISSIPATION SYSTEM (R.G. 4.2/3.4)	10-2
10.6 RADIOACTIVE-WASTE SYSTEMS AND EFFLUENT SOURCE TERMS	10-2
10.6.1 Source Terms (R.G. 4.2/3.5.1)	10-2
10.6.2 Liquid-Radwaste System and Effluent Source Terms (R.G. 4.2/3.5.2)	10-3
10.6.3 Gaseous Radwaste System and Effuent Source Terms (R.G. 4.2/3.5.3)	10-4
10.6.4 Solid-Radwaste System (R.G. 4.2/3.5.4)	10-4
10.7 CHEMICAL AND BIOCIDAL WASTES (R.G. 4.2/3.6)	10-4
10.7.1 Chlorine	10-4
10.7.2 Other Chemical Wastes	10-5
10.8 EFFECTS OF THE HEAT-DISSIPATION SYSTEM (R.G. 4.1/5.1)	10-5
10.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATIONS (R.G. 4.1/5.2)	10-5
10.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES	10-5

TABLE OF CONTENTS (continued)

	Page
10.11 OCCUPATIONAL EXPOSURE	10-5
10.12 EFFECTS OF ALTERNATIVE FUEL CYCLES	10-6
10.12.1 Uranium-Plutonium/Thorium Spiked Recycle	10-6
10.12.2 Thorium-Plutonium/Thorium Spiked Recycle	10-7
10.12.3 Denatured Uranium-233/Thorium Cycle	10-8
REFERENCE FOR CHAPTER 10	10-15
CHAPTER 11--LICENSING STATUS AND CONSIDERATIONS	11-1
REFERENCES FOR CHAPTER 11	11-2
CHAPTER 12--RESEARCH, DEVELOPMENT, AND DEMONSTRATION	12-1
12.1 RESEARCH FACILITIES	12-1
12.1.1 In-Reactor Facilities	12-1
12.1.2 Out-Of-Reactor Facilities	12-1
12.1.3 Out-Of-Reactor "Laboratory-Scale" Facilities	12-2
12.2 ANALYTICAL DEVELOPMENT	12-3
12.2.1 Steady-State Fuel-Rod Characterization	12-3
12.2.2 Whole-Core Analysis	12-3
12.2.3 Subassembly Analysis	12-4
12.2.4 Heat-Transport System Analysis	12-4
12.2.5 Post-Accident Heat Removal (PAHR)	12-4
12.2.6 Sodium Fires	12-5
12.2.7 Aerosol Behavior	12-5
12.2.8 Environmental-Release Analysis	12-5

ADDENDUM I CARBIDE-FUELED 1,000-MWe LMFBR: PLUTONIUM-URANIUM CARBIDE CORE AND URANIUM CARBIDE BLANKETS

APPENDIX A U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

APPENDIX B Responses to Comments by the U.S. Nuclear Regulatory Commission--PSEID, Volume VI, Liquid-Metal Fast-Breeder Reactors

LIST OF TABLES

Table		Page
1-1	Ground-rule parameters for the Proliferation-Resistant LMFBR Core Design Study	1-2
1-2	Fuel-assembly parameters for the oxide-fueled LMFBR	1-3
1-3	Blanket-assembly parameters for the oxide-fueled LMFBR	1-3
2-1	Performance summary for the LMFBR $(Pu,U)O_2/UO_2/UO_2$ homogeneous reference core	2-2
2-2	Summary of main parameters for the LMFBR $(Pu,U)O_2/UO_2$ homogeneous reference core	2-3
2-3	Equilibrium-cycle reactor charge and discharge data for the LMFBR homogeneous U-Pu/U recycle core	2-4
3-1	Summary of main parameters for the $(Pu,U)O_2/UO_2$ heterogeneous LMFBR core	3-2
3-2	Equilibrium-cycle reactor charge and discharge data for LMFBR U-Pu/U spiked recycle, heterogeneous core	3-4
5-1	Summary of main parameters for the $(Pu,U)O_2/ThO_2$ heterogeneous LMFBR core	5-2
5-2	Equilibrium-cycle reactor charge and discharge data for the LMFBR U-Pu/Th spiked recycle, heterogeneous core	5-4
6-1	Performance summary for the $(Pu,U)O_2/ThO_2$ homogeneous LMFBR core	6-2
6-2	Equilibrium-cycle reactor charge and discharge data for the LMFBR U-Pu/Th spiked recycle, homogeneous core	6-3
7-1	Performance summary for the $(Th,Pu)O_2/ThO_2$ homogeneous reference core	7-2
7-2	Summary of main parameters for the $(Th,Pu)O_2/ThO_2$ homogeneous LMFBR core	7-3
7-3	Equilibrium-cycle reactor charge and discharge data for the Th-Pu/Th spiked recycle LMFBR homogeneous core	7-5
8-1	Performance summary for the LMFBR denatured U-233/Th homogeneous core	8-2
8-2	Summary of main parameters for the LMFBR denatured U-233/Th homogeneous core	8-3
8-3	Equilibrium-cycle reactor charge and discharge data for the LMFBR denatured U-233/Th homogeneous core	8-5
10-1	Design data for the LMFBR heat-dissipation system (wet natural-draft cooling tower)	10-9
10-2	Estimated radionuclide releases for a 1,000-MWe LMFBR power plant, and reference LWR	10-9

LIST OF FIGURES

Figure		Page
2-1	Material flow diagram, LMFBR U-Pu/U/U recycle homogeneous core (General Electric design)	2-5
3-1	Material flow diagram, LMFBR U-Pu/U spiked recycle heterogeneous core (Westinghouse large-pin design)	3-5
4-1	Material flow diagram, LMFBR U-Pu/U/U spiked recycle homogeneous core (General Electric design)	4-2
5-1	Material flow diagram, LMFBR U-Pu/U/Th/Th spiked recycle, heterogeneous core (Westinghouse large-pin transmuter)	5-5
6-1	Material flow diagram, LMFBR U-Pu/Th/Th spiked recycle, homogeneous core (General Electric transmuter design)	6-4
7-1	Material flow diagram, LMFBR Th-Pu/Th spiked recycle, homogeneous core (General Electric design)	7-6
8-1	Material flow diagram, LMFBR denatured U-233/Th cycle homogeneous core (General Electric design)	8-6
10-1	Relationship of radioactive waste systems to reactor systems	10-10
10-2	Pathways for tritium release in LMFBR power plants (Ref. 1)	10-11
10-3	Liquid-radwaste-system flow diagram	10-12
10-4	Schematic diagram of the radioactive-gas-processing system	10-13
10-5	Solid-waste-system flow diagram	10-14

Chapter 1

GENERAL DESCRIPTION

The designs considered here are based on a 1,000-MWe oxide-fueled liquid-metal fast breeder reactor (LMFBR) power plant, with the balance of plant undefined. The core designs were developed for the Proliferation-Resistant LMFBR Core Design Study (PRLCDS) program, which was initiated by the U.S. Department of Energy (DOE) in October 1977 and concluded in September 1978.

1.1 DESIGN GROUND RULES

A common set of ground rules developed for the PRLCDS program applies to all core designs considered in this volume. Tables 1-1 through 1-3 summarize several of the more important parameters. Section 3 of Reference 1 presents a complete discussion of the ground rules used as bases for these core designs.

Table 1-1. Ground-rule parameters for the Proliferation-Resistant LMFBR Core Design Study

General parameters	
Reactor lifetime, years	30
Net power, MWe	1,000[a]
Thermal efficiency[b]	0.365
Reactor inlet temperature, °F	650
Core temperature rise, °F	280
Flow parameters	
Maximum pin-bundle coolant velocity, ft/sec	35[c]
Maximum pin-bundle pressure drop exclusive of entry and exit losses, psi	90
Bypass flow, %	5[d]
Fuel management	
Plant capacity factor, %	70
Refueling interval	Multiples of 6 months
Number of core batches	Open
Residence time, years	
Driver fuel assemblies	Open
Blanket fuel assemblies	≤6
Number of enrichment zones	Open
Out-of-reactor time, years	
Plutonium, fissile	1.00
Uranium-233, fissile	1.33
Combined fabrication/reprocessing loss, %	1.0

[a]This value was chosen to allow the use of turbine-generator systems designed during the Prototype Large Breeder Reactor studies.

[b]Defined as the ratio of the gross electrical power (turbine-generator output) to gross thermal power (reactor power plus pumping heat input).

[c]This value represents a moderate advance in technology.

[d]Fraction of the total flow that is unheated; the remainder is available for cooling driver and blanket assemblies.

Table 1-2. Fuel-assembly parameters for the oxide-fueled LMFBR

Subassembly pitch	Open[a]
Spacer type	Wire wrap
Spacer pitch, in.	12
Minimum cladding thickness, mils	12
Minimum cladding thickness-to-outside-diameter ratio	0.039
Minimum driver-pin pitch-to-diameter ratio	Open[b]
Nominal peak linear power, kW/ft	Open
Plenum location	Split or top
Vented ducts	Not allowed
Maximum nominal subassembly outlet temperature, °F	1,075
Maximum core height	Open[c]
Smear density, % of theoretical	90.0
Maximum cladding O.D. temperature, °F	Open

[a]This design option should not be construed as allowing ductless cores. The development of extremely large subassemblies should be avoided.

[b]The designer should justify driver-pin pitch-to-diameter ratios lower than 1.15.

[c]Core-size effects on capital costs should be considered.

Table 1-3. Blanket-assembly parameters for the oxide-fueled LMFBR

Identical designs for internal and radial blanket assemblies	Required
Minimum cladding thickness, mils	12
Minimum cladding thickness-to-outside-diameter ratio	0.0229
Nominal peak linear power, kW/ft	Open
Maximum smear density, % of theoretical	95.0

1.2 DESIGN DESCRIPTION

1.2.1 HOMOGENEOUS CORES (GENERAL ELECTRIC DESIGNS)

The homogeneous-core designs were developed by the General Electric Company. The core-design parameters, fuel-management, and assembly-design parameters are described in Section 4 of Reference 1.

1.2.2 HETEROGENEOUS CORES (WESTINGHOUSE DESIGNS)

The heterogeneous core designs were developed by the Westinghouse Electric Corporation. The design methodology and techniques used are described in Section 2 of Reference 2.

1.3 NUCLEAR ANALYSIS AND PERFORMANCE

1.3.1 HOMOGENEOUS CORES (GENERAL ELECTRIC DESIGNS)

The nuclear performance evaluations for the General Electric homogeneous cores are described in Section 5 of Reference 1.

1.3.2 HETEROGENEOUS CORES (WESTINGHOUSE DESIGNS)

The detailed neutronic parameters of the Westinghouse heterogeneous designs are described in Section 3 of Reference 2.

1.4 THERMAL-HYDRAULIC ANALYSIS AND PERFORMANCE

1.4.1 HOMOGENEOUS CORES (GENERAL ELECTRIC DESIGNS)

The thermal-hydraulic performance evaluations for the General Electric homogeneous cores are described in Section 6 of Reference 1.

1.4.2 HETEROGENEOUS CORES (WESTINGHOUSE DESIGNS)

The thermal-hydraulic analyses for the Westinghouse heterogeneous cores are described in Section 4 of Reference 2.

1.5 MECHANICAL DESIGN AND PERFORMANCE

1.5.1 HOMOGENEOUS CORES (GENERAL ELECTRIC DESIGNS)

The fuel mechanical design and performance evaluations for the General Electric homogeneous cores are described in Section 7 of Reference 1.

1.5.2 HETEROGENEOUS CORES (WESTINGHOUSE DESIGNS)

The fuel mechanical designs and performance evaluations for the Westinghouse heterogeneous cores are described in Section 5 of Reference 2.

REFERENCES FOR CHAPTER 1

1. General Electric Company, Preconceptual Design Study of Proliferation Resistant Homogeneous Oxide LMFBR Cores, GEFR-00392, November 1978.

2. Westinghouse Electric Corporation, Preconceptual Study of Proliferation Resistant Heterogeneous Oxide Fueled LMFBR Core, Final Report, November 1978.

Chapter 2

URANIUM-PLUTONIUM/URANIUM RECYCLE: HOMOGENEOUS LMFBR CORE (LMFBR U-Pu/U/U RECYCLE)

2.1 DESCRIPTION

This reactor/fuel-cycle combination is a liquid-metal fast-breeder reactor (LMFBR) using recycled coprocessed uranium/plutonium mixed oxide in a homogeneous core and recycled uranium mixed with makeup depleted uranium in the axial-blanket and radial-blanket assemblies. The core fuel is reprocessed separately from the blanket assemblies. All of the coprocessed, recovered plutonium/uranium from the core is mixed with makeup uranium and some of the fissile uranium/plutonium recovered from blanket reprocessing for feed material to core fabrication. The remaining excess coprocessed uranium/plutonium from blanket reprocessing is sent to secure storage for later use in light-water reactors (LWRs) or LMFBRs. All other recovered uranium from blanket reprocessing is recycled to blanket fabrication after being mixed with makeup depleted uranium. Wastes from core fabrication and reprocessing are sent to a geologic waste repository. Wastes from blanket fabrication are sent to a low-level shallow land disposal site.

The fuel-cycle facilities associated with this reactor fuel-cycle combination are shown in the mass-flow diagram (Figure 2-1) and are discussed in the following sections of Volume VII:

Blanket fabrication 1	Chapter 4
Core fabrication 2	Chapter 4
Core processing (Purex 2)	Section 5.2
Blanket reprocessing (Purex 2)	Section 5.2
Plutonium storage	Section 6.2
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The key performance results for the homogeneous mixed-oxide LMFBR core are summarized in Table 2-1, and the significant core-design parameters are presented in Table 2-2. This design is identified in the preconceptual design study (Ref. 1) as $(Pu,U)O_2/UO_2$ reference.

2.2 FUEL MANAGEMENT

Charge and discharge data for the equilibrium-cycle reactor are given in Table 2-3 and the mass-flow diagram in Figure 2-1.

Table 2-1. Performance summary for the LMFBR $(PU,U)O_2/UO_2/UO_2$ homogeneous reference core[a]

Breeding ratio	1.32
Doubling time, years	14.8
Fuel-cycle cost, mills/kW-hr[b]	7.1
Total fissile mass at beginning of equilibrium cycle, kg	3,635
Net fissile gain, kg/yr	
Plutonium	231
Uranium-233	0
Total	231
Average core discharge burnup, MWd/kg	61
Peak discharge burnup, MWd/kg	95
Burnup reactivity loss, % $\Delta k/k$	2.0
Core voiding reactivity, % $\Delta k/k$	2.5
Delayed-neutron fraction	0.0036
Core Doppler coefficient, -T(dk/dT)	0.0060

[a]Core fuel is $(Pu,U)O_2$; radial blanket fuel is UO_2; axial blanket is UO_2.

[b]The fuel-cycle costs are based on the assumptions specified in the Proliferation-Resistant LMFBR Core Design Study. They do not necessarily represent General Electric's best estimates of the fuel-cycle costs.

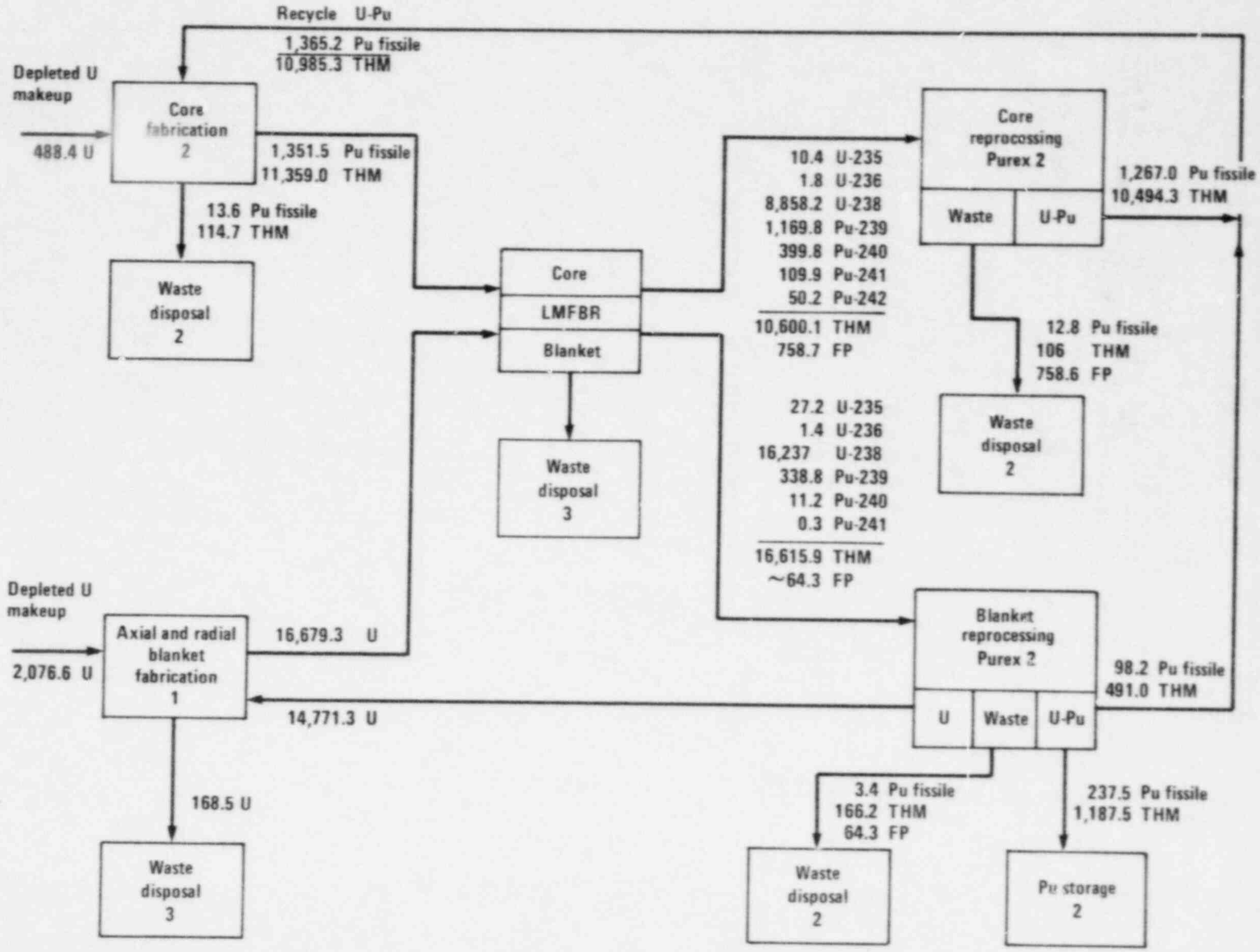
Table 2-2. Summary of main parameters for the LMFBR $(PU,U)O_2/UO_2$ homogeneous reference core

General reactor data	
Reactor power, MWt	2,740
Net electric power, MWe	1,000
Reactor vessel temperature difference, °C	156
Reactor vessel outlet temperature, °C	499
Core fissile enrichment (Pu-239 + Pu-241 + U-233/Pu + U + Th), %	
Inner zone	10.24
Outer zone	14.36
Total fissile inventory at beginning of equilibrium cycle, kg	3,464
Total heavy metal at beginning of equilibrium cycle, kg	85,200
Number of subassemblies	
Drivers, zone 1	150
Drivers, zone 2	102
Internal blanket	0
Control	19
Radial blanket	198
Volume fractions in active core, %	
Fuel	39.22
Sodium	43.93
Steel	16.85
Control	0
Number of core orifice zones	7
Driver residence time, years	2.5
Radial-blanket residence time, row 1, years	3.75
Peak discharge burnup, MWd/kg	92.6
Average discharge burnup, MWd/kg	61.1
Peak neutron flux (E >0.1 MeV), n/cm^2-sec	3.62×10^{15}
Peak neutron fluence (E >0.1 MeV), n/cm^2	1.97×10^{23}
Peak cladding temperature, °C	
Nominal	558
2σ	625
Peak linear power at end-of-equilibrium cycle, W/cm	
Nominal	443
3σ + 15%	564
Sodium void worth, $	
Fresh core	
End-of-equilibrium cycle	7.0
Doppler coefficient, $\Delta k/^{o}C$	0.0060
Breeding ratio	1.32
Fissile gain, kg/cycle	308
Compound system doubling time, years	15
Fuel-cycle cost, mills/kW-hr	6.5
Maximum cumulative damage factor, steady-state	0.03

Table 2-3. Equilibrium-cycle reactor charge and discharge data for the LMFBR homogeneous U-Pu/U recycle core[a]

Isotope	Charge[b] Core	Charge[b] Axial blanket	Charge[b] Radial blanket	Discharge[b] Core	Discharge[b] Axial blanket	Discharge[b] Radial blanket
Thorium-232						
Protactinium-233						
Uranium-232						
Uranium-233						
Uranium-234						
Uranium-235	19.3	13.7	19.2	10.4	11.5	15.7
Uranium-236				1.8	0.6	0.8
Uranium-238	9,611.0	6,937.2	9,709.2	8,858.2	6,793.9	9,443.1
Plutonium-238						
Plutonium-239	1,174.7			1,169.8	121.8	217.0
Plutonium-240	335.3			399.8	3.8	7.4
Plutonium-241	176.8			109.9	0.1	0.2
Plutonium-242	41.8			50.2		
Total	11,358.9	6,950.9	9,728.4	10,600.1	6,931.7	9,684.2
Fission products[c]				758.7	19.7	44.6

[a]General Electric $(Pu,U)O_2/UO_2$ reference.
[b]Mass flows in kilograms per 0.75 GWe-yr.
[c]Total 823.0.



Mass flows in kg per 0.75 GWe-yr. Abbreviations: THM, total heavy metal, FP, fission products.
Data base from Reference 2.

Figure 2-1. Material flow diagram, LMFBR U-Pu/U/U recycle homogeneous core (General Electric design).

REFERENCES FOR CHAPTER 2

1. General Electric Company, Preconceptual Design Study of Proliferation Resistant Homogeneous Oxide LMFBR Cores, GEFR-00392, November 1978.

2. Letter from W. C. Lipinski, Argonne National Laboratory, to Tulwar, General Electric Company, October 2, 1978.

Chapter 3

URANIUM-PLUTONIUM/URANIUM SPIKED RECYCLE: HETEROGENEOUS LMFBR CORE (LMFBR U-Pu/U RECYCLE)

3.1 DESCRIPTION

This reactor/fuel-cycle combination is a liquid-metal fast-breeder reactor (LMFBR) using 14.8% fissile assay mixed uranium-plutonium recycle fuel in the core and depleted uranium in the blanket assemblies. The core and blanket assemblies are reprocessed separately. The core is coprocessed, and all of the recovered uranium and plutonium is mixed with a portion of the uranium-plutonium recovered during blanket reprocessing to provide feed material to core fabrication. The excess uranium-plutonium recovered during blanket reprocessing is adjusted to 20% fissile plutonium content and pre-irradiated before storage or sale. The balance of the uranium recovered during blanket reprocessing is mixed with makeup uranium to provide feed material for blanket fabrication. Wastes from reprocessing and core fabrication are sent to a geologic waste repository. Wastes from blanket fabrication are sent to a low-level shallow land disposal site.

The fuel-cycle facilities associated with this reactor fuel-cycle combination are shown in the mass-flow diagram (Figure 3-1) and are discussed in the following sections of Volume VII:

Blanket fabrication 1	Chapter 4
Core fabrication 2	Chapter 4
Core reprocessing (Purex 2)	Section 5.2
Blanket reprocessing (Purex 2)	Section 5.2
Plutonium storage	Section 6.2
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The significant core-design parameters, including the fissile-mass gain per year, are summarized in Table 3-1. For details of design-data specifications, including performance characteristics, see Table J-0 of Reference 1. This design is identified in Reference 1 as $(Pu,U)O_2$ fuel, UO_2 blanket, Reference 1, 7.9-mm-O.D. fuel.

3.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 3-2, and the mass flow diagram in Figure 3-1.

Table 3-1. Summary of main parameters for the $(Pu,U)O_2/UO_2$ heterogeneous LMFBR core[a]

General reactor data	
Reactor power, MWt	[b]2,740
Net electric power, MWe	[b]1,000
Reactor vessel temperature difference, °C	[b]156
Reactor vessel outlet temperature, °C	[b]499
Core enrichment, Pu/heavy metal, at beginning of first core, %	
Zone 1	Single zone,
Zone 2 wt%	18.9
Total fissile inventory at beginning of equilibrium cycle (fuel/blanket), kg	4,525/678
Total heavy metal at beginning of equilibrium cycle, kg	112.367
Number of subassemblies	
Drivers--zone 1	270
Drivers--zone 2	
Inner blanket	121
Control	30
Radial blanket	318
Removable shield	186
Volume fractions in fuel	
Fuel	0.46
Sodium	0.35
Steel	0.19
Volume fractions in blanket	
Oxide	0.58
Sodium	0.27
Steel	0.15
Number of core orifice zones	13
Driver residence time, years	3
Radial blanket residence time, years	6
Peak discharge burnup, MWd/kg	89.2
Average discharge burnup, MWd/kg	56.3
Peak neutron flux ($E > 0.1$ MeV), n/cm^2-sec	[c]2.6×10^{15}
Peak neutron fluence ($E > 0.1$ MeV), n/cm^2	[d]2.2×10^{23}
Peak cladding temperature for lifetime-limiting rod, end of life, midwall, °C (°F)	
Nominal	566 (1,033)
2σ	629 (1,164)
Peak linear power, W/cm (kW/ft)	
Nominal	466 (14.2)
3σ + overpower	577 (17.6)
Sodium void worth at end of equilibrium cycle, $	
Fuel and axial blankets	+2.9
Inner blankets and extensions	+1.0
Doppler coefficient ($-T(dk/dT) \times 10^4$)	
Fuel	60
Isothermal	117
Breeding ratio	1.44

Table 3-1. Summary of main parameters for the $(Pu,U)O_2/UO_2$ heterogeneous LMFBR core[a] (continued)

General reactor data (continued)	
Fissile gain, kg/cycle	282
Compound system doubling time, years	
Based on beginning-of-equilibrium-cycle fuel fissile mass and without the pre-equilibrium buildup correction	16
With pre-equilibrium buildup correction	18
Fuel cycle cost, mills/kW-hr	9.0
Maximum cumulative damage function	0.14
Fuel-assembly parameters	
Pins per assembly	271
Duct wall thickness, mm	3.81
Duct outside flat to flat, cm	15.99
Fuel-pin pitch-to-diameter ratio, compressed	1.15
Wire diameter, mm	1.18
Assembly pitch, cm	16.32
Rod-bundle pressure difference, kPa (psi)	613 (89)
Maximum mixed mean outlet temperature, nominal, ^{o}C (^{o}F)	568 (1,055)
Driver-pin parameters	
Pin outside diameter, mm	7.874
Cladding thickness, mm	0.3302
Fuel height, cm	122
Axial blanket height, cm	35.56
Plenum volume, cm^3	55.8
Smear density, % of theoretical	91.0
Blanket-assembly parameters	
Pins per assembly	127
Duct wall thickness, mm	3.05
Duct outside flat to flat, cm	15.94
Pin outside diameter, mm	12.48
Pin pitch-to-diameter ratio, compressed	1.07
Assembly pitch, cm	16.32
Assembly fueled height, cm	193.04
Plenum volume, cm^3	145.2
Peak linear pin power, W/cm (kW/ft)	
Inner blanket, nominal	476 (14.5)
Radial blanket, nominal	456 (13.6)
Plutonium gain, kg/yr	282

[a]Core fuel $(Pu,U)O_2$; blanket fuel UO_2.
[b]Ground rule.
[c]Fuel.
[d]Blanket.

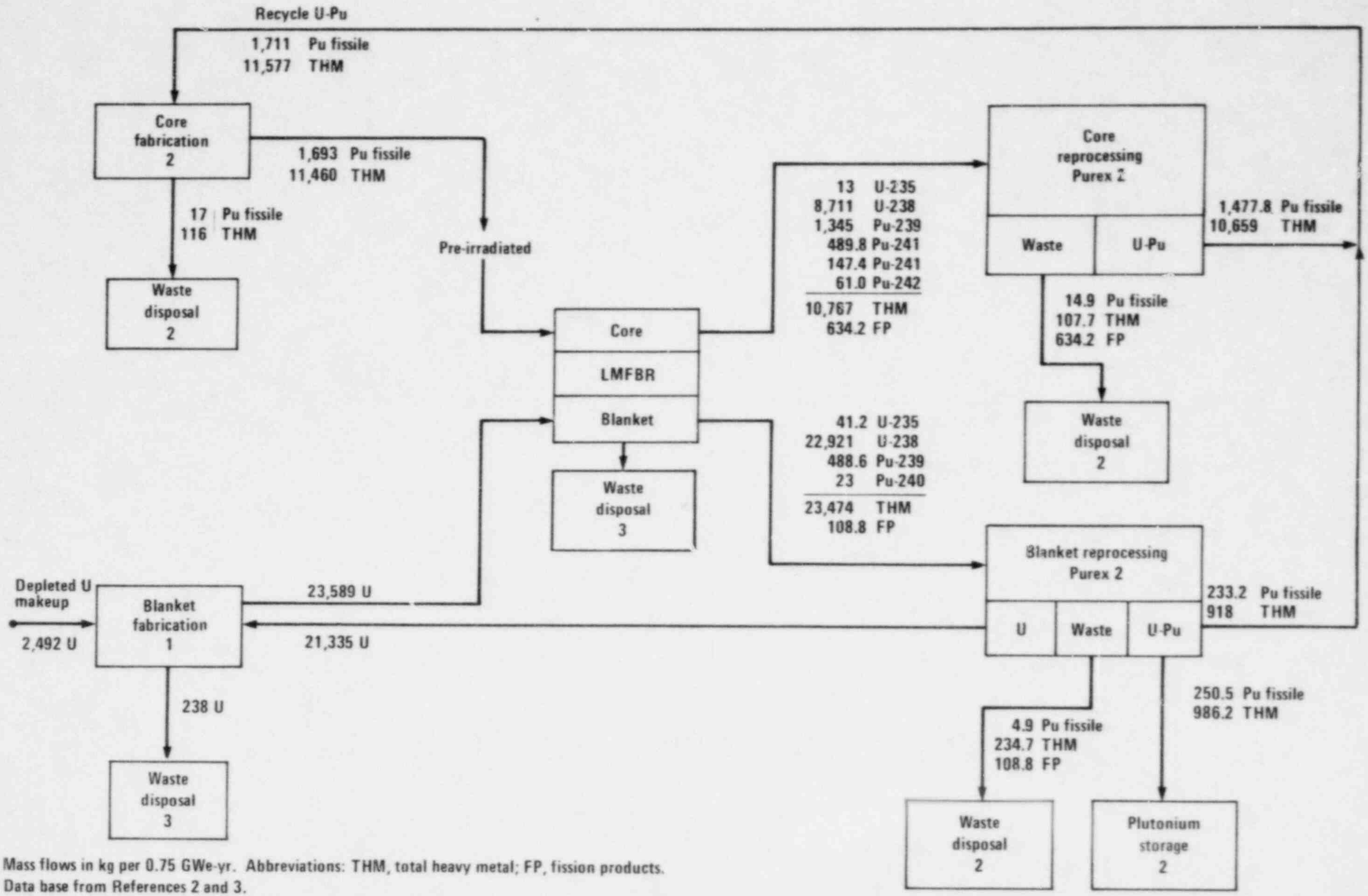
Table 3-2. Equilibrium-cycle reactor charge and discharge data for LMFBR U-Pu/U spiked recycle, heterogeneous core[a]

Isotope	Charge[b]				Discharge[b]			
	Core	AB	IB	RB	Core	AB	IB	RB
Uranium-235	20.1	14.5	24.6	12.2	12.9	13.0	18.6	9.6
Uranium-238	9,249.0	6,653.0	11,284.0	5,600.0	8,711.0	6,575.0	10,911.0	5,435.0
Plutonium-239	1,471.0				1,345.0	68.8	289.1	130.7
Plutonium-240	443.5				489.8	1.4	14.4	7.2
Plutonium-241	222.6				147.4			
Plutonium-242	53.1				61.0			
Total	11,459.3	6,668.5	11,308.6	5,612.2	10,767.0	6,658.0	11,233.0	5,583.0
Fission products[c]					634.2	8.3	68.1	32.4

[a]Westinghouse Reference 1 design, 7.9-mm-O.D. fuel.

[b]Mass flows in kilograms per 0.75 GWe-yr. Abbreviations: AB, axial blanket; IB, inner blanket; RB, radial blanket. Data base from Reference 3 (average for years 22, 23, and 24).

[c]Total = 743.0. From Reference 2.



Mass flows in kg per 0.75 GWe-yr. Abbreviations: THM, total heavy metal; FP, fission products.
Data base from References 2 and 3.

Figure 3-1. Material flow diagram, LMFBR U-Pu/U spiked recycle heterogeneous core (Westinghouse large-pin design).

REFERENCES FOR CHAPTER 3

1. Westinghouse Electric Corporation, Preconceptual Study of Proliferation Resistant Heterogeneous Oxide Fueled LMFBR Core Final Report, Vols. 1 and 2, November 1978.

2. Letter from R. L. Shoup, Oak Ridge National Laboratory, to Project Distribution, November 15, 1978.

3. NASAP/INFCE Reactor Mass Flow Data Base, Oak Ridge National Laboratory, December 1978.

Chapter 4

URANIUM-PLUTONIUM/URANIUM SPIKED RECYCLE: HOMOGENEOUS LMFBR CORE (LMFBR U-Pu/U RECYCLE)

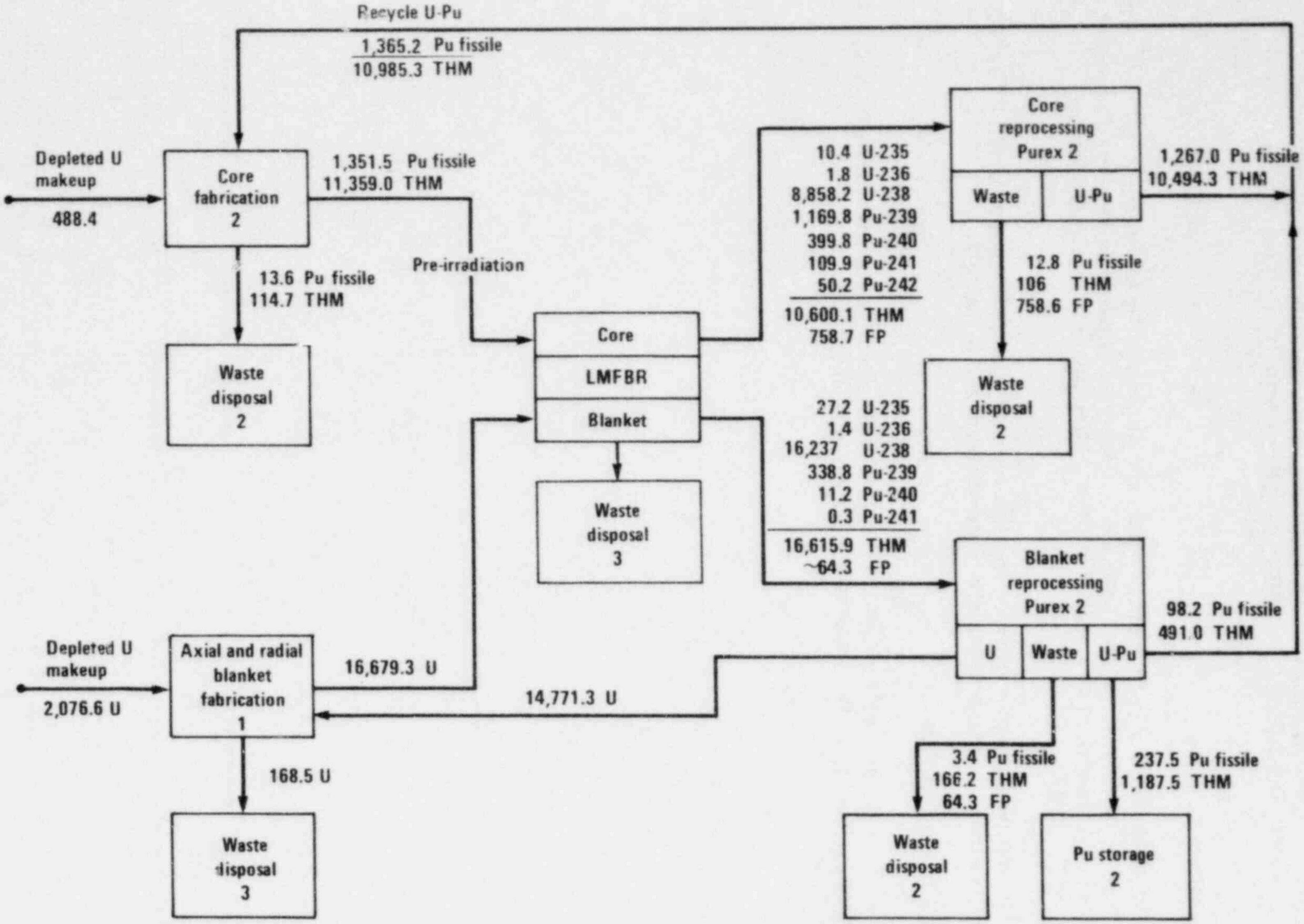
4.1 DESCRIPTION

This reactor/fuel-cycle combination and its mass-flows are identical with those discussed in Chapter 2, except that the excess mixed oxide is pre-irradiated before it is sent to storage and the core fuel assemblies are pre-irradiated before shipment to the power reactor.

The key performance results for the reactor core are summarized in Table 2-1; the significant core-design parameters are summarized in Table 2-2.

4.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 2-3 of Chapter 2. The mass-flow diagram is shown in Figure 4-1.



Mass flows in kg per 0.75 GWe-yr. Abbreviations: THM, total heavy metal; FP, fission products.
Data base from Reference 2 of Chapter 2.

Figure 4–1. Material flow diagram, LMFBR U-Pu/U/U spiked recycle homogeneous core (General Electric design).

Chapter 5

URANIUM-PLUTONIUM/THORIUM SPIKED RECYCLE: HETEROGENEOUS LMFBR CORE (LMFBR U-Pu/U/Th/Th RECYCLE)

5.1 DESCRIPTION

This reactor/fuel-cycle combination is a liquid-metal fast-breeder reactor (LMFBR) using a 16% fissile uranium/ plutonium core and an axial blanket of depleted uranium. In addition this reactor has internal blanket and radial blanket assemblies of thorium oxide. Core fuel assemblies are pre-irradiated before shipment. Core and axial-blanket assemblies are coprocessed, and the recovered uranium/plutonium is mixed with makeup uranium/plutonium as feed to core fabrication. Excess depleted uranium from core reprocessing is used as diluent to the uranium-233 recovered from the internal- and radial-blanket reprocessing and as feed to axial-blanket fabrication. Makeup depleted uranium is required to complete axial-blanket fabrication feed material requirements. The denatured (in process) uranium-233 is stored in an interim storage facility. New thorium is used for internal- and radial-blanket fabrication. Wastes from blanket fabrication are sent to a low-level shallow land disposal site. Reprocessing and core fabrication wastes are sent to a geologic waste repository.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram (Figure 5-1) and are discussed in the following sections of Volume VII:

Axial blanket fabrication 1	Chapter 4
Radial blanket fabrication 1	Chapter 4
Core fabrication 2	Chapter 4
Core and axial-blanket reprocessing (Purex 2)	Section 5.2
Internal- and radial-blanket reprocessing (Thorex 1)	Section 5.4
Thorium storage	Section 6.1
Plutonium storage	Section 6.2
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The significant core-design parameters, including the fissile mass gains per year, are summarized in Table 5-1. Details of design data specifications, including performance characteristics, are given in Table J-0 (Transmuter 1 design), the Westinghouse Electric Corporation preconceptual design study (Ref. 1). This design is identified in Reference 1 as (Pu, U)O_2 fuel, ThO_2 blanket, Transmuter 1, 7.9-mm-O.D. fuel.

5.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 5-2.

The mass flow diagram for the fuel cycles is shown in Figure 5-1.

Table 5-1. Summary of main parameters for the $(PU,U)O_2/ThO_2$ heterogeneous LMFBR core[a]

General reactor data	
Reactor power, MWt	[b]2,740
Net electric power, MWe	[b]1,000
Reactor vessel temperature difference, ^{o}C	[b]156
Reactor vessel outlet temperature, ^{o}C	[b]499
Core enrichment, Pu/heavy metal, at beginning of first core, wt%	
Zone 1	20.5
Zone 2	19.4
Total fissile inventory at beginning of equilibrium cycle (fuel/blanket), kg	4,853/656
Total heavy metal at beginning of equilibrium cycle, kg	109,112
Number of subassemblies	
Drivers--zone 1	222
Drivers--zone 2	48
Inner blanket	121
Control	30
Radial blanket	138
Removable shield	186
Volume fractions in fuel	
Fuel	0.46
Sodium	0.35
Steel	0.19
Volume fractions in blanket	
Oxide	0.58
Sodium	0.26
Steel	0.16
Number of core orifice zones	11
Driver residence time, years	3
Radial blanket residence time, years	6
Peak discharge burnup, MWd/kg	93.6
Average discharge burnup, MWd/kg	56.8
Peak neutron flux (E >0.1 MeV), n/cm^2-sec	[c]2.6×10^{15}
Peak neutron fluence (E >0.1 MeV), n/cm^2	[d]2.1×10^{23}
Peak cladding temperature for lifetime-limiting rod, end of life, midwall, $^{o}C(^{o}F)$	
Nominal	545/(1013)
2σ	617/(1142)
Peak linear power, W/cm (kW/ft)	
Nominal	486 (14.8)
3σ + overpower	601 (18.3)
Sodium void worth at end of equilibrium cycle, $	
Fuel and axial blankets	+2.5
Inner blankets and extensions	+0.6
Doppler coefficient ($-T(dk/dT) \times 10^4$)	
Fuel	57
Isothermal	119
Breeding ratio	1.40

Table 5-1. Summary of main parameters for the $(PU,U)O_2/ThO_2$ heterogeneous LMFBR core[a] (continued)

General reactor data (continued)	
Fissile gain, kg/cycle	236
Fuel cycle cost, mills/kW-hr	11.2
Maximum cumulative damage factor	0.37
Fuel-assembly parameters	
Pins per assembly	271
Duct wall thickness, mm	3.68
Duct outside flat to flat, cm	15.92
Fuel-pin pitch-to-diameter ratio, compressed	1.15
Wire diameter, mm	1.18
Assembly pitch, cm	16.29
Rod-bundle pressure difference, kPa(psi)	538(78)
Maximum mixed mean outlet temperature, nominal, $^{o}C(^{o}F)$	571(1,059)
Driver-pin parameters	
Pin outside diameter, mm	7.874
Cladding thickness, mm	0.3302
Fuel height, cm	122
Axial blanket height, cm	35.56
Plenum volume, cm^3	55.8
Smear density, % of theoretical	91.0
Blanket-assembly parameters	
Number of pins per assembly	127
Duct wall thickness, mm	3.56
Duct outside flat to flat, cm	15.97
Pin outside diameter, mm	12.42
Pin pitch-to-diameter ratio, compressed	1.07
Assembly pitch, cm	16.29
Assembly fueled height, cm	193.04
Plenum volume, cm^3	143.7
Peak linear pin power, W/cm (kW/ft)	
Inner blanket, nominal	400 (12.2)
Radial blanket, nominal	541 (16.5)
Plutonium loss, kg/yr	171
Uranium gain, kg/yr	407

[a]Transmuter 1 design of Ref. 1, 7.9-mm-O.D. fuel. Core fuel is $(Pu,U)O_2$; blanket fuel is ThO_2.

[b]Ground rule.

[c]Fuel.

[d]Blanket.

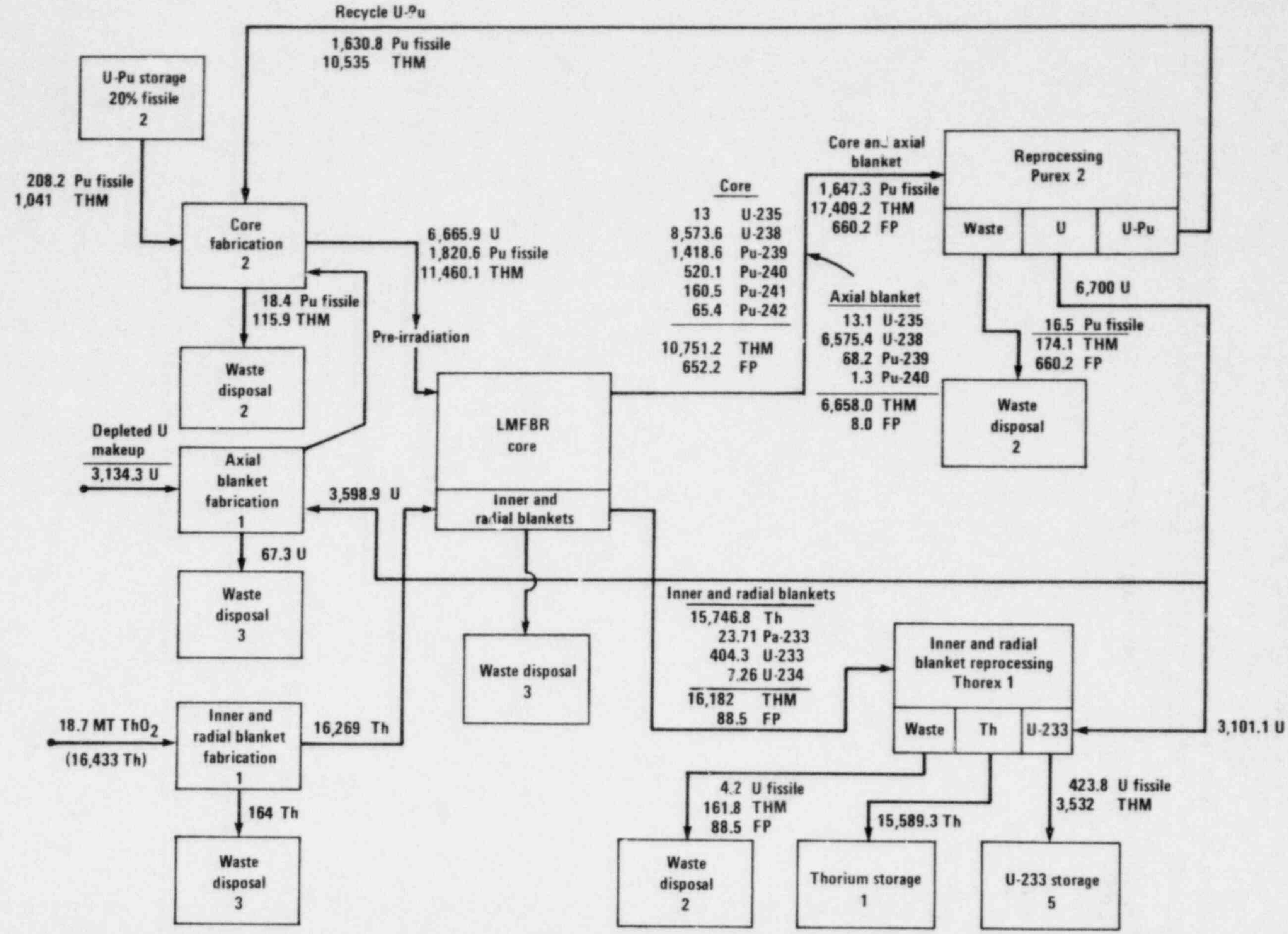
Table 5-2. Equilibrium-cycle reactor charge and discharge data for the LMFBR U-Pu/Th spiked recycle, heterogeneous core[a]

Isotope	Charge[b]				Discharge[b]			
	Core	AB	IB	RB	Core	AB	IB	RB
Thorium-232			10,712	5,557			10,360.7	5,386.1
Protactinium-233							19.1	4.61
Uranium-232								
Uranium-233							266.0	138.3
Uranium-234							4.53	2.73
Uranium-235	19.8	14.5			13.0	13.1		
Uranium-236								
Uranium-238	9,086.8	6,651.4			8,573.6	6,575.4		
Plutonium-238								
Plutonium-239	1,581.4				1,418.6	68.2		
Plutonium-240	476.8				520.1	1.3		
Plutonium-241	239.2				160.5			
Plutonium-242	57.1				65.4			
Total	11,460.1	6,665.9	10,712	5,557	10,751.2	6,658.0	10,650.3	5,531.7
Fission products[c]					652.2	8.0	54.0	34.5

[a]Westinghouse Electric Corporation Transmuter 1 design, 7.9-mm-O.D. fuel.

[b]Mass flows in kilograms per 0.75 GWe-yr. Abbreviations: AB, axial blanket; IB, inner blanket; RB, radial blanket. Data base from Reference 2.

[c]Total = 748.7.



Mass flows in kg per 0.75 GWe-yr. Abbreviations: THM, total heavy metal; FP, fission products.
Data base from Reference 2.

Figure 5-1. Material flow diagram, LMFBR U-Pu/U/Th/Th spiked recycle heterogeneous core (Westinghouse large-pin transmuter).

REFERENCES FOR CHAPTER 5

1. Westinghouse Electric Corporation, Preconceptual Study of Proliferation Resistant Heterogeneous Oxide Fueled LMFBR Core, Final Report, November 1978.

2. Letter from R. L. Shoup, Oak Ridge National Laboratory, to Project Distribution, November 15, 1978.

Chapter 6

URANIUM-PLUTONIUM/THORIUM SPIKED RECYCLE: HOMOGENEOUS LMFBR CORE (LMFBR U-Pu/Th RECYCLE)

6.1 DESCRIPTION

This reactor/fuel-cycle combination is a liquid-metal fast-breeder reactor (LMFBR) using a homogeneous core of 12% fissile uranium-plutonium and axial and radial blankets of thorium oxide. Core assemblies are pre-irradiated before shipment. Core and blanket assemblies are processed separately. Core assemblies are sheared to separate the axial blanket which is processed with the radial blanket. The remainder of the core assemblies are coprocessed, and all of the recovered uranium and plutonium is recycled to fabrication. Makeup plutonium from secure storage and depleted uranium are mixed with the recycled uranium-plutonium as feed to core fabrication. Blanket assemblies are fabricated from new thorium. The uranium-233 recovered during blanket reprocessing is denatured with the addition of depleted uranium in process. The denatured uranium-233 is sent to safe storage. The recovered thorium is stored for 10 years. Wastes from reprocessing and core fabrication are sent to a geologic waste repository. Wastes from blanket fabrication are sent to a low-level shallow land disposal site.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram (Figure 6-1) and are discussed in the following sections of Volume VII:

Blanket fabrication 1	Chapter 4
Core fabrication 2	Chapter 4
Core reprocessing (Purex 2)	Section 5.2
Blanket reprocessing (Thorex 1)	Section 5.4
Thorium storage	Section 6.1
Plutonium storage	Section 6.2
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The key performance results for the reactor core are summarized in Table 6-1. The $(Pu,U)O_2/ThO_2$ transmuter design is identical with the $(Pu,U)O_2/UO_2$ reference design (Ref. 1) since the use of thorium dioxide rather than uranium dioxide in the blankets does not significantly alter the optimum design parameters. Thus, the significant core-design parameters are identical with those summarized in Table 2-2 of Section 2.1. This design is identified in Reference 1 as the $(Pu,U)O_2/ThO_2$ transmuter.

6.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 6-2.

The mass-flow diagram for the fuel cycle is shown in Figure 6-1.

Table 6-1. Performance summary for the $(Pu,U)O_2/ThO_2$ homogeneous LMFBR core[a]

Breeding ratio	1.31
Doubling time, years	15.8
Fuel-cycle cost, mills/kW-hr[b]	7.5
Total fissile mass at beginning of equilibrium cycle, kg	3,526
Net fissile gain, kg/yr	
Plutonium	-79
Uranium-233	298
Total	219
Average core discharge burnup, MWd/kg	62
Peak discharge burnup, MWd/kg	97
Burnup reactivity loss, % $\Delta k/k$	2.0
Core voiding reactivity, % $\Delta k/k$	2.5
Delayed-neutron fraction	0.0036
Core Doppler coefficient, $-T(dk/dT)$	0.0060

[a]Core fuel is $(Pu,U)O_2$; blanket fuel is ThO_2.

[b]The fuel-cycle costs are based on the assumptions specified in the ground rules for the Proliferation-Resistant LMFBR Core Design Study. These results do not necessarily represent the General Electric Company's best estimates of the fuel-cycle costs.

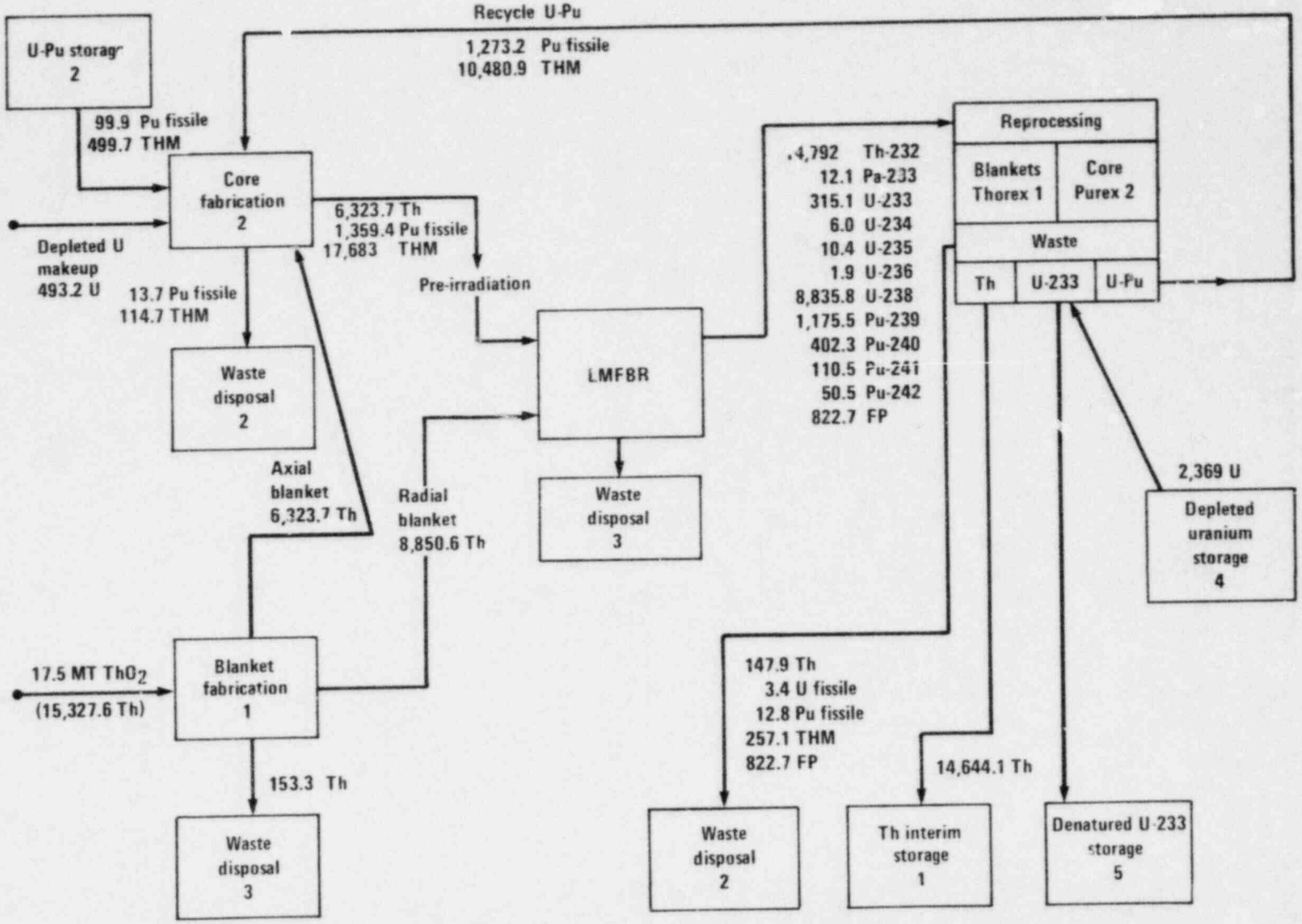
Table 6-2. Equilibrium-cycle reactor charge and discharge data for the LMFBR U-Pu/Th spiked recycle, homogeneous core[a]

Isotope	Charge[b]			Discharge[b]		
	Core	Axial blanket	Radial blanket	Core	Axial blanket	Radial blanket
Thorium-232		6,323.7	8,850.6		6,180	8,612
Protactinium-233					6.17	5.91
Uranium-232						
Uranium-233					119.2	195.9
Uranium-234					2.26	3.74
Uranium-235	19.2			10.2	0.1	0.1
Uranium-236				1.91		
Uranium-238	9,601.1			8,835.8		
Plutonium-238						
Plutonium-239	1,181.5			1,175.5		
Plutonium-240	337.3			402.3		
Plutonium-241	177.9			110.5		
Plutonium-242	42.1			50.5		
Total	11,359.1	6,323.7	8,850.6	10,586.7	6,307.7	8,817.7
Fission products[c]				771.7	16.9	34.1

[a]General Electric's $(Pu,U)O_2/ThO_2$ transmuter design. Data based on Reference 2.

[b]Mass flows in kilograms per 0.75 GWe-yr.

[c]Total 822.7.



Mass flows in kg per 0.75 GWe-yr. Abbreviations: THM, total heavy metal; FP, fission products.

Data base from Reference 2.

Figure 6-1. Material flow diagram, LMFBR U-Pu/Th/Th spiked recycle, homogeneous core (General Electric transmuter design).

REFERENCES FOR CHAPTER 6

1. General Electric Company, Preconceptual Design Study of Proliferation Resistant Homogeneous Oxide LMFBR Cores, GEFR-00392, November 1978.

2. Letter from R. L. Shoup, Oak Ridge National Laboratory, to Project Distribution, November 15, 1978.

Chapter 7

THORIUM-PLUTONIUM/THORIUM SPIKED RECYCLE: HOMOGENEOUS LMFBR CORE

7.1 DESCRIPTION

This reactor/fuel-cycle combination is a liquid-metal fast-breeder reactor (LMFBR) using a 14.2% fissile plutonium-thorium mixed-oxide homogeneous core and thorium oxide blankets. The core and blanket are reprocessed separately. All of the plutonium and part of the thorium recovered during reprocessing are recycled to core fabrication after being mixed with plutonium/thorium make-up material from secure storage. The recycled thorium is highly radioactive and provides the spiking for the plutonium-thorium recycle fuel. The excess thorium recovered during core reprocessing is mixed with the thorium recovered during blanket reprocessing and sent to interim thorium storage for 10 to 20 years' decay. The uranium-233 recovered during core reprocessing is mixed with depleted uranium and the uranium-233 recovered during blanket reprocessing to produce a 12% fissile denatured product that is sent to secure storage. Blanket assemblies are fabricated from new thorium. Wastes from core fabrication and reprocessing are sent to a geologic waste repository. Wastes from blanket fabrication are sent to a low-level shallow land disposal site.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram (Figure 7-1) and are discussed in the following sections of Volume VII:

Blanket fabrication 1	Chapter 4
Core fabrication 3	Chapter 4
Blanket reprocessing (Thorex 1)	Section 5.4
Core reprocessing (Thorex 3)	Section 5.5
Thorium storage	Section 6.1
Plutonium storage	Section 6.2
Depleted uranium storage	Section 6.4
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The key performance results for the reactor core are summarized in Table 7-1, and the significant core-design parameters are summarized in Table 7-2. This design is identified in the General Electric Company preconceptual design study (Ref. 1) as the $(Th,Pu)O_2/ThO_2$ transmuter.

7.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 7-3.

The mass flow diagram for the fuel cycle is shown in Figure 7-1.

The LMFBR fuel is assumed to be spiked by the recycle of processed thorium. The resultant radiation level would be so high as to prohibit handling the fuel without substantial shielding.

Table 7-1. Performance summary for the $(Th,Pu)O_2/ThO_2$ homogeneous LMFBR core[a]

Breeding ratio	1.22
Doubling time, years	29.6
Fuel-cycle cost,[b] mills/kW-hr	11.7
Total fissile mass, at beginning of equilibrium cycle, kg	4,212
Net fissile gain, kg/yr	
Plutonium	-598
Uranium-233	742
Total	144
Average core discharge burnup, MWd/kg	60
Peak discharge burnup, MWd/kg	93
Burnup reactivity loss, % $\Delta k/k$	~0
Core voiding reactivity, % $\Delta k/k$	0.7
Delayed-neutron fraction	0.0030
Core Doppler coefficient (-T(dk/dt))	0.0075

[a]Core fuel is $(Th,Pu)O_2$; blanket fuel is ThO_2.

[b]The fuel-cycle costs are based on the assumptions specified in the ground rules for the Proliferation-Resistant LMFBR Core Design Study. These results do not necessarily represent the General Electric Company's best estimates of the fuel-cycle costs.

Table 7-2. Summary of main parameters for the $(Th,Pu)O_2/ThO_2$ homogeneous LMFBR core

General reactor data	
Reactor power, MWt	2,740
Net electric power, MWe	1,000
Reactor vessel temperature difference, °C	156
Reactor vessel outlet temperature, °C	499
Core fissile enrichment (Pu-239 + Pu-241 + U-233/Pu + U + Th), %	
Inner zone	11.82
Outer zone	17.67
Total fissile inventory at beginning of equilibrium cycle, kg	4,192
Total heavy metal at beginning of equilibrium cycle, kg	87,700
Number of subassemblies	
Drivers--zone 1	150
Drivers--zone 2	102
Internal blanket	0
Control	19
Radial blanket	198
Volume fractions in active core, %	
Fuel	41.45
Sodium	41.58
Steel	16.97
Control	0
Number of core orifice zones	7
Driver residence time, years	2.5
Radial-blanket residence time, row 1, years	3.7
Peak discharge burnup, MWd/kg	91.5
Average discharge burnup, MWd/kg	59.5
Peak neutron flux (E > 0.1 MeV), n/cm^2-sec	3.36
Peak neutron fluence (E > 0.1 MeV), n/cm^2	1.83
Peak cladding temperature, °C	
Nominal	565
2σ	633
Peak linear power at end-of-equilibrium cycle, W/cm	
Nominal	473
3σ + 15%	602
Sodium void worth, $	
Fresh core	
End-of-equilibrium cycle	2.4
Doppler coefficient, $\Delta k/^{o}C$	0.0075
Breeding ratio	1.22
Fissile gain, kg/cycle	203
Compound system doubling time, years	30
Fuel-cycle cost, mills/kW-hr	12.1
Maximum cumulative damage function, steady state	0.30
Fuel-assembly parameters	
Number of pins per assembly	271
Duct wall thickness, mm	3.30
Duct outside flat-to-flat, cm	16.08
Fuel-pin pitch-to-diameter ratio, compressed	1.17

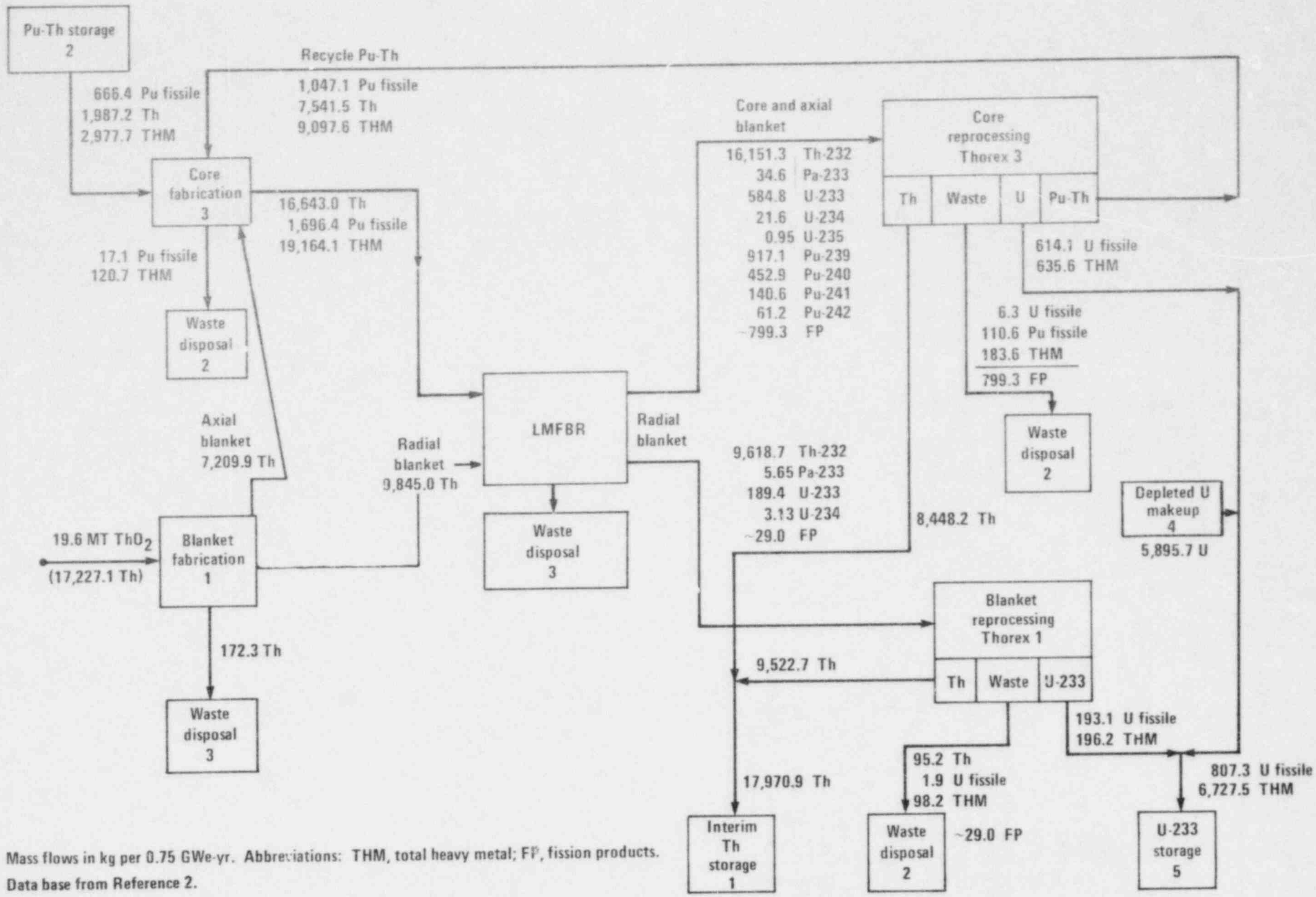
Table 7-2. Summary of main parameters for the $(Th,Pu)O_2/THO_2$ homogeneous LMFBR core (continued)

Fuel-assembly parameters (continued)	
Wire diameter, mm	1.30
Assembly pitch, cm	16.94
Nozzle-to-nozzle pressure difference, kPa	561
Maximum mixed mean outlet temperature, °C	529
Driver-pin parameters	
Pin outside diameter, mm	7.87
Cladding thickness, mm	0.330
Fuel height, cm	121.9
Axial-blanket height, cm	71.1
Plenum volume, cm^3	37.4
Smear density, % of theoretical (fuel/blanket)	90/95
Radial-blanket assembly parameters	
Number of pins per assembly	127
Duct wall thickness, mm	3.30
Duct outside flat to flat, cm	16.08
Pin outside diameter, mm	12.52
Pin pitch-to-diameter ratio, compressed	1.075
Assembly pitch, cm	16.94
Assembly fueled height, cm	193.0
Smear density, % of theoretical	95

Table 7-3. Equilibrium-cycle reactor charge and discharge data for the Th-Pu/Th spiked recycle LMFBR homogeneous core[a]

Isotope	Charge[b] Core	Charge[b] Axial blanket	Charge[b] Radial blanket	Discharge[b] Core	Discharge[b] Axial blanket	Discharge[b] Radial blanket
Thorium-232	9,784.4	7,209.9	9,845.0	9,080.9	7,070.4	9,618.7
Protactinium-233				28.6	6.0	5.65
Uranium-233				466.9	117.9	189.4
Uranium-234				19.8	1.82	3.13
Uranium-235				0.95		
Plutonium-239	1,474.4			917.1		
Plutonium-240	420.9			452.9		
Plutonium-241	221.9			140.6		
Plutonium-242	52.5			61.2		
Total	11,954.1	7,209.9	9,845.0	11,169.0	7,196.1	9,816.9
Fission products[c]				785.6	14.3	29.0

[a]General Electric transmuter design. Data base from Reference 2.
[b]Mass flows in kilograms per 0.75 GWe-yr.
[c]Total = 829.9.



Mass flows in kg per 0.75 GWe-yr. Abbreviations: THM, total heavy metal; FP, fission products.
Data base from Reference 2.

Figure 7-1. Material flow diagram, LMFBR Th-Pu/Th spiked recycle, homogeneous core (General Electric design).

REFERENCES FOR CHAPTER 7

1. General Electric Company, Preconceptual Design Study of Proliferation Resistant Homogeneous Oxide LMFBR Cores, GEFR-00392, November 1978.

2. Letter from R. L. Shoup, Oak Ridge National Laboratory, to Project Distribution, November 15, 1978.

Chapter 8

DENATURED URANIUM-233/THORIUM CYCLE: HOMOGENEOUS LMFBR CORE

8.1 DESCRIPTION

This reactor/fuel-cycle combination is a liquid-metal fast-breeder reactor (LMFBR) using a 10.1% fissile recycle uranium-233 oxide homogeneous core and a thorium oxide blanket. The core and blanket are reprocessed separately. The denatured uranium-233 recovered during core reprocessing is mixed with the highly enriched uranium-233 from blanket reprocessing and make-up denatured uranium-233 that is about 24.7% fissile to provide the feed for fabrication. The plutonium recovered during core reprocessing is diluted with depleted uranium to 20% fissile content and placed in secure storage. Thorium recovered during blanket reprocessing is placed in interim storage for 10 years. Blanket assemblies are fabricated from new or decayed thorium. Wastes from core fabrication and reprocessing are sent to a geologic waste repository. Wastes from blanket fabrication are sent to a low-level shallow land disposal site.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in mass-flow diagram (Figure 8-1) and are discussed in the following sections of Volume VII:

Blanket fabrication 1	Chapter 4
Core fabrication 3	Chapter 4
Core reprocessing (Purex 1)	Section 5.1
Blanket reprocessing (Thorex 1)	Section 5.4
Thorium storage	Section 6.1
Plutonium storage	Section 6.2
Depleted uranium storage	Section 6.4
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The key performance results for the reactor core are summarized in Table 8-1, and the significant core-design parameters are presented in Table 8-2. This design is identified in the General Electric Company preconceptual design study (Ref. 1) as $(U3U8)O_2/ThO_2$ denatured.

8.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 8-3.

The mass flow diagram for the fuel cycle is shown in Figure 8-1.

Table 8-1. Performance summary for the LMFBR denatured U-233/Th homogeneous core[a]

Breeding ratio	1.23
Doubling time, years	21.2
Fuel-cycle cost, mills/kW-hr[b]	7.8
Total fissile mass at beginning of equilibrium cycle, kg	3,893
Net fissile gain, kg/yr:	
Plutonium	462
Uranium-233	-288
Total	174
Average core discharge burnup, MWd/kg	58
Peak discharge burnup, MWd/kg	90
Burnup reactivity loss, % $\Delta k/k$	4.5
Core voiding reactivity, % $\Delta k/k$	0.6
Delayed-neutron fraction	0.0041
Core Doppler coefficient, (-T(dk/dT))	0.0055

[a]General Electric Company's $(U3U8)O_2/ThO_2$ homogeneous denatured core (Ref. 1). The core fuel is uranium-233/uranium-238 dioxide and the blanket is thorium dioxide.

[b]The fuel-cycle costs are based on the assumptions specified in the ground rules for the Proliferation-Resistant LMFBR Core Design Study. These results do not necessarily represent the General Electric Company's best estimates of the fuel-cycle costs.

Table 8-2. Summary of main parameters for the LMFBR denatured U-233/Th homogeneous core[a]

General reactor data	
Reactor power, MWt	2,740
Net electric power, MWe	1,000
Reactor vessel temperature difference, °C	156
Reactor vessel outlet temperature, °C	499
Core fissile enrichment (Pu-239 + Pu-241 + U-233/Pu + U + Th), %	
Inner zone	8.11
Outer zone	11.77
Total fissile inventory at beginning of equilibrium cycle, kg	3,675
Total heavy metal at beginning of equilibrium cycle, kg	106,500
Number of subassemblies	
Drivers--zone 1	150
Drivers--zone 2	102
Internal blanket	0
Control	25
Radial blanket	198
Volume fractions in active core, %	
Fuel	42.31
Sodium	40.47
Steel	17.22
Control	0
Number of core orifice zones	7
Driver residence time, years	3.0
Radial-blanket residence time, row 1, years	4.5
Peak discharge burnup, MWd/kg	88.7
Average discharge burnup, MWd/kg	57.5
Peak neutron flux (E > 0.1 MeV), n/cm^2-sec	2.63×10^{15}
Peak neutron fluence (E > 0.1 MeV), n/cm^2	1.75×10^{23}
Peak cladding temperature, °C	
Nominal	569
2σ	639
Peak linear power at end of equilibrium cycle, W/cm	
Nominal	476
3σ + 15%	606
Sodium void worth, $	
Fresh core	
End of equilibrium cycle	1.4
Doppler coefficient, Δk/°C	0.0055
Breeding ratio	1.23
Fissile gain, kg/cycle	281
Compound system doubling time, years	21
Fuel-cycle cost, mills/kW-hr	8.6
Maximum cumulative damage function, steady state	0.06

Table 8-2. Summary of main parameters for the LMFBR denatured U-233/Th homogeneous core[a] (continued)

Fuel-assembly parameters	
Number of pins per assembly	271
Duct wall thickness, mm	3.43
Duct outside flat to flat, cm	17.29
Fuel-pin pitch-to-diameter ratio, compressed	1.15
Wire diameter, mm	1.26
Assembly pitch, cm	18.10
Nozzle-to-nozzle pressure difference, kPa	454
Maximum mixed mean outlet temperature, ^{o}C	529
Driver-pin parameters	
Pin outside diameter, mm	8.64
Cladding thickness, mm	0.381
Fuel height, cm	121.9
Axial-blanket height, cm	71.1
Plenum volume, cm^3	44.6
Smear density, % of theoretical (fuel/blanket)	90/95
Radial-blanket-assembly parameters	
Number of pins per assembly	127
Duct wall thickness, mm	3.43
Duct outside flat to flat, cm	17.29
Pin outside diameter, mm	13.59
Pin pitch-to-diameter ratio, compressed	1.070
Assembly pitch, cm	18.10
Assembly fueled height, cm	193.0
Smear density, % of theoretical	95

[a]General Electric Company's (U3U8)O_2/ThO_2 homogeneous denatured core (Ref. 1).

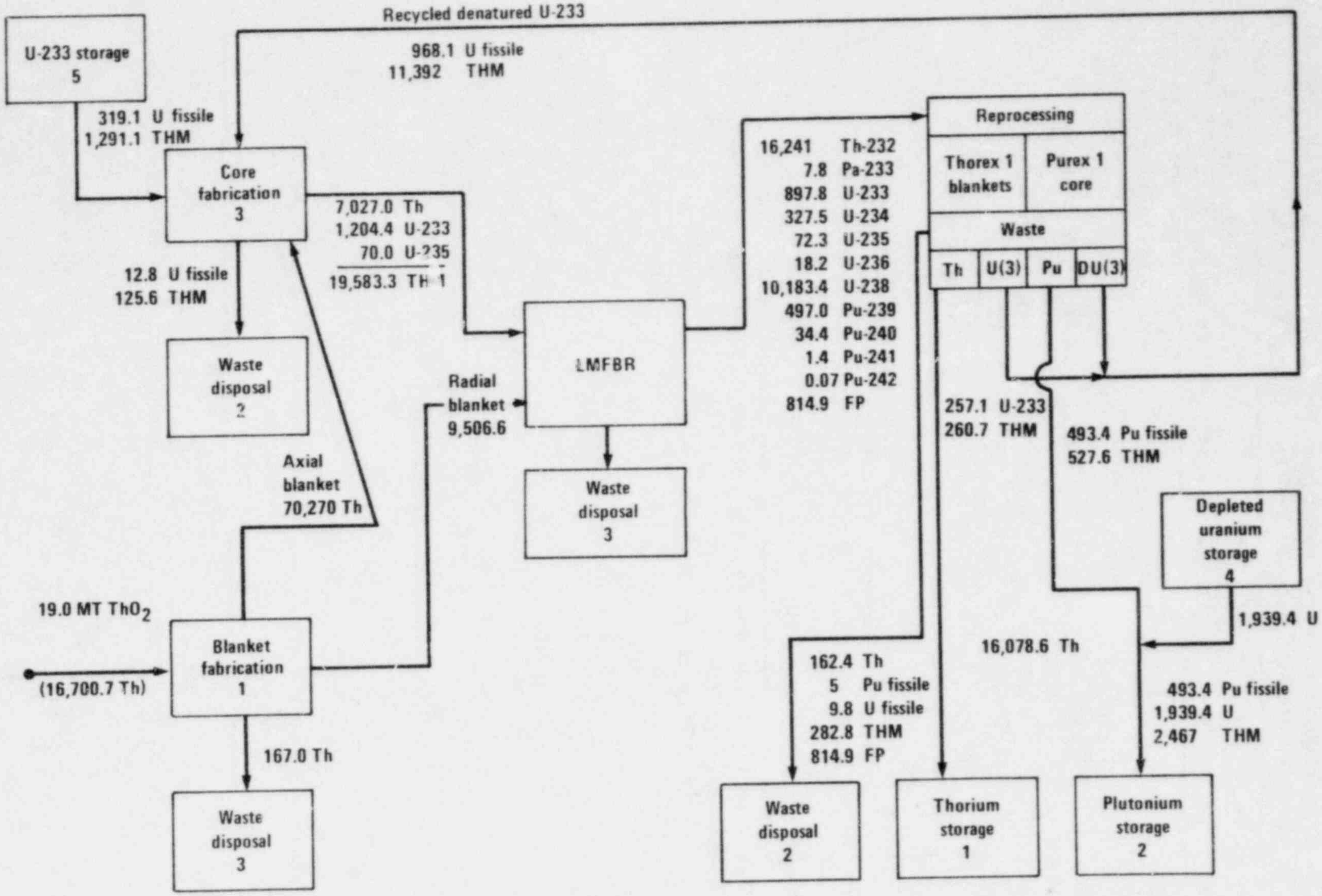
Table 8-3. Equilibrium-cycle reactor charge and discharge data for the LMFBR denatured U-233/Th homogeneous core[a]

Isotope	Charge[b] Core	Axial	Radial	Discharge[b] Core	Axial	Radial
Thorium-232		7,027.0	9,506.6		6,903.5	9,337.5
Protactinium-233					4.3	3.5
Uranium-233	1,204.4			645.9	106.2	145.7
Uranium-234	337.9			324.0	1.5	2.0
Uranium-235	70.0			72.2		0.07
Uranium-236	11.2			18.2		
Uranium-238	10,932.7			10,183.4		
Plutonium-239				497.0		
Plutonium-240				34.4		
Plutonium-241				1.4		
Plutonium-242				0.07		
Total	12,556.2	7,027.0	9,506.6	11,776.6	7,015.5	9,488.8
Fission products[c]				784.3	12.0	18.6

[a]General Electric Company's $(U3U8)O_2/ThO_2$ denatured homogeneous core Data base from Reference 2.

[b]Mass flows in kilograms per 0.75 GWe-yr.

[c]Total = 814.9.



Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Abbreviations: THM, total heavy metal; FP, fission products; DU(3), denatured U-233.
3. Data base from Reference 2.

Figure 8–1. Material flow diagram, LMFBR denatured U-233/Th cycle homogeneous core (General Electric design).

REFERENCES FOR CHAPTER 8

1. General Electric Company, Preconceptual Design Study of Proliferation Resistant Homogeneous Oxide LMFBR Cores, GEFR-00392, November 1978.

2. Letter from R. L. Shoup, Oak Ridge National Laboratory, to Project Distribution, November 15, 1978.

Chapter 9

SAFETY CONSIDERATIONS FOR THE LMFBR

9.1 INTRODUCTION

The most recent detailed safety assessment of a proposed liquid-metal fast-breeder reactor (LMFBR) was the one conducted by the U.S. Nuclear Regulatory Commission (NRC) staff for the Clinch River Breeder Reactor Plant (CRBRP). Before licensing activities associated with the CRBRP were suspended at the request of the U.S. Environmental Research and Development Administration (ERDA) in April 1977, this assessment had progressed through a relatively detailed (though not completed) review of the CRBRP Preliminary Safety Analysis Report (PSAR) and the issuance of the Site Suitability Report (Ref. 1).

As a basis for carrying out this assessment, the NRC staff first developed and issued a set of design criteria, included as Appendix A of Reference 1. These criteria represented the minimum requirements acceptable to the staff for the principal design criteria of the CRBRP. The basic safety approach used by the staff in formulating these criteria was that the CRBRP should achieve a level of safety comparable to that of present-generation light-water reactor (LWR) plants, according to all current criteria for evaluation, and that the design approaches for attaining the required level of safety be similar or analogous to current practice.

In formulating the CRBRP design criteria and applying them to the CRBRP, the NRC staff has identified a number of safety-related issues believed to require special attention during the course of designing and licensing the CRBRP. The common thread running through these issues is a determination that major emphasis be placed on the prevention of accidents that could lead to core melting and disruption and the subsequent loss of containment integrity. At the time the Site Suitability Report was issued, the NRC staff concluded that, though the staff had not reviewed the (as yet incomplete) design sufficiently to determine that the design criteria were satisfied by the CRBRP design, it appeared that no problems existed that would preclude proper satisfaction of the criteria.

Section 9.2 lists and briefly discusses the safety-related issues that the NRC staff identified as requiring special attention. Section 9.3 describes the status of these issues. Section 9.4 addresses the impact that proposed core-design variations and alternative fuels would have on the successful resolution of these issues.

9.2 KEY LMFBR SAFETY ISSUES

Two safety aspects of the LMFBR have historically drawn substantial attention: the potential for the core to be driven into a more critical geometrical arrangement and the presence of large quantities of sodium. This has led to considerable emphasis being placed on accidents that could lead to melting of the core, the so-called core-disruptive accidents. Over the past 10 or so years, much progress has been made in LMFBR system design so as to reduce the probability of initiating events that could lead to core melting. At the same time, much progress has also been made in developing an understanding of the range of possible consequences that could result from core-disruptive accidents. All of this led the NRC staff to conclude that, for the CRBRP, the probability of core-melt and core-disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design-basis accident spectrum.

To provide for this low probability, the NRC staff identified four design-related issues that would have to be resolved favorably for the CRBRP design:

1. The scram systems must be shown to have sufficient redundancy and diversity to make the probability of their failure very small.
2. Sufficient redundancy and diversity must be provided in the heat-transport system design to make the probability of its not being able to remove heat under shutdown conditions very small.
3. Reliable means to detect and cope with fuel-rod failure and subassembly faults must be provided.
4. The continuing high integrity of the heat-transport system must be ensured.

In addition to requiring that the four above issues associated with minimizing the probability of core-melt and core-disruptive accidents be resolved favorably, the NRC staff identified three other issues associated with minimizing the probability of containment failure in the event that an accident did occur that would also have to be resolved favorably. These are as follows:

5. The containment must be able to accommodate the consequences of spillage of large quantities of sodium from the primary or intermediate coolant system.
6. The containment/confinement system must be capable of adequate mitigation of the radioactivity releases that could result from all events within the containment design basis.
7. The containment system should also be so designed that it could maintain its integrity for at least 24 hours in the unlikely event of the occurrence of a broad range of conditions involving the energetic disassembly of the core and production of vaporized fuel and other possible consequences resulting from core-melt accidents.

9.3 STATUS OF KEY ISSUES

9.3.1 SCRAM-SYSTEM RELIABILITY

To provide the necessary scram-system reliability, two redundant and diverse scram systems are required. Each system has to be capable of shutting the reactor down under extreme conditions, and no electrical or other external power can be required for the scram of any control rod. The NRC staff has concluded (Ref. 1) that it is feasible to design such a system and that the CRBRP design has the potential for satisfying the scram-system reliability criterion.

9.3.2 RELIABILITY OF THE RESIDUAL-HEAT-REMOVAL SYSTEM

Because of the importance of being able to remove the decay heat under a wide variety of shutdown conditions, a redundant and diverse shutdown heat-removal capability is required. Although the CRBRP residual-heat-removal design, as submitted for review, had not been shown to provide the necessary redundancy and diversity, the NRC staff has concluded (Ref. 1) that it is technically feasible to provide an adequate residual-heat-removal system for LMFBRs.

9.3.3 ACCOMMODATION OF SUBASSEMBLY FAULTS

The NRC staff has specified that means to detect subassembly faults, to cope with these faults, and to protect against progressive subassembly fault propagation should be provided. These provisions are intended to help insure that the probability of damage to a significant portion of the core due to subassembly-scale initiating events is very remote. Events that could lead to significant subassembly damage include subassembly coolant-inlet blockage, flow obstructions within the subassembly pin array, and random failure of an individual rod or of a few rods.

The NRC staff has concluded (Ref. 1) that there is a substantial basis of analytical and experimental evidence for anticipating that local faults affecting a single or a few rods within a subassembly will not rapidly propagate to adjacent rods. The current LMFBR subassembly inlet designs, having multiple inlet ports at different planes with interposed strainers, should prevent the occurrence of blockages that could significantly reduce flow to a subassembly. As operating experience with failed fuel rods is gained, it is anticipated that it will be possible to set reasonable limits on how much failed fuel can be tolerated while keeping an acceptable limit on failure propagation potential. All of these considerations led the NRC staff to conclude that it is possible to limit the potential for fuel failure propagation beyond a single subassembly to such a level that it need not be considered as an initiator of a whole-core accident.

9.3.4 INTEGRITY OF THE HEAT-TRANSPORT SYSTEM

The sodium in the primary system of an LMFBR is at a low operating pressure since the operating temperature is well below the temperature at which sodium will boil at near atmospheric pressure. Typical peak pressures do not exceed 1.3 MPa. Thus there is no stored energy for flashing to vapor in the event of a pipe break. By proper layout of the piping, including guard pipes around coolant pipes in some areas and the inclusion of check valves in the cold legs, it is possible to prevent the core from being uncovered in the unlikely event that a leak in the system does occur.

Provided that proper leak-detection equipment is installed and that an in-service inspection program is carried out in addition to starting with a proper design, the

NRC staff has concluded (Ref. 1) that the heat-transport system can be designed for a high level of integrity and for continued assurance of this integrity throughout the operating history of the plant. To this end, the staff concluded that for the CRBRP a double-ended rupture of the primary-system cold-leg piping need not be considered as a design-basis event. Because of the higher operating temperatures, the staff determined that hot-leg ruptures should be considered as design-basis events for the CRBRP but that containment design features could be included to cope with the consequence of the resulting large sodium releases.

9.3.5 CONTAINMENT DESIGN TO COPE WITH SODIUM HAZARDS

At the operating temperatures of LMFBRs, sodium will ignite and burn readily if sprayed into the air, even in reduced-oxygen atmospheres. It will also burn as a pool. The heat released from a sodium fire can damage concrete, and water released from the heated concrete reacts exothermally with sodium. The net effect of these reactions is to increase containment-cell temperatures and pressures, with structural degradation of the concrete and the production of potentially explosive hydrogen. However, because of the substantial experience that has been gained in handling sodium, the NRC staff has stated (Ref. 1) that it is possible to design features in the containment system to alleviate the sodium hazards.

9.3.6 CONTAINMENT DESIGN TO MITIGATE RADIOACTIVITY RELEASES FROM EVENTS WITHIN THE DESIGN BASIS

The general safety design criteria for the CRBRP containment issued by the NRC staff state that the reactor containment structure, including access openings and penetrations--if necessary, in conjunction with additional post-accident heat-removal systems--shall be so designed that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences, and any of the postulated accidents. In an LMFBR, the accidents that represent the principal challenges to containment are sodium fires coupled with potential sodium-concrete reactions resulting from failure and subsequent release of sodium from the primary heat-transport-system equipment. The general containment-system design concept that appears to meet these needs is a containment-confinement system in which the steel containment building is surrounded by a thick concrete confinement shell, with the annulus between the two maintained at a reduced pressure. An annulus filter system could be added to reduce radioactivity release from the annulus to the environment while maintaining a reduced pressure in the annulus. Examination of the CRBRP containment/confinement system design and the range of conditions to which it might be subjected led the NRC staff to conclude that it is technically feasible to implement design features to meet their stated criteria.

9.3.7 ACCOMMODATION OF CORE-MELT AND CORE-DISRUPTIVE ACCIDENTS

The NRC staff concluded that the CRBRP design should contain provisions making it extremely unlikely that potential core-melt and core-disruptive accidents could result in early containment-system failure. This requirement arises from their basic position that the CRBRP should achieve a level of safety comparable to that of present-generation LWR plants. Studies of a spectrum of events beyond the design basis reveal that some such accidents, such as the loss-of-flow accident with scram-system failure, have a high probability of leading to large-scale fuel melting in the core and possibly to the generation of significant quantities of fuel vapor. The staff concluded that the

CRBRP containment system should be well enough protected from a broad range of such conditions to maintain its integrity for 24 hours for these conditions. The NRC staff has agreed to reconsider the 24-hour criterion when licensing review is reinitiated.

In considering these events beyond the design basis in the challenges to containment integrity that might result from their occurrence, there are really only two generic types of consequences to be considered: (1) excessive fission-energy release during the accident transient (energetics), and (2) failure to cool the core adequately and to accommodate the molten-core debris resulting from the transient. For those events beyond the design basis in which the control system is assumed to operate and the core melts down because of lack of adequate residual-heat-removal capability, the energetics issue is not relevant and post-accident heat removal (PAHR) is the primary concern. For the class of so-called unprotected transients where scram-system failure is assumed to occur, both energetics and PAHR issues must be examined.

Much attention has been given to the energetics issues associated with unprotected accidents in the LMFBR (Refs. 2-6). For convenience, the energetics issues can be broken down into three areas of concern:

1. The positive sodium void worth associated with medium- to large-size LMFBRs
2. Vapor explosions occurring in molten fuel and coolant interactions (FCIs)
3. Recriticality events in the disrupting cores

For smaller LMFBRs, the sodium-voiding-related energetics issue is not relevant; this was certainly the case for the Fast Flux Test Facility (FFTF) reactor (Refs. 7 and 8). For reactors the size of the CRBRP, the positive sodium void worth results in the system being in a supercritical state and at 5 to 20 times nominal power when the fuel pins begin to disrupt in the unprotected loss-of-flow accident. The subsequent loss-of-flow accident scenario is quite sensitive to the initial motion of this disrupting fuel. There appears to be a strong potential for this early motion to be dispersive, thus dispelling the energetics concern. A substantial research and development program is currently aimed at demonstrating the existence of this early fuel-dispersal mechanism. For LMFBRs with total void worths on the order of \$3, it appears to have a high chance of success, but more stringent requirements for rapid early fuel dispersal exist in large LMFBRs where the total void worth is predicted to be in the range of \$5 to \$6. As discussed in Section 9.4.1, this has caused attention to be focused on core-design alternatives in which the void worth would be reduced to the range of \$2.5 to \$3.5.

For the oxide-fueled systems, the earlier concern about vapor explosions from FCIs that might offer an energetics threat appears to be unwarranted. Research carried out over the past 6 to 7 years shows these energetic events to be unlikely, on the basis that energetic FCIs can be ruled out because the interface contact temperature is well below the spontaneous nucleation limit for sodium (Ref. 10). Thus the NRC staff, in its evaluation of CRBRP safety issues, gave them small concern. Again, for the oxide-fuel system, the energetics potential associated with recriticalities after the initial core disruption has been shown to be small. The arguments used to reduce concern about these events are based on the dispersive effect provided by steel vaporization that precludes energetic recriticalities (Ref. 10). Additional confirmatory work is being done on both the FCI and recriticality issues, to provide further support for the arguments that have been advanced to preclude them for the most part.

Whether or not significant energetics result from these severe accidents, large amounts of molten core material will be produced. The PAHR considerations associated with demonstrating that this debris can be contained either within the reactor vessel or outside it without causing early containment failure have received considerable attention. For the smaller FFTF core, it has been shown that there is a high probability that the post-accident debris could be contained within the vessel (Ref. 7). For larger plants, such as the CRBRP, this does not appear to be possible, and efforts have been made to demonstrate that the debris can be accommodated in the reactor cavity below the vessel without threatening containment integrity.

Although the NRC staff was not convinced that the proposed CRBRP design could accommodate the debris in the reactor cavity for the required 24 hours with the containment intact, it did believe that the technology exists to achieve the 24-hour no-failure criterion. As with the sodium-voiding-related energetics issue, a substantial research and development program is in progress to further develop and refine PAHR technology for large LMFBRs.

9.4 SAFETY CONSIDERATIONS ASSOCIATED WITH ALTERNATIVE CORE DESIGNS AND FUEL CYCLES

In recent years, attention has been focused on alternative core designs that would have lower sodium void worths and on fuel cycles other than the reference uranium/plutonium dioxide system, to improve either breeding performance or proliferation resistance. The next two subsections briefly consider the safety considerations associated with these alternative core designs and fuel systems.

9.4.1 LOW-VOID-WORTH CORE DESIGNS

In order to mitigate concern about the energetics potential of sodium voiding in a large oxide-fueled LMFBR experiencing an unprotected loss-of-flow accident, studies have been made of alternative core-loading arrangements that would result in a lowered void worth. It appears to be possible to achieve so-called heterogeneous core designs, in which blanket subassemblies are placed in the core region, that have much lower void worths (on the order of $2.5 to $3.5) than are predicted for homogeneous commercial-size LMFBRs. Preliminary studies indicate that this lowering of the void worth does result in a lessened energetics potential under unprotected loss-of-flow conditions (Ref. 9). For this reason, these low-void-worth core designs are being carefully studied.

9.4.2 ALTERNATIVE FUEL SYSTEMS

As part of the efforts of U.S. participants in Group V of the International Nuclear Fuel Cycle Evaluation (INFCE) effort, a study was made of the safety implications of alternative fuel types. The results of this study are published in Reference 10. The interested reader is directed to this document.

REFERENCES FOR CHAPTER 9

1. Site Suitability Report by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission in the Matter of the Clinch River Breeder Reactor Plant, Energy Research and Development Administration, Tennessee Valley Authority, Project Management Corporation, Docket No. 50-537, March 4, 1977.

2. H. K. Fauske, "The Role of Core Disruptive Accidents in Design and Licensing of LMFBRs," Nuclear Safety, 17(5) (September-October 1976).

3. H. K. Fauske, "Assessment of Accident Energetics in LMFBR Core-Disruptive Accidents," Nuclear Engineering & Design, 42(19) (1977).

4. W. R. Bohl et al., An Analysis of Transient Undercooling and Transient Overpower Accidents without Scram in the Clinch River Breeder Reactor, ANL/RAS 75-29, Argonne National Laboratory, July 1975.

5. W. R. Bohl et al., An Analysis of the Unprotected Loss-of-Flow Accident in the Clinch River Breeder Reactor with an End-of-Equilibrium-Cycle Core, ANL/RAS 77-15, Argonne National Laboratory, May 1977.

6. J. F. Meyer et al., An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics, U.S. Nuclear Regulatory Commission, NUREG-0122, March 1977.

7. U.S. Nuclear Regulatory Commission, Safety Evaluation Report Related to Operation of Fast Flux Test Facility, Project No. 448, NUREG-0358, August 1978.

8. M. G. Stevenson et al., Report on the Analysis of the Initiating Phase of a Loss-of-Flow (Without Scram) Accident in the FTR, ANL/RAS 74-24, Argonne National Laboratory, September 1974.

9. C. P. Tzanos et al., "Design-Related Inherent Safety Characteristics in Large LMFBR Power Plants," CONF-761001, Proceedings of the International Meeting on Fast Reactor Safety and Related Physics, Chicago, Ill., October 5-8, 1976.

10. H. K. Fauske, INFCE/5 Fast Breeder Reactor Studies: Safety Implications of Alternative Fuel Types, ANL-INFCE/5-TM-5, Argonne National Laboratory, July 1978.

Chapter 10

ENVIRONMENTAL CONSIDERATIONS

10.1 SUMMARY ASSESSMENT

The bases for the environmental assessment of the routine operation of the LMFBR plant are derived from the Environmental Statement--Liquid Metal Fast Reactor Program (WASH-1535), dated December 1974 (Ref. 1). The conceptual design in the above document does not provide the detail required for a rigorous treatment of source terms as was performed for the reference light-water reactor (LWR) and other reactor concepts presented in the Preliminary Safety and Environmental Information Document. Using the then "typical" values for radioactive and non-radioactive effluents for the 1,000-MWe liquid-metal fast-breeder reactor (LMFBR) plant, it is concluded that routine operation of the LMFBR would result in significantly smaller environmental impacts, in some areas, than those associated with the reference LWR. The thermal impact would be smaller due to the higher thermal efficiency, the radiological impact would be substantially reduced as evidenced by the lower release rates of radioactive effluents, and the chemical impacts and occupational exposures would be comparable to those of the LWR.

10.2 REACTOR AND STEAM-ELECTRIC SYSTEM (R.G. 4.2/3.2)

The LMFBR design and plant characteristics used as the basis for this environmental assessment are derived from a conceptual design study of a 1,000-MWe plant. The principal characteristics are as follows (Ref. 1):

Net electrical power, MWe	1,000
Reactor thermal power, MWt	2,740
Fuel type	Uranium/plutonium dioxide
Heat rate, Btu/kW-hr	9,352
Heat-dissipation rate, Btu/hr	5.2×10^9

10.3 STATION LAND USE

Approximately 35 to 50 acres of land will be required for facilities associated with an LMFBR power plant: the reactor buildings, turbine building, switchyard, parking lot, access roads, and cooling towers. An exclusion area of at least 400 acres will probably be needed. This is generally comparable with the areas associated with LWR plants. The average area of present-day LWR power plants is about 1,160 acres, with a range of 84 to over 3,000 acres (Ref. 1). In comparison, the site of the Clinch River Breeder Reactor Plant (CRBRP) is 1,364 acres, including approximately 100 acres for plant facilities (Ref. 2).

The size of the individual site will vary with the type of cooling system employed and other plant-specific factors. However, the basic criteria on site-boundary selection are the requirements set forth in 10 CFR 100 for the control of personnel in the exclusion area and ability to take emergency protective measures in the low-population zone.

10.4 STATION WATER USE (R.G. 4.2/3.3)

The LMFBR plant will use, as do other types of present-day power stations, large amounts of water for makeup to the heat-dissipation system. The proposed design is assumed to use a closed-loop cooling-water system with natural-draft cooling towers to reject heat. As shown in Table 10-1 the maximum and average rates of loss from evaporation and drift are 8,900 and 5,300 gpm, respectively, compared to 11,500 and 6,800 gpm for the reference LWR.

10.5 HEAT-DISSIPATION SYSTEM (R.G. 4.2/3.4)

About 1,740 MWt of waste heat will be rejected from a 1,000-MWe plant, mainly to the atmosphere. Any of several types of heat-dissipation systems may be used, depending on site conditions and other factors. One of the more commonly used is a wet natural-draft cooling tower. That type of system with freshwater makeup was assumed for this report.

A typical natural-draft cooling tower for a 1,000-MWe LMFBR unit will have a single shell with a height of about 510 feet and a maximum shell diameter of about 400 feet. Heat is dissipated to the atmosphere by a combination of evaporation and sensible-heat transfer. Although evaporation predominates, the balance between the two modes of heat transfer depends on air temperature and humidity. The average rate of water use, therefore, will vary from month to month. Blowdown is required to limit the concentration of solids in the circulating water. For the reference plant discussed herein, a maximum concentration of 5 is used, though other values are frequently found. Design data for a heat-dissipation system are shown in Table 10-1 for a site in the north-central United States.

Circulating water will be periodically chlorinated to control algae and other slime-forming microorganisms. Typically, chlorine is added as required to achieve a free residual chlorine content of 0.5 to 1.0 ppm for 1 to 2 hours per day. The cooling-tower blowdown may have a small residual chlorine content during periods of chlorination.

10.6 RADIOACTIVE-WASTE SYSTEMS AND EFFLUENT SOURCE TERMS

10.6.1 SOURCE TERMS (R.G. 4.2/3.5.1)

It should be recognized that the only design information available for a 1,000-MWe LMFBR plant is from a conceptual design. Nevertheless, the results of this conceptual design study, together with data from the CRBRP (Ref. 2) and other research and development programs (Ref. 3), form the basis for this assessment of environmental effects of LMFBR deployment.

Figure 10-1 is a block diagram showing the interconnections between the various plant components and systems and the paths for transfer of radioactivity. As the figure shows, no continuous or intermittent releases of radioactive effluents (other than tritium) to the environment will occur during the normal operation of large commercial LMFBR plants, although some small leakages through seals may be expected.

The 1,000-MWe commercial LMFBR will be designed to collect various radioactive materials produced during plant operation and store them in the plant before the processed radioactive wastes are shipped to offsite storage sites for permanent storage or disposal. This radioactive-waste processing and in-plant storage will essentially eliminate any significant radiation exposure of the public from normal plant operation.

The use of sodium as the reactor coolant is one of the major distinguishing features of the LMFBR. The sodium, in addition to being an excellent coolant, has the ability to retain fission products released from the small number of fuel failures or defects that might occur during reactor operation. The gaseous radioisotopes, particularly xenon and krypton, which are not held by the sodium, will escape to the inert cover gas, whence they will be removed by gas-purification and recovery systems. Some of the metals and halogens will normally plate out on metal surfaces or be removed by the sodium cold-trap purification system. The principal fission-product impurities in the sodium then will be the longer lived isotopes of cesium and of other alkali metals. Thus, accidental spilling or leakage of the primary sodium system will not release large quantities of fission products, as most of these would have already been removed.

The main radioactive source material found in the primary sodium coolant system during normal operation will be sodium-24, which has a half-life of 15 hours. Because of its short half-life, it decays within a few days and thus poses only a minor maintenance problem. An additional activation product is sodium-22, which, though it has a longer half-life of 2.6 years, is produced in much smaller quantities (three orders of magnitude smaller) and is less radioactive than sodium-24. Moreover, the primary sodium system is designed to operate at a lower pressure than is the secondary system, and therefore contamination of the secondary system by leakage from the radioactive primary system is highly improbable.

Most of the tritium formed in the fuel and control elements will diffuse through the cladding into the sodium coolant. However, most of the tritium will be precipitated in the primary sodium-purification system cold traps. Of the small quantity (less than 10%) of the tritium that will diffuse through the intermediate-heat-exchanger tube walls to the secondary sodium system, most will precipitate in the secondary sodium-purification cold traps. Finally, less than 1% is expected to diffuse through the steam-generator tube walls to the steam system and be released as tritiated water in steam-generator blowdown streams. Figure 10-2 shows the tritium-release pathways in LMFBR plants (Ref. 1). One calculation shows that a total annual release rate of tritium (gaseous and liquid) from a 1,000-MWe LMFBR is approximately 120 Ci/plant-year (Ref. 1). In comparison, the tritium release from the 1,000-MWe reference LWR is 850 Ci/plant-year.

10.6.2 LIQUID-RADWASTE SYSTEM AND EFFLUENT SOURCE TERMS (R.G. 4.2/3.5.2)

All potentially contaminated liquids from the plant will be processed in the liquid-radwaste system before discharge. Laundry and laboratory wastes will be processed by this system, as will be liquid-waste streams from fuel-handling areas and the sodium-waste system. The quantity of low-level liquid waste to be processed by the system is expected to be between 200,000 and 40,000 gal/yr (Ref. 1).

Figure 10-3 shows the flow diagram for a typical LMFBR liquid-radwaste system. The system consists of two subsystems. The first subsystem is designed to process liquids with intermediate levels of radioactivity, with the effluent being reused after decontamination. The second subsystem is designed to process liquids with low levels of radioactivity, with the liquid released after the removal of radioactivity. The radioactivity is removed by ion-exchange beds or by evaporation of the liquid. The contaminated ion-exchange resins and evaporator concentrates will be handled by the solid-radwaste system. Therefore, except for tritium, only insignificant amounts of radioactivity will be released to the environment as liquid.

10.6.3 GASEOUS RADWASTE SYSTEM AND EFFLUENT SOURCE TERMS (R.G. 4.2/3.5.3)

There are two principal forms of radioactive materials in the primary coolant of the LMFBR: those that reside in the liquid sodium coolant at some equilibrium condition, and those volatile fission products that escape into the inert cover gas.

The radioactive-gas-removal systems provided in the LMFBR plant are designed to remove virtually all fission products from the primary cover-gas system. Figure 10-4 shows a schematic diagram of the radioactive-gas-removal system. Gases are temporarily stored for decay in a holdup system until the radioactivity from all gases except krypton-85 (half-life 10.8 years) has decayed to an insignificant level. Table 10-2 gives the estimated annual quantities released by leakage to the environment compared to those released from the reference LWR. The quantity of krypton-85 to be removed from the plant, between 2,000 and 6,000 Ci/yr, could be bottled in one standard 50-liter gas cylinder and shipped to a waste repository designed for the long-term storage of gaseous wastes (Ref. 1).

10.6.4 SOLID-RADWASTE SYSTEM (R.G. 4.2/3.5.4)

Solid radioactive wastes consist of spent resins, sludges, filters, clothing, and tools. These wastes are generated in other waste systems, laboratories, fuel-handling operations, and maintenance operations.

The flow diagram for a typical solid-radwaste system is shown in Figure 10-5. Filters and dry solids will be compacted, and moist resins will be dried and combined with other dry solids. Sludges will be mixed with concrete and cast into drums. Tritium wastes will be converted to tritiated calcium hydroxide. All solid wastes will be packaged in drums. The total number of 55-gallon drums required is estimated to be between 135 and 270 per year per plant. In comparison, 1,050 fifty-five-gallon drums of low-level waste are estimated to be shipped off site from the reference LWR each year.

10.7 CHEMICAL AND BIOCIDAL WASTES (R.G. 4.2/3.6)

The largest volume of chemical wastes discharged will originate as blowdown from the natural-draft cooling towers. The chemical constituents of the makeup water will be concentrated as a result of evaporative losses. Other contributors to chemical wastes will be the makeup-water treatment system and the regeneration of cation, anion, and mixed-bed demineralizers. Chemicals and their concentrations in the cooling-tower blowdown will depend on the chemicals and concentrations in the makeup water.

10.7.1 CHLORINE

Chlorination of cooling water is used to control biological slimes within a cooling system. It is essential in power plants that the slimes be removed because a buildup of slimes would seriously interfere with the transfer of heat and the flow of cooling water.

Chemical defouling is accomplished by the intermittent addition of chlorine to cooling water to kill the slime-forming organisms. Unfortunately, the agents that are toxic to slime-forming organisms are also toxic to other aquatic organisms. Therefore, it is desirable to manage defouling treatments to release as little as possible of the toxic substance to natural water bodies. The amount of chlorine discharged to water bodies from an LMFBR plant will be less than that from an LWR power plant because a smaller amount of cooling water is used.

10.7.2 OTHER CHEMICAL WASTES

The process-water system provides the high-purity water that is used in the steam loops of nuclear power plants to minimize corrosion and scale formation in the loop components. The aqueous waste effluent from the process-water system (regeneration waste) is a solution of sodium sulfate plus significant quantities of the dissolved and suspended solids and salts contained in the raw makeup water. These wastes are discharged either to the blowdown stream or once-through cooling streams for dilution before entering the environment. The amount of these wastes discharged from an LMFBR plant will be comparable to or less than that from an LWR power plant.

10.8 EFFECTS OF THE HEAT-DISSIPATION SYSTEM (R.G. 4.1/5.1)

The heat rejected to cooling water in the reference LWR plant is significantly more than that for this LMFBR plant (6.7×10^9 vs 5.2×10^9 Btu/hr at 1000-MWe operation). The impacts will be qualitatively the same as those described for the reference LWR but quantitatively less in proportion to the heat dissipated. However, thermal impacts from the reference LWR do not substantially affect licensability; therefore, this advantage of the LMFBR system will probably not represent a substantial improvement in licensability.

10.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATIONS (R.G. 4.1/5.2)

The environmental impact of radionuclide releases from an LMFBR plant during normal operation will be significantly smaller than that for the reference LWR. This is based on a comparison of the radionuclide releases from the LMFBR with the radionuclide releases from the reference LWR.

The exposure pathways for an LMFBR are similar to those for the reference LWR since there are no special siting requirements for an LMFBR relative to the reference LWR. The licensability of the LMFBR would therefore be at least as advantageous with regard to radiological impacts from routine operation as that of the reference LWR.

10.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES

The level of chlorine and other chemical wastes required for normal LMFBR operation is comparable to those required for normal operation of the LWR. It is concluded that, for the purposes of the NASAP comparison study, the effects of chemicals and biocides are probably not important.

10.11 OCCUPATIONAL EXPOSURE

On the basis of information compiled by the U.S. Nuclear Regulatory Commission (NRC) on past experience from operating nuclear power plants, it is estimated that the average collective dose to all onsite personnel at a 1,000-MWe LWR plant will be approximately 250 man-rem/plant-yr (Ref. 4) to 450 man-rem/plant-yr (Ref. 5). Although no directly relevant operating experience is available for a large LMFBR plant, an evaluation of the yearly exposure using the design parameters for the

CRBP indicates an exposure of approximately 280 man-rem/plant-yr (Ref. 4). On this basis, the exposure level will be comparable to that of a current LWR plant. The plant system designs and operational and maintenance procedures are such that the radiation protection afforded to plant personnel in a commercial LMFBR plant will be consistent with the requirements of 10 CFR 20.

Doses to plant personnel are influenced by many variables, including the following:

1. The ability of fuel elements to retain fission products
2. The extent of deposition of activated corrosion products throughout the primary and auxiliary coolant systems
3. The plant layout
4. Operational and maintenance procedures
5. In-service inspection procedures
6. Radiation protection programs

However, a major portion of the radiation exposure of plant personnel is received during maintenance, radwaste handling, in-service inspection, refueling, and nonroutine operations.

In addition to the fission products, the erosion and corrosion products that become mobile and are activated constitute perhaps the principal source of radiation with respect to the exposures of plant personnel. Specific radionuclides that have been identified in crud in LWR plants are cobalt-58, cobalt-60, manganese-54, zinc-65, and zirconium-95 (Ref. 3). Similar nuclides are expected to be present in LMFBR coolant and loop components.

The LMFBR fuel is assumed to be spiked by pre-irradiation or by the addition of a small quantity of cobalt-60. The resultant radiation level would be so high as to prohibit handling the fuel without substantial shielding. The spikant would not affect plant operation or fuel characteristics during and after fuel irradiation.

The effects of the radioactive spiked (or pre-irradiated) fresh fuel will be to increase the occupational dose to plant personnel during fresh-fuel handling and refueling operations. Since these fuel-handling operations are performed by automated remote-control systems, the actual incremental dose will be relatively small and will depend on the shielding designs for the fuel-handling system and the refueling procedures at a given plant. In the absence of actual data, it is expected that the dose for the operating personnel during fuel-handling operation for the spiked fuel will be, at worst, twice that of the non-spiked-fuel case. However, the fraction of the dose received by plant personnel during fuel-handling operations is only 4% (or 11 man-rem/yr) of the total dose received (Ref. 4). Therefore, the occupational exposure in a commercial LMFBR plant that uses spiked fuel may be about 22 man-rem/yr during fuel-handling operations. Thus, the total occupational exposure will be approximately 300 man-rem/plant-yr.

10.12 EFFECTS OF ALTERNATIVE FUEL CYCLES

10.12.1 URANIUM-PLUTONIUM/THORIUM SPIKED RECYCLE

The major difference between the uranium-plutonium/thorium spiked recycle and the uranium-plutonium/uranium spiked recycle is the substitution of thorium for uranium as the fertile material in the radial-blanket assemblies. The fuel assemblies in both cycles use plutonium-uranium as the fissile material with axial reflectors of depleted uranium. Relatively small changes are made in the amounts of plutonium and uranium in the fuel. The balance of plant (including heat-transport systems, steam and power-conversion systems, and waste-disposal system) is, in concept, identical for the two cycles; thus, the nonradiological environmental considerations related to power-plant operation are identical. These include land use, water use, heat-dissipation systems and effects, and chemical and biocidal wastes. The discussion in Sections 10.1 through 10.11 is therefore applicable to the uranium-plutonium/thorium cycle as well.

As noted above, the fuel assemblies in the two cycles use the same fissile and fertile materials--and in similar relative amounts. The design parameters and mechanical design features that affect fission-product retention and long-term fuel-element integrity are also similar; thus, the overall release of fission products from the fuel to the coolant is about the same for the uranium-plutonium/thorium cycle as for the uranium-plutonium/uranium cycle. Some differences may occur in the blanket; however, the major part (on the order of 90%) of the energy generation and fissions occurs in the fuel, with the balance occurring in the blanket. The fuel assemblies, therefore, far outweigh the blanket assemblies as contributors to fission-product release.

Because of the similarity in the fuel assemblies, reactors, and plants, the fission-product releases and occupational exposures would be about the same for the uranium-plutonium/thorium spiked recycle and the uranium-plutonium/uranium spiked recycle.

10.12.2 THORIUM-PLUTONIUM/THORIUM SPIKED RECYCLE

The major difference between the thorium-plutonium/thorium cycle and the uranium-plutonium/uranium spiked recycle is the substitution of thorium for uranium as the fertile material in the fuel assemblies and in the radial and axial blankets. The fuel assemblies in both cycles use plutonium as the fissile material. Relatively small changes are made in the amounts of plutonium in the fuel. The balance of plant (including heat-transport systems, steam and power-conversion systems, and waste-disposal system) is, in concept, identical for the two cycles; thus, the nonradiological environmental considerations related to power-plant operation are identical. These include land use, water use, heat-dissipation-system effects, and chemical and biocidal wastes. The discussion in Sections 10.1 through 10.11 is therefore applicable to the thorium-plutonium/thorium cycle as well.

As noted above, the fuel assemblies in the two cycles use the same fissile materials in similar relative amounts. The fertile material is changed from uranium to thorium. The design parameters and mechanical design features that affect fission-product retention and long-term fuel-element integrity are also similar.

At present, the performance of thorium in fast-reactor fuel is not well understood. As compared to uranium, thorium has both advantages and disadvantages (Refs. 6 and 7) in fundamental physical properties and in behavior leading to fuel-rod failure. There is virtually no experience with thorium-plutonium fuels for fast-breeder reactors. There is no basis for predicting long-term fission-product-retention properties with any degree of certainty.

For purposes of evaluation, it is therefore assumed that future research, development, and demonstration programs will result in design methods such that equivalent performance can be achieved from plutonium/thorium fuels as well as from plutonium/uranium fuels. Because of the similarity in reactors and plants and the assumed similarity of core fission-product releases, the plants' fission-product releases and occupational exposures should be about the same for the thorium-plutonium/thorium spiked recycle as for the uranium-plutonium/uranium spiked recycle.

10.12.3 DENATURED URANIUM-233/THORIUM CYCLE

The major difference between this fuel cycle and the uranium-plutonium/uranium spiked recycle fuel cycle is the use of denatured uranium-233 as the fuel and the use of thorium as the fertile material in the blankets. The uranium-233 concentration in the denatured uranium fuel is about 10%. The balance of plant (including heat-transport systems, steam and power-conversion systems, and waste-disposal system) is, in concept, identical for the two cycles; thus, the nonradiological environmental considerations related to power-plant operation are identical. These include land use, water use, heat-dissipation-system effects, and chemical and biocidal wastes. The discussion in Sections 10.1 through 10.11 is therefore applicable to the denatured uranium-233/thorium cycle as well.

As discussed earlier, the fuel assemblies for the denatured uranium-233/thorium cycle use uranium-233 rather than plutonium as the fissile material. Depleted uranium is the fertile material in the fuel region for both fuel cycles. The design parameters and mechanical design features that affect fission-product retention and long-term fuel-integrity are also similar. Fission yields of important isotopes are, however, somewhat different for the two fuels.

Thorium is used as the blanket material in the denatured uranium-233/thorium cycle, whereas depleted uranium is used in the uranium-plutonium/uranium cycle. Differences in the blanket are of little consequence to fission-product release since most fissions occur in the fuel region.

Comparing the two cycles, fuel performance (in terms of fission-product release to the coolant) would be expected to be quite similar. Operating experience with LWRs shows that fission-product releases from different plants of the same design can vary widely, and the releases can also be quite different for different fuel batches. Many factors can cause this, including detailed design features, fuel-fabrication differences, and operating transients.

These factors will undoubtedly occur for LMFBRs as well as for LWRs and appear to be more important in determining releases than the small differences in fuel design. It is concluded, therefore, that the releases of radioactivity to the environment would be within the same range for the denatured uranium-233/thorium cycle and the uranium-plutonium/uranium cycle.

Table 10-1. Design data for the LMFBR heat-dissipation system (wet natural-draft cooling tower)

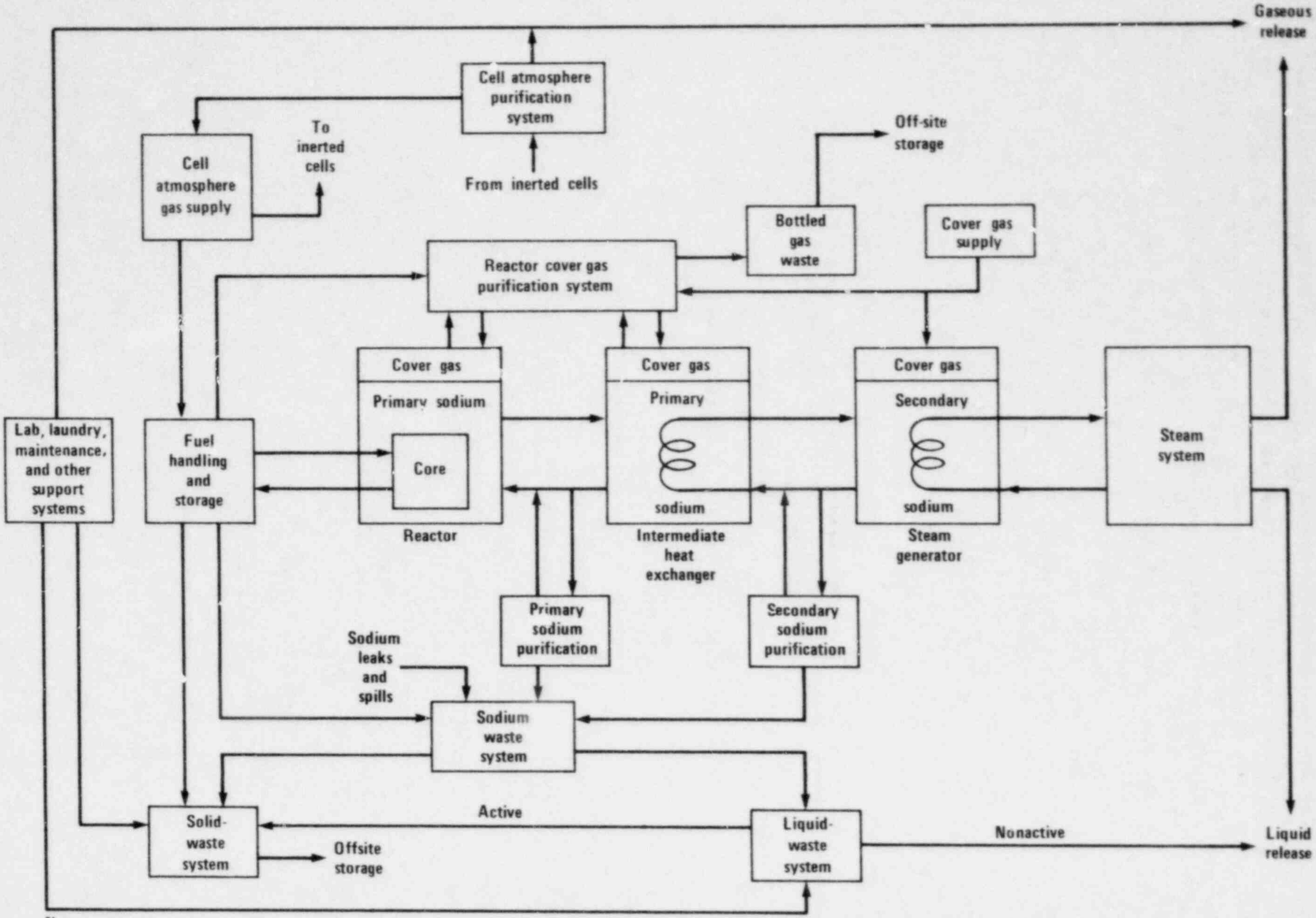
Heat-dissipation rate (maximum, full power), Btu/hr	5.2×10^{9}
Evaporation and drift (maximum, full power), gpm	8,900
Evaporation and drift (annual average), gpm	5,300
Blowdown (maximum), gpm	2,300
Blowdown (annual average), gpm	1,300

Table 10-2. Estimated radionuclide releases for a 1,000-MWe LMFBR power plant; and reference LWR

Nuclide	Release (Ci/plant-year) LMFBR[a]	Release (Ci/plant-year) LWR[b]
	Gaseous releases	
Tritium	56	580
Argon-39	75	---
Krypton-85m	1	11
Krypton-85	1	380
Krypton-87	1	2
Krypton-88	1	14
Xenon-133	1	7,200
Others	1	180
	Liquid releases	
Tritium	56	270

[a]Based on Table 4.2-2, p. 4.2-36, of Reference 1.

[b]1,000-MWe LWR, PSEID, Vol I.



Note:

Mass flows in kg per 0.75 GWe-yr.

Figure 10-1. Relationship of radioactive waste systems to reactor systems.

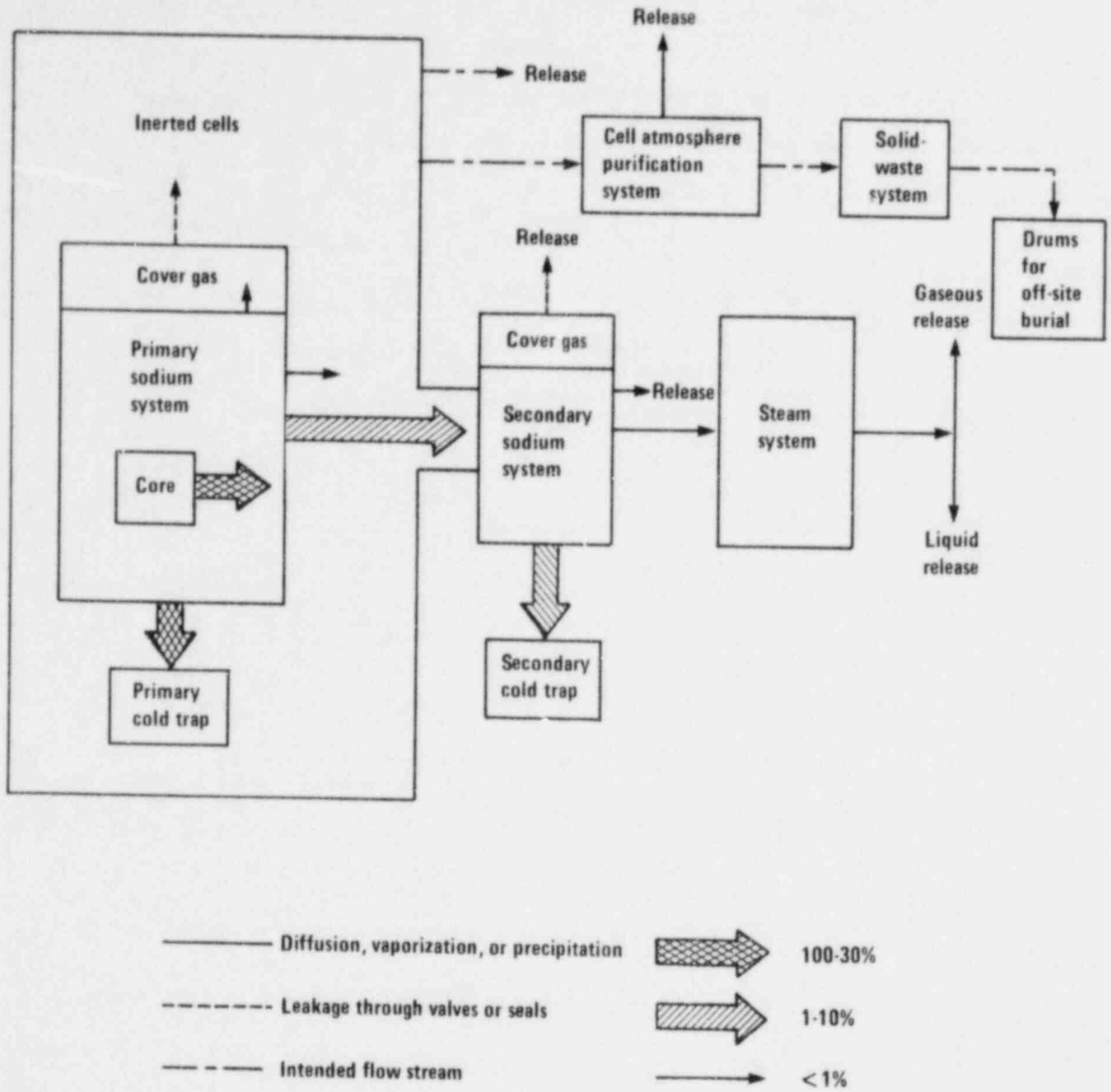


Figure 10-2. Pathways for tritium release in LMFBR power plants (Ref. 1).

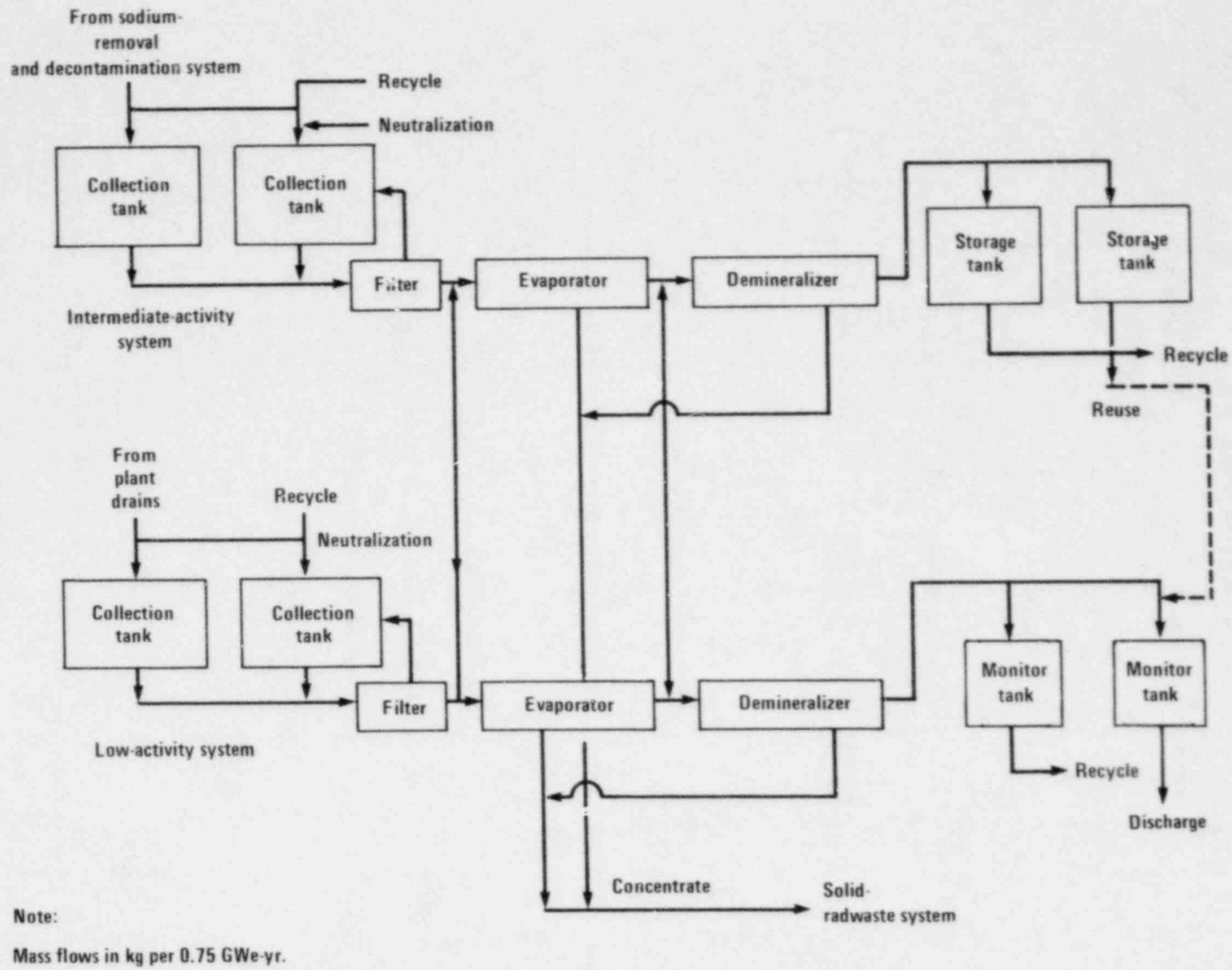
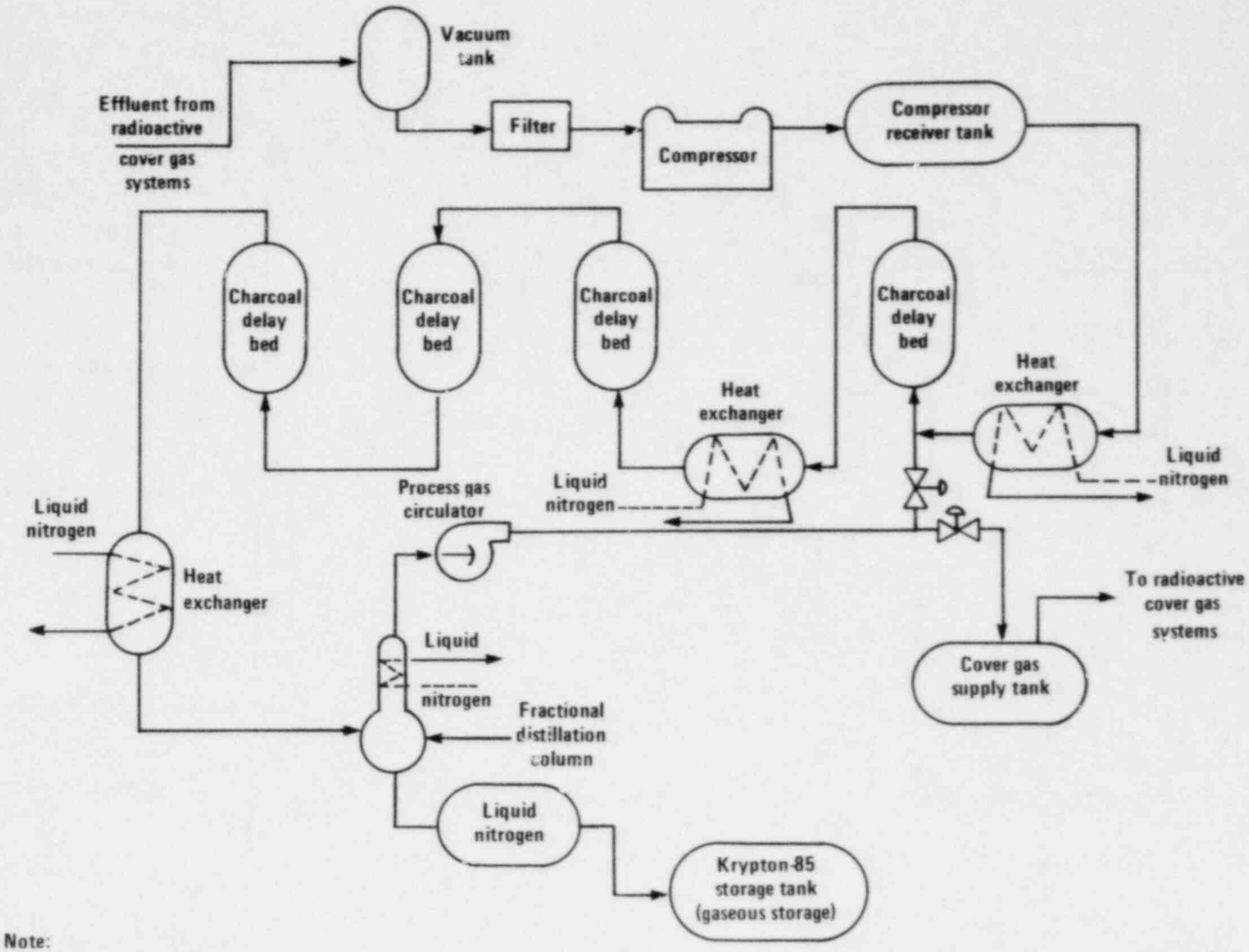


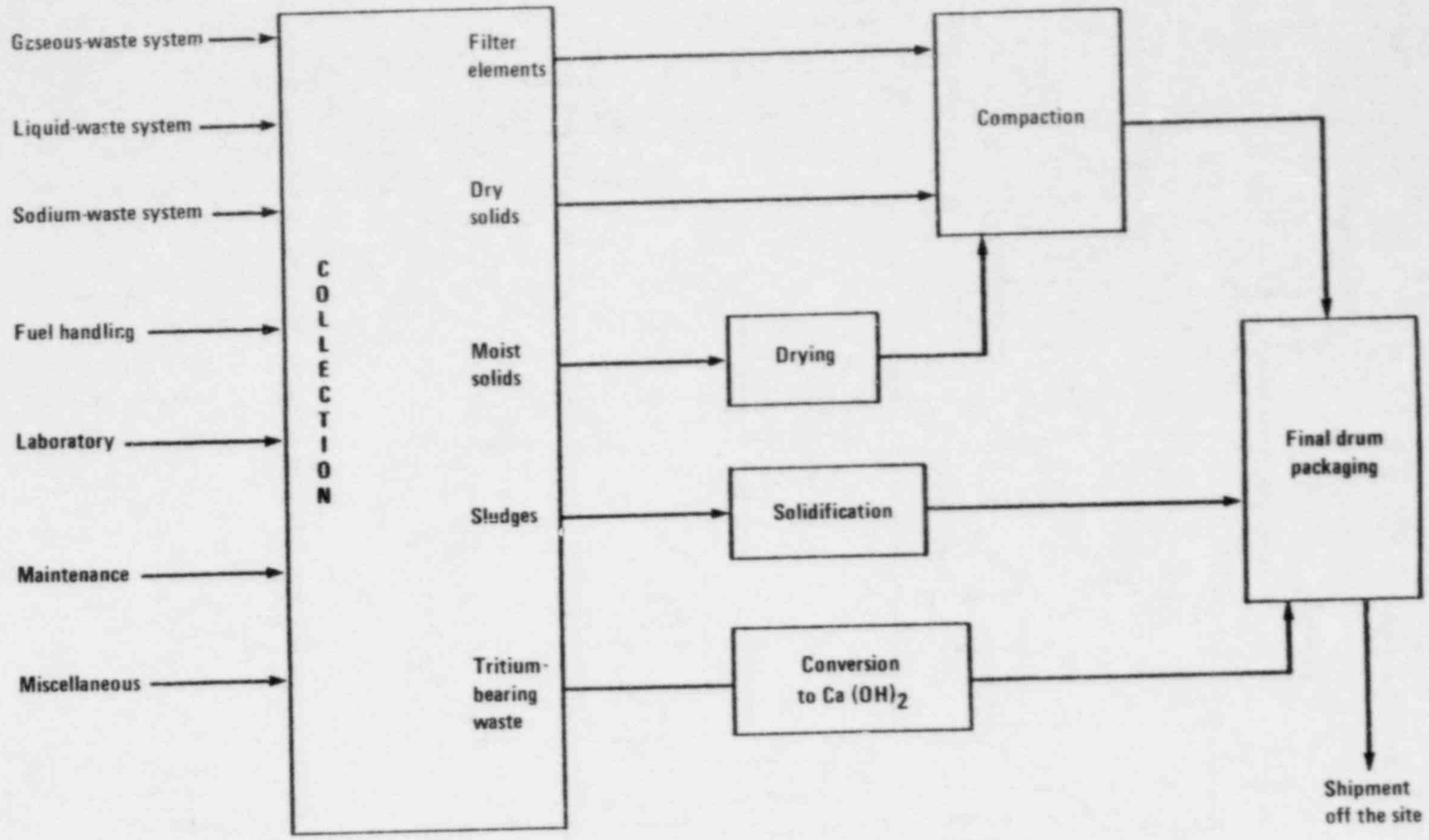
Figure 10-3. Liquid-radwaste-system flow diagram.



Note:

Mass flows in kg per 0.75 GWe-yr.

Figure 10-4. Schematic diagram of the radioactive-gas-processing system.



Note:

Mass flows in kg per 0.75 GWe-yr.

Figure 10-5. Solid-waste-system flow diagram.

REFERENCES FOR CHAPTER 10

1. U.S. Atomic Energy Commission, Environmental Statement--Liquid Metal Fast Breeder Reactor Program, WASH-1535, Section 4.2, December 1974.

2. Clinch River Breeder Reactor--Environmental Report, Vol. 2, Project Management Corporation, NRC Docket No. 50-537, April 1975.

3. U.S. Atomic Energy Commission, Environmental Statement: Liquid Metal Fast Breeder Reactor Demonstration Plant, WASH-1509, April 1972.

4. Clinch River Breeder Reactor--Preliminary Safety Analysis Report, Chapter 12, Project Management Corporation, NRC Docket No. 50-537, April 1975.

5. U.S. Nuclear Regulatory Commission, Final Environmental Statement, Perkins Nuclear Power Station No. 1, 2, & 3, NUREG-75/088, Section 3.5, October 1975.

6. P. R. Kasten et al., Assessments of the Thorium Fuel Cycle in Power Reactors, ORNL/TM-5565, Oak Ridge National Laboratory.

7. B. Talwar, Preconceptual Design Study of Proliferation Resistant, Homogeneous, Oxide-Fueled LMFBR Cores, General Electric Company, Advanced Reactor Systems Department, August 1978.

Chapter 11

LICENSING STATUS AND CONSIDERATIONS

The 200-MWt Enrico Fermi Atomic Power Plant was a licensed reactor (Ref. 1). The 400-MWt government-owned Fast Flux Test Facility (FFTF), soon to be completed, has been evaluated by the U.S. Nuclear Regulatory Commission (NRC) staff and the Advisory Committee for Reactor Safeguards and found acceptable for construction (Ref. 2). An additional safety evaluation pertaining to the operation of this reactor has been performed by the NRC staff (Ref. 3). On the basis of this evaluation, it has been concluded that the FFTF can be operated as a test reactor in a safe manner and with reasonable assurance of not endangering the health and safety of the public. A license application to construct a 975-MWt demonstration reactor, the Clinch River Breeder Reactor (CRBR), had been under review by the NRC until the public hearings were indefinitely suspended as a result of the President's energy message of April 22, 1978.

The licensing evaluation of the CRBR conducted so far (Refs. 4-6) indicates that the state of technology and experience would result in a safe design, but the evaluation is incomplete and the particular design proposed has not yet been found satisfactory. The major concerns that have not been resolved include instrumentation to detect core abnormalities that could lead to accidents, inspection of the reactor system to provide continued confidence in the system integrity, reliability of the decay-heat-removal system, and containment design to withstand the consequences of low-probability accidents. All of these topics are being addressed in the United States and other countries, and it appears that they can be satisfactorily resolved.

Alternative fuel cycles have an effect on the plant design, safety, and licensability. Of significant importance for licensing is the demonstrated technology; this is presently concentrated on the mixed oxides of uranium and plutonium, including most of the safety experiments and analysis models, but could be applicable to other oxide systems as well (Ref. 7).

The licensing of commercial-size LMFBRs is difficult to predict, particularly until the technology has been demonstrated on a large scale; however, no serious obstacles to licensability are currently forecast.

REFERENCES FOR CHAPTER 11

1. Power Reactor Development Company, Enrico Fermi Atomic Power Plant, Technical Information and Hazards Summary Report, Part B, Revised License Application, AEC Docket No. F-16.

2. U.S. Atomic Energy Commission, Safety Evaluation of the Fast Flux Test Facility, Project No. 448, October 31, 1972.

3. U.S. Nuclear Regulatory Commission, Safety Evaluation Report Related to Operation of Fast Flux Test Facility, Project No. 448, NUREG-0358, August 1978.

4. Site Suitability Report by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the Matter of the Clinch River Breeder Reactor Plant, Energy Research and Development Administration, Tennessee Valley Authority, Project Management Corporation, Docket No. 50-537, March 4, 1977.

5. U.S. Nuclear Regulatory Commission, Final Environmental Statement Related to Construction and Operation of Clinch River Breeder Reactor Plant, Project Management Corporation, Tennessee Valley Authority, and Energy Research and Development Administration, Docket No. 50-537, February 1977 (NUREG-0139).

6. J. F. Meyer, L. Lois, J. L. Carter, and T. P. Speis, An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics, U.S. Nuclear Regulatory Commission, NUREG-0122, March 1977.

7. H. K. Fauske et al., "Generic Considerations of LMFBR Hypothetical Accident Energetics," invited paper presented at the ENS/ANS International Topical Meeting on Nuclear Reactor Safety, Brussels, Belgium, October 16-19, 1978.

Chapter 12

RESEARCH, DEVELOPMENT, AND DEMONSTRATION

12.1 RESEARCH FACILITIES

The research facilities for the breeder-reactor safety research and development program provide support for the experimental data requirements. The research facilities can further be categorized as in-reactor test facilities and out-of-reactor safety test facilities.

12.1.1 IN-REACTOR FACILITIES

12.1.1.1 Transient Reactor Test Facility (TREAT)

The TREAT is a pulsed test reactor. Current experiments are designed to examine fuel failure under transient overpower and loss-of-flow conditions, and to provide data for reconstruction of the mechanisms that produced fuel failure and subsequent movements of sodium, fuel, and cladding.

12.1.1.2 TREAT Upgrade (TU)

The TREAT reactor is scheduled to be upgraded in response to the need for in-reactor experimental data to resolve key safety issues for the fast-breeder reactor. The TREAT upgrade will also provide data for the study of questions associated with key issues of fuel dispersal and to the recriticality potential for disrupted fuel. The estimated availability of this facility is FY 1982.

12.1.1.3 Sodium Loop Safety Facility (SLSF)

The SLSF is designed to produce information needed to demonstrate the behavior of fast-breeder-reactor fuel elements, subassemblies, or the core for postulated abnormal or hypothetical accident situations. The SLSF provides for the utilization of full-length rods from the Fast Flux Test Facility and for rod bundles of up to 61 rods.

12.1.2 OUT-OF-REACTOR FACILITIES

12.1.2.1 Thermal-Hydraulic Out-of-Reactor Safety Facility (THORS)

THORS provides an engineering-scale, high-temperature sodium loop for thermal-hydraulic testing of simulated reactor subassemblies at normal and abnormal operating conditions.

12.1.2.2 Sodium Boiling Test Facility (SBTF)

The SBTF is a single-channel sodium loop for the study of free-convection and low-flow forced-convection boiling dynamics and heat transfer.

12.1.2.3 Components and Materials Evaluation Loop (CAMEL)

CAMEL is designed to perform out-of-reactor tests undertaken to examine the hydraulic aspects of fuel sweep out and/or plug formation under simulated transient overpower conditions.

12.1.2.4 Containment Systems Test Facility (CSTF)

The CSTF is designed to obtain baseline information on aerosol behavior in large vessels as well as on the performance of emergency air-cleaning systems. The CSTF is 20.3 meters in height and 7.6 meters in diameter, and the vessel volume is approximately 850 m^3. Typical test conditions have included the generation of aerosols resulting from a simulated sodium spill of approximately 450 kg sodium at 900 K.

12.1.2.5 ANL Zero Power Reactors (ZPR-6, ZPR-9, ZPR)

These existing critical-assembly facilities are used to obtain experimental physics data needed for confirmation of physics aspects of core design and safety analyses.

12.1.3 OUT-OF-REACTOR "LABORATORY-SCALE" FACILITIES

12.1.3.1 Out-of-Pile Expulsion and Reentry Apparatus (OPERA)

OPERA was constructed at the Argonne National Laboratory (ANL) primarily for prototype studies of coolant behavior after postulated pump-coastdown and flow-blockage transients. The apparatus is presently being modified to perform these investigations with a 15-rod triangular section.

12.1.3.2 Large Sodium Fire Facility (LSFF)

The LSFF at the Hanford Engineering Development Laboratory (HEDL) has been modified to perform a variety of investigations of phenomena associated with sodium spills. These include evaluation of cavity-liner designs, sodium-concrete interactions, hydrogen recombination, large-scale "feature" tests, and filter-loading studies on emergency air-cleaning systems.

12.1.3.3 Direct Electric Heating (DEH)

The DEH apparatus at the Argonne National Laboratory relies on the electrical resistance of the fuel pellets to generate time-varying thermal transients in a restrained fuel column. The fuel is restrained in a quartz tube to allow observation of the fuel column during the transients. The data obtained from these experiments include fuel expansion and/or slumping of fuel during transient overpower and loss-of-flow thermal transients. Investigations are now under way to determine the response of fresh uranium carbide fuels to thermal transients.

12.1.3.4 Fuel Cladding Transient Tester (FCTT)

The FCTT at the Hanford Engineering Development Laboratory provides data on the response of unirradiated and irradiated cladding materials to internal pressure and thermal loadings. The pressurized cladding tube sample is inductively heated to simulate a temperature ramp. Data have been obtained on the cladding strength/ductility, fracture mode, failure strain/temperature, and wastage effects as input to the development of the Larson/Miller parameter values. Plans are being made to expand the capability to provide controlled strain-rate and temperature-rate transients.

12.1.3.5 Fission-Gas Release Mechanism (FGRM)

The FGRM facility at the Hanford Engineering Development Laboratory is designed to provide data on transient fission-product release from samples of irradiated fuel.

The transient is simulated by inductive heating of the fuel sample and the gas release is analyzed. Direct observation of the fuel-transient response is provided by a unique gamma-ray scanning technique.

12.2 ANALYTICAL DEVELOPMENT

Along with experiments and phenomenological modeling efforts, computer codes perform an invaluable service in providing for continuing improved resolution of fast-breeder-reactor (FBR) safety and licensing problems. In support of the FBR safety research and development goals there is a continuing program of computer code development and validation that addresses the various proposed stages of hypothetical core-disruptive accidents.

12.2.1 STEADY-STATE FUEL-ROD CHARACTERIZATION

Before a hypothetical core-disruptive accident can be analyzed, it is necessary to characterize the fuel before the accident. The LIFE (ANL) and SIEX (HEDL) codes are capable of providing the information on fuel restructuring that has occurred under operating conditions before the accident, while whole-core HCDA analysis codes, such as SAS (ANL), have their own steady-state characterization routines. These codes must be capable of providing information on such topics as fuel swelling, cladding swelling, and fission-gas release for accurate analysis of the accident.

12.2.2 WHOLE-CORE ANALYSIS

The whole-core-analysis codes model the response of the entire core to a hypothetical core-disruptive accident. Depending on the state of the core during the accident, the analysis can be broken into four phases: initiating phase, transition phase, core-disassembly phase, and reactor-vessel and structure-response phase.

12.2.2.1 Initiating-Phase Analysis

Codes in this category are concerned with the phenomena occurring from accident inception until

1. There is a benign neutronic shutdown with intact core geometry and very little core damage.
2. There is a gradual meltdown of the core.
3. There is gross core disassembly because of the very large pressures generated by vaporized core materials.

Codes that analyze the initiating phase can treat individual phenomena separately, or they can combine the individual phenomena into an integrated whole-core analysis. Examples of the former are BEHAVE, developed by the General Electric Company to study fuel-rod behavior; LAFM, developed by the Los Alamos Scientific Laboratory (LASL) to study fuel-rod behavior; and FRAS and PLUTO, developed by ANL to study fission-gas behavior and fuel and coolant motion, respectively. Examples of the latter are SAS (ANL) and MELT (HEDL).

12.2.2.2 Transition-Phase Analysis

If the negative reactivity feedback during the initiating phase is not sufficient to terminate the accident but is sufficient to preclude a prompt-critical power excursion, it is probable that the core would gradually melt and begin to boil. Codes such as TRANSIT and TRANSIT-HYDRO (ANL), FUMO (HEDL), and SIMMER (LASL) attempt to model the phenomena occurring during this transition phase.

12.2.2.3 Core-Disassembly Analysis

Should there be a prompt-critical power excursion during the course of an accident, a level of energetics sufficient to challenge the energy-absorption capability of the primary containment can be postulated. A prompt-critical power excursion may theoretically occur during

1. The initiating phase, because of reactivity insertions from sodium voiding and/or autocatalytic fuel-rod failures
2. The transition phase, because of molten-fuel-pool compaction and subsequent recriticality

Codes in this category calculate the energetics resulting from such prompt-critical bursts. Examples of disassembly-phase analytical tools are FX II-VENUS III (ANL) and SIMMER (LASL).

12.2.2.4 Reactor Vessel and Structural Response

These codes determine whether the energy-absorption capability of the primary containment is exceeded. The pressures generated by either a prompt-critical power excursion or a thermal interaction between fuel and coolant are used to determine the mechanical loading on the structural members of the primary vessel. Codes in this category include REXCO (ANL) and ICECO (ANL).

12.2.3 SUBASSEMBLY ANALYSIS

Codes have been developed to analyze the response of a single subassembly to accident conditions. These are useful in analyzing safety experiments on a single subassembly. They are also capable of more detailed thermal-hydraulic modeling than are the whole-core-analysis codes. This is important since incoherence effects, such as sodium boiling and fuel-rod failure, can be better assessed with these codes than with the whole-core-accident codes. Examples of such codes are COBRA (ANL, HEDL), which models subassembly thermal-hydraulics; STRAW (ANL), which determines the subassembly structural response to a transient; and PORPLUG (ANL), which can be used to analyze flow blockages.

12.2.4 HEAT-TRANSPORT SYSTEM ANALYSIS

For a complete analysis of a hypothetical core-disruptive accident, it is not enough to know just what happens to the core. Pressure pulses from such accidents could be transmitted through the primary piping loops and could jeopardize their structural integrity. Normal and abnormal design transients that could occur during plant lifetime must also be analyzed to show that the heat-transport system is capable of removing the heat generated during such transients. The ICEPEL (ANL) code performs such analyses.

12.2.5 POST-ACCIDENT HEAT REMOVAL (PAHR)

After neutronic shutdown in a hypothetical core-disruptive accident, there must be a long-term capability to remove the decay heat from any molten debris that may be formed. Analysis of post-accident heat removal has been performed using the "general principles approach" rather than using large, sophisticated computational packages. The GROWS (ANL) code supports PAHR analyses.

12.2.6 SODIUM FIRES

In the unlikely event that a leak should develop from a sodium pipe, the reactions of the sodium with the surrounding equipment-cell atmosphere and the materials of the cell must be assessed. This is required for proper design of the containment cells. SPRAY and CACECO (HEDL), as well as SOFIRE and SOMIX (Atomics International) support such analyses.

12.2.7 AEROSOL BEHAVIOR

After the reactor-vessel structural response has been assessed, it is necessary to determine how much radioactive material is released through the vessel and into the containment building in order to define the radiological source term. Most of the radioactive material generated is in the form of aerosols. The amount of radioactive material that could eventually be released to the environment depends on the behavior of these aerosols. Codes such as HAA-3 (Atomics International) support the modeling of aerosol behavior.

12.2.8 ENVIRONMENTAL-RELEASE ANALYSIS

These codes assess the radioactive dose to man resulting from release of materials from containment to the environment. Many codes have been developed for the analysis of environmental release of radioactive material. Codes such as ACRA (Oak Ridge National Laboratory) and COMRADEX (Atomics International) support analyses in these areas.

ADDENDUM I

ADDENDUM I

TABLE OF CONTENTS

	Page
1.0 GENERAL DESCRIPTION	
1.1 DESIGN GROUND RULES	1
1.2 DESIGN DESCRIPTION	1
1.2.1 Homogeneous Cores (Combustion Engineering Designs)	1
1.2.2 Heterogeneous Cores (Atomics International Designs)	1
1.3 NUCLEAR ANALYSIS	1
1.3.1 Homogeneous Cores (Combustion Engineering Designs)	1
1.3.2 Heterogeneous Cores (Atomics International Designs)	1
1.4 THERMAL-HYDRAULIC ANALYSIS AND DESIGN	1
1.4.1 Homogeneous Cores (Combustion Engineering Designs)	2
1.4.2 Heterogeneous Cores (Atomics International Designs)	2
1.5 MECHANICAL ANALYSIS AND DESIGN	2
1.5.1 Homogeneous Cores (Combustion Engineering Designs)	2
1.5.2 Heterogeneous Cores (Atomics International Designs)	2
	2
2.0 DESIGN DESCRIPTIONS	5
2.1 HOMOGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND URANIUM CARBIDE BLANKETS	5
2.1.1 Description	5
2.1.2 Fuel Management	5
2.2 HETEROGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND URANIUM CARBIDE BLANKETS	5
2.2.1 Description	5
2.2.2 Fuel Management	6
2.3 HOMOGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND THORIUM CARBIDE BLANKETS	6
2.3.1 Description	6
2.3.2 Fuel Management	7
2.4 HETEROGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND THORIUM CARBIDE BLANKETS	7
2.4.1 Description	7
2.4.2 Fuel Management	7
3.0 SAFETY CONSIDERATIONS FOR THE CARBIDE-FUELED LMFBR	23
3.1 INTRODUCTION	23
3.2 KEY SAFETY ISSUES FOR CARBIDE-FUELED LMFBR	23
3.3 STATUS OF KEY ISSUES FOR CARBIDE-FUELED LMFBR	23
3.3.1 Accommodation of Core Melt and Disruptive Accidents in a Carbide-Fueled LMFBR	24
4.0 ENVIRONMENTAL CONSIDERATIONS	27
5.0 LICENSING STATUS AND CONSIDERATIONS	28
6.0 RESEARCH, DEVELOPMENT, AND DEMONSTRATION	29
REFERENCES	32

List of Tables

		Page
1-1	Ground rules for the proliferation-resistant LMFBR core design study	3
1-2	Assembly parameters for the carbide-fueled LMFBR	4
1-3	Blanket-assembly parameters for the carbide-fueled LMFBR	4
2-1	Summary core-design description for a carbide-fueled LMFBR homogeneous core: plutonium-uranium carbide core and uranium carbide blankets	9
2-2	Equilibrium-cycle reactor charge and discharge data for the LMFBR homogeneous (Pu-U) C/UC spiked recycle	10
2-3	Summary of main parameters for (Pu-U) C core, UC blankets, heterogeneous core	11
2-4	Equilibrium-cycle reactor charge and discharge data for the LMFBR heterogeneous (Pu-U) C/UC spiked recycle	13
2-5	Summary core-design description for (Pu-U) C core, ThC blankets, homogeneous core	14
2-6	Equilibrium-cycle reactor charge and discharge data for the LMFBR homogeneous (Pu-U) C/Th spiked recycle	15
2-7	Summary of main parameters for (Pu-U) C core, ThC blankets, heterogeneous core	16
2-8	Equilibrium-cycle reactor charge and discharge data for the LMFBR heterogeneous (Pu-U) C/Th spiked recycle	18
6-1	Experiments with carbide fuel in terms of key issues and facilities	30

List of Figures

		Page
2-1	Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, homogeneous core (uranium carbide blankets, spiked recycle)	19
2-2	Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, heterogeneous cycle (uranium carbide blankets, spiked recycle)	20
2-3	Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, homogeneous core, thorium carbide axial blanket, uranium radial blanket (spiked recycle)	21
2-4	Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, heterogeneous core, thorium carbide blankets (spiked recycle)	22
3-1	Necessary temperatures for carbide fuels to cause spontaneous nucleation on contact with liquid sodium	26

CARBIDE-FUELED 1,000-MWe LMFBR: PLUTONIUM-URANIUM CARBIDE CORE AND URANIUM CARBIDE BLANKETS

1.0 GENERAL DESCRIPTION

The designs considered here are based on a 1,000-MWe carbide-fueled liquid-metal fast breeder reactor (LMFBR) power plant, with the balance of plant undefined. The core designs were developed for the Proliferation-Resistant LMFBR Core Design Study (PRLCDS) program, which was initiated by the U.S. Department of Energy in October 1977 and concluded in September 1978.

1.1 DESIGN GROUND RULES

A common set of ground rules developed for the PRLCDS program applies to all core designs considered in this volume. Tables 1-1 through 1-3 summarize several of the more important parameters. Appendix I of Reference 1 and Section 2 of Reference 2 present a complete discussion of the ground rules used as bases for these core designs.

1.2 DESIGN DESCRIPTION

1.2.1 HOMOGENEOUS CORES (COMBUSTION ENGINEERING DESIGNS)

The homogeneous-core designs were developed by Combustion Engineering, Inc. The designs are summarized in Section 1.4 of Reference 2. The parametric studies for design optimization are described in Section 3 of Reference 2.

1.2.2 HETEROGENEOUS CORES (ATOMICS INTERNATIONAL DESIGNS)

The heterogeneous-core designs were developed by Atomics International Division, Rockwell International. The key features of the designs are summarized in Section 2.1 of Reference 1. The physical characteristics of the reactor-core designs are described in Section 3 of Reference 1.

1.3 NUCLEAR ANALYSIS

1.3.1 HOMOGENEOUS CORES (COMBUSTION ENGINEERING DESIGNS)

The results of detailed neutronics calculations for the Combustion Engineering homogeneous-core designs are described in Section 5 of Reference 2.

1.3.2 HETEROGENEOUS CORES (ATOMICS INTERNATIONAL DESIGNS)

The nuclear designs analyses for the Atomics International heterogeneous cores are described in Section 4 of Reference 1.

1.4 THERMAL-HYDRAULIC ANALYSIS AND DESIGN

1.4.1 HOMOGENEOUS CORES (COMBUSTION ENGINEERING DESIGNS)

The thermal and hydraulic design procedures and characteristics for the Combustion Engineering homogeneous cores are described in Section 6 of Reference 2.

1.4.2 HETEROGENEOUS CORES (ATOMICS INTERNATIONAL DESIGNS)

The thermal-hydraulic analyses for the Atomics International heterogeneous cores are described in Section 5 of Reference 1.

1.5 MECHANICAL ANALYSIS AND DESIGN

1.5.1 HOMOGENEOUS CORES (COMBUSTION ENGINEERING DESIGNS)

The driver-rod and assembly design, radial-blanket-assembly designs, and control-assembly volume fractions for the Combustion Engineering homogeneous cores are described in Section 4 of Reference 2.

1.5.2 HETEROGENEOUS CORES (ATOMICS INTERNATIONAL DESIGNS)

The results of mechanical analyses and evaluations for the Atomics International heterogeneous cores are described in Section 6 of Reference 1.

Table 1-1. Ground rules for the proliferation-resistant LMFBR core design study

General parameters	
Reactor lifetime, yr	30
Net electric power, MWe	1,000[a]
Thermal efficiency[b]	0.365
Reactor inlet temperature, ^{o}F	650
Core temperature rise, ^{o}F	280
Flow parameters	
Maximum rod-bundle coolant velocity, ft/sec[c]	35
Maximum rod-bundle pressure drop, exclusive of entry and exit losses, psi	90
Bypass flow, %	5[d]
Fuel management	
Plant capacity factor, %	70
Refueling interval	Multiples of 6 months
Number of core batches	Open
Residence time, yr	
Driver fuel assemblies	Open
Blanket assemblies	≤6
Number of enrichment zones	Open
Out-of-reactor time, yr	
Plutonium fissile	1.00
U-233 fissile	1.33
Combined fabrication reprocessing loss	0.01

[a]This value was chosen to allow use of the turbine-generator systems designed during the Prototype Large Breeder Reactor studies.

[b]This value is defined as the ratio of gross electric power (turbine-generator output) to gross thermal power (reactor power plus pumping heat input).

[c]This value represents a moderate advance in technology.

[d]This fraction of the total flow is unheated; the remainder is available for cooling driver and blanket assemblies.

Table 1-2. Assembly parameters for the carbide-fueled LMFBR

Subassembly pitch	Open
Spacer type	Wire wrap
Spacer pitch, in.	12
Minimum cladding thickness, mils	12
Ratio of minimum cladding thickness to outside diameter	0.039
Minimum driver rod pitch-to-diameter ratio	Open
Nominal peak linear power, kW/ft	Open[a]
Plenum location	Split or top
Vented ducts	Not allowed
Maximum nominal subassembly outlet temperature, °F	1,075
Maximum core height	Open[b]
Smear density, % of theoretical	82.0
Maximum cladding outside diameter temperature, °F	Open

[a]Limited to 30 kW/ft for sodium-bonded carbide rods.
[b]Core-size effects on capital cost should be considered.

Table 1-3. Blanket-assembly parameters for the carbide-fueled LMFBR

Identical design for internal and radial blanket assemblies	Required
Minimum cladding thickness, mils	12
Ratio of minimum cladding thickness to outside diameter	0.0229
Nominal peak linear power, kW/ft	Open[a]
Maximum smear density, % of theoretical	90.0

[a]Limited to 30 kW/ft for sodium-bonded carbide rods.

2.0 DESIGN DESCRIPTIONS

2.1 HOMOGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND URANIUM CARBIDE BLANKETS

2.1.1 DESCRIPTION

A summary description of the core design is presented in Table 2-1. Details of the reactor design and performance are given in Table C.1 of Reference 2, Appendix C. The design is identified in Reference 2 as Reference-UC Blankets.

The reactor/fuel-cycle combination is fueled with mixed uranium-plutonium carbide and depleted uranium carbide in the axial and radial blanket assemblies. The core and blankets are reprocessed separately, and the core is coprocessed. The blankets are reprocessed to produce partially partitioned products. Part of the recovered uranium-plutonium of 20% fissile assay is sent to core fabrication, where it is mixed with the coprocessed uranium-plutonium and depleted makeup uranium fuel. The excess 20% fissile material is sent to secure storage. The residual uranium from blanket reprocessing is sent to blanket fabrication, where it is mixed with makeup depleted uranium.

2.1.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 2-2. The mass-flow diagram is shown in Figure 2-1. The numerical identifiers in the fuel-cycle steps are correlated with the fuel-cycle descriptions of Volume VII as follows:

Reprocessing (Purex 2)	Section 5.2
Plutonium storage	Section 6.2
Depleted uranium storage	Section 6.4
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The mass-flow data for Table 2-2 and Figure 2-1 were extracted from Table C.12 of Reference 2, Appendix C. The data given in the original table are based on a plant electrical power of 1,095 MWe, a plant capacity factor of 70%, and a 296-day fuel cycle. In accordance with the Nonproliferation Alternative Systems Assessment Program (NASAP) ground rules, the data in Table 2-2 and Figure 2-1 were normalized to 1,000 MWe by multiplying the original data by 1,000/1,095, to a 75% capacity factor by multiplying by 0.75/0.70, and to an annual basis by multiplying by 365/296.

2.2 HETEROGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND URANIUM CARBIDE BLANKETS

2.2.1 DESCRIPTION

The main core-design parameters are summarized in Table 2-3. Detailed specifications for the reactor design are given in Appendix III of Reference 1. This design is identified in Reference 1 as Reference A.

The reactor/fuel-cycle combination is fueled with uranium-plutonium carbide in the core assemblies and uranium carbide in the axial and internal blankets. The core is coprocessed to recover the uranium-plutonium, which is recycled to fabrication;

the blankets are coprocessed with partial partition. Part of the recovered uranium-plutonium is recycled to fabrication, where it is mixed with the uranium-plutonium from core reprocessing. The excess uranium-plutonium, of 20% fissile assay, is sent to secure storage. The uranium recovered during blanket reprocessing is recycled to blanket fabrication, where it is mixed with makeup depleted uranium.

2.2.2 FUEL MANAGEMENT

The mass-flow diagram is shown in Figure 2-2, and the equilibrium-cycle reactor charge and discharge data are given in Table 2-4. The numerical identifiers in the fuel-cycle steps are correlated with the fuel-cycle descriptions of Volume VII as follows:

Reprocessing (Purex 2)	Section 5.2
Plutonium storage	Section 6.2
Depleted uranium storage	Section 6.4
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The mass-flow data for Table 2-4 and Figure 2-2 were extracted from Table A-V to A-XII of Reference 1, Appendix III.A. The original tables report values for a half-core model. Therefore, all values were multiplied by 2 to obtain values for the whole core. Furthermore, since the data given in the original tables are based on a plant capacity factor of 70%, the data in Table 2-4 and Figure 2-2 were normalized to a 75% capacity factor by multiplying the original mass-flow values by 0.75/0.70.

2.3 HOMOGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND THORIUM CARBIDE BLANKETS

2.3.1 DESCRIPTION

A summary description of the core design is presented in Table 2-5. Details of the reactor design and performance data are given in Table C.1 of Reference 2, Appendix C. The design is identified in Reference 2 as low-cost coprocessing.

The reactor/fuel-cycle combination uses mixed uranium-plutonium carbide in the core, thorium carbide in the axial blanket, and thorium-uranium carbide (made from recycled uranium-233) in the radial blanket. The core and blankets are reprocessed separately. The core is coprocessed with partial partition. The uranium-plutonium is partitioned at a 20% fissile content, and most of it is recycled to fabrication, where it is mixed with the rest of the uranium recovered during core reprocessing and makeup depleted uranium. The excess 20% fissile uranium-plutonium is sent to secure storage. The blankets are coprocessed with partial partition. The thorium/uranium-233 is recovered at a 12% fissile content. Most of it is sent to secure storage, but part is recycled to radial blanket fabrication, where it is mixed with part of the recovered thorium and makeup new thorium. The recycled thorium is highly radioactive and provides the axial blanket, and thereby the spiking for the uranium-plutonium fuel elements. The balance of the recovered thorium is sent to axial blanket fabrication.

2.3.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 2-6. The mass-flow diagram is shown in Figure 2-3. The numerical identifiers in the fuel-cycle steps are correlated with the fuel-cycle descriptions of Volume VII as follows:

Reprocessing (Purex 2)	Section 5.2
Reprocessing (Thorex 3)	Section 5.7
Plutonium storage	Section 6.2
Depleted uranium storage	Section 6.4
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The mass-flow data for Table 2-6 and Figure 2-3 were extracted from Table C.14 of Reference 2, Appendix C. The data given in the original table are based on a plant capacity factor of 70% and a 395-day fuel cycle. In accordance with the NASAP ground rules, the data in Table 2-6 and Figure 2-3 were normalized to a 75% capacity factor by multiplying the original values by 0.75/0.70 and to an annual basis by multiplying by 365/395.

2.4 HETEROGENEOUS LMFBR CORE: PLUTONIUM-URANIUM CARBIDE CORE AND THORIUM CARBIDE BLANKETS

2.4.1 DESCRIPTION

The main core-design parameters are summarized in Table 2-7. Detailed specifications for the reactor design are given in Appendix III of Reference 1. This design is identified in Reference 1 as Reference B.

The reactor/fuel-cycle combination uses uranium-plutonium carbide in the fuel elements and recycled thorium carbide in the axial, radial, and inner blankets. The core and blankets are reprocessed separately. The core is coprocessed, and all recovered uranium-plutonium is recycled to fabrication, where it is mixed with makeup uranium-plutonium from secure storage. The blanket materials are coprocessed to partially partition the uranium-233 and the thorium. The thorium/uranium-233, 12% fissile, is sent to secure storage. The excess thorium is recycled to blanket fabrication, where it is mixed with new makeup thorium. The recycle thorium is highly radioactive and provides the axial blanket, and thereby the spiking for the uranium-plutonium fuel elements.

2.4.2 FUEL MANAGEMENT

The equilibrium-cycle reactor charge and discharge data are given in Table 2-8. The mass-flow diagram is shown in Figure 2-4. The numerical identifiers in the fuel-cycle steps are correlated with the fuel-cycle descriptions of Volume VII as follows:

Reprocessing (Purex 2)	Section 5.2
Reprocessing (Thorex 1)	Section 5.4
Plutonium storage	Section 6.2
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The mass-flow data for Table 2-8 and Figure 2-4 were extracted from Tables B-V to B-XII of Reference 1, Appendix II.B. The original tables report values for a half-core model. Therefore, all values were multiplied by 2 to obtain values for the whole core. Furthermore, since the data given in the original tables are based on a plant capacity factor of 70%, the data in Table 2-8 and Figure 2-4 were normalized to a 75% capacity factor by multiplying the original mass-flow values by 0.75/0.70.

Table 2-1. Summary core-design description for a carbide-fueled LMFBR homogeneous core: plutonium-uranium carbide core and uranium carbide blankets[a]

General parameters	
Reactor power, MWt	3,000
Core volume, 10^3 liters	11.1
Core height, cm	106.7
Fuel residence time, yr	2.4
Driver assembly	
Number of rods per assembly	169
Rod pitch-to-diameter ratio	1.20
Lattice pitch, cm	16.48
Duct-wall thickness, mm	3.81
Rod diameter, mm	9.40
Cladding thickness, mm	0.38
Bond type	Sodium
Smear density, % of theoretical	77
Performance	
Peak linear power (3σ + 15% OP), kW/m	120
Peak cladding temperature, end of life (2σ midwall), ^{o}C	658
Fissile inventory at beginning of equilibrium cycle, kg	
Uranium fissile	--
Plutonium fissile	3,155
Fissile production/destruction, kg/yr	
Uranium fissile	--
Plutonium fissile	321
Fuel cycle cost, mills/kW-hr	7.5
Symbiotic system doubling time, yr	--
Sodium void worth at end of equilibrium cycle, $	5.02

[a]Combustion Engineering reference design with uranium carbide blankets (Ref. 1).

Table 2-2. Equilibrium-cycle reactor charge and discharge data for the LMFBR homogeneous (Pu-U)C/UC spiked recycle[a]

Isotope	Charge[b]			Discharge[b]		
	Core	AB	RB	Core	AB	RB
Uranium-235	18.1	18.1	32.6	8.45	15.7	29.0
Uranium-236	0.0	0.0	0.0	$<$1.2	0.0	0.0
Uranium-238	9,028.7	8,947.9	16,598.7	8,249.3	8,776.5	16,383.9
Plutonium-238	13.27			8.45		
Plutonium-239	943.5			999.0	147.2	181.0
Plutonium-240	270.3			327.0	4.83	6.03
Plutonium-241	142.4			86.9	1.2	1.2
Plutonium-242	33.8			43.4		
Plutonium (fissile)	1,085.9			1,085.9	147.2	181.0
Total	10,450.0	8,966.0	16,631.3	9,723.7	8,944.3	16,599.9
Fission products[c]				713.1	24.13	31.4

[a]Combustion Engineering reference design.
[b]Mass flows in kilograms per 0.75 GWe-yr. Abbreviations: AB, axial blanket; RB, radial blanket.
[c]Total - 768.6.

Table 2-3. Summary of main parameters for (Pu-U)C core, UC blankets, heterogeneous core[a]

General reactor data	
Reactor power, MWt	2,740
Net electric power, MWe	1,000
Reactor vessel ΔT, °C	156
Reactor vessel outlet temperature, °C	499
Fissile feed enrichment, %	
Driver ring 1	13.9
Driver ring 2	13.9
Driver ring 3	15.5
Total fissile inventory at beginning of equilibrium cycle, kg	4,671
Total heavy metal at beginning of equilibrium cycle, kg	129,930
Number of assemblies	
Driver ring 1	36
Driver ring 2	72
Driver ring 3	132
Internal blankets	115
Control	24
Radial blanket	204
Volume fractions in active core	
Fuel	0.367
Sodium	0.445
Steel	0.188
Number of core orifice zones (drivers/ blankets)	5/7
Driver residence time, calendar year	3
Radial blanket residence time, calendar year[b]	3/6
Peak discharge exposure, MWd/kg	88
Average driver discharge exposure, MWd/kg	61
Peak neutron flux, $E > 0.1$ MeV, n/cm^2-sec	28×10^{14}
Peak fluence, $E > 0.1$ MeV, n/cm^2	19×10^{22}
Peak cladding temperature at beginning of life, °C[c]	
Nominal	581
2σ	651
Peak linear power at beginning of life, kW/m[c]	
Nominal	78.4
$3\sigma + 15\%$	99.6
Sodium void worth at end of equilibrium cycle (driver regions), $	2.74
Coupling coefficient at beginning of equilibrium cycle (K_{ii}-K_{ij}), maximum	0.869
Doppler coefficient, $\Delta K/°C$	-3.27×10^{-6}
Breeding ratio, middle of equilibrium cycle	1.55
Fissile gain, kg/cycle	353
Compound system douling time, year	12.9

Table 2-3. Summary of main parameters for (Pu-U)C core, UC blankets, heterogeneous core[a] (Continued)

Fuel-cycle cost, mills/kW-dr[d]	8.9/7.1/9.1
Maximum CDF (steady-state, driver)	0.013
Fuel assembly parameters	
Number of rods per assembly	169
Duct-wall thickness, mm	3.6
Duct outside flat to flat, mm	152
Fuel-rod pitch-to-diameter ratio, compressed	1.17
Wire diameter, mm	1.57
Assembly pitch, mm	159
Bundle ΔP, kPa	433
Maximum mixed mean outlet temperature at beginning of life, ^{o}C	551
Driver-rod parameters	
Rod outside diameter, mm	9.40
Cladding thickness, mm	0.38
Fuel height, mm	1168
Axial blanket height, mm	787
Plenum volume at beginning of life (cold), cm^3	45
Smear density, % of theoretical	81
Radial and internal blanket assembly parameters	
Number of rods per assembly	91
Duct-wall thickness, mm	2.5
Duct outside flat to flat, mm	152
Rod outside diameter	14.17
Rod pitch-to-diameter ratio, compressed	1.07
Assembly pitch, mm	159
Assembly fueled height (radial), mm	1,651
Assembly fueled height (internal), mm	1,956
Plenum volume at beginning of life (cold), cm^3	70
Peak linear rod power at end of life (radial), kW/m	38.4
Peak linear rod power at end of life (internal, kW/m	82.7

[a]Atomics International reference A design (Ref. 1).

[b]Blankets adjacent to drivers have a 3-year residence time.

[c]Peak fresh assembly at beginning of equilibrium cycle.

[d]Fuel-cycle costs: Unit fabrication costs from the Hanford Engineering Development Laboratory (HEDL) modified for carbide fuel, Combustion Engineering unit costs, and revised HEDL costs.

Table 2-4. Equilibrium-cycle reactor charge and discharge data for the LMFBR heterogeneous (Pu-U)C/UC spiked recycle[a]

Isotope	Charge[b]				Discharge[b]			
	Core	AB	IB	RB	Core	AB	IB	RB
Uranium-235	16.3	13.7	23.1	24.2	10.07	12.64	16.93	20.36
Uranium-238	8,134.3	6,775.7	11,532.9	12,049.3	7,650.0	6,705.0	11,093.6	11,820
Plutonium-238	19.29				14.14			
Plutonium-239	1,302.0				1,166.6	63.2	328.3	195.9
Plutonium-240	371.8				411.4	0.94	17.36	5.14
Plutonium-241	196.1				127.9	0.011	0.62	0.12
Plutonium-242	46.3				39.6		0.021	
Plutonium (fissile)	1,498.1				1,294.5	63.2	328.9	196.1
Total	10,084.3	6,788.6	11,556.4	12,072.9	9,430.7	6,780.0	11,455.7	12,040.7
Fission products[c]					653.6[d]	8.6[d]	100.7[d]	32.2[d]

[a]Atomics International reference A design.

[b]Mass flows in kilograms per 0.75 GWe-yr. Abbreviations: AB, axial blanket; IB, inner blanket; RB, radial blanket.

[c]Total = 795.1.

[d]Calculated from difference between charge and discharge total heavy metal.

Table 2-5. Summary core-design description for (Pu-U)C core, ThC blankets, homogeneous core[a]

General parameters	
Reactor power, MWt	2,740
Core volume, 10^3 liters	11.1
Core height, cm	91.4
Fuel residence time, year	2.2
Driver assembly	
Number of rods per assembly	169
Rod pitch-to-diameter ratio	1.17
Lattice pitch, cm	16.12
Duct-wall thickness, mm	3.81
Rod diameter, mm	9.40
Cladding thickness, mm	0.38
Bond type	Sodium
Smear density, % of theoretical	83
Performance	
Peak linear power (3σ + 15% OP), kW/m	130
Peak cladding temperature at end of life (2σ midwall), °C	--
Fissile inventory at beginning of equilibrium cycle, kg	
Uranium fissile	294
Plutonium fissile	2,949
Total	3,243
Fissile production/destruction, kg/year	
Uranium fissile	267
Plutonium fissile	43
Total	310
Fuel-cycle cost, mills/kW-hr	9.6
Symbiotic system doubling time, yr	14
Sodium void worth at end of equilibrium cycle, $	4.63

[a]Combustion Engineering low-cost coprocessing design (Ref. 1).

Table 2-6. Equilibrium-cycle reactor charge and discharge data for the LMFBR homogeneous (Pu-U)C/Th spiked recycle[a]

Isotope	Charge[b]			Discharge[b]		
	Core	AB	RB	Core	AB	RB
Thorium-232		7,662.0	9,633.2		7,470.0	9,512.4
Protactinium-233			2.97			1.98
Uranium-233			24.7		175.2	135.6
Uranium-234			<1		2.97	1.98
Uranium-238	8,504.6			7,664.0		
Plutonium-239	893.0			948.5		
Plutonium-240	347.5			384.1		
Plutonium-241	67.3			58.4		
Plutonium-242	26.7			31.7		
Plutonium (fissile)	960.3			1,006.9		
Total	9,839.1	7,662.0	9,660.9	9,086.7	7,648.2	9,652.0
Fission products[c]				741.5	15.84	10.89

[a]Combustion Engineering low-cost coprocessing core design.
[b]Mass flows in kilograms per 0.75 GWe-yr. Abbreviations: AB, axial blanket; RB, radial blanket.
[c]Total = 768.2.

Table 2-7. Summary of main parameters for (Pu-U)C core, ThC blankets, heterogeneous core[a]

General reactor data	
Reactor power, MWt	2,740
Net electric power, MWe	1,000
Reactor vessel ΔT, °C	156
Reactor vessel outlet temperature, °C	499
Fissile feed enrichment, %	
Driver ring 1	14.6
Driver ring 2	14.6
Driver ring 3	16.3
Total fissile inventory at beginning of equilibrium cycle, kg	4,780
Total heavy metal at beginning of equilibrium cycle, kg	108,090
Number of assemblies	
Driver ring 1	36
Driver ring 2	72
Driver ring 3	132
Internal blankets	115
Control	24
Radial blanket	204
Volume fractions in active core	
Fuel	0.367
Sodium	0.445
Steel	0.188
Number of core orifice zones (drivers/blankets)	4/7
Driver residence time, calendar year	3
Radial blanket residence time, calendar year[b]	3/6
Peak discharge exposure, MWd/kg	94
Average driver discharge exposure, MWd/kg	64
Peak neutron flux, E >0.1 MeV, n/cm^2-sec	29×10^{14}
Peak fluence, E >0.1 MeV, n/cm^2	19×10^{22}
Peak cladding temperature at beginning of life, °C[c]	
Nominal	580
2σ	651
Peak linear power at beginning of life, kW/m[c]	
Nominal	85.6
3σ +15%	108.8
Sodium void worth at end of equilibrium cycle (driver regions), $	1.99
Coupling coefficient at beginning of equilibrium cycle (K_{ii}-K_{ij}), maximum	0.852
Doppler coefficient, ΔK/°C	-3.40×10^{-6}
Breeding ratio, middle of equilibrium cycle	1.42
Fissile gain, kg/cycle	263
Compound system doubling time, year	18.3

Table 2-7. Summary of main parameters for (Pu-U)C core, UC blankets, heterogeneous core[a] (continued)

Fuel-cycle cost, mills/kW-hr[d]	11.4/9.6/11.7
Maximum CDF (steady-state, driver)	0.012
Fuel assembly parameters	
Number of rods per assembly	169
Duct-wall thickness, mm	3.6
Duct outside flat to flat, mm	152
Fuel-rod pitch-to-diameter ratio, compressed	1.17
Wire diameter, mm	1.57
Assembly pitch, mm	159
Bundle ΔP, kPa	506
Maximum mixed mean outlet temperature at beginning of life, ^{o}C	546
Driver-rod parameters	
Rod outside diameter, mm	9.40
Cladding thickness, mm	0.38
Fuel height, mm	1168
Axial blanket height, mm	787
Plenum volume at beginning of life (cold), cm^3	45
Smear density, % of theoretical	81
Radial and internal blanket assembly parameters	
Number of rods per assembly	91
Duct-wall thickness, mm	2.5
Duct outside flat to flat, mm	152
Rod outside diameter, mm	14.17
Rod pitch-to-diameter ratio, compressed	1.07
Assembly pitch, mm	159
Assembly fueled height (radial), mm	1,651
Assembly fueled height (internal), mm	1,956
Plenum volume at beginning of life (cold), cm^3	70
Peak linear rod power (radial) at end of life, kW/m	25.8
Peak linear rod power (internal) at the end of life, kW/m	72.7

[a]Atomics International reference B design (Ref. 1).

[b]Blankets adjacent to drivers have a 3-year residence time.

[c]Peak fresh assembly at beginning of equilibrium cycle.

[d]Fuel-cycle costs: Unit fabrication costs from the Hanford Engineering Development Laboratory (HEDL) modified for carbide fuel, Combustion Engineering unit costs, and revised HEDL costs.

Table 2-8. Equilibrium-cycle reactor charge and discharge data for the LMFBR heterogeneous (Pu-U)C/Th spiked recycle[a]

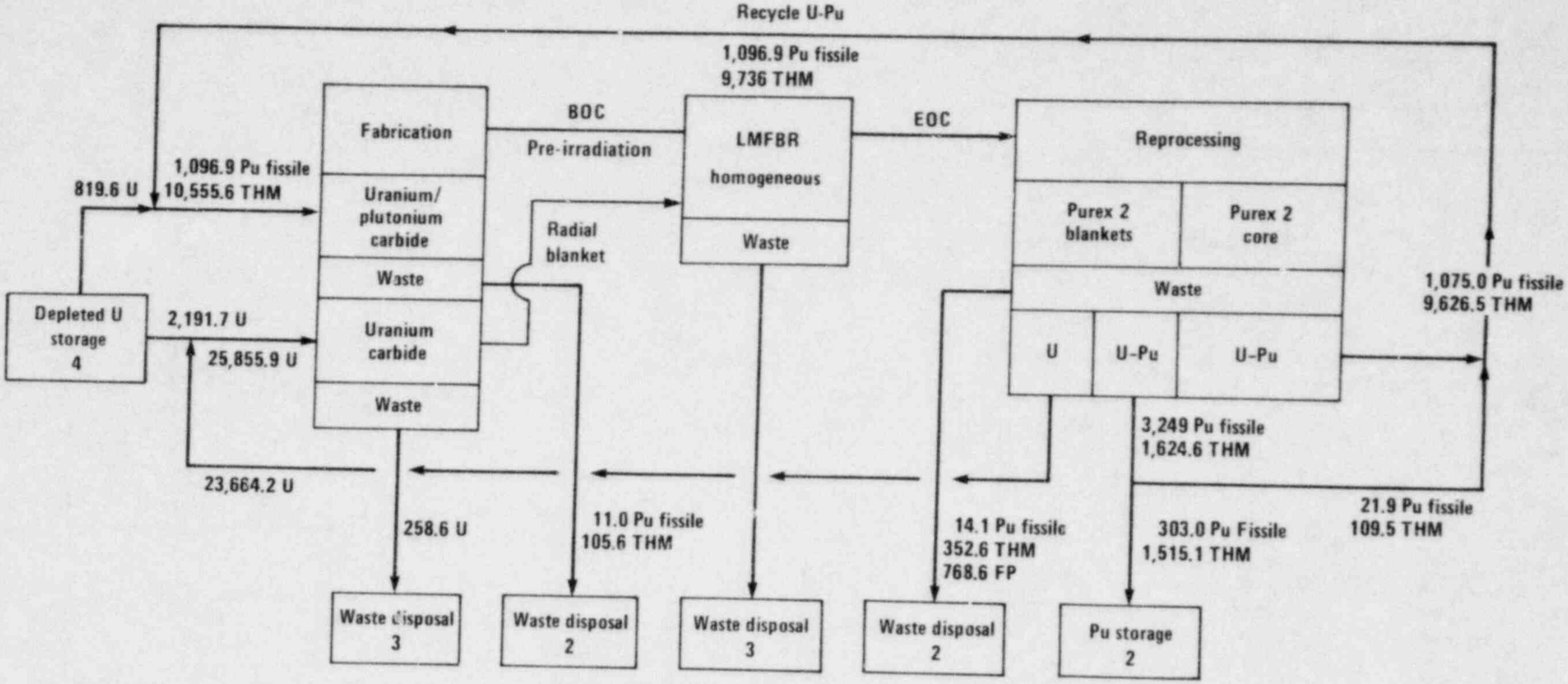
Isotope	Charge[b] Core	AB	IB	RB	Discharge[b] Core	AB	IB	RB
Thorium-232		5,342.1	9,083.6	9,490.7		5,280	8,710.7	9,285
Protactinium-233						(4.07)	(23.4)	(10.5)
Uranium-232						55.07+4.07 =59.1	266.1+23.4 =289.5	170.6+(10.5) =181.1
Uranium-234						0.26	9.34	2.55
Uranium-235	16.07				9.86	0.004	0.39	0.056
Uranium-238	8,037.9				7,553.6			
Plutonium-238	20.4				15.0			
Plutonium-239	1,367.6				1,200			
Plutonium-240	390.4				351.9			
Plutonium-241	205.9				132.9			
Plutonium-242	48.6				41.8			
Plutonium (fissile)	1,573.5				1,332.9			
Uranium (fissile)						59.1	289.9	181.1
Total	10,086.4	5,342.1	9,083.6	9,490.7	9,394.3	5,337.9	9,008.6	9,469.3
Fission products[c]					692.1[d]	4.2[d]	75[d]	21.4[d]

[a]Atomics International reference B design.

[b]Mass flows in kilograms per 0.75 GWe-yr. Abbreviations: AB, axial blanket; IB, inner blanket, RB, radial blanket.

[c]Total = 792.7.

[d]Fission products calculated from difference between charge and discharge total heavy metal.

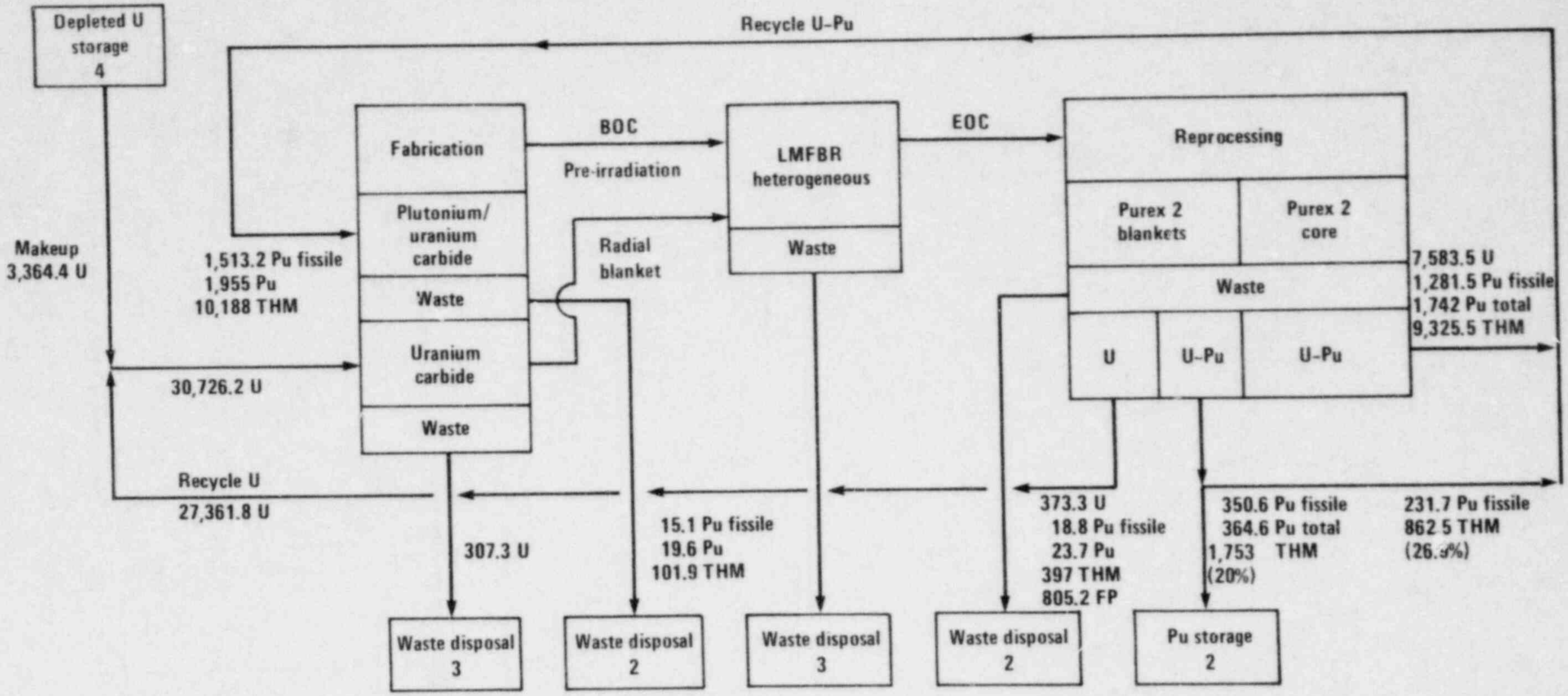


	BOC		EOC	
	Core	Blankets	Core	Blankets
U-235	18.1	50.7	8.45	44.7
U-Total	9,046.8	25,597.3	8,259	25,205.1
Pu fissile	1,085.9	–	1,085.9	328.2
Pu total	1,403.3	–	1,464.8	339
THM	10,450	25,597.3	9,723.7	25,544.2
FP	–	–	713.1	55.5

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Data base: Table 2-2 of this addendum, data from ANL (March 6, 1979).
3. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.

Figure 2-1. Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, homogeneous core (uranium carbide blankets, spiked recycle).

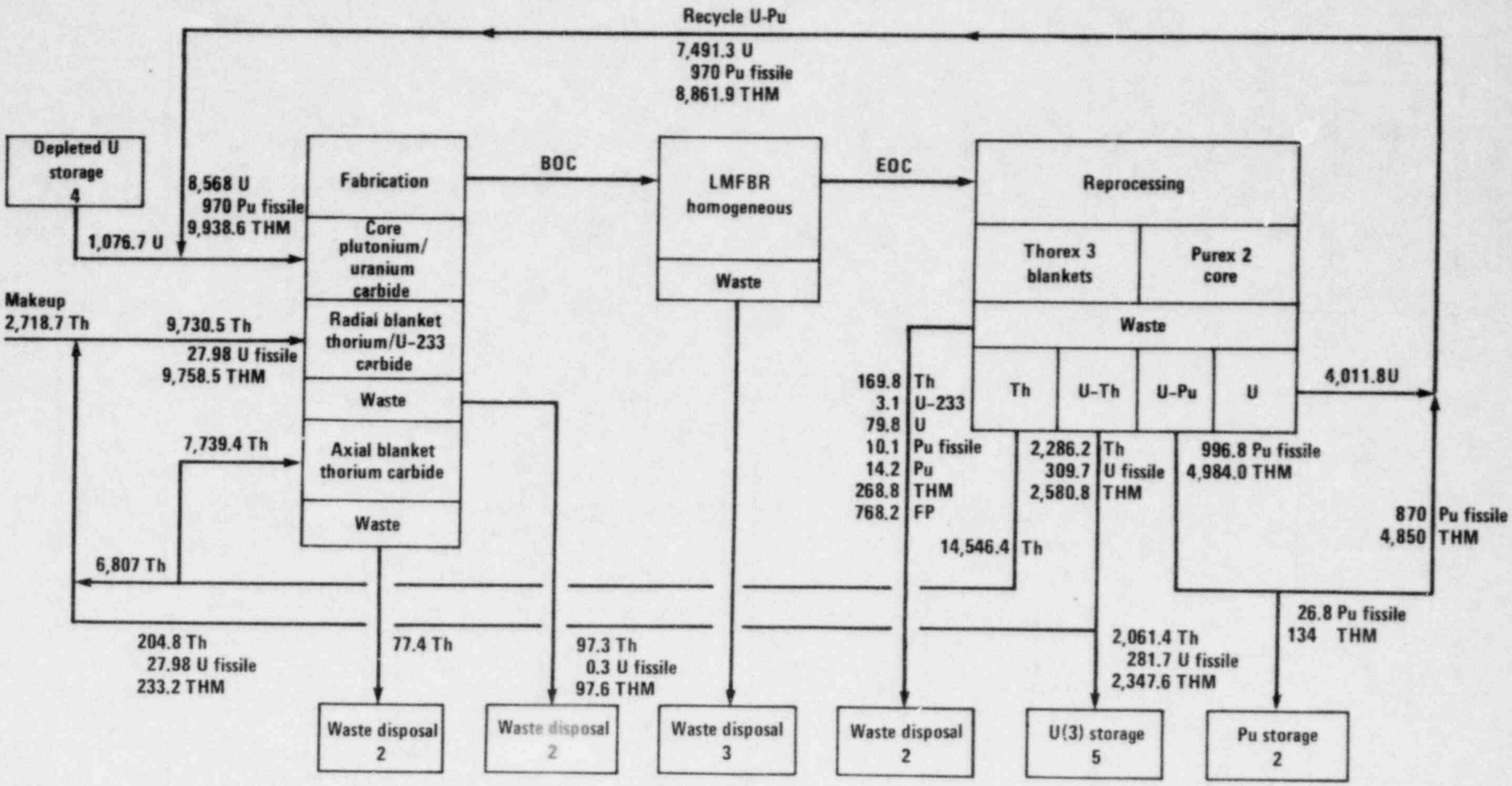


	BOC		EOC	
	Core	Blankets	Core	Blankets
U-235	16.3	61.0	10.1	49.9
U total	8,150.6	30,418.9	7,660.1	29,668.5
Pu fissile	1,498.1	–	1,294.5	588.2
Pu total	1,935.5	–	1,759.6	611.6
THM	10,086.1	30,418.9	9,419.7	30,280.1
FP	–	–	666.4	198.8

Notes:

1. Mass flow: in kg per 0.75 GWe-yr.
2. Data base: Table 3-2 (corrected) of this addendum, data from ANL (March 6, 1979).
3. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.

Figure 2-2. Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, heterogeneous cycle (uranium carbide blankets, spiked recycle).

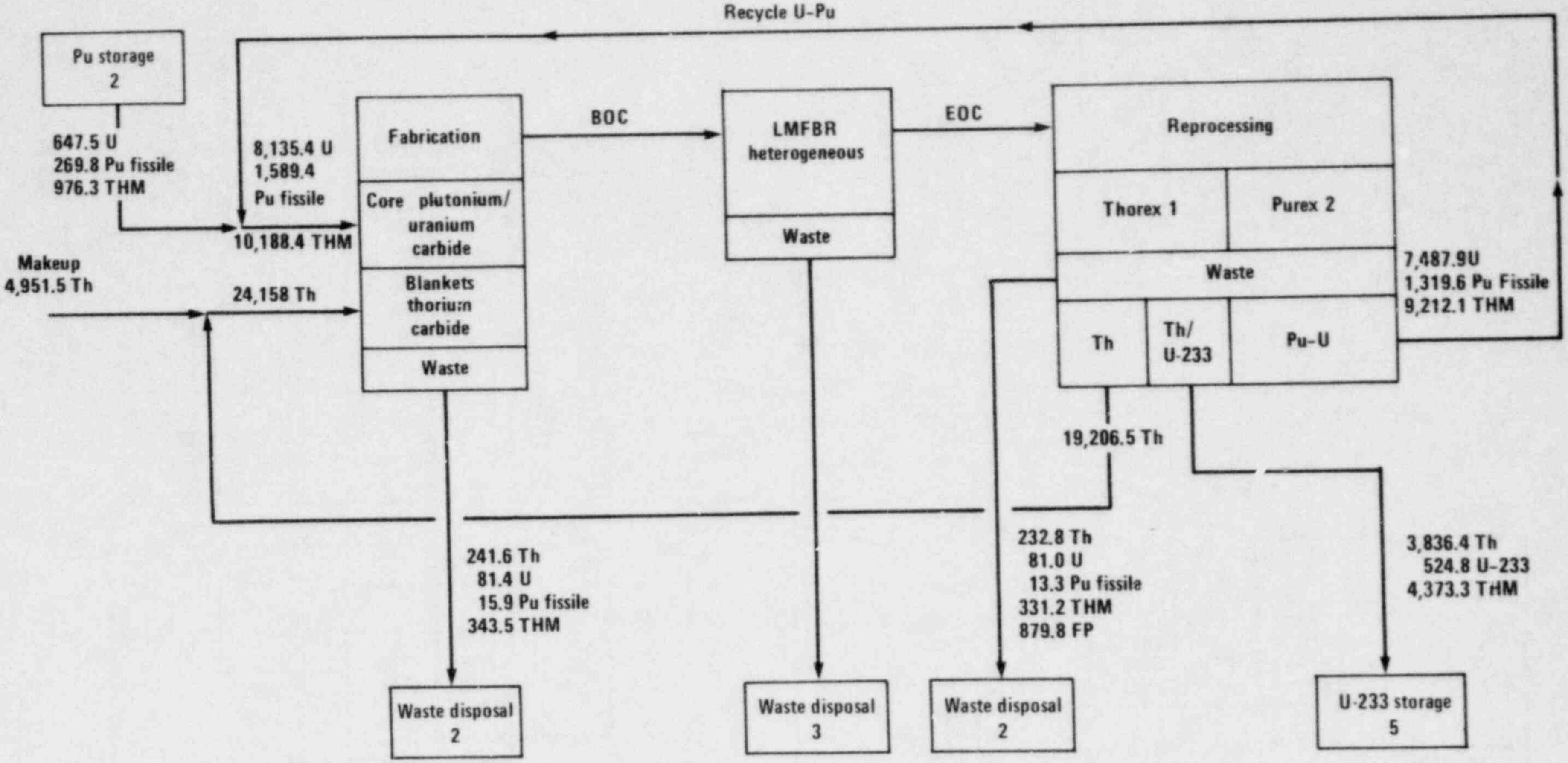


Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Data base: Table 4-2 of this addendum, data from ANL (March 6, 1979).
3. Abbreviations: AB, axial blankets; RB, radial blankets; BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.
4. All assemblies out of fabrication are highly radioactive, thus the term "spiked recycle."

	BOC			EOC		
	Core	AB	RB	Core	AB	RB
Th		7,662	9,633.2		7,470	9,512.4
U-233			27.7		175.2	137.6
U	8,504.6		27.7	7,664	178.2	139.6
Pu fissile	960.3			1,006.9		–
Pu total	1,334.5			1,422.7		–
THM	9,839.1	7,662	9,660.9	9,086.7	7,648.2	9,652
FP				741.5	15.8	10.9

Figure 2-3. Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, homogeneous core, thorium carbide aixial blanket, uranium radial blanket (spiked recycle).



Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Data base: Table 5-2 of this addendum, data from ANL (March 6, 1979).
3. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.
4. All assemblies out of fabrication are highly radioactive, thus the term "spiked recycle."

	BOC		EOC	
	Core	Blankets	Core	Blankets
Th	–	23,916.4		23,275.7
U fissile	16.1	–	9.9	530.1
U	8,054	–	7,563.5	542.3
Pu fissile	1,573.5	–	1,332.9	–
Pu total	2,032.5	–	1,741.6	–
THM	10,086.5	23,916.4	9,305.1	23,818
FP		–	781.4	98.4

Figure 2-4. Mass-flow diagram for an LMFBR fueled with uranium/plutonium carbide, heterogeneous core, thorium carbide blankets (spiked recycle).

3.0 SAFETY CONSIDERATIONS FOR THE CARBIDE-FUELED LMFBR

3.1 INTRODUCTION

Chapter 9 of Reference 3 presents a review of the relevant safety and licensing considerations for the LMFBR. This assessment focuses on the oxide-fueled LMFBR to the extent that a specific fuel type is considered. This section reviews only those safety considerations for which there are significant differences between carbide-fueled systems and oxide-fueled systems.

3.2 KEY SAFETY ISSUES FOR THE CARBIDE-FUELED LMFBR

Section 9.2 of this volume lists seven safety issues that have been identified by the NRC as having to be resolved favorably before an LMFBR can be licensed. The first four of these are design-related issues and are as follows:

1. The scram systems must be shown to have sufficient redundancy and diversity to make the likelihood of failure of the system very small.
2. Sufficient redundancy and diversity must be provided in the heat-transport system design to make the probability of its not being able to remove heat under shutdown conditions very small.
3. Reliable means to detect and cope with fuel-rod failure and subassembly faults must be provided.
4. The continuing high integrity of the heat-transport system must be insured.

Favorable resolution of the above issues would insure that the possibility of core melt and disruptive accidents is minimized. The three remaining issues identified by the NRC are associated with minimizing the probability of containment failure in the event of a severe accident. They are as follows:

5. The containment must be able to accommodate the consequences of spillage of large quantities of sodium from the primary or intermediate coolant system.
6. The containment/confinement system must be capable of adequately mitigating the radioactivity releases that could result from all events within the containment design basis.
7. The containment system should also be so designed that it could maintain its integrity for at least 24 hours in the unlikely occurrence of a broad range of events involving the energetic disassembly of the core and the production of a vaporized fuel and other possible consequences resulting from core-melt accidents.

3.3 STATUS OF KEY ISSUES FOR CARBIDE-FUELED LMFBR

The status of the four design-related issues listed first above is insensitive to whether the system of concern is carbide-fueled or oxide-fueled. Thus, the status reported in Section 9.3 of this volume is fully applicable to the carbide-fueled system of concern here. Likewise, issues 5 and 6 above are not sensitive to fuel type to any significant extent. Only issue 7 is somewhat sensitive to fuel type and is discussed further here.

3.3.1 ACCOMMODATION OF CORE-MELT AND DISRUPTIVE ACCIDENTS IN A CARBIDE-FUELED LMFBR

The basic requirement stated by the NRC staff is that there should be an extremely low likelihood that potential core-melt and core-disruptive accidents could result in early containment system failure. As discussed in Section 9.3.7 of this volume, early containment failure could possibly result from two generic phenomena, excessive fission-energy release during the accident transient (energetics) and failure to adequately cool and accommodate the molten core debris resulting from the transient (post-accident heat removal). Furthermore, the energetics issue can be broken into three areas of concern:

1. The energetics potential arising from the presence of the positive sodium-void worth associated with medium-to-large-sized LMFBRs
2. The energetics potential associated with the potential for vapor explosions occurring when molten fuel and coolant interact
3. The energetics potential associated with the possible occurrence of recriticality events in the disrupting cores

For smaller LMFBRs, the sodium voiding-related energetics issue is not relevant. For reactors the size of the Clinch River Breeder Reactor Plant (CRBRP), the positive sodium-void worth results in the system being in a supercritical state and at 5 to 20 times nominal power when fuel rods begin to disrupt in the unprotected loss-of-flow accident (LOCA). The subsequent LOCA scenario is quite sensitive to the initial motion of this disrupting fuel. There appears to be a strong potential for this early motion to be dispersive in an oxide-fueled system, thus dispelling the energetics concern. Relative to this early-fuel-dispersal potential, there does not seem to be a significant difference between oxide and carbide fuels. Carbide fuel has a lower melting point (about 2,400 vs. 2,800°C) but operates at lower temperatures, so that the two fuels would become molten and mobile at about the same time in an accident, all other factors being equal. In addition, carbide fuel retains more fission gas, which increases its fission-gas-induced dispersive potential in comparison with oxide fuel. In either case, however, the requirements for early fuel dispersal become quite stringent for LMFBRs with \$5 to \$6 of voiding reactivity. This has caused attention to be focused on core-design concepts that would have void worths in the \$2.50 to \$3.50 range.

For the oxide-fueled systems, the earlier concern about vapor explosions from fuel-coolant interactions (FCIs) that might offer an energetics threat appears to be unwarranted. Research carried out over the past 6 to 7 years shows these energetic events to be unlikely, on the basis that energetic fuel-coolant interactions can be ruled out because the interface contact temperature is well below the spontaneous nucleation limit for sodium. Thus the NRC staff, in its evaluation of CRBRP safety issues, gave them small concern. Again, for the oxide-fuel system, the energetics potential associated with recriticalities after the initial core disruption has been shown to be small. The arguments used to reduce concern about these events are based on the dispersive effect provided by steel vaporization, which precludes energetic recriticalities. Additional confirmatory work is being done on both the FCI and recriticality issues, to provide further support for the arguments that have been advanced, to largely preclude them.

For the carbide fuel, with its higher thermal conductivity, it is not possible to rule out more energetic fuel-coolant interactions on the fundamental physical principles mentioned above. This is illustrated in Figure 3-1, taken from Reference 3. The basic concept is that vapor explosions are not possible if the interface temperature between the hot and cold liquids is below the homogeneous-nucleation temperature of the cold liquid. Although this criterio is generally satisfied in the oxide system, Figure 3-1 shows that there is a large region of fuel and coolant temperature combinations for the carbide fuel systems in a core-disruptive accident where interface temperatures are above the homogeneous-nucleation temperature.

The increased potential for an energetic fuel-coolant interaction in the uranium-plutonium carbide system, as it affects the problem of the positive sodium-void coefficient, has been illustrated by some recent tests at Sandia Laboratories. A test series with single rods of 15.24-cm length was carried out under NRC sponsorship in the annular core power reactor (ACPR) subjected to prompt-burst transients. The oxide-fuel-rod tests confirmed the benign results obtained with the earlier TREAT tests (Ref. 4). However, limited testing with carbide fuel rods subjected to essentially the same conditions as the oxide fuel rods in the ACPR resulted in relatively energetic fuel-coolant interactions (Ref. 5). It is of interest to note that the efficiency obtained in the carbide tests was considerably lower than the maximum thermodynamic value (Ref. 6). In this regard, it is important to recognize certain inherent features of the current LMFBR design, including the following:

1. The boiling point of liquid sodium is well below the fuel-cladding melting temperature.
2. The time constant for the cladding is much larger than the period associated with the nuclear transient of interest (prompt burst).

Molten fuel and liquid sodium therefore cannot both be present in the core without being largely separated by solid cladding (Ref. 7). This condition could restrict considerably the potential for mixing, although the condition for film boiling, a necessary requirement for intermixing, is satisfied for the carbide system. The presence of the solid cladding appears to reduce the vapor-explosion potential somewhat. In any case, a substantial experimental program would be required to establish the real vapor-explosion potential for carbide-fueled LMFBRs.

Carbide fuels also warrant more concern than do oxide fuels in the recriticality area. A molten mass of carbide fuel and stainless steel is inherently less dispersive because the melting point of the carbide is some 400°C below that of the steel, so that the fuel would have to be heated more before production of steel vapor could become a dispersive force. In the meanwhile, a compaction of this molten mass would be possible. A more significant recriticality threat in the carbide system is that of a pressure-driven compaction. If a high-thermal-conductivity carbide core undergoes a mild disassembly, then conditions favorable for a pressure-driven recompaction induced by a fuel-coolant interaction--on a scale much smaller than the whole core--are present and must be considered along with potential barriers to this possibility.

In summary, the question of accident energetics potential is not as easily resolved for the carbide system as it is for the oxide system, because there are at least two areas in which fundamental physical principles do not offer strong support to the carbide case and also because much less experimental work has been done on carbide fuels.

The question of post-accident heat removal is not substantially different for the two fuel types. Although substantial research would have to be carried out for carbide fuel to confirm this, it is believed that the technology for meeting the NRC criterion of 24 hours with no containment failure is available.

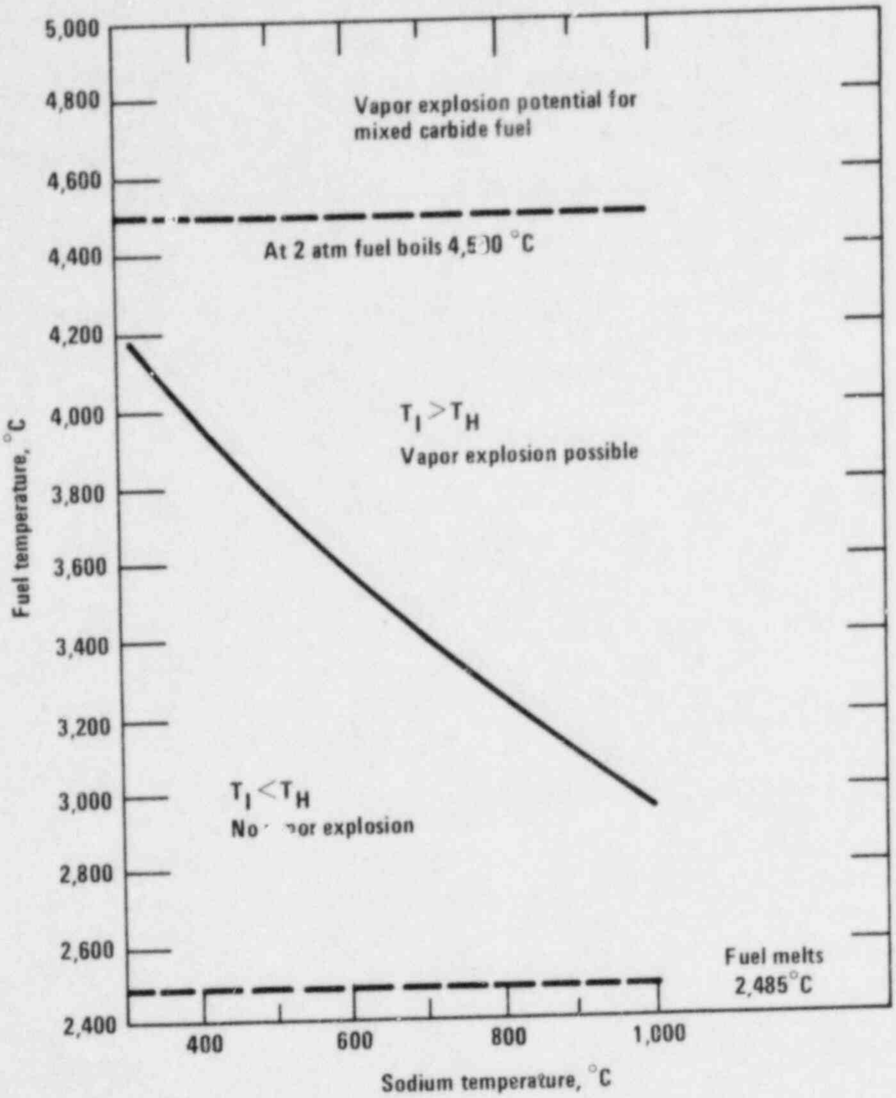


Figure 3-1. Necessary temperatures for carbide fuels to cause spontaneous nucleation on contact with liquid sodium.

4.0 ENVIRONMENTAL CONSIDERATIONS

Chapter 10 of this volume discusses the environmental impacts associated with the normal operation of the liquid-metal fast-breeder reactor (LMFBR). All the reactor fuel-cycle combinations considered in Chapter 10 are based on oxide fuels; the reactor/fuel-cycle combinations discussed in this addendum are based on carbide fuels.

The use of carbide fuels in the LMFBR should not introduce any significant environmental differences. The nonradiological impacts--such as those associated with heat dissipation, water use, land use, and chemical and biocidal effluents--would be the same because, by definition, the plant designs are the same.

The radiological impacts would also be similar if not identical. The design of radioactive process systems are the same, and the only difference may be in the level of activity present in the primary coolant or the cover-gas system because oxide and carbide fuels have different fission-product retentions.

This difference is not, however, expected to be significant in terms of effluent release from the plant under normal operating conditions. The discussion of environmental considerations in Chapter 10 of this volume is therefore applicable to the LMFBR concepts discussed in this addendum.

5.0 LICENSING STATUS AND CONSIDERATIONS

General aspects of the licensing status of liquid-metal fast-breeder reactors (LMFBRs) are discussed in Chapter 11 of this volume.

Of significant importance for licensing is demonstrated technology; this is presently concentrated on the mixed oxides of uranium and plutonium, including most of the safety experiments and analysis models. The licensing application of an LMFBR with a carbide-fueled core and would have to be supported by a safety analysis report covering carbide-fuel behavior under normal and accident conditions.

6.0 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

A general discussion of liquid-metal fast-breeder reactor (LMFBR) research, development, and demonstration is presented in Chapter 12 of this volume.

The test facilities being planned for oxide fuel (Refs. 3 and 8) should also be adequate for carbide fuel. Considerably more detailed testing may be required for the carbide fuel, however, in view of relatively unfavorable safety characteristics as compared with the oxide fuel. A summary of desirable experiments in terms of key issues and facilities is given in Table 6-1.

Table 6-1. Experiments with carbide fuel in terms of key issues and facilities

Issue	CAMEL	EBR-II	TREAT	TREAT-Upgrade	SLSF
Fuel-failure propagation		Run Beyond Clad Breach tests including operational transients to demonstrate ability to operate with failed fuel.			Tests to study the potential for blockage propagation as a result of limited fuel release. Measurements include rate of propagation and potential for detection.
Limited core damage	Tests to study the potential for fuel sweepout. Includes characterization of fuel-coolant interactions, freezing, and plugging.		Small-bundle tests ($\leq$7 rods) to simulate transient-overpower conditions. Data include fuel failure (time and location), internal fuel motion and the potential for fuel-coolant interactions. Burnup, and ramp rate are the principal variables.	Transient-overpower tests with large bundles to assess the possibility of early shutdown under prototypical hydraulic conditions. Tests should include incoherence effects and their potential mitigating effect on voiding and fuel plugging induced by fuel-coolant interactions.	Transient-overpower tests with small ramp rates (<10¢/sec) desirable. Loss-of-fluid tests simulating conditions typical of heterogeneous core designs to study the potential for early fuel dispersal by fission-gas release.
Energetics Sodium void			Small-bundle tests ($\leq$7 rods to simulate a loss-of-fluid-driven transient overpower. Drive TREAT as hard as possible to explore potential early autocatalytic effects. Shorter period tests are desirable.	Larger bundle tests desirable to explore incoherence effects.	
Recriticality			Small-sample tests to explore fuel-dispersal possibilities at decay-heat power levels. Special emphasis needs to be given to the large potential for pressure-driven recompaction.	Large-bundle tests (37-61 rods) to explore the potential for monotonic fuel dispersal by high fission-gas retention at nominal power level. Large-sample tests to further explore fuel behavior at decay-heat power levels.	

Table 6-1. Experiments with carbide fuel in terms of key issues and facilities (Continued)

Issue	CAMEL	EBR-II	TREAT	TREAT-Upgrade	SLSF
Fuel-coolant interactions			See energetics--sodium void. Subsantial additional testing may be required in view of the apparent high potential for fuel-coolant interactions. The role of cladding as a mitigating effect needs clarification.	See energetics--sodium void. Shorter period tests desirable.	
Loss of heat sink					Test(s) to study long-term behavior of fuel rods expecting sodium boiling at decay-heat power levels to demonstrate the absence of significant fuel disruption under these conditions.

REFERENCES

1. Rockwell International, Atomics International Division, Summary Report Preconceptual Study of 1000-MWe Carbide-fueled LMFBR Designs, ESG-DOE-13244, September 30, 1978.

2. Combustion Engineering, Inc., Homogeneous Carbide Fueled Cores for the Proliferation Resistant LMFBR Core Design Study, CE-FBR-78-380, COO-2426-149, August 24, 1978.

3. H. K. Fauske, INFCE/5 Fast Breeder Reactor Studies: Safety Implications of Alternative Fuel Types, ANL-INFCE/5-TM-5, Argonne National Laboratory, July 1978.

4. C. E. Dickerman et al., "Status and Summary of TREAT In-pile Experiments on LMFBR Response to Hypothetical Core-Disruptive Accidents," Symposium on the Thermal and Hydraulic Aspects of Nuclear Reactor Safety, Vol. 2, Liquid Metal Fast Breeder Reactors, O. C. Jones, Jr., and S. G. Bankoff, eds., American Society of Mechanical Engineers, 1977.

5. K. O. Reil, M. F. Young, and H. Plitz, "Prompt Burst Energetics Experiments, Uranium Carbide Series," Trans. Am. Nucl. Soc., 28, 487, 1978.

6. E. P. Hicks and D. C. Menzies, "Theoretical Studies on the Fast Reactor Maximum Accident," Safety, Fuels, and Core Design in Large Fast Power Reactors, ANL-7120, Argonne National Laboratory, 1965.

7. D. R. Ferguson et al., "The SAS4A LMFBR Accident Analysis Code System: A Progress Report," Fast Reactor Safety and Related Physics, CONF-761001.

8. R. Avery et al., "The SAREF Program," Fast Reactor Safety and Related Physics, CONF-76001.

APPENDIX A

U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

BACKGROUND

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for the physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage, and theft. The U.S. Nuclear Regulatory Commission (NRC) has considered whether strengthened physical protection may be required as a matter of prudence (Ref. 1). Proposed upgraded regulatory requirements to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgraded rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM SAFEGUARDS BASIS

The desired basis for the NRC review of safeguards systems for the Nonproliferation Alternative Systems Assessment Program (NASAP) alternative fuel-cycle materials containing significant quantities of strategic special nuclear material (SSNM),[a] greater than 5 formula kilograms,[b] during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I[c] material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, proposals have been made to increase radioactivity of strategic special nuclear materials (SSNMs) that are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh-fuel material to require that, during the period after export from the United States and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover low-radioactivity SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time required in obtaining material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements imposed by the Nuclear Non-Proliferation Act of 1978 include application of International Atomic Energy Authority (IAEA) material accountability

---

[a] $\geq$20% U-235 in uranium, $\geq$12% U-233 in uranium, or plutonium.

[b] Formula grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium); Ref. 10 CFR 73.30.

[c] IAEA definitions of highly enriched uranium (>20%).

requirements to nuclear-related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by the IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative that could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC REVIEW

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches (such as upgraded physical protection, improved material control and accountancy, dilution of SSNM, decreased transportation requirements, few sites handling SSNM, and increased material-handling requirements as applied to each fuel material type) should be assessed. The evaluation should consider, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb-making purposes; the relative impacts on domestic and on international safeguards, the impact of radioactive contaminants on detection for material control and accountability, measurement, and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tampering or breaching; the increased public exposure to health and safety risk from acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, the NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation, we request that the NRC assess the differences in the licensing requirements for the domestic facilities, transportation systems to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel-cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, and on the international and national safeguards systems of typical importers for protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, as well as the potential advantages in detection or deterrence should be described in detail. The potential role, if any, that added radioactivity could or should play should be clearly identified, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its acquisition by foreign countries for weapons purposes. Licensability issues that must be addressed by research, development, and demonstration programs also should be identified.

Table A-1 presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import, and export:

Table A-1. Minimum radiation levels for various fuel material types

Fuel Material Type	Minimum radiation level during 2-year period, rem/hr at 1 meter (Ref. 6)	
	Mixed[a]	Mechanically attached[b]
PuO_2, $HEUO_2$ powder or pellets[c]	1,000/kgHM	10,000/kgHM
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[c]	100/kgHM	10,000/kgHM
LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[a]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[b]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[c]HEU is defined as containing 20% or more U-235 in uranium, 12% or more of U-233 in uranium, or mixtures of U-235 and U-233 in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are indicated in the following table and references.

Table A-2. Candidate methods and radiation levels for spiking fuel materials

Fuel material type	Minimum 2-year radiation level, (rem/hr at 1 m)	Process	Minimum initial radiation level, (rem/hr at 1 m)	References
PuO_2, $HEUO_2$ powder or pellets	1,000/kgHM	Co-60 addition	1,300/kgHM	2, 3, 5, 6
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	Co-60 addition	130/kgHM	2, 3, 5, 6
		Fission product addition (Ru-106)	400/kgHM	2, 3, 5, 6
LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4
LMFBR or GCFR fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4

REFERENCES FOR APPENDIX A

1. Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat, NUREG-0414, U.S. Nuclear Regulatory Commission, May 1978.

2. J. E. Selle, Chemical and Physical Considerations of the Use of Nuclear Fuel Spikants for Deterrence, ORNL TM-6412, Oak Ridge National Laboratory, October 1978.

3. J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen, Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence, ORNL TM-6483, Oak Ridge National Laboratory, October 1978.

4. G. F. Pflasterer and N. A. Deane, Pre-Irradiation Concept Description and Cost Assessment, GEFR 00402.

5. Modification of Strategic Special Nuclear Materials to Deter Their Theft or Unauthorized Use, IRT-378-R, (The Spiking of Special Nuclear Materials as a Safeguards Measure, Vol. 2, BNL File No. 5.9.1 for Vol. 1).

6. E. A. Straker, Material Radiation Criteria and Nonproliferation, SAI-01379-50765, Science Application, Inc., January 8, 1979.

APPENDIX B

Responses to Comments by the U.S. Nuclear Regulatory Commission
PSEID, Volume VI, Liquid-Metal Fast-Breeder Reactors

Preface

This appendix contains comments and responses resulting from the U.S. Nuclear Regulatory Commission (NRC) review of the preliminary safety and environmental submittal of August 1978. It should be noted that the NRC comments are the result of reviews by individual staff members and do not necessarily reflect the position of the Commission as a whole.

RESPONSES TO GENERAL COMMENTS

1. Regarding the NRC's request to reduce the number of reactor concepts and fuel cycle variations, the NASAP set out to look at a wide variety of reactor concepts and fuel cycles with potential nonproliferation advantages. These various concepts have differing performance characteristics in other important respects, such as economics, resource efficiency, commercial potential, and safety and environmental features. The relative importance of these other characteristics and trade-offs have been determined and are incorporated in the NASAP final report.

2. Regarding the comment on the need to address safeguards concepts and issues, some concepts for providing protection by increasing the level of radioactivity for weapons-usable materials have been described in Appendix A to each PSEID. Appendix A has been revised to reflect NRC comments.

   An overall assessment of nonproliferation issues and alternatives for increasing proliferation resistance is provided in Volume II of the NASAP final report and referenced classified contractor reports.

RESPONSES TO SPECIFIC COMMENTS

Question 1

Since the 15 variations submitted as part of the NASAP PSEID package are variations on core design and fuel cycles only, it will be difficult, if not impossible, to perform a comparative evaluation of "integral" reactor systems (i.e., nuclear steam supply and balance-of-plant systems). We believe that it will be unfair to the liquid-metal fast-breeder reactor (LMFBR) assessment for the staff to assume and use an extrapolation of the Clinch River Breeder Reactor (CRBR) design for making these judgments. The CRBR, being a loop design of the early 1970s, does not reflect recent design innovations or improvements. Also a number of key safety issues associated with the Clinch River Breeder Reactor Plant (CRBRP) remained outstanding at the time of the suspension of the safety review (Spring 1977).

It is important for the U.S. Department of Energy (DOE) to recognize that any one of these LMFBR conceptual designs must be consistent with and conform to the spirit and intent of the staff licensing positions as reflected in the regulatory guides, critiera in the Standard Review Plan, the General Design Criteria, and other licensing regulations. Some of the key areas that must be addressed include fuel-system design, in-service inspection, control system diversity and independence, decay-heat-removal-system diversity and independence, and finally, containment system design. Due to the importance of containment-system design, we have included a recapitulation of recent licensing staff positions on containment design in a separate enclosure for your information.[a] Before we can proceed with the LMFBR portion of our NASAP review, we need to know your basic safety approach; to what extent this approach conforms with accepted practice; how and when you will decide on specific design concepts (e.g., loop vs. pot); and the level and direction of the research and development (R&D) effort for reactor safety. It is important that the DOE be reminded that, in the past, the staff and the DOE differed in basic safety approach and implementation for both the CRBR and the Fast Flux Test Facility (FFTF) reactors. These differences have been documented in great detail for the CRBR and the FFTF in correspondence between the staff and projects, and in the CRBR site-suitability report and final environmental statement (FES) and in the FFTF safety evaluation report (SER). It is imperative that the DOE recognize and understand these differences and factor them into their overall planning, in particular in their formulation of and commitment to a much-needed safety R&D program. Anything short of this could have serious implications for licensing.

Question 2

It appears that the only substantive reference regarding LMFBR core-disruptive accidents (CDAs) for these alternative fuel types and core designs is the report by

---

[a]The NRC position on containment design is incorporated in this Appendix after the response to Questions 1 and 2.

H. K. Fauske, Safety Implications of Alternative Fuel Types, INFCE/5-TM-5. Several general comments and questions are in order:

a. The NRC Office of Nuclear Reactor Regulation (NRR) does not necessarily agree with some of the basic conclusions, methodology, and basis for design features presented in this report, among which are the following:

   (1) The methodology of using the "first principles," listed, for example, on page 26, to draw global conclusions on the relative merits of oxide vs. carbide or metal fuels.

   (2) The conclusion that metal fuels are inherently safer fuels than carbides, drawn from application of these first principles.

   (3) The conclusion drawn (page 29) that, for a loss-of-heat-sink accident, fuel melting is initiated only if the coolant level drops below the core.

   (4) The conclusion that sodium bonding of metal or carbide fuel has only safety advantages in CDA sequences.

   (5) Based on some of the above conclusions, the author recommends certain design options, such as the removal of the upper structure removed from lead subassemblies (S/As); perforation of S/A ducts; and sodium-bonding for carbide and metal fuels.

b. This report has an outline for "experimental resolution of key issues" for all three fuel types. To what extent will the DOE rely on the definition of problems and resolution approaches as outlined in this report? (It is important for the DOE to recognize that the technical judgments and opinions in this report are not necessarily those of the technical community either within the NRC or without. Thus the DOE should proceed with caution in implementing the research programs described therein.) More generally, what would be the DOE experimental and analytical program to resolve key safety issues if, for example, metal-fueled LMFBRs are a major part of the U.S. LMFBR program?

c. Can the DOE supply any analysis in the area of CDAs for the design options including large homogeneous vs. heterogeneous cores; carbide and metal vs. oxide; and ThO_2 blankets vs. UO_2 blankets?

d. Does the DOE have a position on the homogeneous vs. heterogeneous core? And, if so, why? Provide analysis including CDA transition phase analysis.

e. In a number of reports supplied to the NRC, there is a design constraint that the positive sodium void coefficient be less than $3.00. Provide the basis for this constraint and its effect on consideration of the LMFBR-variants in the NASAP study.

Response to Questions 1 and 2

The DOE recognizes the validity of most of the NRC comments relevant to the LMFBR variants presented in Volume VI of the PSEID documents. The present status of the LMFBR design does not allow for responses that would provide the depth of information reflected in the NRC comments; thus, no specific responses can be provided.

ENCLOSURE

Bases for Containment Design in Large LMFBRs

In the past the NRC staff took the position that an LMFBR containment system should be able to withstand not only design-basis events such as sodium fires, but also the consequences of low-probability or Class 9 accidents (Refs. 1, 2, 3). Specifically, for the case of CRBR, the staff took the position that the containment system should be protected from the effects of low-probability accidents (commonly referred to as core-disruptive accidents in LMFBRs or CDAs) such that, comparability to the inherent protection of light-water reactor (LWR) containment systems to core-melt events is achieved. This resulted in the 24-hour-containment integrity requirement for the CRBR which can be found in the above given references. Since the termination of the CRBR review in April 1977, the staff completed the FFTF review and also completed a comparative study of the radiological consequences of core-meltdown events between land-based and offshore-sited floating nuclear plants. On the basis of this study (Ref. 4, the staff recommended the issuance of a manufacturing license for barge-mounted plants subject to the condition that "the applicant shall replace the concrete pad beneath the reactor vessel with a pad constructed of magnesium oxide (MgO) or other equivalent refractory material that will provide increased resistance to melt-through by the reactor core in the event of a highly unlikely core-melt accident and which will not react with core-debris to form a large volume of gases..." (Ref. 5).

For the case of FFTF, the staff analysis indicated that overpressurization and the generation of hydrogen resulting from sodium and core-debris interaction with concrete are the principal challenges to containment. The quantity of hydrogen generated that could create a potentially explosive or highly energetic flammable mixture in the FFTF containment building atmosphere, or portions of the building, preceded the point of threatening containment integrity by overpressurization. Although the staff in the FFTF SER, NUREG-0359, August 1978, considered various means to alleviate the buildup of pressure and hydrogen in the containment building following postulated core-meltdown events, some of the recommended steps to deal with the problem were probably not appropriate in view of the facility being essentially constructed. For example, even though refractory materials (e.g., similar to the MgO recommended for the floating nuclear plant design) which are highly resistant to molten core debris and do not generate hydrogen could have been used in the reactor cavity and in the containment subcavity of the FFTF, their use would have been difficult, expensive, and maybe detrimental from an overall safety viewpoint, since the cavity and subcavity were already built and sealed.

For future large fast-reactor designs, the approach should be to integrate the necessary features and designs in the containment system design from the start; thus, the containment will be able to withstand and mitigate not only the consequences of design-basis events but also the consequences of lower-probability, higher-consequence accidents. Accordingly, three broad classes of accidents should be taken into consideration in the designs of large fast-reactor containment: (1) those postulated accidents considered in the design basis of plants (i.e., 10 CFR 50); (2) hazards not exceeded by those from any accidents considered credible (i.e. 10 CFR 100 of Site Suitability Source Term); and (3) low-probability or Class 9 accidents. Because the information provided in the LMFBR PSEID relates primarily to the core (i.e., various fuel cycles) and it has not been integrated into a system design, the following staff comments on these three classes are somewhat generic in nature and are primarily based on the staff's experience with previous reviews of LMFBRs and LWRs, as well as on the recent staff position mentioned earlier regarding floating nuclear plants.

Design-Basis Accidents (10 CFR 50)

In an LMFBR, the accidents that represent the principal challenges to containment are sodium fires coupled with potential sodium-concrete reactions that result from failure of pipes and vessels containing sodium and the subsequent release of the sodium. Following sodium release, combustion with oxygen (even for those areas which are inerted) will result in increasing pressures and temperatures. The specific initiating events, as well as consequences, will be very system dependent. Based on the staff review of the CRBR and the FFTF, the sodium releases were based on a spectrum of postulated component and piping failures of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated sodium fire accidents is covered. Some of the specific challenges to the containment presented by sodium release accidents that should be considered in a containment design are as follows:

a. Mechanical. The deterioration of concrete by sodium can weaken structures, cause cracking, and enlarge leak paths; therefore, means should be used to prevent or reduce the likelihood of direct contact between sodium and concrete. For the FFTF and the CRBRP, cell liners were used to accomplish this.
b. Thermal. The chemical heat of sodium reactions with oxygen or concrete can build up pressures within inerted cells or the containment building which must be included as part of the containment design basis.
c. Explosive. The generation of hydrogen from reactions between sodium and water (or concrete) can lead to explosive mixtures in the air atmospheres of the reactor containment building; therefore, water should be kept to a minimum in buildings containing large amounts of sodium. Hydrogen recombiners are provided in LWRs to control hydrogen. For LMFBRs (the FFTF and the CRBR), the applicants have claimed in the past that the presence of sodium oxide has a catalytic effect in promoting recombination of hydrogen and oxygen and in keeping the hydrogen concentration below the explosive limit. Based on the available information, the staff has previously been unwilling to accept the view that hydrogen can be depended upon to burn benignly under the natural processes associated with these accidents.
d. Nonradiological toxicity. If released from containment or the steam generator building, large quantities of nonradioactive sodium could be an inhalation and environmental hazard. Effective methods can be used to suppress or extinguish sodium fires; isolation can prevent the release of the hazardous smoke.
e. Filters. The dense smoke from sodium fires can rapidly plug ventilation filters. Scrubbers or prefilters are generally required to eliminate this problem.

In recognition of the above, the NRR staff during the review of the CRBRP issued general safety design criteria for the CRBRP, including Criterion 41, "Containment Design Basis," which stated in part . . . "the reactor containment structure, including access openings and penetrations, and if necessary, in conjunction with additional post-accident heat removal systems including ex-vessel systems, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences and any of the postulated accidents."

Site-Suitability Source Term (10 CFR 100)

The site-suitability source term (SSST) is nonmechanistic, and its use is intended to represent an assumed radiological release from the core, the consequences of which would result in potential hazards not exceeded by those from any accident considered credible (see footnote 1 to 10 CFR 100.11 (a)). A primary objective of the staff's safety review is to assure that no other accident sequences within the design envelope result in the release of fission products to the environment greater than those postulated for the SSST. As part of this review, the staff has in the past examined very carefully such factors as core physical and geometrical configuration, including the type and quantity of fissionable material, control system(s), decay-heat-removal system(s), and amount of redundancy and diversity in important safety systems. Consideration has been given also to the manner in which the as-designed plant responds and interacts to a spectrum of accidents, including very severe ones.

Without a particular detailed design description, such as presented in a PSAR, it is not clear from the PSEIDs that the consequences of <u>all</u> credible events would be enveloped by an SST, nor is it apparent that "generic" attenuation mechanisms would apply in all scenarios. At present, both the design concepts for large fast reactors and the analytical methods for examining accidents are in a state of development.

We would recommend, therefore, that the containment design be based on sufficiently conservative source terms that encompass all the uncertainties in presently available data, analyses, and design concepts. As an example, the staff reviewed the bases provided by the CRBR project for its source term and concluded that insufficient information had been furnished to establish that it met the requirements of 10 CFR 100.11 (including footnote 1). As a result, a more conservative radiological source term was adopted. (See CRBRP Site Suitability Report, p. III-14.)

Additional materials not included in an SSST for LWRs, or even the CRBR, might have to be included for the NASAP concepts to account for the introduction of alternative materials (e.g., U-233, U-232, . . .).

Additional design requirements imposed on containment systems, such as filtration, fission-product removal, and containment-heat-removal systems, will have to be considered very carefully. In these areas, additional R&D and proof testing will almost certainly be needed.

Core Melt and Disruptive Accidents

In an LMFBR, the low-probability accidents that represent the principal challenge to containment are associated with core melt and disruption with the potential for concurrent energy release. The energy release is a result of either core vaporization (direct core disassembly and/or recriticality), or sodium vaporization from the transfer of heat from the molten core to the sodium coolant. Energetics could lead to early (order of minutes) containment failure if the containment system is not designed to accommodate the generated loads; on a longer time scale, failure of the containment would occur from the evolution of the meltdown accident progression. This latter evolution could involve chemical reaction products and/or sodium vapor resulting from the inadequate postaccident decay heat removal of a molten and/or disrupted core and could lead to hydrogen explosions, or overpressurization and/or thermal and structural degradations, either one or a combination being able to cause containment failure. Without a particular design description as presented in a preliminary safety analysis report (PSAR), it is not possible to evaluate either the potential evolution of an accident scenario and its consequences or whether it will or can be mitigated and/or contained. Based on the staff's past involvement and experience with the safety analyses and reviews of LMFBRs, containments should be designed to mitigate or to reduce significantly the consequences of core melt and disruptive accidents. From the viewpoint of the two major accident sequences (i.e. early accident energetics and longer time meltdown consequences) that can threaten containment integrity, the following should be considered:

a. Accident Energetics (Direct Disassembly, Recriticality and Fuel Coolant Interactions)

In the past, some LMFBR designers have relied on the primary heat transport system (PHTS) to accommodate the potential energetics; this was especially true for the CRBRP. At the time of suspension of the CRBRP licensing review, the staff and applicant had not resolved the question of whether the design was adequate to accommodate the value of the energetics described in NUREG-0122. Other designers (e.g., the United Kingdom in the case of the commercial fast reactor (CFR) design) have considered prestressed concrete vessels with inherent capability to accommodate large energetics. The choice of a particular vessel/containment system would depend, among other things, on the requirements derived from a specific design. Some of the key considerations (note NUREG-0122) that infuenced the selection of the level of energetics for the CRBRP were:

1. The potential for large work-energy release during the "initiating phase" (direct disassembly) due to the autocatalytic, positive-sodium-void effect without the presence of the mitigating effect of timely and substantial fuel dispersion
2. The potential for large work energy release during the "transition phase" (recriticality)
3. The many uncertainties and unknowns associated with CDA phenomena including the potential for sodium as a working fluid; fuel pin failure dynamics; freezing, plugging, and remelting of molten fuel and fuel/steel mixes; and molten pool boiling dynamics.

Areas and parameters that will influence accident scenarios and consequences for the design(s) and fuel cycle(s) considered in the NASAP study are:

1. The effect of a heterogeneous core (compared to a homogeneous core such as the CRBR) on accident progressions.
2. The effect of core size.
3. The effect of fuel type such as carbides and metals vs. mixed oxides (e.g., on Doppler Coefficient). In the area of fuel-coolant interactions (FCIs), the effect may be major for both carbide and metal fuels because the potential for sodium becoming a working fluid is considerably enhanced.
4. The effect of various bondings for metal and carbide fuels (either helium or sodium).
5. The effect of fuel cycle types such as Pu/Th with Th blankets vs. Pu/U with uranium blankets.
6. The effect of a pot design vs. a CRBR-type loop design.

7. The effect of design specifics such as upper fission gas plena vs. lower plena, perforated subassembly ducts, and temperature profiles across subassemblies.

An aggressive and comprehensive experimental and analytical research and development program will be necessary in order to understand the above effects and their relevance to the safety of a particular LMFBR variant. We need to understand the DOE policy and planning, time frame, and resource commitment for these safety-related areas.

b. Core Meltdown

As was previously mentioned, a benign (i.e., nonenergetic) core meltdown can result in hydrogen explosions, overpressurization due to sodium vapor and noncondensable gas generation, and thermal/structural degradation. All of these effects can lead separately or contribute jointly to containment failure. The FFTF containment failure, for example, was predicted to occur either from hydrogen explosions in the time interval of 10 to 20 hours, or from overpressurization in the interval of 30 to 60 hours.

Evaluations performed by the staff for the CRBRP and FFTF, as well as the floating nuclear plant, indicate that containment integrity can be extended substantially or even indefinitely with the addition of refractory sacrificial materials and/or cooling systems in the lower reactor cavity area. In other areas outside the reactivity cavity, steel liners constructed as engineered safety features can be used to protect the concrete from sodium attack. For both cases, the objective is to reduce or eliminate the potential for the buildup of hydrogen and other noncondensable gases, as well as sodium vapor, that can threaten the containment integrity. Areas of work that should be pursued within the framework of future large LMFBR design(s) are:

1. Examination of refractory sacrificial materials that are highly resistant to core melt debris and do not interact to form a large volume of gases;
2. Examination of cooling systems, both active and passive, to prevent sodium from evaporating following a core meltdown and to remove decay heat from the outer extremities of the refractory material, such that containment of molten core debris can be assured;
3. Investigate methods to monitor and control the hydrogen concentration in the containment building following postulated core meltdown events; and
4. Examine means to further reduce radiological releases from containment following postulated core-meltdown events, such as the addition of sand and gravel filters.

In summary, the licensing staff believes that positive and clearly identifiable actions should be taken in large fast-reactor designs to mitigate significantly the consequences of core melt and disruptive accidents.

REFERENCES FOR ENCLOSURE

1. U.S. Nuclear Regulatory Commission, Final Environmental Statement, Perkins Nuclear Power Station Nos. 1, 2, & 3, NUREG-75/088, Section 3.5, October 1975.
2. Clinch River Breeder Reactor--Preliminary Safety Analysis Report, Chapter 12, Project Management Corporation, NRC Docket No. 50-537, April 1975.
3. U.S. Atomic Energy Commission, Environmental Statement: Liquid Metal Fast Breeder Reactor Demonstration Plant, WASH-1509, April 1972.
4. Liquid Pathway, Generic Study Accidental Radioactive Release to Bodies of Water From Floating and Land-Based Nuclear Power Plants, NUREG-0440, U.S. Nuclear Regulatory Commission, February 1978.
5. Final Environmental Statement, Part III, "Floating Nuclear Power Plants," NUREG-0502, U.S. Nuclear Regulatory Commission, December 1978.

*U.S. GOVERNMENT PRINTING OFFICE : 1980 0-310-912/173