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MASTER

1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

ATOMICS INTERNATIONAL
TASK I REPORT

Volume 2

Prepared for ARGONNE NATIONAL LABORATORY

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United States Atomic Energy Commission Division of Technical Information

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ANL 1000-Mwe LMFBR
(LIQUID METAL FAST BREEDER REACTOR)
FOLLOW-ON STUDY
TASK I REPORT
VOLUME II

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CONTENTS

	Page
I. Introduction	I-1
II. Engineering Studies	II-1
A. Trade Studies	II-1
1. Introduction	II-1
B. Pool-vs-Loop	II-9
1. Requirements and Assumptions	II-15
2. Description of Alternatives	II-16
3. Analyses of Comparison of Alternatives	II-23
4. Conclusions	II-36
C. Number of Heat-Transfer Circuits	II-37
1. Summary	II-37
2. Systems Studied	II-38
3. Analysis and Results	II-39
D. Fuel-Handling System	II-42
1. Introduction	II-42
2. Requirements and Assumptions	II-43
3. Description of Alternates	II-45
4. Analysis and Comparison of Alternates	II-52
5. Conclusions	II-59
E. Vented vs Nonvented Fuel Elements	II-60
1. Summary	II-60
2. Introduction	II-62
3. System Requirements	II-62
4. Systems Studied	II-62
5. Ground Rules of Study	II-62
6. Analyses and Results	II-63
F. Core Design Point Selection	II-69
1. Fuel-Management Study	II-83
2. Core/Height, ΔT Study	II-83
3. Radial and Axial Blanket Studies	II-90

CONTENTS

	Page
4. Midequilibrium Nuclear Characteristics of Selected Design	II-93
5. Nuclear Methods of Analysis	II-93
6. Thermal Hydraulics Methods of Analysis	II-104
G. System Design Point Selection	II-105
H. Materials Selection	II-116
1. Cladding	II-116
2. Structural Materials	II-119
3. Neutron-Absorber Materials	II-125
4. Steam Generator Materials	II-128
III. Safety Considerations	III-1
A. Safety Guidelines	III-1
B. Description of Design Basis Accident	III-5
1. Sequence of Events	III-5
C. Containment Approach	III-6
D. Site Evaluation.	III-10
1. Radiological Evaluation of DBA.	III-12
Appendix	
A. Justification of Fuel-System Preselection	A-1
B. Justification of Core Geometry.	A-2
C. Code Description	A-4
1. CAESAR-IV.	A-4
2. AILMOE.	A-5
3. GRISM	A-5
4. COST-I	A-6
5. SIZZLE	A-7
6. ANISN	A-7
7. PABUC	A-7
8. FAITH	A-8
9. FTB-3	A-8
10. CURIE-DOSE-THUNDERHEAD.	A-9
11. COMRADEX	A-10
12. System Optimization Code	A-10
References	R-1

TABLES

	Page
II-1. Trade Studies Influenced Significantly by Safety Considerations.....	II-5
II-2. Variables in Economic Evaluation.....	II-6
II-3. Overall System Trade Studies.....	II-8
II-4. Reactor Subsystem Trade Studies.....	II-10
II-5. Heat-Transfer Subsystem Trade Studies.....	II-11
II-6. Fuel-Handling Subsystem Trade Studies.....	II-12
II-7. Fuel-Element Trade Studies.....	II-12
II-8. Control-Rod Trade Studies.....	II-13
II-9. Steam Generator Trade Studies.....	II-14
II-10. Key Requirements and Assumptions Applicable to the Pool-vs-Loop Trade Study.....	II-17
II-11. Comparison of Design, Pool-vs-Loop Trade Study.....	II-22
II-12. Inherent Safety Differences, Pool-vs-Loop.....	II-24
II-13. Safety Comparison, Minor (Unprotected) Accidents.....	II-25
II-14. Safety Comparison, Major (Unprotected) Accidents.....	II-26
II-15. Pool-vs-Loop Plant Capital Cost Comparison.....	II-28
II-16. Operation and Maintenance Cost Summary.....	II-31
II-17. Plant Unavailability Summary.....	II-32
II-18. Availability/Reliability Comparison.....	II-33
II-19. Overall Economic Summary.....	II-34
II-20. Loop Flowrates and Piping Sizes.....	II-38
II-21. Heat-Transfer System Differential Costs.....	II-39
II-22. Heat-Transfer System Unavailabilities Based on Number of Circuits.....	II-40
II-23. Economic Comparison Matrix (mills/kwh).....	II-40
II-24. 1000-Mwe vs Demonstration-Plant Equipment Sizes.....	II-41
II-25. Capital Cost and Availability Evaluation, 1000-Mwe Plant.....	II-54
II-26. Results of Trade Study on Vented vs Nonvented Fuel Elements.....	II-60
II-27. Important Considerations for Vented and Nonvented Fuel Elements.....	II-64

TABLES

	Page
II-28. Economic Incentive for Vented Fuel Elements	II-65
II-29. Risks and Advantages of Using Vented Fuel Elements Before Completion of the R&D Program	II-66
II-30. Economic Incentive for Designing 1000-Mwe Plant for Nonvented and Vented Fuel Elements and Using Vented Elements after Ten Years	II-67
II-31. Summary of Breeding Ratio, Power Peak-to-Average, and Peak Linear Heatrate Data During Approach-to-Equilibrium Operation	II-70
II-32. Equilibrium Midcycle Reactor Performance Data from Fuel-Management Study	II-75
II-33. Startup and Equilibrium-Cycle Fuel Loadings	II-77
II-34. Mass Balances	II-78
II-35. Core Preliminary Thermal-Hydraulic Performance Data	II-84
II-36. Basic Cores Calculated	II-85
II-37. Nuclear Results of Core-Height ΔT Survey	II-88
II-38. Equilibrium Fuel-Cycle Costs for Base Case and Variations of Economic Ground Rules	II-89
II-39. 1000-Mwe Core Geometry	II-95
II-40. Nuclear Characteristics of 1000-Mwe Reference Core at Midequilibrium Cycle	II-96
II-41. Study Ground Rules	II-106
A-1. FBR Fuel-Cycle Cost vs Core Burnup (1986-1990 Economy) . . .	A-2
A-2. Gas-Bubble Reactivity Ramprates	A-4

FIGURES

	Page
I-1. Core Fabrication Costs for 180 KgH/day Throughput	I-3
I-2. Fuel Cycle Block Diagram	I-4
I-3. Radial Blanket Reprocessing Cost as a Function of Residence Time	I-5
I-4. Axial Blanket Reprocessing Cost as a Function of Axial Blanket Thickness	I-6
II-1. Trade Study Process	II-2
II-2. Trade Study Approach	II-3
II-3. Loop Concept	II-18
II-4. Pool Concept	II-20
II-5. Double-Rotating Plug with In-Vessel Fuel Storage	II-44
II-6. Double-Rotating Plug with Out-of-Vessel Fuel Storage	II-46
II-7. Single-Rotating Plug with Rotating-Arm Fuel-Handling Machine and In-Vessel Fuel Storage	II-47
II-8. Fixed Plug with Three Rotating-Articulating-Arm Fuel-Handling Machines and In-Vessel Fuel Storage	II-49
II-9. Removable Plug in Hot Cell with Out-of-Vessel Fuel Storage	II-50
II-10. Single Rotating Plug with a Single Nonrotating-Articulating-Arm Fuel-Handling Machine and In-Vessel Fuel Storage	II-51
II-11. Removable Plug in Hot Cell with In-Vessel Fuel Storage	II-53
II-12. Horizontal Cross-Section (50-in. Core)	II-72
II-13. Vertical Cross-Section Dimensions	II-73
II-14. Core Fuel Average Discharge Exposure: Approach to Equilibrium Cycle Operation	II-80
II-15. Reactivity Changes: Approach to Equilibrium Cycle Operation	II-81
II-16. Blanket Power Fraction: Approach to Equilibrium Cycle Operation	II-82
II-17. Horizontal Cross-Section of 41.4-in. Core	II-86
II-18. Horizontal Cross-Section of 35.6-in. Core	II-86
II-19. Total Midequilibrium Fuel Cycle Costs for Core Height ΔT Parameter Survey	II-90
II-20. Total Gain and Net Present Worth vs Residence Time of Inner and Outer Blankets	II-92

FIGURES

	Page
II-21. Optimum Residence Time vs Reprocessing Cost for the Inner and Outer Blanket Elements	II-93
II-22. Net Present Worth vs Residence Time for the Inner Blankets as a Function of Reprocessing Cost	II-94
II-23. Net Present Worth vs Residence Time for the Outer Blankets as a Function of Reprocessing Cost	II-97
II-24. Optimum Residence Time vs Pin Diameter for the Inner and Outer Blanket Elements	II-98
II-25. Net Present Worth vs Residence Time for Inner and Outer Blankets as a Function of Pin Diameter	II-99
II-26. Differential Power Costs as a Function of Core Height	II-108
II-27. Power Cost, 2400 psig, 850°/850°F	II-109
II-28. Power Cost, 2400 psig, 900°/900°F	II-110
II-29. Power Cost, 2400 psig, 950°/950°F	II-111
II-30. Fuel Cycle Costs	II-112
II-31. Partial Plant Costs, 2400 psig, 850°/850°F	II-113
II-32. Partial Plant Costs, 2400 psig, 900°/900°F	II-114
II-33. Partial Plant Costs, 2400 psig, 950°/950°F	II-115
II-34. The Three Periods of Cladding Life	II-118
II-35. Comparison of Typical Curves to Determine a Design Curve for Type 304 Stainless Steel	II-123
II-36. Comparison of Typical Curves After Extrapolation to 250,000 Hr, Type 304 Stainless Steel	II-124
II-37. AI Structural Member Design Stress-Allowable Curve for Irradiation to 10^{23} nvt	II-126
III-1. Reactor Containment	III-8
III-2. Design Basis Accident Two-Hour Dose	III-13
III-3. Design Basis Accident Thirty-Day Dose	III-14

I. INTRODUCTION

This volume of the ANL 1000-Mwe Liquid Metal Fast Breeder Reactor (LMFBR) Follow-On, Task I report lists the technical bases for the 1000-Mwe design described in Volume I. The technical bases consist of trade studies (to arrive at an optimum selection between various engineering alternates), core and system parametric studies, materials selection studies, and safety considerations.

To perform the trade and parametric studies, economic ground rules are required. ANL, acting as overall program manager for the Follow-On Study, specified certain basic ground rules for economic analyses. Additional ground rules required in the study and not specified by ANL were established by AI. ANL ground rules are as follows:

Cost Data

Fissile Pu value	\$10/gm
Charge on core Pu	10%/year
Reprocessing plant Pu criticality limit	67% of U-235 limit
Reprocessing plant charge	\$30,000/day (including turnaround)
Shipping cost, core and axial blanket	\$20/kg
Shipping cost, radial blanket	\$10/kg
Fabrication cost	Adjusted for design
Fuel fabrication charge	10% of book value
Nitrate conversion to feed material	Adjusted for design
Fabrication losses	1% (included in Pu credit)
Reprocessing losses	1% (included in Pu credit)

Capitalization

Plant capacity factor	90% of availability factor
Annual capital charge	13%
Plant availability factor equals	
100% less downtime for refueling.	

The ground rules established by Atomics International are as follows:

Fuel Cycle Cost Data

Core fabrication costs including powder preparation, 1% Pu losses, and new-fuel shipping	(See Figure I-1.)
Axial blanket fabrication costs including powder preparation and new-fuel shipping	\$50/kg
Radial blanket fabrication costs including powder preparation and new-fuel shipping	$\$/\text{kg} = 60 \left(\frac{0.4}{d} \right) \left(\frac{48}{H} \right)^{0.4}$, where d = pellet diameter, in. H = fuel height, in.
Reprocessing plant throughput as function of U-235 concentration	Determined from TID-7025
Ex-core residence time	(See Figure I-2.)

Capitalization

Plant capacity factor	80% (tentative)*
Plant availability factor equals 100% less downtime for refueling, scheduled outages, and non-scheduled outages.	90% (tentative)*
Present worth factor	7%

*To be established at end of Task III, based on final availability analysis.

Based on the preceding ground rules the reprocessing costs were further broken down to allow differential costs to be assigned to each region. Assuming that the fractions of the core and axial and radial blankets removed are reprocessed together, a cost was assigned to the separate regions by accounting for the total cost, both with and without the separate regions.

The reprocessing costs for the inner and outer radial blankets are given in Figure I-3. These costs are a function of the residence time of both blankets, since the mixed enrichment determines the incremental cost, and a change in residence time of one affects the enrichment in the other. The bands indicate the effect on reprocessing cost of variation in residence time from one blanket to the other. It is recommended that a single mean value for radial

Figure I-1. Core Fabrication Costs for
180 kgH/day Throughput

blanket reprocessing be used for each blanket independent of the residence time of the other. This recommendation involves at most $\pm 10\%$ error in reprocessing cost for the residence times of interest for the radial blanket.

The axial blanket reprocessing cost as a function of axial blanket thickness is shown in Figure I-4. This cost is also a function of radial blanket residence time, but for simplicity the single curve of Figure I-4 drawn for respective residence times of 2 and 3 yr for the inner and outer radial blankets was used. The decrease with thickness of incremental cost of reprocessing the axial blanket is due to the increasing dilution of the reprocessing batch caused by the thicker blanket. At thicknesses greater than 17 in. the cost of reprocessing the axial blanket is actually negative, reflecting the total increased throughput allowable.

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Figure I-2. Fuel Cycle Block Diagram

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Figure I-3. Radial Blanket Reprocessing Cost as a Function of Residence Time

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Figure I-4. Axial Blanket Reprocessing Cost
as a Function of Axial Blanket Thickness

II. ENGINEERING STUDIES

A. TRADE STUDIES

1. Introduction

The basic trade studies which have determined the configuration of the ANL Follow-On 1000-Mwe reference design were made as a part of the North American Rockwell 500-Mwe Demonstration Plant Engineering Program. In that program the "system engineering" approach was followed for making the basic engineering decisions. This assures that a disciplined technique is used, incorporating all the basic goals and objectives in the design and considering all important technical and economic factors in making a decision. The trade studies form the basis of this system engineering approach. Their purpose is to consider the various design options available and point out the optimum approach, one which will meet the plant requirements with a minimum of technical risk. Because the many design options available on each component and subsystem within the plant would make it difficult to consider all possible combinations, a simple iterative procedure has been used as a part of the Conceptual Design efforts.

a. Sequence

Before any trade studies are performed, overall requirements for the plant are determined. These come from a number of sources including North American Rockwell, AEC, potential customers, etc. Plant requirements (documented in the System Design Descriptions) form the basis for trade studies within the plant. They may change as engineering proceeds and as contacts with the utilities and AEC proceed, and must be considered highly flexible.

After overall plant requirements, the system is broken down into major subsystems and components. System trade studies are performed, taking into consideration interactions between the various subsystem alternatives. They involve choice of:

- 1) Configuration of component/subsystem
- 2) Design point operation for component/subsystem
- 3) Materials for component/subsystem.

When complete, the system trade studies form the basis for requirements for the subsystem studies, which in turn establish requirements for the components and their trade studies. This is an iterative process, since later trade studies may affect some of the requirements and assumptions made in the earlier ones. Fortunately, however, it is a converging process and has been used effectively in the conceptual design of the 500-Mwe Demonstration Plant. Figure II-1 illustrates this iterative process.

The conceptual design of the 500-Mwe Demonstration Plant will become complete when all of the trade studies have been completed and the plant design is defined down to a level including all critical components which require developmental testing. All of the requirements and component designs will be described in System Design Descriptions (SDD's) and the trade studies documented in Trade Study Reports (TSR's).

b. Approach

An overall approach to performing a trade study is illustrated in Figure II-2. This same approach is used for system, subsystem or component level trade studies. The steps illustrated in Figure II-2 are described as follows:

- 1) The requirements are first established. These reflect the performance, safety, availability, maintenance, operation, design, environmental and interface features required of the subsystem or component. These basic requirements must be met by all of the alternatives and are not considered as design trade variables.

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Figure II-1. Trade Study Process

Figure II-2. Trade Study Approach

- 2) Alternative designs are then formulated. Many are considered and the number is only limited by the ingenuity of the designers. These alternatives are reduced to a few of the most promising ones which can be studied within the existing budget and schedule, and which meet the established requirements.
- 3) Analysis and design of the alternatives then proceeds, and a conceptual design of each alternative is formulated. The depth of detail required is dependent on the particular study. Analysis of the various aspects of the design, including economics, safety, reliability, availability, operation, maintenance, technical risks, development required, schedule, etc. (as listed in Figure II-2) is made. The design is iterated as required until it meets all of the established

requirements. In general, the most extensive iterations have been in the safety area, as safety requirements have represented the most difficult ones to meet. Trade studies in which safety has played a major role are listed in Table II-1. This safety iteration goes on in parallel with the trade study and is not completed until all designs have "acceptable" safety (i. e. , meet established requirements). In this way, safety is not a trade variable but shows up in the economic and technical risk penalties required to make each design acceptable.

- 4) When analysis and design of the alternatives are complete, an evaluation of all factors then proceeds. All of the technical advantages and disadvantages of each concept are compared and reduced to the following three considerations:

- a) Safety

All alternatives must meet the safety design criteria as well as all design requirements. Alternate designs may have safety differences whose weighting is difficult to establish. Such differences are recognized but are not weighted strongly if the designs meet the safety requirements.

- b) Economics

Such factors as capital cost, fuel-cycle costs, plant availability, operation and maintenance costs, etc. , must be normalized to permit a relative comparison of economic factors. For this purpose an economic model was formulated which co-relates all of these factors.

The major elements of this economic model and corresponding weighting factors are shown in Table II-2. The variables in their specified units applicable to a specific study are multiplied by their weighting factor to give the correct consideration for the trade study. The three periods correspond to different fuel-cycle cost periods. The fuel-cycle costs decrease with time in an attempt to reflect

TABLE II-1
TRADE STUDIES INFLUENCED SIGNIFICANTLY BY
SAFETY CONSIDERATIONS

Trade Study	Safety Considerations
Core geometry: Square* vs Modular	Sodium-void coefficient of reactivity
Pool vs Loop*	Sodium containment, DBA effects
Refueling: On-line vs Off-line*	DBA containment integrity
Core height: 50-in.*	Negative power coefficient
Loop elevations: Elevated Loop*	Sodium containment, convection flow
Cover Gas: Argon vs Helium*	Cleanup of contaminated cover gas
Fuel-element plenum location: Top* vs Bottom	Fuel-pin failure propagation
Core clamping method	Negative bowing coefficient of reactivity
Fuel-element size: 217 pins*	Fuel-element cooling and reactivity worth

*Concept chosen

anticipated fuel-cycle cost trends. Note that a given trade study may not require all of these elements (e. g., investment and availability terms in the Pool-vs-Loop trade study do not change fuel-cycle cost, thus simplifying the particular comparison. The final result is an equivalent differential levelized (over 30 yr at a 7% discount factor) energy cost for each of the alternatives being considered.

c) Technical Risks

These represent the significant technical problems or uncertainties impossible to evaluate quantitatively in terms of their effect on economics, safety, or other requirements. They are considerations

TABLE II-2
1000-Mwe PLANT ECONOMIC MODEL

Description of Variable	Weighting Factor Value		
	Years 1-5	Years 6-10	Years 11-30
Direct construction, \$	2.933×10^{-8}	2.666×10^{-8}	2.666×10^{-8}
Startup, \$	2.095×10^{-8}	1.904×10^{-8}	1.904×10^{-8}
Initial sodium and gases, \$	2.095×10^{-8}	1.904×10^{-8}	1.904×10^{-8}
Availability, effect on construction,	2.000	2.000	2.000
R&D and plant risk, \$	1.458×10^{-7}		
Startup delay, days	6.286×10^{-4}	5.714×10^{-4}	5.714×10^{-4}
<u>Fuel-Cost Alternative No. 1*</u>			
Availability, effect on fuel cost	-6.579×10^{-1}	-4.898×10^{-1}	-4.893×10^{-1}
Non-inventory cost, mills/kwh	1.000	1.000	1.000
Inventory cost, mills/kwh	1.053	9.569×10^{-1}	9.569×10^{-1}
<u>Fuel-Cost Alternative No. 2</u>			
Availability, effect on fuel cost	-6.609×10^{-1}	-5.276×10^{-1}	-5.089×10^{-1}
Thermal efficiency	-9.636×10^{-1}	-7.031×10^{-1}	-5.078×10^{-1}
Fabrication and reprocessing, \$/kg (U+Pu)	1.302×10^{-3}	1.303×10^{-3}	1.303×10^{-3}
Burnup, Mwd/kg (U+Pu)	-9.114×10^{-3}	-7.812×10^{-3}	-6.836×10^{-3}
Plutonium value, \$/gm	1.850×10^{-2}	1.206×10^{-2}	1.041×10^{-2}
Breeding Gain	-1.042	-1.042	-1.042
Mass of Pu in-core, kg Pu	2.407×10^{-4}	2.112×10^{-4}	2.038×10^{-4}
Ex-core time/in-core time	3.300×10^{-1}	3.000×10^{-1}	3.000×10^{-1}

Levelized totals (to 1980) = $0.15852[(2.0843) (\text{Total}_{1-5}) + (1.4861) (\text{Total}_{6-10}) + (2.737) (\text{Total}_{11-30})]$

*Simplified model for known fuel-cycle cost

which, if they cause problems, could result in considerable plant downtime, large schedule delays, licensing problems, excessive costs, or large required component development programs. In considering technical risks much engineering judgment is involved, both in terms of estimates of the probability of the problem arising and the consequences of having the problem. A careful consideration of plant simplicity, available design margins, potential failure modes, past AI (and others') experience, availability of adequate data, backup designs, state-of-the-art, etc., must be made in judging the magnitude of these risks.

- 5) The decision on each trade study is reached by comparison of the safety, economics, and technical risks, involving considerable engineering and management judgment. A firm quantitative evaluation is very difficult to make. In general the goals of the first plant are strongly weighted in terms of a low-risk proven approach. As such, trade study decisions are heavily weighted by AI in terms of the low-risk approach. The economic penalties must, however, be considered along with the risks. If a large economic penalty is involved, then a higher-risk approach may be allowable as long as a carefully defined development and test program is set up to minimize this risk.

The economic model (mills/kwhr) usually indicates the same results for both the 500- and 1000-Mwe plants when the cost of development is excluded. Some of the trade studies made have been on 1000-Mwe plants and others on a 500-Mwe Demonstration Plant, reflecting changes in the goals and emphasis as the program developed. If other small inconsistencies appear in any trade studies because they were performed at different times it is a reflection of the iterative process previously described. In general all of the trade studies are continuously reviewed as others are performed, to assure that these inconsistencies do not affect the final results.

TABLE II-3
OVERALL SYSTEM TRADE STUDIES

Type	Trade Study	"Reference" Selection	Percent Complete
Configuration	Pool vs loop	Loop	100
	Plant containment	Double containment	90*
		Primary: dome plus cells	
		Secondary: steel-lined reinforced concrete building	
	Reheat vs no reheat	Reheat	100*
Design point parameters	Plant control method	Variable flow swing load	80
	After-scrum temperature control	Convection flow with valves, no hot restart	100
	Number of coolant circuits	3 primary, 3 secondary	100
	Reactor outlet temperature	1060°F	100*
	Steam conditions	2400 psi/900°F/900°F	100
	System ΔT 's	300°F primary, 250°F secondary	100
	System ΔP 's	117 psi primary loop	100
Materials	System availability allocation	See detailed allocation	100
	Cover gas: argon vs He	Helium	100
	Primary/secondary loop materials	304 SS	100
	Na vs NaK	Na	100

*Being reconsidered

c. Summary

A summary of the various system, subsystem, and component trade studies which have been and are being performed as a part of Company-funded programs are summarized in Tables II-3 through -9. These tables also indicate the status of each trade study and the decision made. They form the basis of the reference 1000-Mwe Follow-on Plant.

In the following sections are summaries of some of the major trade studies for the conceptual design. The first four are configuration type and are summaries of the results of studies on Company programs. The next two are design-point type. The last is a summary of materials trade studies made on the Company programs.

B. POOL-VS-LOOP

The components of a fast-breeder reactor primary loop can be arranged in one of several distinct ways. The first method, similar to that used in light water reactor power plants, is called the Loop Concept. In this concept the reactor core is contained within a relatively small reactor vessel. The main primary coolant (sodium) is pumped through the vessel into external heat-transfer loops located in individual shielded vaults outside the reactor. These vaults can be filled with inert gas and contain the pumps, heat exchangers, valves, and instrumentation for the primary heat-transfer circuit. A number of liquid metal systems have been constructed using this concept, both in this country and abroad. Examples of such systems are EBR-I, Fermi, SRE, HNPF, SEFOR, DFR, BR-5, and Rapsodie. Also, several advanced type fast-breeder reactor plants are designed to use the Loop Concept, including the FFTF, plants proposed by AI and Westinghouse, the German SNR and Na-1, and the Russian BN-350.

An alternate method of configuring the main heat-transfer system is the Pool Concept. In this concept all of the main primary heat-transfer equipment is contained within a relatively large reactor vessel, which also includes the reactor core (and the necessary in-vessel fuel storage, if desired). All of the primary sodium is within this vessel. The large vessel requires a minimum

TABLE II-4
REACTOR SUBSYSTEM TRADE STUDIES

Type	Trade	"Reference" Selection	Percent Complete
Configuration	Open vs closed lattice	Closed lattice	100*
	Vented vs nonvented fuel	Nonvented	100
	Control rod cooling	Thermo-siphon	100
	Blanket cooling	Forced convection	100
	Cover-gas cleanup	Charcoal absorption system	80
	Vessel support method	Top support	100
	Core geometry	Regular (nonspoiled) core	100
	Core clamping method	Thermal-bowing tightening with core clamp	100*
	Core instrumentation	Thermocouple core and blanket element	100*
	Vessel inspection and repair	"Skate" Concept	20
	Sodium secondary containment	Containment vessel surrounding reactor vessel, elevated loop	100
	Emergency core cooling	Loop natural convection plus cooled cavity liner	100*
	Na level control	Overflow system	100
	Top shield cooling	Insulation/high temperature concrete	50
	Failed element location system	Not required	100
	DBA pressure relief	Top shield lifting, rupture disc from dome to cells	100*
	Top shield sealing	Double seal with pumpback	100*
Design Point	Allowable number of element failures	1%	50
	Core dimensions	4.2-ft core height, 5.8-ft diameter	100*
	Core ΔP	90 psi	100*
	Core ΔT	300°F	100*
	Blanket dimensions	1.0-ft axial blanket, 0.84-ft radial blanket	100
	Blanket residence times	Inner (1.5 yr), outer (3.0 yr)	90
	Number of orifice zones in core	5 zones, orifice for constant clad temperature	100*
	Fuel-pin size	0.25 in.	100*
	Number of enrichment zones	2	100
Materials	Neutron-shielding material	Serpentine	50
	Cavity liner	Carbon steel	90

*Being reconsidered

TABLE II-5
HEAT-TRANSFER SUBSYSTEM TRADE STUDIES

Type	Trade Study	"Reference" Selection	Percent Complete
Configuration	Sodium containment	Elevated loop concept	100
	Pump location: hot vs cold leg	Hot leg	100
	Pump type: EM vs mechanical	Mechanical	0
	Pump speed control method	EM coupling	100
	Valve types	Primary loop: throttling (butterfly)	80
		Check (balanced), blocking (double wedge)	
	System gas removal	Loop layout, gas venting, diffusion barrier, degassing system	100*
	Emergency cooling	Convection flow (5%)	100
	IHX type	Vertical, bellows type shell and tube	100*
Design point	Piping size	Primary: 36 in. (suction) 32 in. (discharge)	100
		Secondary: 32 in. (hot leg) 30 in. (cold leg)	
	Number of pumps	One per loop	100
	Number of IHX's	One per loop	100

* Being reconsidered

TABLE II-6
FUEL-HANDLING SUBSYSTEM TRADE STUDIES

Type	Trade Study	"Reference" Selection	Percent Complete
Configuration	Fuel-handling method 5 methods considered	Rotating plug, in-vessel storage	100
		Elements in pots	
		Separate in-core and ex-core machines	
	Double vs triple plug	Double plug	100
	Concentric vs eccentric	Eccentric	100
	On-line vs off-line removal from storage	Off-line	100
Design point	Single vs double barrel ex-core machine	Single	100
	Refueling interval	Flexible 2 to 6 month; desirability of 3-day weekend refueling	100

TABLE II-7
FUEL-ELEMENT TRADE STUDIES

Type	Trade Study	"Reference" Selection	Percent Complete
Configuration	Fuel-element spacer	"Eggcrate" with several backups	50
	Fuel-element hold-down	Hydraulic	100
	Gas-plenum location	Above core	100
	Pellet configuration	Annular pellets	80
	Fuel-shipping methods	Ship in Na	100
Design point parameters	Gas-plenum length	30 in.	100
	Clad thickness	0.015 in.	100
	Fuel-element size	217 pins, 5.3 in. across flats	100*
	Linear power and centerline temperature	15 kw/ft	100*
	Smear density	80%	50
	Pellet/clad gap	2.5 mil (cold-as fabricated)	80
Materials	Fuel type	Hot pressed-sintered-coprecipitated	100*
	Clad material	304 SS (316 SS backup)	50

*Being reconsidered

TABLE II-8
CONTROL-ROD TRADE STUDIES

Type	Trade Study	"Reference" Selection	Percent Complete
Configuration	Top vs bottom latch	Bottom	100
	Main seal type	Gas-purged seal	100
	Scram latch seal type	Bellows with gas backup	100
	Rod types	Separate shim-safeties	100
		Separate safeties	
	Cooling method	Thermo-siphon	100
	Absorber pins vs cylinders	Pins	100
	Immersed vs non-immersed bellows	Immersed	100
	Alternate sealing methods		100
Design point	Number of rods		
	Shim-safeties	9	100
	Safeties	6	100
Materials	B ₄ C vs tantalum	Tantalum	100

TABLE II-9
STEAM GENERATOR TRADE STUDIES

Type	Trade Study	"Rcference" Selection	Percent Complete
Configuration	Modular vs single unit	Modular	100
	Recirculating vs once-through	Once-through	100
	Separate evaporator and superheater vs combined	Separate	100*
	Alternate means of expansion compensation	None	80
	Hockey stick		
	Bellows in shell		
Sine wave tube			
Design point	Module size		100*
	Superheater	79 tubes	
	Evaporator	91 tubes	
	Reheater	161 tubes	
Materials	Superheater material	2-1/4 Cr - 1 Mo	100
	Incoloy vs		
	Austenitic vs		
	Ferritic (2-1/4 Cr - 1 Mo and 5 Cr - 1/2 Mo)		

*Being reconsidered

of penetrations. Nonradioactive secondary sodium flows out of the intermediate heat exchanger (IHX) to a secondary heat-transfer system essentially identical to that of the Loop concept. One Pool reactor (EBR-II) is currently in operation. A number of advanced reactors and design studies have adopted the Pool concept as the reference approach. In the U. S. these include the General Electric and B&W Advanced 1000-Mwe Studies; in Europe they include the British PFR design (now under construction) and the French Phenix design.

The incentives for considering the Pool concept as an alternate design are its safety against primary pipe leaks and ruptures and its compact design — which may lead to lower overall capital costs. AI considered these potential advantages worth investigating early in the conceptual design phase. Potential advantages of the Loop relate mainly to the ease of maintenance and access to primary system components, thus reflecting potential increases in overall plant availability. Also, this concept has had more experience behind it. It provides a relatively clear-cut separation of primary system components and places components in a more controllable and predictable environment.

The Pool-vs-Loop trade study, which was one of the first performed on the Company-funded program (because of its importance in setting the requirements and configuration for other elements of the plant), was recently reviewed and updated to reflect the latest changes in plant design and requirements. It was originally made for a 1000-Mwe design to ensure that economics and technical risks for an ultimate plant were adequately considered.

1. Requirements and Assumptions

These were established before the trade study was started and served as a conceptual design basis for both the Pool and Loop designs. The requirements reflect the overall plant requirements (Section III, Vol I) as well as other requirements and assumptions based on results of other trade studies. Table II-10 summarizes the principal ones applicable to the Pool-vs-Loop trade study.

2. Description of Alternatives

During the course of the trade study (and subsequent updatings) a number of loop and pool concepts were considered. Alternate loop concepts have considered both the elevated and nonelevated loops and the use of blocking valves and siphon breakers as safeguards against major pipe rupture. For the Pool concepts six different alternative designs were conceived in which the pump, core, and heat exchanger were arranged in a number of different configurations. After initial screening studies, chosen reference Loop and Pool concepts were used as a basis of this trade study. These chosen designs are described below.

a. Reference Loop Concept

This consists of a 3-circuit plant, each containing one pump and two intermediate heat exchangers as shown in Figure II-3. It is of the elevated loop design which prevents a pipe leak from siphoning the core below a minimum level, assuring cooling in the remaining loops. It also provides adequate afterglow cooling using a single loop. A separate reactor vessel contains the core, blankets, fuel storage, and a neutron shield for limiting reactor vessel irradiation damage. The reactor vessel is surrounded by a close-fitting sodium-containment vessel which prevents lowering of Na to an unsafe level in a vessel rupture. The heat-transfer loops are contained in nitrogen-filled shielded vaults. Each primary loop contains a pump in the hot leg, two vertical tube-and-shell heat exchangers (with bellows in the shell), piping expansion bends, and primary-loop instrumentation. The reactor vessel closure at the top consists of a double-rotating plug used also for fuel handling. Removal of stored fuel from the vessel is done off-line. Each primary loop has two blocking valves to permit maintenance, a check valve to prevent backflow in a single-loop pump failure or trip, and a throttling valve for after-scam temperature control and startup flexibility.

Maintenance of components in the primary loop may be performed by isolating and draining a primary loop after closure of the blocking valves and allowing the Na-24 in that loop to decay. Reactor operation can continue at approximately two-thirds power during this period. Then upon reactor

TABLE II-10
KEY REQUIREMENTS AND ASSUMPTIONS APPLICABLE TO THE
POOL-VS-LOOP TRADE STUDY

Function	Requirement
<u>Performance:</u> Primary circuit must remove heat from the reactor and transfer to the secondary loop.	Power: 2500 Mw (thermal)
<u>Safety:</u> The primary system must meet all plant safety requirements.	<ol style="list-style-type: none"> 1. Must contain a DBA (nuclear excursion plus sodium-vapor explosion) within primary boundary. 2. Primary boundary must serve as a leak barrier against fission-product release following DBA. Leak-rate < 10%/day @ 20 psi. 3. Must be designed to assure core cooling under all emergency conditions. At least two different systems (redundance), preferably using two different principles, (diversity) should be available. 4. No single coolant system rupture shall interfere with the ability to cool the core. 5. The system must be designed to provide adequate core cooling in the event of a double-ended guillotine pipe rupture. 6. Safeguards must be provided to preclude Na fires. 7. Excessive radioactivity in secondary loop should be prevented.
<u>Availability:</u> Overall plant must meet a mature availability of 90%.	Allocation to heat-transfer components in this trade study (pumps, vessel, IHX, valves, instrumentation) 3.7% (maximum) forced outage rate.
<u>Maintenance:</u> Design should allow ease of inspection, replacement, and repair of components, including core internals.	Loop: Maintenance of primary components consists of <ol style="list-style-type: none"> (a) Valving off loop. (b) Allowing Na²⁴ decay while running at 2/3 power. (c) Open heat-transfer cell and perform maintenance. Pool: Remote removal and repair of all components while reactor is shut down.
<u>Configuration (From other trade studies):</u> Fuel-handling method Core geometry System design point Primary outlet temperature Primary inlet temperature Sodium flowrate Pump Type	Double-rotating plug, in-vessel storage, off-line stored fuel removal Regular, nonspoiled 1060°F 760°F 94x10 ⁶ lb/hr Centrifugal, removable internals

4-24-68 UNCL

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Figure II-3. Loop Concept

shutdown the vault can be entered for any maintenance required, including repair of the heat exchanger, instrumentation, valves, piping, etc. The pump internals are removable and can be removed by means of a plastic bag after decay of Na-24 if it is not contaminated. It is not necessary to wait for Na-24 decay if a shielded cask is provided.

Safety requirements of the Loop concept are met with the elevated loop design. There are always at least two different means of cooling the core (two or three loops), and because of the elevated loop concept there is adequate natural circulation on a single loop to assure adequate cooling even with complete pump failure in all loops. Thus both coolant redundancy and diversity are assured. Loop elevations are such that large or small leaks do not siphon the sodium below the loop outlets. Reactor scram and pump trip are provided for loss of Na level or flow to assure that the upper pool is not pumped dry during a major pipe failure. Valve action is not required but acts as a backup to the other factors. A close-fitting containment jacket around the reactor vessel and pumpwell assures that leaks will not drain the reactor to an unsafe level. Inert gas will prevent Na fires in the cells in the unlikely event of leakage. The primary-system vaults contain the products from a Design Basis Accident (DBA) and serve as a primary-containment barrier. A dome over the top shield also serves as an extension of this primary barrier. The pressure from a DBA is relieved into the large expansion volume in the primary vaults, thus holding this primary-containment pressure to a safe level (~20 psi).

b. Reference Pool Concept.

The chosen Pool design consists of a single large tank, as shown in Figure II-4. The core, storage, neutron shielding, pumps, heat exchangers, valves, and primary instrumentation are all contained within this tank. The flowpath is upward through the core into an intermediate exchanger, then downward emptying into the sodium pool. The pumps then take their suction from the Na pool and return the higher-pressure Na to the reactor inlet. Six pumps and six heat exchangers have been chosen for the reference design. The Na pool, and hence the vessel, is at the cold or inlet temperature. The pumps are in the cold leg. Two Na levels exist within the pool, their difference being determined by the pressure drop across the intermediate heat exchanger.

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Figure II-4. Pool Concept

This IHX pressure drop must therefore be kept low to minimize level fluctuations. The fuel-handling system consists of a double-rotating plug with in-vessel fuel storage and off-line removal of stored fuel from the vessel. An internal neutron shield is of sufficient size to prevent excessive activation of secondary sodium, pumps, or IHX's. The 50-ft vessel is supported on the bottom, thus requiring a bellows for expansion at the top. Piping slip joints are used to accommodate piping thermal expansion. The pumps and heat exchangers all have removable internals, and a shielded cask is provided for removing and replacing them. Components are removed only with the reactor shut down.

The reference Pool design meets all of the safety requirements. A large vessel is contained within a close-fitting outer liner so that a Na leak will not drain it to an unsafe level. There are six coolant loops; emergency cooling is provided by pony motors on the pumps. Emergency power is available to the motors. As in the Loop concept a containment dome may be needed over the entire closure to serve as the primary-containment barrier. Further safety analysis is necessary to assess the design of the top closure as a containment barrier and establish the need for the dome. If required, the hot-cell refueling concept may be a much more attractive refueling concept for the Pool. Because some expansion-gas volume is provided in the top of the pool to relieve DBA pressures, the adequacy of this amount of gas to relieve pressure also needs further analysis. The need for this cover-gas volume in the Pool, as opposed to the Loop, is established by a lack of expansion volume space available to the latter by rupture to the large cells. Such additional space could be provided in the Pool concept, however, only with an increase in building size and labor costs. Further trades could be made in this area. The cover gas in contact with the Na surface in the Pool concept could cause significant gas-solubility problems which have been precluded in the design of the Loop by means of a restricted cover-gas surface.

Table II-11 gives a comparison of some of the pertinent data for the reference Pool and Loop designs.

TABLE II-11
COMPARISON OF DESIGN, POOL-VS-LOOP TRADE STUDY

Design Feature	Pool	Loop
Number of:		
Primary loops	6	3
Primary pumps	6	3
IHX's	6	6
Blocking valves	0	6
Check valves	6	3
Throttle valves	6	3
Pump:		
Location	Cold leg	Hot leg
NPSH, ft	45	28
Capacity, gpm	36,250	75,900
IHX		
Type	Removable bundle, floating head	Vertical, fixed bundle, repairable in place
Length, ft	18	45
Number of tubes	5000	1300
ΔP (primary), psi	~1	10 to 12
Vessel support:	Bottom, with bellows at top	Top
Diameter, ft	50	28
Building diameter, ft	110	145
Neutron shielding:		
Material	SS	SS
Thickness, in.	18	8
Total sodium inventory, lb	3×10^6	1×10^6

3. Analysis and Comparison of Alternatives

The two alternative reference designs described in the previous section were analyzed to assess the technical advantages and disadvantages of each approach. Because of the conceptual nature of each design, only major features are compared; the results of the trade study did not show that a more detailed comparison was warranted. The basic analysis and comparisons are divided into safety, economics, and technical risks, as discussed in the following paragraphs.

a. Safety

Both the Pool and Loop concepts have individual safety characteristics not displayed by the other. A comparison of these inherent safety features is given in Table II-12. Their effects were compared on both minor operational and major (unprotected) type accidents. Minor accidents include incidents which might be expected to occur with reasonable probability over the life of a plant. The major ones are low-probability occurrences of major magnitude studied mainly for licensing purposes, e. g. , the (DBA), the guillotine pipe rupture, etc. Comparison of the two concepts from the standpoint of these accidents is made in Tables II-13 and -14. The principal safety advantages of the Pool concept are:

- 1) Vessel simplicity and reliability of coolant boundary.
- 2) Minimum effect of major primary pipe ruptures on core cooling.
- 3) Minimizing of potential sodium fire hazard in an open vault during some maintenance operations in which piping must be cut.
- 4) Large de-entrainment (low-velocity) surface for gas in system and lack of a number of potential gas-collection locations in piping system.
- 5) Large thermal inertia and hence lesser transients resulting from operational incidents.

TABLE II-12
INHERENT SAFETY DIFFERENCES, POOL VS LOOP

	Pool	Loop
Thermal inertia, Mw sec/ ^o F	1300	400
Pipe failures:		
Feet available	0	800
Failures/30 yr (estimated)	0	~ 4
Structural (earthquake):		
Natural frequency, vessel, cps	18	17
Natural frequency, loop, cps		145
Blast effects:		
Cover-gas volume, ft ³	3500	1000
Vault expansion volume, ft ^{3*}	0	300,000
Seal length, ft	370	170
Shock effects:		
Relative shock		
Vessel walls	0.3	1.0
Vessel bottom	0.8	0.8
IHX	0.3	0.05
Natural convection:		
Head available, ft	7	21
Full flow, %	4.6	8
Na bubble transport:		
Time in plenum, sec	60	10
Na volume above exit nozzles, ft ³	21,000	6,100

*If hot cell refueling is used, add 80,000 ft³ in both pool and loop concepts.

TABLE II-13
SAFETY COMPARISON, MINOR OPERATIONAL ACCIDENTS

Analysis	Preference		Comments
	Pool	Loop	
1. <u>Fault tree comparison</u> (accident probability)	X (slight)		Fault tree for pool is essentially identical to that of loop system. Differences arise from omission of external loop from pool system.
2. <u>Thermal considerations:</u>			
(a) Excursion effect on system temperatures	X		Effect of transient is greater in the loop design. Absence of loops and nozzles are an asset in the pool concept.
(b) Steady-state structural		X	Temperature gradients can be kept small in the loop concept through use of insulation.
(c) Vessel failure and resulting leakage paths	X		More leakage paths occur in loop design due to larger number of access penetrations.
3. <u>Structural considerations:</u>			
(a) Vibrational shock and seismic loading effects			No advantage in either concept.
(b) Thermal stress and cycling effects	X		Attenuation of large Na pool would reduce temperature fluctuation seen by IHX's and pumps.
(c) Fabrication and Quality Control			No clear-cut advantage for one system over the other; loop concept seems to lend itself better to operational surveillance.
4. <u>Controls, kinetics, and instrumentation:</u>			
(a) Incident excursion estimates			About the same for both concepts.
(b) Instrumentation		X	The higher primary instrument temperatures associated with the pool concept make the instrumentation problem more complex.
(c) Control and protective system			Essentially identical for both concepts.
5. <u>Operational safety:</u>			
(a) Small gas bubbles in core due to entrainment	X		Pool concept is slightly favored due to large de-entrainment surface and minimization of gas collection pockets.
(b) Small gas bubbles in core due to solubility of gas		X	Loop concept is superior because of limited area of contact of cover gas with pool.
(c) Maintenance		X	Potential for single-loop shutdown during maintenance without full plant shutdown.
(d) Sodium purification		X	It appears that a larger purification system will be necessary for the pool concept.
(e) Component handling		X	Slight advantage for loop concept resulting from smaller surface areas contacting radioactive primary coolant.
(f) Released fission gases			No advantage is obvious for either system.
(g) Accommodation of vented fuel elements			No advantage is obvious for either system.
(h) Solid-fission-product release and deposition	X		Fission-product deposition in primary loop could restrict maintenance activities in loop concept.
(i) In-core storage of failed fuel elements			No advantage is obvious for one system over the other.
(j) Piping leaks	X		Less piping in pool concept gives it an edge.

TABLE II-14
SAFETY COMPARISON, MAJOR
(UNPROTECTED) ACCIDENTS

Type Accident	Preference	
	Pool	Loop
Primary pipe failure	X	
Gas in core:		
Solubility		X
Entrainment/collection	X	
Core "explosion":		
IHX integrity		X
Seal integrity		X
Blast pressure	X	
Primary containment pressure		X
Pump or cooling failure:		
Convection flow		X
Time to boil	X (5.1 hr)	(1.5 hr)
Primary tank rupture:	No difference	
Earthquake	No difference	
Sodium fire in open vault	X	

The principal safety advantages of the Loop concept are:

- 1) Natural convection emergency cooling eliminates need for pony motors, thus providing diversity in emergency cooling and meeting the safety criteria for emergency core cooling. In the Pool

concept certain common failures (e. g. , a common failure of all pump bearings) could disable all emergency core cooling.

- 2) The IHX is isolated from the effects of a DBA, providing a higher assurance of adequate heat removal following a DBA.
- 3) The Loop concept provides a large expansion volume to suppress the resulting pressure rise, thus assuring lower pressure containment of the DBA.
- 4) The Loop concept allows a limited contact of cover gas with primary Na, thus minimizing gas-solubility problems.
- 5) Protective instrumentation is located in a less-severe more-maintainable environment. This may enhance the reliability of protective equipment.

The net conclusion of the safety evaluation is that, in spite of significant differences between the two designs, both can meet the established safety criteria. The economic penalties, e. g. , pony motors (for the pool) and large cell volume (for the loop), thus are involved in the economic comparison.

b. Economics

The principal economic differences between the two concepts are capital, operation and maintenance, and unavailability costs. Other factors such as fuel-cycle cost appear to be unaffected by the choice of configuration. The cost comparison was made on a differential basis using the economic model discussed in the Introduction, and only factors which vary between the Pool and Loop design are considered, as discussed in the following paragraphs.

Items involved in capital cost estimates (See Table II-15) are reactor building and substructure, reactor vessel, neutron shield, cavity liner/insulation, primary piping system, pumps, intermediate heat exchangers, primary component preheating, sodium inventory, and miscellaneous unrelated costs.

Because of the relatively compact Pool design, the reactor building is approximately 110 ft diam as opposed to 145 ft for the Loop, resulting in a cost of \$2,000,000 less than for the Loop concept.

TABLE II-15
POOL VS LOOP
PLANT CAPITAL COST COMPARISON

Item	Loop	Pool Cost (\$)	
		Differential	Uncertainty
Reactor building and substructure	Base	-2,000,000	±400,000
Reactor vessel	Base	+600,000	±150,000
Neutron shield	Base	+985,000	±200,000
Cavity liner/insulation	Base	+70,000	±14,000
Primary piping system	Base	-2,450,000	±280,000
Pumps	Base	-570,000	±110,000
IHX's	Base	+2,600,000	±1,000,000
Preheating	Base	-300,000	±100,000
Unevaluated items (see text)	Base	0	
Total		-1,065,000	±1,100,000 (Combined statistically)
Sodium inventory	Base	+400,000	±50,000

The large vessel used in the Pool concept is more costly than the smaller one in a Loop concept. It must also be field-fabricated, whereas the Loop vessel may be shop-fabricated, depending on local transportation conditions at the site. This differential is estimated at \$600,000 in favor of the Loop concept.

The Pool requires a neutron shield within the vessel to prevent activation of the IHX, pumps and check valves. This shield is 18 in. of SS estimated to cost \$985,000.

The pool concept has a larger cavity surrounding the vessel which must be insulated, at an estimated cost differential of \$70,000.

The primary piping is a significant cost in the comparison of the two concepts. The following additional costs are required with the Loop:

Piping	\$1,100,000
Valves (block)	450,000
Vault liners	200,000
Vault shielding (concrete)	400,000
Vault cooling equipment	300,000
Total	<u>\$2,450,000</u>

The primary pumps for both concepts have similar design; they can, however, differ in cost because of several factors. First, the net positive suction head (NPSH) is slightly different, as shown in Table II-11. Also, there can be a difference in the design complexity of the pump for the Pool concept because of the possibility of thermal distortions and misalignment within the Pool, and the increased difficulty in controlling the thermal environment. The difference in NPSH is primarily due to the pressure drop across the suction piping, and accounts for a difference of approximately \$500,000 on pump costs in favor of the Pool. Location of the pump in the hot leg also requires additional pumping power for equal mass flow rates; this requires additional installed capacity for the Loop, at an estimated penalty of \$100,000. The pony motors are required in the Pool along with emergency power to provide decay-heat removal, at an estimated penalty of \$30,000.

Intermediate heat exchangers for the Pool concept are more costly than the Loop concept because: first, the pressure drop must be low, as discussed previously, second, the units must be made removable so that they may be remotely repaired and handled by means of a cask; third, the tubes can not be as long as in the Loop concept, resulting in more tubes and hence an off-optimum and more expensive design; and last, special insulation and thermal-shock protection is required for the Pool environment. The Loop IHX also has cost associated with structures, insulation, and manifolding; however, the overall differential cost is estimated to be \$2,600,000 in favor of the latter.

The Loop concept requires a more extensive preheating system for primary components, viz. the vessel, piping, pumps, valves, and IHX's. The Pool preheat is considerably simpler and requires preheat only for the vessel and the components in it. The net cost advantage for the latter is \$300,000.

The Pool requires considerably more Na inventory than the Loop, as shown on Table II-11, resulting in an additional \$400,000 cost.

A number of technical areas have not been completely evaluated at this time, and could result in significant differences between the two concepts. These include the dome, shield hold-down structure, top rotating shield closure sodium-purification system, and drain tanks. Because of the conceptual nature of the design and lack of a detailed safety evaluation for the two concepts, these are uncertainties which could influence the cost comparison. They are reflected into the cost-sensitivity study discussed later.

A detailed operation and maintenance analysis (see Table II-16) showed that the difference in operation and maintenance costs between the two concepts is small, amounting to approximately \$22,000 per year.

An assessment of the energy availability difference of the two concepts was made. This study combined component failure-rate estimates from the meager existing data with repair times for each component to assess the difference in forced outage rates of the two systems (see Tables II-17 and -18). As shown in Table II-17, forced outage rate for a Pool concept is higher than for the Loop, reflecting the increased maintainability of the latter's components. A major contributor to unavailability of the Pool concept is the IHX. This is based on an assumed tube-failure rate of 3×10^{-9} /hr due to such causes as tube vibration, faulty tube/tube-sheet welds and tube erosion, corrosion, leakage, and flow blockage. Failures attributed to the shell are estimated to be less than 20% of the total. The tube-failure rate is assumed to be generally independent of tube length since the only failure mode which may increase in probability with length of tube is tube leakage (due to such things as manufacturing defects in the tube wall), and this is estimated at less than 10% of the total tube-failure rate. The failure rate is also assumed to be independent of tube diameter. Under these assumptions the reliability of the IHX unit can be improved by lengthening the unit and enlarging the tube diameter. The Loop IHX used can be made

TABLE II-16
OPERATION AND MAINTENANCE COST SUMMARY

Equipment	Yearly Costs/Yr (\$)	
	Pool	Loop
Core and primary system instrumentation	20,000	-
Safety monitoring system	-	1,000
Vault inert gas system	-	5,000
Sodium cover-gas systems	-	3,000
Sodium purification systems	10,000	-
Reactor concrete shield coolant systems	4,000	-
Valves	-	25,000
IHX and piping	-	5,000
Sodium heaters	-	6,000
Pumping power	-	11,000
Total	34,000	56,000

approximately 40-ft long without compromising the concept's ability to remove heat from the reactor by free convection. The Pool IHX, on the other hand, is limited in length by the depth of the tank, and for this study was taken to have an effective tube length of about 18 ft. Combined with the requirement for very low pressure drop, this results in a unit with 5000 tubes vs about 1300 for the Loop.

Under the above considerations, it is believed that an IHX for the Loop system will have fewer tube failures than the Pool. The exact magnitude of the reliability difference is, however, uncertain and can be established only during actual reactor operation. AI is continuing engineering studies in IHX optimization.

TABLE II-17
PLANT UNAVAILABILITY SUMMARY

Equipment	Unavailability (%) [*]	
	Pool	Loop
Piping and valves	1.1	0.9
Pumps	0.9	0.9
IHX	2.3	1.6
Primary instruments	0.8	0.4
Auxiliaries	0.2	0.2
Total	5.3	4.0 [†]

^{*} Uncertainty $\pm 4.5\%$

[†] Loop availability advantage = 1.3%

All of the previous cost factors were combined by means of the economic model (see Table II-19), and the differential cost of power is shown in mills/kwhr. While the Pool concept has an advantage in capital cost and operation and maintenance, and the Loop concept in the areas of sodium inventory and unavailability, the net effect is a slight overall economic advantage to the Loop. Nevertheless, sensitivity studies, made because these cost estimates are highly uncertain at this time, showed that even though the economics are in favor of the Loop concept, the potential-error uncertainty is greater. Even a detailed system design and safety analysis of both concepts would not eliminate all uncertainty.

c. Technical Risks

In the economics and safety evaluations, possibility of a number of technical uncertainties exist which could considerably influence the results of this study. These items are referred to as technical risks.

TABLE II-18
AVAILABILITY/RELIABILITY COMPARISON

Factors Favoring Loop Concept	Factors Favoring Pool Concept
<u>Piping System and Valves</u> Simplified repair due to easier access, less radiation. No slip joints to degrade flow and temperature characteristics.	<u>Primary Piping System, Heavy Wall, Including Slip Joints</u> Length of runs and welds, approximately 1/8 that of loop. Leakage threshold approximately 20x higher than loop.
<u>Pumping System, Speed Control, Check Valve</u> Each pump can be isolated for decay. Scroll leakage can be repaired. Simpler and more reliable flow measurements. Operating flowrates are not uncertain due to a possible increase in long-term leakage. Better gamma and neutron shielding can be provided.	Eliminates primary nozzles at vessel. No blocking valves. Fewer instrument wells or welds. Smaller thermal transients. Gas entrainment is less likely to produce bubbles in core. No complex preheat problems.
<u>Heat Exchangers</u> Can be made longer and with fewer tubes. Shell can be repaired. Sodium drain is feasible to facilitate repairs. Better shielding is possible. Each IHX can be isolated for decay and postponement of replacement to scheduled plant shutdown. Similar design in a more controlled environment. Need not have a removable bundle. May have a high primary side ΔP	<u>Pumping System, Speed Control, Check Valves</u> Pump scroll leakage is not critical. Pumps operate in 300°F colder sodium. Simpler pump-bearing overflow provision. Simpler requirement for variable suction-head control. Identical flowpaths through symmetry.
<u>Instrumentation and Monitoring System</u> Fewer thimbles, headers, and wire-runs under hot sodium. Malfunctioning and failed equipment can be located and diagnosed with greater speed and precision. Simplified repair.	<u>Heat Exchangers</u> Shell leakage is not critical. Shell heaters may be eliminated.
<u>Auxiliary Systems</u> Less sodium purification and smaller storage facility. Smaller cover-gas systems. Smaller requirement for vessel sodium heating at startup and restarts.	<u>Auxiliary Systems</u> No primary vault problems (gas, nitrogen cooling, etc.). Preheat system less complex.

TABLE II-19
OVERALL ECONOMIC SUMMARY

Item	Differential Power Costs (mills/kwhr)		
	Loop	Pool	Uncertainty
Direct construction	Base	-0.027	±0.027
Sodium inventory	Base	+0.008	±0.001
Operation and maintenance	Base	-0.003	±0.002
Unavailability	Base	+0.026	±0.090
Total	Base	+0.004	±0.094 (Combined statistically)

The principal technical risks of the Loop concept are the valves, piping, gas problems, and fission-product deposition.

Large low-leakage-blocking valves are required for maintenance operations on this concept. Liquid metal valves in this size are at present not available. For this study it was assumed that both the Pool and Loop require throttling and check valves.

Large (48-in.) piping systems must be developed for the Loop. Problems here can include piping expansion compensation, piping support, steady-state and transient stress compensation, inspection and surveillance, and weld reliability.

The extensive piping length in the Loop system provides opportunities for gas-collection pockets and sources of gas entrainment. The relatively high velocities exiting from core and the small vessel may not be adequate to de-entrain small gas bubbles. Running vents at high spots in the piping system are an additional source of complexity and potential problems. Gas collection in the high spots could conceivably prevent natural circulation flow.

Failed fuel elements and long-term corrosion of the primary system can cause deposition of radioactive products in the primary-loop piping system. Although local shielding for maintenance is provided in the vaults, if radioactivity gets too high maintenance operations may be precluded. Because little is understood on deposition of fission and corrosion products, this can represent a significant penalty to the Loop concept.

The principal technical risks of the Pool concept are instrumentation, pumps, IHX's, slip joints, reactor vessel, gas solubility, temperature distribution and thermal stress, and bellows.

Primary system flowmeters, pressure transducers, and neutron-flux instrumentation must be immersed in the pool. Whether the instrumentation is designed to be in cooled thimbles or to withstand high-temperature immersion, considerable uncertainties and penalties may be incurred.

The centrifugal sodium pumps for this system are beyond current pump sizes. Because pumps immersed in the pool must be as small in diameter as possible, readily removable from the pool, and not require increasing the vessel size, design compromises are likely. Also, the pump is more closely thermally coupled to other pool components. The pump casing and bearings may be more subject to distortion, misalignment, and thermal shock because of immersion in the pool. The pump environment is more controllable and predictable in the Loop concept.

Pool-type IHX's must have a very low pressure drop on the shell side, the complete bundle must be removable for repair, the height must be limited because of immersion in the pool, and the shell must be insulated from the Na pool by some type of insulation not yet proven. Integrating all these factors into an adequate design leaves many uncertainties about tube-and-shell reliability, tube-bundle bypass flows, thermal shock problems, and temperature stratification (particularly at low flow). Present availability studies show that the reliability of the IHX is one of the key factors in meeting availability objectives of the plant.

Slip joints provide piping expansion in the reference Pool design. They must not self-weld or bind, and the leakage must not be excessive or produce cavitation damage. Such slip joints have not been developed and it may be possible to engineer a Pool system without them.

Site fabrication of the 50-ft diam vessel for this concept may be difficult, may introduce quality assurance problems, and turn out to be more expensive than anticipated.

Cover-gas solubility problems may occur because of gas contact with the Na surface in the Pool. This situation can cause gas to be liberated in the cold leg, possibly de-priming the pump, or be carried through the core.

The flow patterns of hot and cold Na in this concept are difficult to predict and may cause problems with thermal gradients, equipment distortion, fuel-handling misalignment, and temperature stratification in the Na pool.

A large bellows closure of the bottom-supported reactor vessel is required. Such a bellows was used on the Hallam Nuclear Power Facility but none has been produced of the required size for the Pool concept. It is possible to engineer a top-supported vessel for this concept which does not require a large bellows. Trades are required in this area.

4. Conclusions

In consideration of the safety, economics and technical risks discussed previously, AI has chosen the Loop concept for the following reasons:

- 1) The economics, while highly uncertain at this time, appear to favor a Pool concept slightly. A more detailed design could reduce some of the uncertainties, but only building and operating two types of plants will ever be able to effectively prove any economic advantage of either concept.
- 2) Neither concept appears to have a strong and over-riding safety advantage; both have other advantages and disadvantages. Both designs seem capable of meeting safety criteria and are licenseable.

- 3) From the evaluation of the basic technical problems, it appears to AI that there are more risks and uncertainties in the Pool concept. AI has liquid-metal experience in the design of Loop-type reactors, providing higher confidence of first-plant success for this concept.
- 4) The Loop system offers a high degree of primary-coolant component decoupling. The strong coupling in the Pool concept results from the compact packaging arrangement and strong thermal hydraulic and structural interactions between components, further complicated by the difficulty of accurately predicting thermal gradients and flow patterns. To establish a quantitative economic advantage for this decoupling in the Loop is difficult, but it simplifies the design, allows more realistic representation of the environment in component tests, and enhances the ease of plant improvement and modifications for early plants.
- 5) The Loop concept has the eventual capability to eliminate the secondary-sodium circuit, thus significantly reducing plant capital cost.
- 6) Even though AI has chosen the Loop concept, it believes that both concepts appear to offer acceptable configurations for ultimate FBR plants.

C. NUMBER OF HEAT-TRANSFER CIRCUITS

1. Summary

Heat-transfer systems for a 1000-Mwe LMFBR plant consisting of two, three, and four sodium-coolant circuits were compared on the basis of expected capital cost and plant availability. A two-circuit plant is estimated to cost less than three by approximately \$2,800,000, and less than a four-circuit plant by about \$5,100,000. A three-circuit type has an availability of approximately 1% better than a two- and 0.3% better than a four-circuit. The availability advantage of a three-circuit is insufficient, however, to justify the additional capital cost; therefore, a two-circuit is the economic choice. Equipment sizes in such a plant are larger than those for the demonstration plant; hence the risks associated with components of larger size and with

pipe rupture accidents are greater. On the basis of these risks the selection of a three-circuit configuration for the 1000-Mwe base plant was made.

2. Systems Studied

The alternatives in this study were three types of heat-transfer systems for a 1000-Mwe sodium-cooled reactor, the differences being in the number of heat-transfer circuits used to remove heat from the reactor and generate steam. Two-, three-, and four-circuit systems were studied, each having a primary, and secondary loop. The primary consists of piping, two block valves to isolate the loop from the reactor, a throttle valve for flow control, a free-surface centrifugal pump, and two IHX units. The secondary consisted of piping, a free-surface centrifugal pump mounted in an expansion tank, a steam-generating system consisting of evaporators, superheaters, and reheaters, a block valve, a combination block/throttle valve, and a reheater sodium throttle valve. The flowrates per loop and approximate piping sizes are given in Table II-20.

TABLE II-20
LOOP FLOWRATES AND PIPING SIZES

Number of Circuits	Primary Flow (gpm)	Secondary Flow (gpm)	Primary Pump Suction Line (in.)	Secondary Pump Suction & Discharge Lines (in.)	Primary and Secondary Return Lines (in.)
4	57,000	65,000	42	38	36
3	76,000	87,000	48	44	42
2	113,000	130,000	58	54	52

In the case of the four- and three-circuit systems, a failure in one circuit (e. g. a pump-drive failure) does not prevent continued reactor operation at reduced load using the remaining circuits. If access to the pipe vaults is required, the affected circuit can be shut down, and the primary loop isolated from the reactor by closing the block valves. After decay of Na-24 which requires about 10 days, the reactor can be shut down and the vault entered for inspection and repairs.

In a two-circuit plant both circuits must be in operation. If one circuit must be shut down for repairs the entire plant must be shut down. This approach is based on the philosophy that there must always be one reserve heat-transfer circuit available for decay-heat removal.

3. Analysis and Results

The two-, three-, and four-circuit heat-transfer systems were compared by estimating system cost and plant availability differentials associated with them. Cost favors the two-circuit concept because overall heat-transfer system costs decrease (although not strongly) with the number of circuits, and because a decrease in reactor building size is expected to accompany a reduction in the number of heat transfer-circuits. It is assumed that a reduction of 20 ft in building diameter would accompany a reduction from three to two in the number of circuits, and that a 10-ft increase would accompany an increase from three to four in the number of circuits. Table II-21 gives a cost summary.

TABLE II-21
HEAT-TRANSFER SYSTEM DIFFERENTIAL COSTS

Number of Circuits	Heat-Transfer System Cost Differential (\$)	Building Cost Differential (\$)	Total Cost Differential (\$)
4	$+1.9 \times 10^6$	$+0.4 \times 10^6$	2.3×10^6
3	Base	Base	Base
2	-2.0×10^6	-0.8×10^6	-2.8×10^6

The preceding table indicates that a savings of approximately \$2,800,000 can be made in a 1000-Mwe FBR by reducing the number of heat transfer-circuits from three to two. Even if building size cannot be decreased, the savings would still be about \$2,000,000.

Plant unavailability due to forced outages of the heat-transfer system equipment was also estimated. As would be expected, the two-circuit plant suffers a greater unavailability than either of the others, in spite of the fewer number

of critical components, because of the requirement that the entire plant be shut down if one loop shuts down. A summary of estimated heat-transfer system unavailabilities is shown in Table II-22.

TABLE II-22
HEAT-TRANSFER SYSTEM UNAVAILIBILITIES BASED ON NUMBER
OF CIRCUITS

Number of Circuits	Unavailability	Differential Unavailability
4	0.062	+0.003
3	0.059	Base
2	0.069	+0.010

Assuming that the cost of unavailability is 2 mills/kwhr, the cost comparison in Table II-23 can be made.

TABLE II-23
ECONOMIC COMPARISON MATRIX (mills/kwh)

Parameter	Multiplier	2-Circuit	3-Circuit	4-Circuit
Capital cost	2.564×10^{-8}	-0.077	Base	+0.063
Unavailability	2.000	+0.020	Base	+0.006
Total		-0.057	Base	+0.069

Thus a two-circuit 1000-Mwe plant is estimated to have 0.057 mills/kwh lower overall energy cost than the reference three-circuit plant.

The pumps, valves, and piping for a two-circuit 1000-Mwe plant are very large, compared to those expected to be needed by the mid-1970's. The approximate capacities and sizes are compared in Table II-24.

It is anticipated that the design and fabrication of the pumps, valves, and piping for the 1000-Mwe plant will be based primarily on the experience from early operation of FFTF and from the first demonstration plant. (Six months of operating experience from a three-loop demonstration plant is equivalent to three years of operation of a single pump in a test loop). It is further believed

TABLE II-24

1000-Mwe vs DEMONSTRATION-PLANT EQUIPMENT SIZES

	350-Mwe Plant (3- circuits)	500-Mwe Demonstration Plant, Mid-1970's (3-circuit)	1000-Mwe Plant(3- circuits)	1000-Mwe Plant(2- circuits)
Primary-pump flow- rate, gpm	27,000	42,000	80,000	120,000
Primary pump suction Line Diam, in.	30	36	48	58

that to maintain an acceptable level of risk, further extrapolation of component sizes would have to be within a factor of two of the then-proven designs.

The AI demonstration plant is a three-loop plant. The use of three loops for a 1000-Mwe plant would provide for maximum similitude between the two plants on thermal hydraulics, piping stresses, and plant control. This should minimize many potential design, construction, and operational difficulties for the larger plant.

In the unlikely event of a pipe rupture, the rate at which the reactor vessel Na level is lowered increases with increasing primary-loop flowrate. Such accident can probably be handled in a three-circuit 1000 Mwe-plant by braking the pumps immediately upon loss of level in the vessel. Further study will be required to determine whether this procedure is feasible for two-circuit cases.

Although the preceding analysis indicates an economic advantage to the two-circuit concept, three circuits were retained in the design for the 1000-Mwe plant primarily because of the problems of extrapolation of equipment size and of sheared-pipe accident uncertainties.

D. FUEL-HANDLING SYSTEM

1. Introduction

A variety of fuel-handling concepts have been considered for FBR systems, the main differences being in the manner in which spent fuel is removed, stored, and replaced in the reactor. Major types considered were rotatable or fixed plugs with either fixed or articulating-arm-type fuel-handling machines, and a hot cell over the vessel within which all fuel handling takes place.

Because special attention must be given to removing spent fuel from the reactor to prevent overheating of the elements, in-vessel storage of spent elements is usually provided to allow for decay before their removal from the Na coolant pool. However, to avoid a significant inventory cost for excessive storage time this fuel must be removed from the reactor as soon as possible. Thus provisions are made where possible to allow "on-line" fuel removal from the in-vessel storage area for reprocessing as soon as possible. Safety considerations, however, do not always allow this on-line spent-fuel removal during operation.

The study discussed here was performed as part of the Company-sponsored 500-Mwe Demonstration Plant design effort but has been updated to reflect the trade-offs for a 1000-Mwe plant. Although the study was started with the assumption that on-line removal from in-vessel storage could be accomplished safely, the subsequent safety evaluation required off-line removal until the on-line method could be shown to be safe under the loads imposed by the DBA. Follow-on studies directed at choosing the best method of achieving the off-line refueling with the selected concept led to the selection of the present reference approach. The original comparison was not repeated for off-line refueling, and the increased inventory charge due to the longer in-vessel storage period could significantly change the overall economic comparison. However, methods of eliminating the need for in-vessel storage are now under investigation. Ex-vessel storage would eliminate any excessive fuel-storage inventory charges associated with the off-line case and reconfirm the reference design selected.

2. Requirements and Assumptions

The principal requirements for the fuel-handling system are:

- 1) Remove spent-fuel and blanket elements from the reactor and place in spent-fuel storage.
- 2) Place new-fuel and blanket elements in the reactor core.
- 3) Maintain the temperature of the elements within acceptable limits during fuel handling.

These operations must be done without endangering the safety of plant personnel or the general public. Operations must also be done without damage to the fuel elements.

The basic assumptions in the initial on-line refueling study were:

- 1) A complete refueling cycle consists of handling a prescribed fraction of core and blanket loadings.
- 2) An in-vessel handling machine requires minimum shielding. An ex-vessel fuel-handling machine requires heavy shielding.
- 3) The reactor will be on-line during transfer of spent fuel out of and new fuel into in-vessel storage.
- 4) The ex-vessel fuel-handling machine will not compromise the ability of the system to handle the DBA.
- 5) No backup reactivity control system is required.
- 6) The reactor will be shut down for transfer of fuel between the reactor and in-vessel storage.
- 7) Out-of-reactor fuel transfer requires sodium cooling.
- 8) Out-of-storage fuel transfer requires gas cooling.
- 9) Offsite shipment or reprocessing of fuel elements is not a part of this study.

4-24-68 UNCL

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Figure II-5. Double-Rotating Plug with In-Vessel Fuel Storage

3. Description of Alternates

Seven different design concepts were evaluated, some with in-vessel spent-fuel storage and some with ex-vessel storage. A brief description of these is given in this section. Only on-line removal of fuel was considered in cases where in-vessel storage was used.

Concept I: Double-Rotating Plug with In-Vessel Fuel Storage. The top shield of the reactor vessel consists of two large rotating plugs, shown schematically in Figure II-5, each designed to rotate 360° . All in-core fuel handling is accomplished through two penetrations in the smaller plug, and access to the fuel-storage thimbles is through penetrations near the periphery of the larger plug. Spent-fuel elements can be removed from the storage thimbles and replaced with fresh elements with the reactor on-line. Before refueling can proceed, the control-rod actuators and instrument "trees" must be elevated to permit rotation of the plugs.

Concept II: Double-Rotating Plug with Out-of-Vessel Fuel Storage. The top shield of the reactor vessel, Figure II-6, consists of two large rotating plugs, similar to Concept I except that for the same-size core the reactor vessel is smaller because there is no in-vessel storage. A heavily shielded liquid-sodium-cooled fuel-handling machine is used to transfer spent-fuel elements directly to ex-vessel storage. The fuel-handling machine is supported on a gantry and must be indexed over the fuel-handling port each time an element is transferred from or to the reactor. The fuel-handling system must handle the shield plug, new fuel, and spent fuel. Since this fuel-handling machine will be large, the control-rod drive units must be removed from the reactor cover to provide working space for the machine.

Concept III: Single-Rotating Plug with a Rotating-Arm Fuel-Handling Machine and In-Vessel Fuel Storage. The top shield of the reactor vessel is a single, large rotating plug. All in-core fuel handling is accomplished by a rotating-arm in-vessel handling machine mounted under the plug as shown in Figure II-7. The arm and the plug are each designed to rotate a total of 360° ; the fuel-storage thimbles are serviced through penetrations near the periphery of the plug. Spent-fuel elements can be transferred from,

4-24-68 UNCL

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Figure II-6. Double-Rotating Plug with Out-of-Vessel Fuel Storage

4-24-68 UNCL

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Figure II-7. Single-Rotating Plug with Rotating-Arm Fuel-Handling Machine and In-Vessel Fuel Storage

and fresh elements transferred to, in-vessel storage with the reactor on-line. Before refueling, the control-rod actuators and all in-core instrumentation must be raised to the underside of the reactor cover to provide working space for the fuel-handling arm. This avoids interference of the handling arm with the control-rod actuators and instrumentation, which are mounted on the plug and project downward from it.

Concept IV: Fixed Plug with Three Rotating-Articulating-Arm Fuel-Handling Machines and In-Vessel Fuel Storage. The top shield of the reactor is a large fixed plug. All in-core fuel handling is accomplished by three articulating-arm fuel-handling machines installed 120° apart in plug penetrations, as shown in Figure II-8. These machines provide radial movement in addition to rotation, and also raise and lower fuel elements into reactor and in-vessel storage positions. The machines can be folded up into their central structure for removal from the reactor. The fuel-storage thimbles are serviced through penetrations near the periphery of the plug. Spent-fuel elements can be transferred from reactor storage and be replaced with fresh elements while the reactor is on-line. Before refueling, the control-rod actuators and all in-core instrumentation must be raised to the under side of the plug to provide working space for the articulating arms.

Concept V: Removable Plug in Hot Cell with Out-of-Vessel Fuel Storage. The top shield of the reactor vessel is a removable plug which must be removed for fuel handling. A liquid-sodium-cooled fuel-handling machine mounted on a gantry, as shown in Figure II-9, transfers spent-fuel elements to ex-vessel storage. The entire system is enclosed in a shielded gastight cell containing an inert gas. The fuel-handling machine is simpler than that required for Concept II. It has no small shield plug nor seal to the reactor cover; the control-rod drives and in-core instrumentation will be removed as a unit with the reactor cover.

Concept VI: Single-Rotating Plug with a Single Nonrotating-Articulating-Arm Fuel-Handling Machine and Ex-Vessel Fuel Storage. The top shield of the reactor vessel is a single large rotating plug. All in-core fuel handling is accomplished by a nonrotating-articulating-arm fuel-handling machine installed through a penetration in the plug, as shown in Figure II-10.

4-24-68 UNCL

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Figure II-8. Fixed Plug with Three Rotating-Articulating-Arm Fuel-Handling Machines and In-Vessel Fuel Storage

4-24-68 UNCL

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Figure II-9. Removable Plug in Hot Cell with Out-of-Vessel Fuel Storage

4-24-68 UNCL

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Figure II-10. Single Rotating Plug with a Single Nonrotating-Articulating-Arm Fuel-Handling Machine and In-Vessel Fuel Storage

The plug is designed to rotate a total of 300 degrees. A combination of plug rotation and radial movement of the articulating arm allows access to any fuel element or storage cell. No interference exists between the articulating arm and the control rods because both are in a fixed position relative to the rotating plug. The articulating arm can be folded into its central structure for removal from the reactor. The fuel-storage thimbles are serviced through penetrations near the periphery of the plug. Spent-fuel elements can be transferred from in-vessel storage and replaced with fresh elements while the reactor is on-line. Before refueling, the control actuators and instrument trees must be elevated to rotate the plug.

Concept VII: The Hot-Cell Fuel Handling. The top shield of the reactor vessel is a removable plug, as shown in Figure II-11. The control-rod drives and in-core instrumentation are removable as a unit with the reactor cover. The entire fuel-handling system is enclosed in a shielded, heated gas-tight cell containing a suitable inert gas. Fuel handling is done as follows:

- 1) The plug is removed from the reactor;
- 2) The sodium level is lowered until tops of the fuel elements are visible;
- 3) Extensions are attached to the fuel elements to be transferred;
- 4) The sodium level is then raised; and
- 5) A grappling mechanism transfers the spent-fuel elements through Na to in-vessel storage.

A "lazy-susan" in-vessel storage concept is postulated. Ex-vessel transfer of fuel elements to and from in-vessel storage is through a loading port which is unrelated to the top shield. The principal difference between this concept and Concept V is that no wet-fuel-handling machine is required here. Manipulation of the reactor Na level provides sufficient depth of primary Na to cool the spent fuel during transfer.

4. Analysis and Comparison of Alternates

Following a rough comparison of all seven fuel-handling schemes for safety, availability, and economics, Concepts II, III, and V were eliminated from further consideration.

4-24-68 UNCL

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Figure II-11. Removable Plug in Hot Cell with In-Vessel Fuel Storage

a. Economic Evaluation

The economic evaluation of the other four was based on the trade-offs of capital cost and availability. These estimates for the 1000-Mwe plant are given in Table II-25.

The equivalent energy costs were combined using the economic model discussed previously. The capital costs and availability estimates are extrapolations from the Company-sponsored 500-Mwe plant study. The double-rotating plug concept is the best choice from overall economic considerations. Although having lower capital costs, the fixed- and rotating-plug with-articulating-arms concepts have a much poorer availability. Both capital cost and unavailability are higher for the hot-cell concept than the base case. As a result of this evaluation the double-rotating plug was selected as the reference fuel-handling concept. A subsequent design study reduced the number of small plug penetrations in the reference design from two to one by an adjustment in the relative diameters of the two plugs.

TABLE II-25.

CAPITAL COST AND AVAILABILITY EVALUATION, 1000-Mwe PLANT

Concept	Capital Cost (\$)	Equivalent Energy Cost (mills/kwh)	Unavail- ability	Equivalent Energy Cost (mills/kwh)	Differential Energy Cost (mills/kwh)
I. Double-Rotating Plug	base	base	base	base	base
IV. Fixed Plug, Three Articulating Arms	-955,000	-0.027	+0.030	+0.060	+0.033
VI. Single-Rotating Plug, One Articulating Arm	-700,000	-0.019	+0.037	+0.074	+0.055
VII. Hot Cell	+480,000	+0.013	+0.003	+0.006	+0.019

b. Safety Considerations

Based on early estimates of the DBA loads on the top shield, it appeared that no problem would occur in providing a tight top-shield hold-down structure to protect against significant leakage during this accident. After economics-based selection of the reference design, estimates for the DBA loads increased to where the ability to hold the leaktightness of the top shield became questionable. This led to the provision of a backup containment dome over the top shield during reactor operation, requiring the ex-vessel transfer of spent and new fuel to take place with the reactor off-line. The major effect of this on the economics is the fuel-inventory penalty in leaving the fuel in the reactor vessel for the next six-month burnup cycle rather than the minimum (~ two-month) decay required for normal ex-vessel fuel transfer. The difference would be eliminated however if the ex-vessel machine were redesigned to handle fuel directly, eliminating the need for in-vessel storage. This approach, now under evaluation, appears feasible.

c. Evaluation of Ex-Vessel Fuel-Handling Schemes.

Because with off-line refueling, it is extremely important to minimize reactor downtime, a trade study was made which compared thirteen possible ex-vessel fuel-handling schemes. These differed in the number of chambers in the fuel-handling machine, type of ex-vessel fuel storage used, and the number of floor valves. The specific cases considered were as follows:

Case 1: One-chamber ex-vessel fuel-handling machine;
Stationary ex-vessel fuel-storage facility;
One floor valve at the reactor fuel-transfer port; and
One floor valve at the fuel-storage facility.

Comment: This is a near-minimum-equipment and
presumably near-minimum-cost system.

Case 2: One-chamber ex-vessel fuel-handling machine;
Stationary ex-vessel fuel-storage facility;
One floor valve at the reactor storage facility; and
Two floor valves at the fuel-storage facility.

Comment: The additional floor valve considerably reduces the amount of time that the ex-vessel machine is required to spend at the fuel-storage facility. If a third floor valve is used, the time saving is clearly greater if used at the storage facility than at the reactor. Thus the case of two valves at the reactor and one at the storage facility can be disregarded.

Case 3: One-chamber ex-vessel fuel-handling machine;
External rotary ex-vessel fuel-storage facility;
One floor valve at the reactor fuel-transfer port; and
One floor valve at the fuel-storage facility.

Comment: The use of an external rotary fuel-storage facility reduces the time for preparations at the storage facility before the ex-vessel machine arrives there. A second floor valve at either location would eliminate the necessity to move the rotary shields but would require that the machine be moved. The time difference would be so small, however, that the use of the second valve would be uneconomical.

Case 4: One-chamber ex-vessel fuel-handling machine;
Internal rotary ex-vessel fuel-storage facility;
One floor valve at the reactor fuel-transfer port; and
One floor valve at the fuel-storage facility.

Comment: The internal rotary design eliminates the need to move the ex-vessel machine at the storage facility.

Case 5: Two-chamber ex-vessel fuel-handling machine;
Stationary ex-vessel fuel-storage facility;
No floor valve at the reactor fuel-transfer port; and
No floor valve at the fuel-storage facility.

Comment: This is the minimum-equipment system using a two-chamber ex-vessel fuel-handling machine.

Case 6: Two-chamber ex-vessel fuel-handling machine;
Stationary ex-vessel fuel-storage system;
One floor valve at the reactor fuel-transfer port; and
No floor valve at the fuel-storage facility.

Comment: A floor valve at the reactor eliminates the need to move the ex-vessel machine. A single floor valve provides greater time saving if the valve is at the reactor than at the storage facility because of the differences in vertical hoisting distances; thus the case of one valve at the storage facility and none at the reactor may be disregarded.

Case 7: Two-chamber ex-vessel fuel-handling machine;
Stationary ex-vessel fuel-storage system;
One floor valve at the reactor fuel-transfer port; and
One floor valve at the fuel-storage facility.

Comment: A floor valve at the storage facility eliminates the need to move the ex-vessel machine.

Case 8: Two-chamber ex-vessel fuel-handling machine;
External-rotary ex-vessel fuel-storage facility;
No floor valve at the reactor fuel-transfer port; and
No floor valve at the fuel-storage facility.

Comment: This is essentially the same as Case 5 for fuel-handling cycle time. The differences are primarily in the initial construction cost, i. e., a stationary facility with many individual shield plugs vs a rotary one with only a few.

Case 9: Two-chamber ex-vessel fuel-handling machine;
External rotary ex-vessel fuel-storage facility;
One floor valve at the reactor fuel-transfer port; and
One floor valve at the fuel-storage facility.

Comment: Floor valves make it unnecessary to move either the ex-vessel machine or the rotating shields during fuel exchanges. If the use of a floor valve is economical at one location it will be economical at both; thus, the cases of one valve at one location and none at the other may be disregarded.

Case 10: Two-chamber ex-vessel fuel-handling machine;
Internal rotary ex-vessel fuel-storage facility;
No floor valve at the reactor fuel-transfer port; and
No floor valve at the fuel-storage facility.

Comment: The internal rotary fuel-storage facility permits fuel exchanges to be made at one setting without a floor valve.

Case 11: Two-chamber ex-vessel fuel-handling machine;
Internal-rotary ex-vessel fuel-storage facility;
One floor valve at the reactor fuel-transfer port; and
No floor valve at the fuel-storage facility.

Comment: A floor valve at the reactor makes it possible for fuel exchanges to be made at one setting. A single valve at either the reactor or the storage facility provides greater benefit in the former case, where a move is avoided; in the latter case only the removal and re-insertion of a shield plug is avoided.

Case 12: Two-chamber ex-vessel fuel-handling machine;
Internal rotary ex-vessel fuel-storage facility;
One floor valve at the reactor fuel-transfer port; and
One floor valve at the fuel-storage facility.

Comment: The addition of a floor valve at the fuel-storage facility makes fuel exchanges possible without the necessity for removing and re-inserting a shield plug.

Case 13: Three-chamber ex-vessel fuel-handling machine;
Any type ex-vessel fuel-storage facility;
No floor valve at the reactor fuel-transfer port; and
No floor valve at the fuel-storage facility.

From the standpoint of capital cost alone the least expensive scheme would be composed of:

A single-chamber fuel-handling machine;
A fixed-top ex-vessel storage tank with individual shield plugs for each storage position;
One reactor valve; and
One storage valve.

However, when the availability penalties are included by means of the economic model, the minimum-energy-cost system results with the following combination:

A single-chamber fuel-handling machine;
A rotating-top-shield ex-vessel storage tank;
One reactor valve; and
One storage valve.

This was selected as the reference approach.

5. Conclusions

Based on an economic evaluation of a number of fuel-handling schemes, the reference selected was the double-rotating plug with in-vessel spent-fuel storage.

A reassessment of the safety aspects of top-shield hold-down during the DBA led to the inclusion of a reactor containment dome over the top shield, and the necessity to change to an off-line plan for transfer of spent fuel from in-vessel to ex-vessel storage.

An evaluation of the most economic approach to off-line in-vessel to ex-vessel transfer led to the selection of a single-chamber ex-vessel machine, a rotating top shield on the ex-vessel storage tank, and a single floor valve each for the reactor and ex-vessel-storage shield plugs.

Future possible perturbations to the fuel-handling design criteria, i. e. , elimination of the need for a top-shield containment dome, and direct spent-fuel removal from in-vessel storage, other than allowing on-line spent-fuel removal, should not change the economic conclusions of the basic reference system selected.

E. VENTED vs NONVENTED FUEL ELEMENTS

1. Summary

The vented versus nonvented fuel-element trade study considered two basic concepts, namely, venting of fission gases to the coolant and trapping of fission gases within an integral plenum. For vented fuel elements, two values of reactor cover-gas leakage through the top shield and pump seals ($1 \text{ ft}^3/\text{day}$ and $10 \text{ ft}^3/\text{day}$) were also considered.

The results of the study are summarized in Table II-26.

TABLE II-26
RESULTS OF TRADE STUDY ON VENTED vs NONVENTED
FUEL ELEMENTS

Fuel Element Type and Use	Net Cost Difference (mills/kwh)*†	
	Top-Shield Leakage	
	at $1 \text{ ft}^3/\text{day}$	at $10 \text{ ft}^3/\text{day}$
Vented	- 0.036	+ 0.072
Nonvented or Vented, Using Vented After First 10 Years of Operation	+ 0.011	+ 0.119
Nonvented or Vented, Using Vented	- 0.026	+ 0.082

*Relative to plant designed for and operated with nonvented fuel elements.

†Cost increase = +
Savings = -

A summary of the risks and gains in designing a 1000-Mwe plant for vented fuel elements is given below. An R&D program is required to reduce these risks to acceptable values.

- Risks:
- (1) Leakage to building exceeding 1 ft³/day.
 - (2) Plugging of hypodermic vent.
 - (3) Delay in schedule due to safeguards report.
 - (4) Reduction in burnup due to thinner cladding.
 - (5) Fission-gas entrainment.
 - (6) Fission products plate out with ruptured fuel elements.

Gain: (1) Lower Fuel-cycle costs.

As Table II-26 shows, there is an incentive of 0.036 mills/kwh to design for and operate the plant with vented fuel elements, based on the leakage through the top shield of 1 ft³/day or less and a proven vented fuel-element design. Without an R&D program for the latter, total risk would exceed the incentive.

Table II-26 shows likewise that there is no incentive to design the plant for both nonvented and vented fuel elements, using the latter after ten years. Whether the additional fission-gas reprocessing equipment is installed initially or after ten years does not affect this fact.

The flexibility of the third-alternative is obtained by an initial expenditure of \$300,000 in building and structure construction over that required for a plant designed for vented fuel elements (see Section E-6, following) and \$100,000 for additional building space over that required for a plant designed for nonvented. The \$100,000 value is obtained by assuming that if the final decision in the plant design is to use nonvented, the additional fission-gas reprocessing equipment for vented elements would not be used and the only loss would be that related to the additional building space.

In consideration of the above the 1000-Mwe LMFBR reference plant was designed for both nonvented and vented fuel elements. Adequate building space was allocated for the additional fission-gas reprocessing equipment required by the vented. The reference design uses nonvented elements and only the fission-gas reprocessing equipment necessary for nonvented.

2. Introduction

In the design of fuel elements, one of the major considerations is the accommodation of the fission gas generated and released. The usual approach is to contain it by allowing the pressure to build up within the fuel cladding. This buildup then requires either thicker cladding or a large fission-gas plenum, which imposes penalties on the core. An alternative is to vent the fission gas to the sodium and then accumulate the released gas in a reprocessing system, which allows thinner cladding and shorter fission-gas plenums. However, penalties are incurred due to the need for a larger reactor core cover-gas reprocessing system.

3. System Requirements

Production of energy at specified rates and conditions is common to both concepts. The nonvented elements must contain the solid and gaseous fission products. The vented elements must:

- 1) Release fission-product gases to the sodium stream.
- 2) Contain solid fission products within the cladding.
- 3) Exclude sodium from the fuel region under normal or transient conditions.

4. Systems Studied

The basic system was a 217 nonvented pin hexagonal array with 0.25-in. OD rods. The vented design was based on using the "diving bell" concept. Since one of the important items is the fission-gas reprocessing equipment cost, which is strongly related to the reactor cover-gas leakage through the top shield, two values of top-shield leakage were considered.

5. Ground Rules of Study

The ground rules in the study were as follows:

- 1) Plant size was 1000 Mwe.
- 2) A loop-type primary system was assumed.
- 3) A regular geometry core configuration was assumed.

- 4) The containment dome over the top shield was not considered, since leakage into the dome would still be an important factor.
- 5) The fuel was $\text{PuO}_2\text{-UO}_2$ with an assumed fission-gas release of 35% for the vented fuel element and 70% for the nonvented.
- 6) The nonvented fuel element has a 64-in. gas-plenum length.
- 7) The vented fuel-element design was identical to the nonvented except that the gas plenum was vented to the Na stream and its length was 24 in. This length was based on the maximum Na pressure variation adjacent to the vent.
- 8) The top-shield gas leakage was assumed to be $1 \text{ ft}^3/\text{day}$ and $10 \text{ ft}^3/\text{day}$. It was felt this covered the leakage range which might be expected.
- 9) The safety regulations complied with 10 CFR 20 and the AEC General Design Criteria (10 CFR 50).
- 10) The vented fuel-element design used in the evaluation was the diving bell concept since this was felt to be representative.

Evaluation of the various concepts was based on, first, the use of an economic model to determine the value of the benefits or penalties, and second, a qualitative evaluation of the risks associated with the various concepts.

6. Analyses and Results

To arrive at a recommendation for preference of vented or nonvented design, some basis had to be established. It was desirable that this be on economics vs risks rather than on intuitive reasons such as improvement in fuel-cladding reliability. In an attempt to establish this basis, the important considerations were determined in a qualitative manner (see Table II-27), evaluated on an economics basis where possible, and summarized (see Tables II-28 through -30).

In the economic model for the 1000-Mwe plant (discussed earlier) the cost of money was assumed at 7% and the reactor operationg period at 30 years.

TABLE II-27
IMPORTANT CONSIDERATIONS FOR VENTED AND NONVENTED
FUEL ELEMENTS

Item	Qualitative Effect	
	Vented	Nonvented
<u>Capital Costs</u>		
Reactor cover-gas reprocessing System		
Leakage 1 ft ³ /day	Higher.	--
Leakage 10 ft ³ /day	Much higher.	--
Fuel handling	--	--
Shipping casks	Safety question due to loss of one barrier.	Longer; high-pressure fission-product gases.
Core construction	--	Increase in core height results in higher capital costs.
Fuel storage	--	--
Failed-element location system	More difficult to detect failed elements.	--
Pumps	--	Higher pumping power because of greater ΔP .
<u>Fuel Costs</u>		
Fuel-element fabrication (core and blanket)	Possibly higher.	
Burnup potential	Not limited by fission-gas release.	Allowance for fission-gas results in higher capital costs.
Specific power		Lower due to cladding ΔT and temperature.
Shipping	--	Higher costs due to larger casks.
Reprocessing	--	Higher due to contained fission-product gases.
<u>Plant Operation and Maintenance</u>		
Loop accessibility due to contamination of Na loops	In the event of fuel-cladding failure, the probability of fission products washing out of fuel pins would be higher.	Greater probability of fuel-cladding failure.
<u>Fuel-Handling Plant Availability</u>		
Fuel handling (transfer time of fuel elements)	--	Larger due to longer fuel elements. Less stringent for fuel handling.
<u>Plant Efficiency</u>		
Pumping power loss	--	Higher pumping power; temperature and ΔT limited due to combined thermal and pressure stresses.
Sodium temperature		
<u>Fuel-Failure Modes</u>	Plugging of hypodermic vent.	Defect in cladding; combined thermal and pressure stresses; gas blanketing; high temperature.
<u>Safety</u>	Loss of one barrier; larger fission-gas de-entrainment system.	Cladding failure could cause extensive core damage.
<u>R&D Engineering, Etc.</u>	Capsule tests; prototype element tests; fission-gas cleanup; hazard report.	

TABLE II-28

ECONOMIC INCENTIVE FOR VENTED FUEL ELEMENTS

Item	Incremental Costs (mills/kwh)*†	
	Top-Shield Leakage	
	at 1 ft ³ /day	at 10 ft ³ /day
Fission-Gas Reprocessing-Equipment Cost	+ 0.040	+ 0.148
Building and Structure Construction (decrease in fuel-element height)	- 0.010	- 0.010
Fuel-Cycle Cost (decrease in fuel-cladding thickness)	- 0.050	- 0.050
Fuel Handling (decrease in fuel-element height)	- 0.016	- 0.016
Net Difference	- 0.036	+ 0.072

* Relative to plant designed for and operated with nonvented fuel elements

† Cost increase = +
Savings = -

The variation in direct construction costs was based on those for the fission-gas reprocessing equipment and changes in plant construction with changes in core size. The reactor cover-gas reprocessing equipment cost for a 1000-Mwe plant was estimated as follows:

- 1) Nonvented fuel elements (1% ruptured), \$500,000.
- 2) Vented fuel elements (1 ft³/day leakage through top shield)
\$1,970,000.
- 3) Vented fuel elements (10 ft³/day leakage through top shield)
\$5,900,000.

TABLE II-29

RISKS AND ADVANTAGES OF USING VENTED FUEL ELEMENTS BEFORE
COMPLETION OF THE R&D PROGRAM

Risk	Penalty	Probability*
Top-shield leakage exceeding 1 ft ³ /day	Would require installation of additional cover-gas reprocessing equipment.	M
Plugging of hypodermic vent	Would require design change and modification of all fuel elements.	L
Delay in schedule to safeguards report	Estimated delay 180 days.	M
Reduction in burnup due to thinner cladding	Possible reduction of about 25% in burnup.	M
Fission-gas entrainment	Would require system modifications, estimated delay of about 180 days.	L
Fission products plate out with ruptured fuel elements	Would require acid cleaning of IHX.	L
Advantage	Benefit	Probability*
Increase in burnup	Burnup increased from 75,000 Mwd/Mt average to 100,000 Mwd/Mt average.	L

* M = medium probability, L = large. These are judgment determinations obtained from estimates of knowledgeable personnel and should be considered only as indicative of the amount of risk.

The reactor structure and building construction cost variation was estimated as follows:

- 1) A decrease in fuel pin gas-plenum height of 1 ft results in a decrease in plant capital cost of \$110,000.
- 2) The gas-plenum height for the nonvented fuel element was 64 in. and for the vented 24 in.

TABLE II-30

ECONOMIC INCENTIVE FOR DESIGNING 1000-Mwe PLANT
FOR NONVENTED AND VENTED FUEL ELEMENTS
AND USING VENTED ELEMENTS AFTER
TEN YEARS

Item	Incremental Costs (mills/kwh)*†	
	Top-Shield Leakage	
	at 1 ft ³ /day	at 10 ft ³ /day
Fission-Gas Reprocessing-Equipment Cost	+ 0.040	+ 0.148
Fuel-Cycle Cost (decrease in fuel-cladding thickness)	- 0.022	- 0.022
Fuel Handling (decrease in fuel-element height)	- 0.007	- 0.007
Net Difference	+ 0.011	+ 0.119

* Relative to plant designed for and operated with nonvented fuel elements

† Cost increase = +
Saving = -

The variation in availability was based on the change in fuel-handling time due to the vented fuel elements' being shorter, assuming equal reliability for both the vented and nonvented.

The availability variation was estimated as follows:

- 1) During fuel-transfer operations (raising and lowering), the speed is 0.625 ft/minute.
- 2) 202 elements are transferred per year.

The variation in fuel-cycle cost was based on a decrease in fuel-cladding thickness. The variation in fuel-cycle cost was estimated on the basis that a reduction in fuel-pin cladding thickness from 0.015 to 0.010 in. results in a

0.05-mill/kwh reduction in fuel-cycle cost. ^{(1)*} The results of the economic analysis are shown in Table II-28.

As Table II-28 shows, there is an incentive of 0.036 mills/kwh for using vented fuel elements if the leakage through the top shield is 1 ft³/day or less. An evaluation of risks and advantages of using vented fuel elements in the 1000-Mwe plant without a backup R&D program shows that it would not be advisable to design the plant without the backup of such a program (see Table II-29).

An R&D program should largely eliminate the risks itemized in the table. That is, the risks with the vented fuel elements would be the same as those with nonvented; therefore, the cost incentive in Table II-28 would apply.

If the vented-fuel-element R&D program is not complete at the time design is started on the 1000-Mwe plant, alternatives are possible.

One is that the plant could be designed for both nonvented and vented fuel elements and operated initially with nonvented. With nonvented fuel elements used for the first ten years and vented thereafter, the economics in Table II-30 would then apply.

As Table II-30 shows, the initial cost of the fission-gas reprocessing equipment for vented fuel elements is not compensated for by fuel-cycle and fuel-handling cost savings over the last 20 years of plant operation.

A second alternative would be to design the plant for both nonvented and vented fuel elements and operate the plant with vented elements which had been proven out by the R&D program. In this case, according to Table II-28, the only item which would be affected would be the core construction (there would be no decrease since the plant would be designed for nonvented fuel elements). This would give a net difference of -0.026 mills/kwh for 1-ft³/day leakage through the top shield and +0.082 mills/kwh for 10-ft³/day leakage through the top shield. This alternative would also allow the use of nonvented fuel elements initially (if the vented fuel elements were not proven out). The cost would be

*Superscript numbers are for References at back of this volume.

about \$100,000 due to the building space provided for the additional fission-gas reprocessing equipment required by the vented fuel elements, which would not be installed.

F. CORE DESIGN POINT SELECTION

Several studies were made to select a core design point, including some on fuel management, core height and ΔT , and axial and radial blanket effects to provide a solid base for selecting the reference design. These are, however, independent studies performed at different times, the results accordingly not being entirely consistent with each other.

1. Fuel-Management Study

Calculations were performed to determine the fuel-cycle characteristics of the reference core by a step-by-step determination of changes in reactivity, power distribution, and isotopic composition of the fuel throughout the period from startup to equilibrium-cycle operation. The effects of fuel burnup and fuel-element relocation were accounted for.

The calculations here represent a preliminary effort. The startup feed enrichment was estimated, based upon data from other studies. The startup cycle was not iterated with the better data derived from this particular study. The calculations have not been refined to the extent that all anomalies have been removed. Thus, for instance, it is seen that the final unpoisoned k_{eff} at end-of-cycle is approximately 1.01. Since the required k_{eff} at end-of-cycle is specified to be 1.0016, this implies that the feed fuel enrichment can be reduced. A reduction from 13.75% fissile plutonium to 13.4% is indicated. Such a change would not alter the conclusions of this study in a significant way.

The indicated peak linear heat rating (Table II-31) is a maximum (17.2 kw/ft) at startup and decreases to 15.7 kw/ft for midcycle equilibrium operation. The value at startup does not appear to be a significant result. That is, by adjusting the enrichment ratios in the startup loading it should be possible to reduce the peak linear heat rating to a value approximately equal to that for midcycle equilibrium operation. The value of 15.7 kw/ft for the equilibrium core is sufficiently close to the nominal heat rating of about 15 kw/ft to be acceptable

TABLE II-31

SUMMARY OF BREEDING RATIO, POWER PEAK-TO-AVERAGE, AND PEAK
LINEAR HEATRATE DATA DURING APPROACH-TO-EQUILIBRIUM
OPERATION

Time*	Breeding Ratio	Power Peak-to-Average		Peak Linear Heat Rating
		Axial	Radial	
BOC (1)	1.405	1.29	1.36	17.15
MOC (1)	1.398	1.27	1.34	16.45
MOC (3)	1.293	1.25	1.28	15.35
MOC (6)	1.288	1.25	1.28	15.51
MOC (9)	1.288	1.25	1.28	15.65

* BOC = Beginning of cycle

MOC = Middle of cycle

() = Cycle number

for these studies. This value could probably be reduced slightly by changes in core zone sizes and power splits. Further fuel-management analysis would be required to develop the optimum-design core and fuel-management scheme. The results of these studies represent a good set of working numbers which indicate the trends as equilibrium is approached.

The fuel-management scheme chosen consists of discharging one-third (85) of the core-region fuel elements, one-sixth (10) of the inner-ring radial-blanket elements, and one-ninth (7) of the outer-ring radial-blanket elements and replacing them with fresh elements each 144 full-power days (6 months at 0.8 plant factor) of operation.

The results of the study indicate that equilibrium-cycle operation is achieved following the 9th operating interval. The 10th operating and all subsequent intervals are postulated to have identical characteristics. At equilibrium the net fissile plutonium gained per cycle (after a 2% processing loss) is 97.8 kilograms. This, coupled with a total system fissile inventory

of about 2500 kg, yields a compound doubling time of 9 yr. The core-fuel average discharge exposure is 74,000 Mwd/MT.

An average feed fuel enrichment (wt % fissile Pu) of 13.75% provides sufficient excess reactivity to maintain the reactor critical during equilibrium fuel-cycle operation. The overall fuel-element peak-to-average power ratio at midcycle equilibrium is 1.605 and the corresponding peak linear heat rating is 15.7 kw/ft.

a. Method of Analysis

The fuel burnup calculations were performed with the PABUC computer program. The calculations were performed in one-dimensional geometry using neutron-diffusion theory. The PABUC code incorporates a modified version of ANISN for flux and eigenvalue (k_{eff}) calculation and the NUCY(2) subroutine for solution of the nuclide production and depletion chain.

The principal calculations were performed in cylindrical geometry. Two core enrichment regions were employed to facilitate control of the peak-to-average power ratio. Each enrichment region was subdivided into three burnup zones to provide greater detail in the calculations. Each of the burnup zones was divided into three equal-volume fuel batches to simulate the fuel-management scheme, and the time history of the changes in the isotopic composition of each batch was calculated on a cycle-by-cycle basis. At each refueling one fuel batch from each burnup zone was discharged and replaced with fresh fuel. The radial blanket was similarly treated with six fuel batches in the inner-ring blanket and nine fuel batches in the outer-ring blanket. A uniformly distributed tantalum poison was used in the outer-core enrichment region to simulate the control rods. The reactor geometry is illustrated in Figures II-12 and -13. Figure II-12 is a sketch of the horizontal cross-section of the reactor showing the arrangement of the various elements, and Figure II-13 is a sketch of the vertical cross-section. The fuel-burnup and fuel-move calculations were repeated until equilibrium operation was achieved at the 10th operating interval.

Figure II-12. Horizontal Cross-Section (50-in. Core)

The fuel in the core region consists of a mixture of depleted U and Pu oxides. The isotopic Pu composition for initial loading is assumed to be 60% Pu-239, 24% Pu-240, 12% Pu-241, and 4% Pu-242. This composition is representative of early LWR discharge compositions. The isotopic composition for the feed fuel in subsequent batches is assumed to be 66% Pu-239, 27% Pu-240, 5% Pu-241, and 2% Pu-242 to reflect a mixture of LWR and LMFBR discharge compositions.

The startup core employs three different enrichments in each of the two major enrichment zones to minimize the swing in reactivity during the approach to equilibrium. The three enrichments correspond to the three equal-volume fuel batches within each burnup zone of the core region. These fissile-Pu enrichments are: 9.1, 9.9, and 10.7 wt % for the inner-core region and 14.1, 15.3, and 16.5 wt % for the outer-core region. In managing the fuel, the lower-enrichment batches are replaced first to provide a smooth transition to high-burnup high-enrichment fuel.

4-24-68 UNCL

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Figure II-13. Vertical Cross-Section Dimensions

Slightly enriched (less than 3 wt%) uranium is used in the startup radial blanket to minimize the swing in blanket power during the approach to equilibrium. Depleted U is used as startup fuel in the axial blanket and as feed fuel in both the radial and axial blankets.

Secondary fuel-burnup and fuel-move calculations were performed in slab geometry (Figure II-13) to determine isotopic compositions for the axial blankets. The masses charged to and discharged from the axial blankets were added to those from the core and radial blankets to obtain the complete mass balance data. The axial blankets reach equilibrium during the 4th operating interval. In these calculations the Ta poison was distributed over the core and the upper axial blanket. The amount of poison in the upper blanket was fixed at a concentration of 0.0003683×10^{24} atoms/cm³.

b. Results

The results of the study consist of the mass balances describing the isotopic masses charged to and discharged from the reactor and of the time histories of the reactor performance characteristics. A summary of performance characteristics for equilibrium-cycle midcycle operation based upon these fuel-management calculations is given in Table II-32.

The startup and equilibrium-cycle midcycle fuel loadings are shown in Table II-33. The total initial fissile mass is 2214 kg and the total fissile mass at midcycle equilibrium is 2286 kg. The initial fissile inventory is 1708 kg. The initial fissile loading is increased approximately 26% by slightly enriched U in the radial blankets.

Table II-34 shows the mass balance for each major isotope on a cycle-by-cycle basis until equilibrium. Each cycle corresponds to 144 days of full-power (1000-Mwe) operation. This shows that during equilibrium operation 610 kg of fissile Pu are charged to the reactor and 721 kg are discharged at each refueling shutdown. This corresponds to an annual excess fissile production (after a 2% processing loss) of 0.196 kg/Mwe. The total system fissile Pu inventory based on the assumed 1986-90 ex-core time of 235 days is 2530 kg. This includes 1708 kg of fissile Pu required for startup plus the ex-core fissile Pu required to sustain operation ($235/180 \times 610$ kg); it also includes a 1% fabrication loss allowance. Based on these values, the compound

TABLE II-32
EQUILIBRIUM MIDCYCLE REACTOR PERFORMANCE DATA FROM
FUEL-MANAGEMENT STUDY

<u>Thermal-Hydraulic Performance</u>	
Total reactor thermal power, Mwt (nominal)	2500
Core power fraction (midcycle)	0.903
Radial blanket power fraction	0.070
Axial blanket power fraction	0.027
Total reactor flowrate, 10^6 lb/hr	93
Number of coolant loops	3
Peak linear rod power, kw/ft	15.7
Radial peaking factor	1.28
Axial peaking factor	1.25
Reactor mix-mean temperature rise, $^{\circ}\text{F}$	300
Reactor outlet coolant temperature, $^{\circ}\text{F}$	1060
Maximum nominal clad inside temperature, $^{\circ}\text{F}$	1126
Hot-spot clad inside temperature, $^{\circ}\text{F}$	1201
Average coolant velocity highest-powered element, ft/sec	27.0
Maximum coolant velocity (at spacer), ft/sec	32.8
Core pressure drop, psi	100
<u>Midcycle Equilibrium Nuclear Characteristics</u>	
Fuel elements, number (enrichment, at. %):	
Inner zone	118 (11.55)
Outer zone	138 (16.00)
Radial blanket elements, number (enrichment, at. %):	
Inner	60 (2.21)
Outer	66 (2.19)
Core height, in.	50
Axial blanket height, in.	12
Radial blanket height, in.	62
Number of control rods:	
Shim-safety	9
Safety	6

TABLE II-32 (continued)
EQUILIBRIUM MIDCYCLE REACTOR PERFORMANCE DATA FROM
FUEL-MANAGEMENT STUDY

Midcycle Equilibrium Nuclear Characteristics (continued)

Fissile masses, kg:			
	<u>Total</u>	<u>Pu</u>	<u>U-235</u>
core	1,800	1,778	22
radial blanket	372	335	37
axial blanket	114	96	18
Total	2,286	2,209	77
Breeding ratio:			
core		0.816	
radial blanket		0.276	
axial blanket		0.195	
Total		1.287	
Initial loading, kg Pu + U:			
core		13,308	
radial blanket		15,198	
axial blanket		7,406	
Specific inventory, $\frac{\text{kg/fiss}}{\text{Mwe}}$		2.5	
Doubling time, yrs		9	

doubling time is 9.0 years. The annual excess fissile production during the first five years of operation is only 0.116 kg/Mwe. The 30-year average is therefore 0.181 kg/Mwe, implying a compound doubling time of 9.7 yr compared to 9.0 yr for equilibrium operation.

The variations in average fuel exposure, k_{eff} , and in blanket power fractions are shown in Figures II-14, -15, and -16, respectively. The breeding ratio, power peak-to-average, and peak linear heat rating data are summarized in Table II-35. Note that core-region and axial-blanket discharge exposures are nearly at equilibrium at the end of the 4th operating cycle. The inner radial

TABLE II-33

STARTUP AND EQUILIBRIUM-CYCLE FUEL LOADINGS

	U-235	U-238	Pu-239	Pu-240	Pu-241	Pu-242	>Pu-242	Total
<u>Startup fuel loading, kg</u>								
Core I	16	5356	517	207	104	35		6235
Core II	17	5545	905	363	182	61		7073
Axial blanket	21	7398	0	0	0	0		7419
Inner radial blanket	239	7850	0	0	0	0		8089
Outer radial blanket	213	6968	0	0	0	0		7181
Total	506	33,117	1422	570	286	96		35,997
<u>Equilibrium fuel loading, kg</u>								
Core I	10	4984	647	268	43	20	1	5973
Core II	12	5212	1013	452	75	34	1	6799
Axial blanket	18	7290	96	2	0	0	0	7406
Inner radial blanket	20	7841	178	6	0	0	0	8045
Outer radial blanket	17	6973	157	6	0	0	0	7153
Total	77	32,300	2091	734	118	54	2	35,376

blanket reaches equilibrium exposure on the 7th cycle and the outer radial blanket on the 10th cycle, or 4.5 yr after startup. The core-region fuel average-discharge exposure is 73.9 Mwd/kg and the peak exposure, based on an effective peak-to-average ratio of 1.605, is 118.6 Mwd/kg.

c. Calculation of Specific Inventory and Doubling Time

The doubling time (D. T.) in this report is the compound doubling time. It refers to the length of time it takes a population of reactors of this type to produce sufficient excess fissile plutonium to allow the startup and operation of an equal number of new reactors. The amount of fissile Pu required for each new reactor is equal to that in the initial fuel loading of the reactor plus the additional material held up in the processing cycle. Specifically, the U-235 in the initial loading of the radial blankets is ignored in the determination of the D. T. as is explained later.

The doubling time is calculated using the relationship:

$$D. T. = 0.693 \times SI/P_{ex} ,$$

TABLE II-34
MASS BALANCES

Cycle	U-235	U-238	Pu-239	Pu-240	Pu-241	Pu-242	>Pu-242	Total
<u>Feed fuel</u>								
Core	10	3565	567	232	43	17	0	4434
Axial blanket	7	2468	0	0	0	0	0	2475
Inner radial blanket	4	1345	0	0	0	0	0	1349
Outer radial blanket	2	796	0	0	0	0	0	798
Total	23	8174	567	232	43	17	0	9056
<u>Discharge end of cycle 1</u>								
Core	9	3585	452	179	70	30	1	4326
Axial blanket	6	2443	23	0	0	0	0	2472
Inner radial blanket	37	1296	11	0	0	0	0	1344
Outer radial blanket	23	769	4	0	0	0	0	796
Total	75	8093	490	179	70	30	1	8938
<u>Discharge end of cycle 2</u>								
Core	7	3420	483	198	63	33	2	4206
Axial blanket	6	2419	44	1	0	0	0	2470
Inner radial blanket	35	1283	21	0	0	0	0	1339
Outer radial blanket	21	766	8	0	0	0	0	795
Total	69	7888	556	199	63	33	2	8810
<u>Discharge end of cycle 3</u>								
Core	5	3261	500	216	57	36	3	4078
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	32	1271	30	1	0	0	0	1334
Outer radial blanket	20	761	12	0	0	0	0	793
Total	62	7687	605	219	57	36	3	8669
<u>Discharge end of cycle 4</u>								
Core	5	3249	534	244	37	18	2	4089
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	30	1259	38	2	0	0	0	1329
Outer radial blanket	18	757	16	0	0	0	0	792
Total	59	7659	651	248	37	18	2	8674
<u>Discharge end of cycle 5</u>								
Core	5	3248	534	244	37	18	2	4088
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	28	1246	47	2	0	0	0	1323
Outer radial blanket	18	752	19	1	0	0	0	790
Total	56	7640	663	249	37	18	2	8665
<u>Discharge end of cycle 6</u>								
Core	5	3246	534	244	37	18	2	4086
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	26	1236	54	3	0	0	0	1319
Outer radial blanket	17	746	22	1	0	0	0	786
Total	53	7622	673	250	37	18	2	8655

TABLE II-34 (continued)

MASS BALANCES

Cycle	U-235	U-238	Pu-239	Pu-240	Pu-241	Pu-242	>Pu-242	Total
<u>Discharge end of cycle 7</u>								
Core	5	3245	534	244	37	18	2	4085
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	3	1270	55	3	0	0	0	1331
Outer radial blanket	16	742	26	1	0	0	0	785
Total	29	7651	678	250	37	18	2	8665
<u>Discharge end of cycle 8</u>								
Core	5	3244	533	244	37	18	2	4083
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	3	1269	55	3	0	0	0	1330
Outer radial blanket	15	740	29	1	0	0	0	785
Total	28	7647	680	250	37	18	2	8662
<u>Discharge end of cycle 9</u>								
Core	5	3243	533	244	37	18	2	4082
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	3	1268	56	3	0	0	0	1330
Outer radial blanket	14	736	31	2	0	0	0	783
Total	27	7641	683	251	37	18	2	8659
<u>Discharge end of cycle 10</u>								
Core	5	3243	533	244	37	18	2	4082
Axial blanket	5	2394	63	2	0	0	0	2464
Inner radial blanket	3	1268	56	3	0	0	0	1330
Outer radial blanket	1	755	32	2	0	0	0	790
Total	14	7660	684	251	37	18	2	8666

where SI is the initial specific fissile - Pu inventory (kg/Mwe) for a new reactor, P_{ex} is the annual specific excess fissile-Pu production (kg/yr per Mwe), and 0.693 is a factor relating compound and simple interest doubling times.

The initial fissile Pu inventory for a new reactor is 2530 kg. This includes:

- 1) 1708 kg of fissile Pu in the initial fuel loading, plus,
- 2) 797 kg of fissile Pu held up in the processing cycle, plus,
- 3) 25 kg of fissile Pu representing a 1% process loss.

For a 1000-Mwe reactor the initial specific fissile-Pu inventory = 2.53 kg. The quantity 797 kg representing the ex-core inventory corresponds to the quantity $235/180 \times 610$, where 235 days is the ex-core holdup time, 180 days

7950-5709

Figure II-14. Fuel Average-Discharge Exposure Approach to Equilibrium-Cycle Operation

is the reactor-cycle time, and the 610 kg is the amount of fissile Pu charged to the reactor at each refueling shutdown. The net fissile Pu gained per equilibrium reactor cycle is 97.8 kg. This corresponds to the quantity $(721/1.01 - 610/0.99)$, where 721 kg is the amount of fissile Pu discharged from the reactor each cycle, 610 kg is the amount charged to the reactor, and the divisors account for a 2% processing loss. Since there are two cycles per year and the reactor power level is 1000 Mwe, the annual specific excess

Figure II-15. Reactivity Changes: Approach to Equilibrium-Cycle Operation

fissile-Pu production is $97.8 \times 2/1000$ equal to 0.196 kg/yr/Mwe. Substituting into the equation above we have

$$D. T. = \frac{0.693 \times 2.53}{0.196} = 9.0 \text{ yr}$$

The annual specific excess fissile-Pu production averaged over the first 10 cycles (5 yr) of operation is 0.116 kg/yr/Mwe, compared to the value 0.196 kg/hr/Mwe for equilibrium fuel-cycle operation; from this comes the 30-yr average value of 0.182 kg/yr/Mwe. Using this value, the doubling time becomes 9.7 instead of 9.0 years.

7950-5711

Figure II-16. Blanket Power Fraction: Approach to Equilibrium-Cycle Operation

For this report the breeding ratio is defined as the ratio of the neutron-capture rate by fertile isotopes (U-238 and Pu-240) to the neutron-absorption rate by fissile isotopes (Pu-239 and Pu-241). The absorptions in U-235 are specifically excluded in this definition. This is consistent with the theory that (BR-1) is approximately proportional to the doubling time

$$D. T. = \frac{SI}{(BR-1)} = \frac{2.53}{0.29} = 8.7 \text{ yr, where BR = breeding ratio.}$$

In calculating the doubling time it is assumed that it is not necessary to replace the U-235 in the feed fuel. (In the calculations U-235 exists in the feed fuel as a 0.3% constituent of depleted U.) This is equivalent to the hypothesis that the excess fissile-Pu production rate remains the same regardless of whether or not there is any U-235 in the feed fuel. This hypothesis relates to the fact that in a U and Pu recycle program the U-235 in the initial U will soon be destroyed.

2. Core-Height, ΔT Study

The nuclear effects of core height, reactor ΔT , and system design points on plant economics were studied to permit selection of the reference core and system design points. Nine core configurations were investigated, consisting of core heights of 35.6, 41.4, and 50 in., and core ΔT 's of 260, 280 and 300°F.

The preliminary thermal-hydraulic results of the core height, ΔT study are given in Table II-35. The Company-developed FAITH computer code which was used for the analysis is based on a peak linear power of 15.25 kw/ft, a maximum velocity at the spacer of 32.8 ft/sec, and a reactor outlet temperature of 1060°F. A radial peak-to-average core power of 1.25 was assumed for all cases. The fuel elements consist of 217 pins of 0.25-in. OD having total axial blanket lengths (top and bottom) of 2 ft. The geometry and spacing of the "cookie-cutter" spacer design were used. The resulting core configurations are: the reference case of 256 fast elements with an active core height of 50 in., the intermediate case of 298 fast elements with an active core height of ~ 3.5 ft, and a short-core case of 340 fast elements with an active core height of ~ 3 ft.

The degradation of the core outlet temperature due to the blanket region was determined on the basis that the blanket elements would be turned 180° at their midcycle, reducing their elemental peaking factors. The fast elements are orificed for constant element-coolant temperature rise. Velocity and hot-spot temperature values reported correspond to the highest-powered fast element.

The bases for the nuclear calculations are given in Table II-36 and Figures II-12, -17 and -18. The basic nuclear calculations were performed with one-dimension 18-group diffusion theory using the CAESAR-IV Code. The

TABLE II-35
CORE PRELIMINARY THERMAL-HYDRAULIC PERFORMANCE DATA

Parameter	Core Height (in.)								
	50			41.4			35.6		
Gas plenum length, in.	64			53			45.6		
Power split, %									
Fast region	90.18			90.03			89.51		
Axial blanket	2.65			4.01			5.52		
Radial blanket	7.17			5.96			4.97		
Axial peak-to-average	1.261			1.225			1.203		
Radial peak-to-average	1.25			1.25			1.25		
Number of fast elements	256			298			340		
Reactor ΔT , $^{\circ}F$	260	280	300	260	280	300	260	280	300
Core ΔT , $^{\circ}F$	272	293	314	271	291	312	269	289	310
Element volume fraction, %*									
Sodium	50.6	49.1	47.9	48.2	46.8	45.3	46.0	44.5	43.2
Fuel	30.9	32.0	32.8	33.0	34.0	35.0	34.9	35.9	36.8
Steel	18.5	18.9	19.3	18.8	19.2	19.7	19.1	19.6	20.0
Lattice pitch, in.	5.550	5.462	5.389	5.369	5.293	5.219	5.228	5.155	5.093
Element pressure drop, psi	100	99	100	89	90	93	84	87	88
Inside clad hot spot, $^{\circ}F$	1195	1199	1201	1198	1200	1204	1198	1201	1204
Average Na velocity in highest-powered element, ft/sec	27.7	27.4	27.0	27.0	26.7	26.5	26.6	26.4	25.9

*Based on maximum velocity (at spacer) of 32.8 ft/sec (fuel elements only)

TABLE II-36
BASIC CORES CALCULATED

Parameter	Core Height, (in.)								
	50			41.4			35.6		
Reactor ΔT , °F	260	280	300	260	280	300	260	280	300
Circumscribed OD, in. including reflector	138.75	136.55	134.73	136.16	134.23	132.35	141.16	139.19	137.51
Number of elements									
Inner zone	118	118	118	142	142	142	160	160	160
Outer zone	138	138	138	156	156	156	180	180	180
Radial Blanket 1	60	60	60	66	66	66	72	72	72
Radial Blanket 2	66	66	66	72	72	72	78	78	78
Reflector	72	72	72	78	78	78	84	84	84
Control	15	15	15	15	15	15	15	15	15
Diametral dimensions, in. (cylindricized)									
Active core	95.92	94.41	93.15	99.74	98.33	96.95	103.43	101.99	100.76
Core + radial blankets	116.12	114.28	112.75	119.73	118.03	116.39	123.36	121.65	120.18
Plus reflector	126.21	124.20	122.55	129.67	127.83	126.05	133.23	131.37	129.79
L/D (core)	0.521	0.530	0.537	0.415	0.421	0.427	0.344	0.349	0.353
Active core volume, liters	5923	5736	5584	5301	5152	5009	4907	4772	4657
Axial blanket length, in. (top and bottom)	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0
Radial blanket length, in.	62.0	62.0	62.0	53.4	53.4	53.4	47.6	47.6	47.6
Volume fraction, core (includes control rods)									
Fuel	0.292	0.302	0.310	0.314	0.324	0.333	0.334	0.344	0.352
Steel	0.182	0.185	0.189	0.185	0.189	0.194	0.188	0.193	0.197
Sodium	0.526	0.513	0.501	0.501	0.487	0.473	0.478	0.463	0.451

4-24-68 UNCL

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CORE REGIONS:

- INNER: 142 FUEL ELEMENTS AND 9 CONTROL RODS
- ▨ OUTER: 156 FUEL ELEMENTS AND 6 CONTROL RODS

● RADIAL BLANKET

- INNER: 66 ELEMENTS
- OUTER: 72 ELEMENTS

○ REFLECTOR:

78 ELEMENTS

● SHIM ROD

● SAFETY ROD

Figure II-17. Horizontal Cross-Section of 41.4-in. Core

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CORE REGIONS:

- INNER: 160 FUEL ELEMENTS AND 9 CONTROL RODS
- ▨ OUTER: 180 FUEL ELEMENTS AND 6 CONTROL RODS

● RADIAL BLANKET

- INNER: 72 ELEMENTS
- OUTER: 78 ELEMENTS

○ REFLECTOR:

84 ELEMENTS

● SHIM ROD

● SAFETY ROD

Figure II-18. Horizontal Cross-Section of 35.6-in. Core

equilibrium distribution of Pu isotopes was taken as 49/40/41/42 = 67/26/5/2% as established by the study ground rules. For this particular set of calculations a peak burnup of 120,000 Mwd/T was set.

Specific axial calculations were run for each case to establish the axial bucklings, axial peak-to-average power ratios, and neutron balances in the axial blankets. With the axial bucklings obtained, radial cases were run with two core zones to determine the specific minimum peak-to-average core for each height and ΔT . Overall neutron balances were then calculated to determine absolute breeding ratios and power splits. Equilibrium enrichments for the blankets in these calculations were estimated by curves developed on previous studies of blankets.

The results of the nuclear analysis are presented in Table II-37. The shorter cores are characterized by lower fissile masses and higher breeding ratios. This is due primarily to the lower sodium fractions and consequent higher fuel fractions in these cores.

Midcycle-equilibrium fuel-cycle cost calculations were performed with the Company-developed COST-II Code. The economic ground rules used are given in Section I of this report. The results of these calculations, shown in Table II-38, are plotted as a function of core height in Figure II-19. Table II-38 also includes the results of varying the Pu value by +50%, varying the core fabrication costs $\pm 30\%$, and varying the reprocessing cost $\pm 30\%$.

Over the range of variables studied the 3-ft-high core with a 300°F ΔT has the lowest fuel-cycle cost (0.791 mills/kwh), and the costs appear to bottom out for each ΔT at this height. The cost differential between 3 and 4.17 ft, however, is only ~0.03 mills/kwh and the differential between ΔT 's of 300 and 260°F is also only ~0.03 mills/kwh. Thus it is apparent that fuel-cycle cost for the Follow-on Study design is a weak function of both core height and ΔT .

In evaluating these results it was seen that most of the variation in cost with core height is due to: (1) increased average core burnup with shorter cores since the axial peak-to-average power is less for the shorter cores; and (2) variation in the axial-blanket-cost contribution because a constant axial blanket

TABLE II-37
NUCLEAR RESULTS OF CORE-HEIGHT ΔT SURVEY

Core Height (in.)	50			41.4			35.6		
Reactor ΔT ($^{\circ}F$)	260	280	300	260	280	300	260	280	300
Kilograms, Fissile *									
Core	1786	1765	1745	1723	1704	1684	1699	1676	1660
Radial blanket 1	181	175	170	149	144	140	125	122	119
Radial blanket 2	247	239	233	203	197	192	173	168	164
Axial blanket	100	100	100	146	146	146	195	195	195
Total	2314	2279	2248	2221	2191	2162	2192	2161	2138
Power Split *									
Core	0.9007	0.9018	0.9024	0.8994	0.9003	0.9013	0.8940	0.8951	0.8957
Radial blanket 1	0.0454	0.0447	0.0443	0.0379	0.0374	0.0369	0.0318	0.0313	0.0311
Radial blanket 2	0.0271	0.0270	0.0270	0.0223	0.0222	0.0221	0.0185	0.0184	0.0184
Axial blanket	0.0268	0.0265	0.0263	0.0404	0.0401	0.0397	0.0557	0.0552	0.0548
Discharge Burnup, Mwd/kg									
Core	78.5	78.5	78.5	80.68	80.68	80.68	82.49	82.49	82.49
Radial blanket 1	11.50	11.70	11.90	10.84	11.01	11.17	9.85	9.97	10.15
Radial blanket 2	12.50	12.85	13.20	11.69	11.98	12.26	10.58	10.83	11.09
Axial blanket	4.23	4.18	4.15	6.24	6.20	6.14	7.78	7.72	7.66
Initial Kilograms, U + Pu									
Core	13,339	13,339	13,339	12,890	12,890	12,890	12,650	12,650	12,650
Radial blanket 1	8,636	8,368	8,142	7,657	7,441	7,235	7,068	6,872	6,708
Radial blanket 2	9,500	9,201	8,956	8,353	8,119	7,893	7,656	7,444	7,266
Axial blanket	7,349	7,349	7,349	8,557	8,557	8,557	9,588	9,588	9,588
Enrichment, wt % *									
Inner core zone	12.02	11.85	11.74	12.21	12.06	11.92	12.36	12.20	12.07
Outer core zone	15.56	15.33	15.20	15.65	15.45	15.26	15.52	15.30	15.16
Breeding ratio *	1.283	1.293	1.299	1.287	1.295	1.303	1.303	1.311	1.318
Doubling time, yr	12.9	12.3	11.9	12.3	11.8	11.3	11.5	11.1	10.7

*Midcycle equilibrium conditions

length was used for all cases. According to Table II-37, in the 50-in. core the axial blanket receives an exposure of only about 4100 Mwd/MT while in the 36-in. core it is about 7700. Because of the higher axial leakage in the shorter cores the optimum axial blanket length will be longer. The optimization of axial blanket length will slightly increase the fuel-cycle cost differential due to core height.

TABLE II-38

EQUILIBRIUM FUEL-CYCLE COSTS
FOR BASE CASE AND VARIATIONS
OF ECONOMIC GROUND RULES

	ΔT ($^{\circ}F$)	Core Height (in.)		
		50	41.4	35.6
Basic Cost Results (mills/kwh)	260	0.85	0.82	0.82
	280	0.83	0.81	0.80
	300	0.82	0.79	0.79
Plutonium Value plus 50% (mills/kwh)	260	0.94	0.89	0.88
	280	0.91	0.87	0.86
	300	0.89	0.85	0.84
Core Fabrica- tion plus 30% (mills/kwh)	260	0.97	0.95	0.95
	280	0.96	0.93	0.93
	300	0.94	0.92	0.92
Core Fabrica- tion minus 30% (mills/kwh)	260	0.76	0.72	0.72
	280	0.74	0.71	0.70
	300	0.73	0.69	0.69
Reprocessing and Shipping plus 30% (mills/kwh)	260	0.90	0.87	0.86
	280	0.88	0.85	0.85
	300	0.86	0.83	0.83
Reprocessing and Shipping minus 30% (mills/kwh)	260	0.82	0.79	0.78
	280	0.80	0.78	0.77
	300	0.79	0.76	0.76

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Figure II-19. Total Midequilibrium Fuel-Cycle Costs for Core-Height, ΔT Parameter Survey

3. Radial and Axial Blanket Studies

One key variable in the evaluation of the optimum radial blanket design and management is the determination of the optimum residence time for the radial blanket elements. A method developed by ANL⁽³⁾ was used to determine the optimum residence times for the inner and outer blankets of the current reference 1000-Mwe FBR. The techniques used involved a determination of the optimum reloading frequency for the blanket elements using the criterion that the net income from the blanket region under consideration be maximized during the reactor lifetime.

In addition to evaluating the total gain in \$/kg of radial blanket, calculations were also made using present worth effects. The basic equations to determine the total gain were altered to yield the present worth of the net income derived at the end of each refueling cycle for the blanket elements. A final net present worth was then determined as a function of cycle time for each of the cases studied. Figure II-20 shows a comparison of the total gain method and the present worth modification. It is seen that although the present worth method gives slightly lower credits, there is very little change in the optimum residence time. Present worth caused a slight shift toward longer residence times. The remainder of the cases for this study used the present worth method.

Figure II-21 summarizes the investigation of the effect of reprocessing cost on the optimum residence time of radial blankets. As reprocessing costs decrease, the optimum blanket residence time decreases. This same trend would hold true for fabrication cost. Figures II-22 and -23 show the net present worth of the radial blanket gain as a function of residence time for the reference pin diameters.

Figure II-24 shows the effect of pin diameter on optimum residence time for two reprocessing costs. The pin diameter has little effect on optimum time. Figure II-25 shows the net present worth of the blanket gain as a function of residence time for three pin diameters.

A preliminary axial blanket length optimization study was performed on the 50-in. core to determine whether the initial choice of 12 in. on top and bottom was well-founded. Equilibrium fuel-cycle mass balances were determined for 9-, 12-, and 15-in. axial blanket lengths with PABUC, and fuel-cycle costs associated with the axial blanket material only were calculated. The results indicated that there is a small incentive to increase the axial blanket length beyond 12 in. The fuel-cycle cost decreased by 0.024 mills/kwh in going from 9 to 12 in., and an additional 0.006 in going to 15.

The basic trade-off here is between the increased breeding of the longer blanket and the increased fabrication, reprocessing, and shipping charges due to the increased axial blanket mass. Because the incremental fuel-cycle benefit of a longer axial blanket was so small it was decided to stay with the 12-in. basic design since so many other facets of the design are keyed to that length.

Figure II-20. Total Gain and Net Present Worth vs Residence Time of Inner and Outer Blankets

Figure II-21. Optimum Residence Time vs Reprocessing Cost for the Inner and Outer Blanket Elements

4. Midequilibrium Nuclear Characteristics of Selected Design

Based upon the studies reported in this section and in G., System Design Point Selection (following), the basic reference design was chosen as the 50-in. active core with: (1) 300°F ΔT for the reactor core; (2) 12-in. axial blankets on top and bottom; and (3) inner and outer radial blanket residence times of 3 and 4.5 yr, respectively. Isotopic ratios for both core and blankets were taken from the fuel-management studies reported earlier, but core enrichments were adjusted slightly for the correct reactivity and minimum radial peak-to-average power at midcycle.

The basic core geometry is summarized in Table II-39. The nuclear parameters for the reference core are summarized in Table II-40.

5. Nuclear Methods of Analysis

Most of the nuclear calculations reported in this section were made using one-dimensional 18-group diffusion theory. The analyses consisted of the systematic application of a series of computer codes. Those codes used and short abstracts describing their functions are listed in Appendix B.

Figure II-22. Net Present Worth vs Residence Time for the Inner Blankets as a Function of Reprocessing Cost

TABLE II-39
1000-Mwe CORE GEOMETRY

Active core height, in.	50
Axial blanket length, in.	24 (12 each at top and bottom)
Radial blanket height, in.	62
Number of Elements	
Inner core zone	118
Outer core zone	138
Inner radial blanket	60
Outer radial blanket	66
Reflector	72
Control	15
Active core diameter, in.	93.15
Height/diameter (core)	0.527
Circumscribed diameter (core, blanket and reflector), in.	122.55
Lattice pitch, in.	5.389

a. Formation of Multigroup Library of Cross-Sections.

Regardless of the specific techniques used in the later analyses of the core, the most important preliminary item is the preparation of the library of nuclear constants to be used. To accomplish this requires that (1) a detailed set of energy-dependent microscopic data for all elements of interest must be available, (2) some method is needed to reduce these data into group cross-sections compatible with the later codes to be used, (3) a desired multigroup energy structure must be specified, and (4) some check must be made of the accuracy of the derived multigroup set in predicting experimental lattice parameters.

TABLE II-40
NUCLEAR CHARACTERISTICS OF 1000-Mwe REFER-
ENCE CORE AT MIDEQUILIBRIUM CYCLE

Characteristic	Value
Kg Fissile	
Core	1757.7
Inner Radial Blanket	174.2
Outer Radial Blanket	157.0
Axial Blanket	97.6
Total	2186.5
Kg U-235 (Total)	76.4
Breeding Ratio, U-235 Destruction Ignored	1.313
U-235 Destruction Included	1.277
Power Splits	
Core	0.8988
Inner Radial Blanket	0.0475
Outer Radial Blanket	0.0215
Axial Blanket	0.0322
Doppler Constant, $T dk/dT$	-0.00960
Isothermal Sodium Temperature Coefficient, $\Delta k/^{\circ}F \times 10^{-6}$	3.32
Sodium Voiding, %	
Maximum	6.82
Total Core	6.03

The basic set of microscopic cross-section data used was that of AI Evaluated Nuclear Data File⁽⁴⁾ (AIENDF), current as of August 30, 1966. With these data the basic procedure for constructing a library is as follows:

- 1) The AILMOE Code is run on a core composition typical of the reactor of interest to obtain a fine-energy spectrum characteristic of the composition and elastic scattering cross-sections for the light elements, sodium, oxygen, and steel (here steel is taken as 74% Fe, 18% Cr, and 8% Ni).
- 2) The fine-energy spectrum thus obtained is then fed into the GRISM set of codes and a multigroup set of absorption, fission, ν -fission and transport cross-sections is obtained. In addition a complete scattering transfer matrix and resonance self-shielding in the fissile and fertile elements are derived.

4-24-68 UNCL

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Figure II-23. Net Present Worth vs Residence Time for the Outer Blankets as a Function of Reprocessing Costs

4-24-68 UNCL

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Figure II-24. Optimum Residence Time vs Pin Diameter for the Inner and Outer Blanket Elements

- 3) The GRISM cross-sections are combined with the AILMOE elastic scattering cross-sections to form the preliminary working set.

Once this preliminary multigroup library is obtained, several additional important steps must take place, viz, the checkout of the library against known critical experiments and the adjustment of certain of the group cross-sections to conform with the latest most accurate experimental data. The adjustment process on the cross-sections is described in great detail by Specht in References 5 and 6. The highlights of these adjustments are as follows:

- 1) σ_f and α for Pu-239 between 2 kev and 10 Mev were modified to incorporate the Greebler evaluation for the BNL central data file.⁽⁷⁾

Figure II-25. Net Present Worth vs Residence Time for Inner and Outer Blankets as a Function of Pin Diameter

- 2) σ_a and σ_f for U-238, Pu-239, and Pu-240 were resonance shielded at the average fuel temperature by use of TRIX. These TRIX data were renormalized to the infinitely dilute group cross-sections reported in Reference 6.
- 3) The lumped fission-product cross-sections and those for Pu-242 were taken as in Reference 5.
- 4) Pu-240 absorption and fission cross-sections were modified according to the recent work of Otter⁽⁸⁾ which recommends lower (25 to 30%) absorption cross-sections in the resonance region (groups 8-16 of the multigroup library structure) and slightly higher fast-fission cross-sections than previously used.

As a checkout of the basic library, a similar set of cross-sections was derived from the ZPR-III-48 critical. For the experimental critical mass, a k_{eff} of 1.001 was calculated, thus lending a good deal of confidence to the FBR set derived for predicting critical masses. These latter calculations are described by Specht.⁽⁶⁾

b. Calculational Procedure

For specific core statics calculations, all parameters derived were obtained with one-dimensional diffusion theory, using CAESAR. For setting up these calculations, cylindrical geometry was used in the radial plane and slab geometry in the axial plane. Simple volume-weighted homogenization of core materials was assumed. Transverse leakage effects, which will be described in greater detail later, were simulated by the use of an effective buckling. In radial calculations each of the cores investigated was broken into two fast-region zones, each ~50% of the total fast-section volume, surrounded by two rows of radial blankets and one of reflector elements. The two-zone fast-region scheme was used to reduce the peak-to-average power of the fast core. Most core calculations were made at the midequilibrium-cycle point with $k_{eff} = 1.0$; thus, fission product concentrations characteristic of ~37.5 mwd/kg were homogeneously dispersed throughout the core.

The isotopic distribution of the Pu isotopes for the midequilibrium state was taken from the fuel-management studies described earlier in this report.

The general procedure for specifying the midequilibrium core zone enrichments is basically a trial and error technique. Using the CAESAR concentration search option, the outer-zone enrichment is specified and the center-zone enrichment is searched for $k_{\text{eff}} = 1.0$. Several outer-zone enrichments are chosen and that combination of inner and outer yielding the minimum core power peak-to-average ratio is selected. This point always occurs when the peaks in either zone compared to the average for the entire fast core are equal. For a wide range of two-zone cores investigated, this procedure usually yields a radial core peak-to-average ratio of 1.20 to 1.22. Very few cores have fallen outside this range.

For axial calculations the volume-weighted core materials as determined from the radially selected case are used. Once again the enrichment yielding $k_{\text{eff}} = 1.0$ is chosen. Axial cases in general are only used to determine the axial peak-to-average power ratio and the neutron balance characteristics of the axial blankets. Most other core characteristics are derived from radial cases. For these calculations, the axial blanket compositions were specified from the midequilibrium values obtained in the fuel-management studies.

In any one-dimensional cylindrical or slab calculation some account must be made of events which take place perpendicularly to the dimension being calculated. In particular, the perpendicular neutron leakage must be accounted for, i. e., in a radial calculation the axial neutron leakage must be considered and vice versa. This leakage is usually defined in terms of a product, DB^2 , where D is the diffusion coefficient of the medium and B^2 is the "buckling". These bucklings are defined as:

$$B^2_{\text{radial}} = \left(\frac{2.405}{R + \delta_r} \right)^2 ,$$

$$B^2_{\text{axial}} = \left(\frac{\pi}{H + 2\delta_z} \right)^2 ,$$

where R and H are the assembly radius and height, and δ_r and δ_z are the radial and axial reflector savings, respectively. The reflector savings

represents the decrease in dimension for a given reactivity due to the reflector (in an FBR, the blankets are effective pseudoreflectors) as opposed to the required dimension for that given reactivity if the reflector were not present. Because of the small size of an FBR core leakage effects are very important, and reliable one-dimensional calculations must include careful attention to transverse-leakage effects. Thus it is important to establish a "consistent" set of radial and axial reflector savings that will produce the same reactivity when calculations are made in either dimension for the same core.

A set of consistent reflector savings for a given core may be determined in the following manner:

- 1) A set of radial cases is run to determine the reactivity as a function of axial buckling, B_z^2 ;
- 2) In a similar manner a set of axial cases is run to determine the reactivity as a function of radial buckling, B_r^2 ;
- 3) From Steps 1 and 2, since the total buckling, $B_T^2 = B_z^2 + B_r^2$, a curve of k_{eff} vs B_T^2 may be constructed;
- 4) A set of zero-leakage cases is then run to determine independently the reactivity of the fast section of the assembly as a function of B_T^2 ;
- 5) When the curve from Step 3 is cross-plotted with the curve from Step 4, a unique value of k_{eff} and B_T^2 is determined.
- 6) With this unique value of k_{eff} and B_T^2 , one may return to the curves developed in Steps 1 and 2 and determine the unique values of B_r^2 and B_z^2 that will yield the same reactivity for axial and radial cases;
- 7) The radial and axial reflector savings, δ_r and δ_z , may then be obtained from their respective definitions.

This procedure is further discussed in Reference 9.

Sets of consistent reflector savings were developed for unvoided reduced sodium density and fully voided cores for subsequent use. It is re-emphasized at this point that the major nuclear characteristics were calculated in the radial

direction as this is thought to be a more representative configuration. Unless otherwise specified, all characteristics were determined from radial calculations.

Isothermal Doppler calculations were made with first-order perturbation theory using the PERT option available in the CAESAR code. Cross-section changes between 1862 and 3141^oF were used. The perturbation approach for Doppler calculations has been checked against the more accurate eigenvalue differencing technique and found to give answers within 5% of the latter. Only changes in U-238 cross-sections were used since the changes due to Pu-239 and -240 are small and opposite in sign, thus tending to cancel out each other. The range of temperatures is irrelevant since the Doppler coefficient of reactivity is linear in $1/T$ (T in degrees absolute). The calculation was run one time for the entire core. The use of U-238 Doppler effects only is conservative since that effect for Pu-239 has been shown experimentally to be essentially zero. Thus calculations do not include the negative Pu-240 Doppler effect; its inclusion would make the overall Doppler effect ~7-10% more negative.

Sodium-voiding coefficients were obtained with CAESAR by the eigenvalue differencing technique, i. e. , multiple core cases were run with successive regions voided. Axial leakage changes in the voided regions were simulated by introducing axial reflector savings changes from the aforementioned consistent reflector savings technique. Voiding was assumed to occur for the core and top axial blanket and to proceed radially outward.

For the calculations reported herein, only changes in the microscopic absorption cross-sections for U-238 with voiding were made. Changes in microscopic scattering cross-sections for the light elements were not made, since calculations made on similar cores have shown that with the group structure used, there is no appreciable effect of light element scattering cross-section changes on the reactivity change with voiding.

The isothermal sodium-temperature coefficient was obtained by running successive CAESAR cases accounting for the change in Na density, microscopic U-238 cross-sections, and resultant axial reflector savings changes for the entire core.

The critical mass was obtained by summing the total inventory of Pu-239 and -241 in the core and blanket regions. For most of the calculations this parameter was taken at the midequilibrium point in time with $k_{\text{eff}} = 1.0$. Axial and radial blanket inventories for this point were obtained from the fuel-management studies.

The overall breeding ratio and region power fractions were determined directly from absolute neutron balances which are direct output of CAESAR calculations. This breeding ratio is the static ratio of the rate of capture of neutrons in fertile isotopes (U-238 and Pu-240) to their rate of destruction in fissile isotopes (Pu-239 and -241). In this calculation the core and both blankets were considered.

Equilibrium fuel-cycle costs were obtained with the COST-I code. This code was developed on the Company-funded program for parameter survey work, and its intent is to show fuel-cycle cost differentials between various cores in the equilibrium condition. With this code its fuel-cycle cost calculations are made on the static equilibrium rates of throughput and production and of destruction rates of fissile isotopes, not on a material balance over the lifetime of the core.

6. Thermal-Hydraulic Methods of Analysis

The primary limitations on the sizing of the fuel elements are thermal and hydraulic. These limitations, assumed for both core and blanket elements are:

- 1) Maximum cladding temperature, 1200°F.
- 2) No fuel melting at 110% power, that is, maximum fuel temperatures are not to exceed 5070°F (2800°C). For the reference design this was further restricted to a maximum temperature 30°C below the melting point.
- 3) Peak sodium velocity, 30 ft/sec (32.8 ft/sec at the spacers).
- 4) Maximum linear heat ratings, ~15 kw/ft.
- 5) Fuel-pin diameter, 0.25 in. OD (0.015 clad, 0.003 cold gap).

Core-region fuel elements were sized by using the FAITH computer code developed on the Company-funded program.

Maximum fuel temperatures were calculated by the FTB-3 (Fuel Thermal Behavior) computer code also developed on the Company-funded program which describes the thermal behavior of gas-bonded ceramic fuel rods. The code calculates the expansion of the cladding, thermal conductivity of the gas bond, thickness of the fuel-clad gap, temperature drop across the gap, temperature distribution in the fuel, redistribution of the fuel due to sintering, and thermal expansion.

A maximum temperature of 2724°C at 110% of rated power was calculated for beginning-of-life conditions when the 0.0025-in. radial gap is filled with pure helium. Two competing thermal effects occur during reactor operation: 1) reduction in radial-gap width due to fuel growth; and 2) reduction of the effective gap conductivity due to fission-gas release. It is assumed for these studies that the fuel-growth effect predominates, thus decreasing fuel temperatures with lifetime.

G. DESIGN POINT SELECTION

The combined reactor heat-transfer energy-conversion system has been analyzed to determine the effects of core height, core ΔT , secondary-sodium system ΔT , and the steam cycle on system costs. The purpose of the analysis was to confirm that the design selected for Task I was near-optimum and provide a satisfactory base for Task III work. A matrix of nine core configurations and three steam cycles was studied. Core heights of about 3, 3-1/2 and 4 ft with reactor ΔT 's of 260, 280, and 300°F were considered, as were steam cycles of 2400 psi, 850/850 $^{\circ}\text{F}$, 900/900 $^{\circ}\text{F}$, and 950/950 $^{\circ}\text{F}$. The study ground rules are summarized in Table II-41. The differential costs as a function of the system variables were determined by an AI-developed system survey computer code.

The fuel-cycle costs, and core thermal and hydraulic characteristics reported previously in Section F were used as input for the code. The code first calculates capital costs for each major portion of the plant, i. e., reactor and building, primary and secondary loops, heat exchangers, and steam plant, and then computes a differential power cost based on summing the annual plant

TABLE II-41
STUDY GROUND RULES

Basic System

1060°F reactor Na outlet
950°F steam generator Na inlet
2400 psig, reheat steam conditions

Core

~15-kwt/ft maximum linear heatrate
~30-fps maximum coolant velocity
6-month refueling cycle
~75,000-Mwd/te average fuel burnup
3-yr inner blanket residence time
6-yr outer blanket residence time
12-in. axial blankets

Economic

13% annual capitalization rate
2-mill/kwh unavailability cost
0.80 plant factor

capitalization charge and the differential-unavailability (fuel-handling) and fuel-cycle costs. For each core height and steam condition the code iterates on primary- and secondary-loop ΔT 's until a minimum power cost is obtained.

The parameters of principal interest in this study were core height and reactor ΔT . Because of the unknowns involved in establishing a model to determine the comparative fuel-burnup relationship for the variation in reactor characteristics three burnup models were established. Most of the analysis was based on the first model, i. e., assumption of a constant peak burnup, which favored the shorter cores because there is an increase in the calculated average burnup due to lower peak-to-average power ratio. In the second model, in which a constant average burnup was assumed, no benefit was gained from

the improved peak-to-average. The third model considered fuel burnup limited by cladding ductility and strain. This considered fuel growth, fission-gas pressure, and clad temperature in establishing the limit. These three models probably bracket the uncertainty.

Figure II-26, with differential power costs plotted as a function of core height, shows the three fuel burnup models and the differential power costs for the demonstration plant. The capital-cost differences with core height are small and the main effect is due to the fuel-cycle cost differences. The cladding-ductility-limited burnup model is considered the most realistic, and the figure shows that with it there is only a small variation in costs with core height. The analysis shows a slight economic incentive for cores of about 40- to 42-in. high (>0.01 mills/kwh) instead of the reference 50-in. This difference, however, is within the limits of uncertainty.

Another factor influencing the selection of core height is the demonstration plant costs. Since the demonstration plant fuel element design might be used for the 1000-Mwe plant, the demonstration plant costs were considered. The dashed curve shows these costs. Because of the higher near-term fuel-fabrication costs, the demonstration plant is penalized significantly by shorter cores. Consideration of all these factors led to a decision to retain the 50-in. core for the Task I base design.

The detailed results of each case studied are given in Figures II-27, -28, and -29, which show the variation of the differential power cost with reactor core height, reactor ΔT , secondary ΔT , and steam temperature. The curves show that the range of ΔT 's selected was not broad enough to give the minimum-cost point, and extrapolation of the data shows that minimum costs would probably be at ΔT 's of 20 to 30°F higher. Because of the uncertainties in the input cost information, the base design primary ΔT of 300°F and secondary ΔT of 260°F were considered acceptable as near-optimum design. ΔT changes of this magnitude will not affect LMFBR planning. Further studies of plant parameters with improved cost estimates are planned in Task III.

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Figure II-26. Differential Power Costs as a Function of Core Height

Figure II-30 shows the variation of fuel cycle with core height, reactor ΔT , and steam temperature. These fuel-cycle costs include a correction for plant efficiency. Figures II-31, -32, and -33 show the variation of the plant capital cost with core height, reactor ΔT , and steam temperature. The capital costs shown are partial values intended for determination of cost differences; they include the differential costs of the heat exchangers, the primary and secondary loop, the reactor structure, the reactor building, the steam plant, and the differential power capability costs to correct for plant output as efficiency changes.

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Figure II-27. Power Cost, 2400 psig, $850^{\circ}/850^{\circ}$ F

4-24-68 UNCL

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Figure II-28. Power Cost, 2400 psig, $900^{\circ}/900^{\circ}\text{F}$

4-24-68 UNCL

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Figure II-29. Power Cost, 2400 psig, $950^{\circ}/950^{\circ}\text{F}$

4-24-68 UNCL

7692-4930

Figure II-30. Fuel Cycle Costs

4-24-68 UNCL

7692-4931

Figure II-31. Partial Plant Costs, 2400 psig, 850°/850°F

4-24-68 UNCL

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Figure II-32. Partial Plant Costs, 2400 psig, 900 $^{\circ}$ /900 $^{\circ}\text{F}$

4-24-68 UNCL

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Figure II-33. Partial Plant Costs, 2400 psig, 950°/950°F

From the analysis it was concluded that the selected base design was sufficiently near-optimum to provide a good basis for the Task III studies and for establishing LMFBR R & D requirements.

H. MATERIALS SELECTION

Selection of materials for the 1000-Mwe LMFBR has been based on studies performed on the North American-Rockwell-funded 500-Mwe demonstration plant design program. Major materials-selection studies were required for the fuel cladding, structural members, control-rod absorber, and steam generator. These are discussed in the following pages.

1. Cladding

The requirements for LMFBR cladding, even without the complicating effects of radiation, are severe. Cladding, fuel, and coolant are expected to interact chemically and/or mechanically. The primary requirement for cladding is that it remain intact during operation to nominal burnup, under both normal reactor operation and transients and abnormal conditions. Basic factors which have an impact on both the LMFBR parameters and R & D needs will be briefly reviewed here; these factors affect the ability to build core assemblies and the subsequent performance of the cladding in-pile. The relative importance of all factors must be judged against the backdrop of economic requirements for fuel performance.

Fabrication of core assemblies involves inputs from several sources which bear a direct relation to alloy selection. The procurement of cladding materials involves the sensitivity of fabrication operations and performance to alloy chemistry, specifications on previous treatment, those which affect reproducibility of mechanical properties, and those on dimensions.

Handling must also be considered a separate requirement in fabrication processes for fuel assemblies. In the actual pin-fabrication activity, due consideration to welds will involve specifications for inspection and flaw detection. AI's experience in design and fabrication of liquid-metal-cooled reactors has shown that an awareness of the factors which affect the ability to build components ultimately controls the success of the end product.

In contrast, the performance of cladding is directly related to the reactor environmental conditions. The conditions under which the cladding must survive will become more severe for the future generation of reactors. This increased demand stems from higher operating temperature, larger temperature gradients, longer residence time, and higher neutron fluence. The normal in-pile conditions that cladding experiences for as long as three years are:

- 1) Superimposed thermal and mechanical stresses under high thermal gradients and cyclic conditions;
- 2) Temperatures from 700 to 1400°F;
- 3) Neutron fluxes of 10^{16} neutrons/cm²-sec;
- 4) Neutron fluences up to 10^{24} neutrons/cm²;
- 5) Flowing sodium at velocities up to 40 ft/sec and under thermal gradients of 600°F in the sodium and 100°F between the cladding and the coolant; and
- 6) Chemical and mechanical interactions with the fuel.

Performance, like manufacture, is controlled by many factors. Unlike fabrication, however, performance deals with the response of the cladding to the reactor environment. The engineering approach to the performance of cladding is to compare extrapolated performance data with performance calculated from material behavior and design parameters. Two models were developed by AI for the response of fuel cladding to the service environment.

The first evaluates the performance of cladding materials in-pile as a composite phenomenon (see Figure II-34). The different periods of life as defined in the figure are:

IA: $t=0$ to $t = t_1$, is that time necessary for the smear density to increase to an as yet undetermined value, but before which the fuel mechanically interacts with the cladding.

IB: $t = t_1$ to $t = t_2$ is the time required to reach a critical stress necessary to match fuel-growth rate with the cladding strain rate.

II: $t = t_2$ to $t = t_f$ is the time the cladding will strain at the equilibrium rate leading to failure.

Figure II-34. The Three Periods of Cladding Life

The model serves to (1) extrapolate current information to more severe conditions such as those expected in LMFBR and (2) provide a means of assessing the relative importance of the environmental factors which influence performance of the cladding. For example, different cladding properties are important at different times in the life of the cladding. The effects of irradiation, sodium corrosion (possibly including the effect of ferrite layer formation), fuel-cladding compatibility, void formation, and temperature and time-temperature must be considered in the design model, since all of these can modify the effective creep strength of the cladding and influence the equilibrium growth rate. Fortunately, most of these phenomena can only modestly influence the equilibrium growth rate, based on existing data.

The other AI model is the CLADRS computer model for predicting fuel-pin performance. This builds upon the individual factors which may influence the behavior of fuel and cladding. Using appropriate equations which describe the individual properties and behavior of fuel and cladding, the fuel-pin behavior is computed as a function of service life.

This model has the added flexibility of changing the design parameters or the material characteristics to meet the design requirements with the optimum combination of materials and design parameters. The fuel properties and the

associated cladding interactions can be studied as well. The reliability of fuel-pin performance can be assessed from the variability of material properties and design tolerances.

Both analytical approaches help determine the factors which influence performance. For example, the effect of heat chemistry and prefabrication material treatment has been the subject of much evaluation in the various FBR programs. The degree of cold work acceptable and/or required in fuel-pin cladding is open to much conjecture. The information generally available indicates that the strength increase due to cold work is relatively short-lived for Type 304, while Type 316, if heat-treated properly, may retain its strength for the life of the cladding. Considerations in evaluating the importance of cold work are: the economic trade-off of cost of material and expected scrap rate during fabrication of the fuel elements for cold worked vs milled annealed tubing; the effect of increased strength of fuel performance due to cold work in the cladding; the effect of cold work on the elastic strain available; and the effect of time-at-temperature on the amount of residual cold work in the cladding at the time the fuel-growth is building up significant stresses in the fuel. The last item relates to the fact that recovery (removal by annealing) of cold work, which is diffusion-dependent, is aided by temperature and possibly high flux; therefore, essentially complete annealing could occur in the higher-temperature areas of the core before cladding restraint is involved in the fuel growth.

While many other examples can be cited, they can only suggest the need for trade studies and R & D work. The main point is that the LMFBR core-design formula should achieve a cost-effective balance between fabrication and performance. The economic sensitivity of a reactor to fuel-cladding performance warrants firm establishment of information permitting reliability assessment.

2. Structural Materials

Structural materials in the LMFBR may be subjected to the same environmental effects as the cladding, but generally to less severe levels. Service lifetimes, however, are expected to be much greater. The major material activity is to interpret relatively short-term corrosion, mechanical property,

and irradiation-effects data in a manner that permits reliable prediction of long-term effects. It is generally thought that unirradiated mechanical properties and sodium-corrosion effects in the LMFBR temperature range can be accommodated with little difficulty. Conservative design criteria generally will cause only minor penalties.

The major uncertainty requiring careful attention is the effect of irradiation. Classification of structural materials into three groups according to lifetime and neutron flux is possible. The first group consists of materials in the active core in such components as control rod and reactor-instrumentation guide-tubes; this group is subject to high flux, but can be replaced with reasonable effort. The second group includes the materials near the active core, such as grid, plenum, and vessel structures; these are subjected to a lower flux but are expected to last the life of the plant, and replacement may be difficult or time-consuming. The last group of structural materials is located in the region remote from the core and for which irradiation effects may be disregarded.

In the first group another problem besides radiation damage effects is present, viz., seizing and galling of moving parts. This is not only dependent on the material properties, but also on operating conditions and environment. Conditions associated with control-rod drive mechanisms and fuel-handling devices must be examined in terms of cold-welding in sodium, wear, seizing, and galling. Materials-selection criteria for moving parts in Na or the upper plenum should include this problem as one of the prime requirements.

In the second group, apart from the radiation problem, is that of joining for such components as the grid and other core structural members. Brazing has been considered an alternative to welding in many of the core components. The materials for these components and the brazing alloys to be used have an effect on plant availability, as component repair and maintenance are both costly and time-consuming. For the austenitic stainless steels, good brazing alloys for use in high-temperature sodium and under radiation are available. However, the high temperatures necessary to perform the joining operation will prevent the use of cold work as a means of giving added strength. Brazing

is not expected to cause extreme difficulties and may be a convenient means of preventing internal stress buildup associated with welding.

The third group, while far from radiation sources, generally involves the containment and associated piping ex-core for heat transfer to the steam generator. These materials are still subject to the non-nuclear aspects of the environment. Therefore many of the same factors which influence the selection of materials for the second group should also apply to this group. Based on AI's experience in successful fabrication and performance for Na and that of others, Type 304 stainless steel should be used for Na piping, valves, IHX's, pumps, auxiliary equipment, and structural core members. It should be possible to use medium-carbon steel of specified impact properties for the reactor containment vessel and superstructure which are isolated from the reactor environment and severe thermal conditions. The steam generator will be discussed separately.

Basic stress-allowable curves must be developed for each of the austenitic stainless steels used within the core. The basis for development of the curves will be the ASME boiler code or an interpretation of its intent, as appropriate. The values in the boiler code are intended as a base point for the design engineer at 100,000-hr life. Adjustments to the values are required to determine code values, after due consideration of the environmental effects on the properties such as irradiation, vibration, corrosion, time, and other parameters unique to a specific component. Thus each component may require that a unique adjustment be made to the base curve. This is done by working closely with stress analysis and by applying design-margin analysis techniques.

Several base curves may be required from the standpoint of life in the reactor. Core components which will spend different times in the reactor and probably be exposed to different flux levels will be categorized. Different base curves will be used for each category. In addition, the base curves may require adjustment for different components within specific categories.

At the reactor outlet the vessel temperature can reach 1060°F during normal operation in the final phase. In this temperature range the effects of creep may be significant. The design-allowable stress provides for a minimum creeprate of 0.01%/1000 hr, and this is usually less than 60% of the average

stress to cause rupture in 100,000 hr, or 80% of the minimum stress to rupture. Data was reviewed to insure that these stress levels are also less than 60% of the stress to cause rupture at a fluence of up to 5×10^{21} nvt. In this evaluation it is important to consider strain-rate effects, i. e., the data should be obtained from long-term creep rupture test if possible. Suitable allowances in the design should be provided for corrosion/erosion effects.

An example of an analytical technique developed at AI to provide a basic stress-allowable for Type 304 stainless steel in 30-yr service in LMFBR is given below. The treatment is based on an interpretation of the intent of the ASME boiler code and is useful in establishment of criteria. The application of the criteria to design is the function of stress analysis and design engineering.

The ASME boiler code specifies the allowable design stress to be the lowest of several stress criteria as illustrated in Figure II-35. By obtaining and cross-plotting the most recent ASTM data it is possible to extrapolate to 250,000 hr, resulting in the curves shown in Figure II-36.

A comparison of Figures II-35 and -36 shows that the creep curve intersects the yield stress curve at a lower temperature for the 250,000-hr case. Thus the design stress for 250,000 hr is limited by creep at 985°F while the 100,000-hr design stress is not limited by creep until 1080°F .

A dashed line with the notation "suggested thermal stability factor (10% arbitrary selection)" has been shown on Figure II-36. This additional conservatism is suggested as insurance against the probable deterioration of the strength of the alloy with very long times at temperature under tensile stress conditions.

The lower temperature portion of the allowable-stress curve in the figure is shown to be determined by the yield strength. Normally this portion of the curve would be conservative for an application in an irradiation field, since neutron radiation hardens the material with a resulting increase in ultimate and yield strength. In the absence of helium this effect is removed by annealing at higher temperatures ($>1000^{\circ}\text{F}$). The yield strength above 1000°F however, is not the determining limit at those temperatures. Data have not been obtained for the effect of irradiation on yield strength where the test sample

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Figure II-35. Comparison of Typical Curves to Determine a Design Curve for Type 304 Stainless Steel

has been stressed for long periods (>25,000 hr) while in the irradiation field. It is likely that the thermal-stability factor mentioned previously is necessary for long-term reactor applications. Since it is not known at this time what the effect of irradiation is on the thermal stability of the material, an arbitrary 10% thermal-stability factor is included in that portion of the allowable-stress curve which is determined by the yield strength of the material.

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Figure II-36. Comparison of Typical Curves After Extrapolation to 250,000 Hr, Type 304 Stainless Steel

It appears that irradiation has little effect on the creeprate of 304 stainless steel. However, the time to rupture is decreased significantly for a given stress. When this knowledge was applied to the balance of the allowable-stress curve, it was seen that the higher-temperature end of the curve was now dependent on the minimum stress-to-rupture curve and not the 1%-creep curve as was the case with unirradiated curves. A factor of one-half the minimum stress to produce rupture at 250,000 hr in unirradiated material was used as the best

irradiation factor for the upper end of the curve. The resultant curve (Figure II-37), though quite arbitrary, does provide a basis for establishing a conservative allowable design stress for 304 stainless steel (structural application) to 10^{23} nvt fast (>1 Mev). To test the curve current in-pile stress-rupture data of 304 stainless-steel bar stock were extrapolated to 250,000 hr (from <1000 hours). At 1200°F they gave the same degree of conservatism as is normal for a boiler code stress-allowable. At the lower temperatures, however, results are very conservative and may possibly be adjusted to higher values, but longer-time in-pile data are required before this action is fully justified.

3. Neutron-Absorber Materials

The control-poison materials can be selected from only a few candidates due to temperature-imposed limitations and material cross-section requirements. The control materials must retain their key properties in fluxes of 10^{15} nvt to exposures of 10^{23} nvt (>0.1 Mev) over a two- to three-year service life, withstand the radiation-induced thermal stresses, and have satisfactory properties under the conditions of operation. An ideal neutron-absorber material would have a high thermal conductivity and low coefficient of thermal expansion and elastic modulus to minimize internal stresses. It should remain inert to its environment and retain its reactivity worth and adequate strength for long periods of operation.

The primary candidates for meeting the control-material requirements are in two groups: (1) boron-containing materials, most specifically boron carbide; and (2) selected metallic materials having effective cross-section energy dependence such as tantalum or tantalum-tungsten mixtures.

a. Boron Carbide.

Boron carbide containing various enrichments of B-10 has been extensively evaluated for use as a control-rod material in many reactor systems. Using various approaches, it can be fabricated in a functional form including (1) powder with densities up to about 75 to 80% of theoretical, (2) hot-pressed up to nearly 100% of theoretical density, and (3) a composite warm-pressed with graphite and sintered to a boron content equivalent to about 40% of theoretically dense B_4C .

Figure II-37. AI Structural Member Design Stress-Allowable Curve for Irradiation to 10^{23} nvt

At operating temperatures of interest, all forms of B_4C will exhibit some swelling subject to a number of parameters, viz., temperature, particle size, compaction density, degree of graphitization, orientation of the structure, boron burnup, and thermal cycling. The preferred form due to its structural-integrity thermal properties and potential stability is the B_4C graphite mixture. The powder form which has been used both in loose-packed and hard-packed arrays frequently presents problems due to a low and not necessarily stable effective thermal conductivity. In addition full gas release cannot be expected. Swelling has been reported in irradiation tests of powder systems although temperatures and conditions were not directly applicable.

Whichever form of boron-containing neutron-absorber material is chosen, a cladding requirement is anticipated for both structural and environmental compatibility reasons. Destructive reactions between B_4C and 347 stainless steel, Multimet, Inconel-X, Haynes Alloy-25, and titanium have been reported at

temperatures above 1600°F. On the other hand, copper has been found to be an effective diffusion barrier between B_4C and stainless steel at or below 1300°F for periods up to 10,000 hr. In addition to the compatibility question, control-rod cladding penetration by Na raises a number of possible problems including carburization of core components, chemical-reaction-induced rod distortion, and control-margin uncertainties.

Neutron absorption in B_4C produces He and Li; thus it is a generator of a gas and a highly active alkali metal. Numerous instances are reported of large amounts of swelling in B_4C -containing materials; it is believed that application of this material will require tailoring of the containment structure on the B_4C support matrix to accommodate, restrain, and/or compensate for the swelling which can be erratic and unpredictable. Design of the B_4C matrix for operating conditions favoring gas release, accompanied by a mechanism for venting the released gases from the containment structure, is one suggested design approach. A dense B_4C matrix, on the other hand, will tend to retain the He, throwing some doubt on the stability of B_4C at the high burnups required. The composite B_4C -graphite system, while offering the advantage of providing a matrix which shrinks initially under the influence of irradiation and appears to enhance gas release, may still allow an unstable system to develop at sufficiently high burnups.

b. Tantalum

Neutron absorption in tantalum leads to the generation of tantalum and tungsten isotopes, which are useful absorber materials in the neutron spectrum of interest in fast reactors. Thus there is no problem with gas generation due to neutron absorption in a Ta control-rod system, and the nuclear-reaction products of this system increase the burnup life of the absorber column. In the temperature region of interest the irradiation stability of Ta is expected to be satisfactory if a cladding is used to provide environmental compatibility protection.

Austenitic stainless steel may be a suitable cladding, based on a limited number of experimental data points reported for the interdiffusion of Ta and austenitic stainless steel. While extrapolation of the data points to lifetimes and temperatures of interest is not certain, diffusion theory predicts that

interdiffusion of tantalum and austenitic stainless-steel cladding are not a serious design problem under LMFBR operating conditions.

A possible disadvantage of Ta may be an increase in heat generation to B_4C , thereby increasing thermal stresses. However, the high conductivity (compared to packed B_4C powder) and the expected irradiation stability of the material indicate that sufficient design options exist to alleviate this problem. A second possible disadvantage of a Ta absorber column may be its initial cost. The ultimate choice between alternate control materials must, however, be based on total costs for the life cycle of the control system and on the control worth required for long-time reactor operation. On this basis relative development costs may play a significant role.

In summary, each of the two classes of control-rod materials presents advantages and disadvantages. In the final comparison tantalum was selected based on the fact that no significant gas is released during the neutron-absorption processes and it would therefore have a longer operational life. Also, Ta is compatible with Na and has good thermal properties.

4. Steam Generator Materials

Safety and plant availability demand that the steam generator operate for the life of the reactor without major difficulties. Balanced against this demand is the cost of designing and building the steam generator to meet the desired economic objectives of the plant. Thus as in the case of fuel assemblies, fabrication and performance of the steam generator are to be engineered in terms of fixed and operational tests. Materials must meet the requirements of low-cost fabrication and reliable operation. From this point of view factors which affect the ability to build the steam generator and affect its performance have considerable impact on selection of the materials as well as on the basic design.

Of the commercial alloys available, several have potential for the steam generator. These are Inconel 600, Incoloy 800, austenitic stainless steels, and the ferritic 2-1/4 chrome-1Mo alloy steels. The cost of these alloys proceeds downward in the order of presentation. Procurement of material for

the steam generator must consider the sensitivity of both performance and fabrication processes to alloy chemistry specifications, specifications which affect reproducibility of mechanical and corrosion behavior, and dimensional tolerances. For Incoloy 800 it may not be possible because of the manufacturer's proprietary commitments to control the specifications on chemistry and previous treatment.

In addition to specifications on procurement, fabrication processes such as welding must be qualified and tested. From the standpoint of reliability, past experience in joining has a strong influence on the method and materials selected for service. This restriction tends to favor either the austenitic stainless steels or the ferritic alloys. Again, the ease and reliability of manufacture of the steam generator is sensitive to the alloy selected and therefore relevant to its successful operation.

Performance is dependent on three main factors; stress-corrosion cracking in the water environment, mass transfer on the sodium side, and heat-transfer characteristics. Inconel 600, Incoloy 800, and the 2-1/4Cr-1Mo alloy steel are all resistant to chloride-ion effects, particularly stress-corrosion cracking. Because the reliability of the steam generator would be severely impaired by chloride-ion attack, the austenitic stainless steels are marginal candidates. In terms of heat-transfer characteristics the ferritic alloy holds the edge on better properties. The problem with the ferritic alloys is the possibility of carbon transport in the Na can leave the steam generator material decarburized and cause the piping from the heat exchanger to be carburized. In addition, weldability of the ferritic alloy is not as good as in the other materials. Careful analysis has revealed, however, that mass transfer does not appear to be a significant problem if the loss in strength is factored into the design-allowables. In addition, the weld problem can be overcome by design and quality control. Since Inconel and the ferritic alloy have essentially the same stress-allowables at temperature-pressure, cost becomes the over-riding factor in favor of the 2-1/2 Cr-1Mo alloy.

In summary the ferritic alloy of the 2-1/4Cr-1Mo type is the best potential alloy for use in the steam generator of LMFBR. Lower cost, resistance to stress-corrosion cracking, and minimal loss of properties due to carbon transport in the sodium, together with ease and reliability in fabrication, form the basis of this selection.

III. SAFETY CONSIDERATIONS

In this section are discussions of safety criteria for fast reactors, DBA description, and containment approach. The loss-of-flow accident is discussed as a potential DBA candidate, and the consequences of a DBA are then discussed in more detail. These preliminary results indicate that if the containment integrity can be maintained the radiological hazard for the postulated site is well within the AEC guidelines.

Detailed evaluation of all major DBA candidates and radiological consequences are in progress as a part of Task III studies.

A. SAFETY GUIDELINES

The fundamental guidance for design safety considerations in the 1000-Mwe FBR powerplant is provided by the "General Design Criteria for Nuclear Power Construction Permits," 10 CFR 50, Appendix A.⁽¹⁰⁾ However, the implementation of the intent, if not the words, of these criteria will differ somewhat in form or emphasis in sodium-cooled fast-breeder reactors from that usually found in water power reactors. To provide a concise statement of the key or principal safety design requirements for the 1000-Mwe FBR facility, a subset of safety-oriented guidelines was developed. Although stated in relatively general terms, these guidelines generally have greater specificity than those in 10 CFR 50, and are not intended to be comprehensive or a replacement of the AEC General Design Criteria. They are intended to concisely highlight design criteria which recognize the principal special safety considerations or inherent features of a sodium-cooled FBR, have a major design influence, or describe major plant safety characteristics. These guidelines are listed, and are followed by a brief summary of the implementation approach taken in this design.

Guideline No. 1

The reactor building complex shall be designed to maintain mechanical integrity under DBA conditions so as to satisfy the radioactivity-release guidelines of applicable federal regulations.

Implementation

Containment of radioactivity release is achieved with a reactor building having two containment barriers, described below, to protect the public in a severe accident.

1) Inner Containment Barrier

The inner containment barrier, defined by the metal lining on the primary equipment vaults and the reactor containment dome, is designed to maintain a nitrogen atmosphere during reactor operation. The structure is designed to contain an energy release and accident-caused missiles greater than those of the DBA, and within leakage limits of 10%/day at 20 psig.

2) Outer Containment Barrier

A conventional all-welded steel containment building is provided for the outer containment pressure barrier. The containment building is designed for conditions which exceed the transients of pressure and temperature of the DBA and within leakage limits of 0.1%/day at 10 psig.

3) Engineered Safeguards

Engineered Safeguards are provided to maintain the effectiveness and integrity of the containment boundaries during and following severe accidents. Two systems are provided: the core-cavity-liner cooling system and the containment air-cleanup system.

Guideline No. 2

Design provisions shall be made to assure highly reliable and redundant normal and emergency core cooling systems.

Implementation

1) Reactor Cooling

Three independent heat-transfer loops are provided for normal and emergency reactor cooling. Each system is capable of independently providing 37% of rated full-load capacity. Each system is designed to provide the capability for independently removing decay heat by natural convection.

2) Core Coolant-Loss Protection

The primary-coolant system is designed to assure that the core and exit nozzles remain covered with coolant in a pipe rupture or reactor-vessel failure. This feature, together with an elevated-loop design, assures free-convection-cooling capability.

Guideline No. 3

The reactor core shall meet the following requirements:

- 1) The core shall be designed to provide an overall negative power coefficient at all operating conditions.
- 2) The integrated Doppler effect, from operating conditions to the onset of fuel damage, shall be sufficient to counteract all credible rapid-reactivity perturbations.
- 3) The reactor and fuel elements shall be designed to minimize interaction between fuel elements which could lead to significant fuel element-to-element failure propagation.

Implementation

The core is so designed that the combination of the positive coolant-temperature and negative fuel coefficients results in a net negative overall

power coefficient during startup and at power operating conditions. It is so designed that fuel subassembly bowing does not result in a net positive reactivity insertion. The power coefficient is negative in all core positions. The core and reactivity-control system are designed so that all credible rapid-reactivity perturbations are less than those which can be compensated by the negative Doppler effect. Each fuel element is provided with exit thermocouples to detect abnormal temperatures, permit control, and minimize damage.

Guideline No. 4

The control and protective systems shall be sufficiently independent, redundant, and rapid to prevent significant damage to the core from all credible accidents.

Implementation

Two independent systems of poison rods are provided for reactivity control, as described below.

1) Shim-Safety Control-Rod System

A redundant shim-safety-rod system is provided in which the rods will have the capability of being driven in or out of the core by positive electromechanical means and of being released for gravity drop. The maximum worth of the most effective shim-safety rod is limited so that a continuous rod withdrawal at full power will not cause fuel-rod damage in the hot channel.

2) Safety-Rod System

A redundant safety-rod system is provided which uses gravity drop for emergency shutdowns and provides a large refueling shutdown margin.

Guideline No. 5

The system design will provide highly reliable containment and control over liquid metals to prevent or limit fire and other chemical interactions.

Implementation

All sodium is contained in high-integrity piping systems equipped with appropriate relief devices and associated vent or containment vessels. Potential chemical interactions are further prevented or limited by the exclusion of water from areas containing sodium, appropriate selection of construction materials, and the use of reduced oxygen atmospheres. Isolation barriers and relief systems are provided where the use of water cannot be avoided.

B. DESCRIPTION OF DBA

A definite selection of the DBA cannot be made at this time, as the analysis of various hypothesized accidents in the 1000-Mwe FBR has not been made. It is expected, however, that the 1000-Mwe plant will share many characteristics with the demonstration plant, for which a preliminary analysis has been performed. For the Task I work it will be assumed that the DBA is initiated in the same manner as for the demonstration plant. Accident analysis for the 1000-Mwe plant will be made in Task III.

1. Sequence of Events

Under normal conditions the core is cooled by forced-convection flow through the three main heat-transfer circuits. Each contains a primary pump which circulates the sodium through the core, driven by a motor through a coupling which can be disengaged. The reactor will not be operated with less than two main heat-transfer circuits functioning; thus serious flow coastdowns would require a simultaneous loss of electric power to all the operating primary sodium-pump motors or a simultaneous decoupling of the pumps from their drives.

Each flow coastdown results in an increasing mismatch between reactor power generation and heat-rejection rate, which is normally corrected by the control system. If the transient exceeds the capability of the control system or it fails, the protective system will take action to reduce reactor power. The protective-system actions applicable to this transient are inhibit-alarm, setback, and reactor trip. Inhibit-alarm action warns the operator that an off-normal operating condition exists which, if left unchecked, may lead to a

reactor trip. Inhibit-alarm action also prevents control-rod withdrawal. (The control rods, however, are not withdrawn during a flow coastdown.) Setback action causes all shim-safety rods to be driven into the core at a rapid rate and reduces reactivity by as much as 50¢/sec. Setback by itself would probably be capable of shutting the reactor down without damage during a flow coastdown. Reactor trip causes the reactor to be rapidly shut down by the gravity insertion of control rods, which rapidly reduces reactivity by approximately \$24. Two sets of protective-system sensors based on different principles, flow and temperature, are each capable of causing a reactor trip in time to prevent core damage during a flow coastdown.

In a flow coastdown with no operator, control, or protective-system action the reactor sodium-outlet temperature would rise to the boiling point in approximately 10 to 16 sec. During that time the reactor power would decrease moderately due to the overall negative temperature coefficient of reactivity. The high two-phase pressure drop and self-pressurization effect of the boiling Na would tend to progressively void the core, starting from the highest coolant temperature regions. Due to the positive reactivity effect of voiding, reactor power would increase rapidly. This would cause very rapid fuel-pin centerline melting, leading to partial vaporization of the fuel. The transient would be terminated by disassembly of the core.

C. CONTAINMENT APPROACH

Containment as applied to nuclear power reactors means the control of radioactive effluents from "potential reactor accidents of exceedingly low probability of occurrence"⁽¹⁰⁾. These accidents, often referred to as maximum hypothetical accidents, have a probability of occurrence several orders of magnitude below those for which protection of the plant investment is provided. The control of the radioactive effluents is generally provided by one or more containment barriers along with several other "engineered safeguards" to assure that these barriers provide the necessary degree of containment with the desired margins of safety.

No inherent limitations have as yet been identified that would limit the site of a sodium-cooled FBR more severely than for an equivalent LWR.

Indeed, consider an analysis of the site dose to the critical organs, which used the best estimates for sodium retention of iodine, plateout, agglomeration, precipitation, and solubility of plutonium and fission products.

It can be shown that such an analysis would support the use of a single containment barrier with a leakage rate of the order of 0.1%/day at 10 psig. However, even though significant advances have been made in FBR safety technology in the last few years, a great deal of uncertainty still exists in both the analytical models and the physical constants involved in such an analysis.

This uncertainty is large enough to require the use of an additional containment barrier to provide an effective reduction of three or more orders of magnitude in the radioactive material release. Two orders of magnitude would be provided by an inner barrier with a leakage as high as 10%/day at peak accident pressures. An additional order of magnitude reduction would be provided by increased agglomeration, plateout, and precipitation if the volume of the inner containment barrier is considerably smaller than the outer containment barrier.

The specific containment scheme given here precedes much of the safety analysis to be made and as such is only tentative. Accordingly the proposed reference containment for the 1000-Mwe FBR is shown schematically in Figure III-1.

Because the reaction to a sudden reactor excursion may depend on whether the expansion is similar to a TNT explosion or a burning propellant, the system should be designed to contain both types of excursions.

In a TNT-type explosion, the following events occur:

- 1) A shock wave is generated, which, unless attenuated severely by the blanket elements and thermal shields, will rupture the vessel.
- 2) The ensuing blast pressure will force liquid fuel and sodium and fuel vapor into the inverted reactor-vessel cavity burning what oxygen (~2% maximum) is present.
- 3) Simultaneously the top shield will lift and release sodium and fuel to the containment-dome inner barrier increasing the pressure in the inverted dome to under 40 psig.

7-D8-291-29 A

Figure III-1. Reactor Containment

- 4) The rise of the top shield is controlled by a group of crush columns which transmits the load to a "strong back." The strong back in turn transmits the load directly to the bottom of the building structure through support columns.
- 5) As the pressure in the reactor cavity rises, the low-pressure diaphragms (3 to 5 psig) in the pipe chases rupture, relieving the gases to the upper equipment galleries.
- 6) As the pressure in the upper galleries rises, primarily due to the burning of the small amount of oxygen present ($\sim 2\%$ maximum O_2), the rupture diaphragms to the lower galleries give way resulting in a final gallery pressure under 20 psig.
- 7) The gallery pressure is reduced by wall cooling and operation of redundant gallery heat exchangers.
- 8) Leakage to the outer containment of fission products and plutonium occurs at the nominal rate of about 10%/day.
- 9) The outer-containment pressure rises to a pressure less than 0.2 psi. However, due to changing weather conditions the driving pressure for leakage from the outer containment barrier may be as high as 1 psig.
- 10) Leakage to the site occurs at the nominal rate of 0.1%/day at 10 psig.
- 11) The reactor cavity liner contains the remains of the molten core.
The cavity liner is cooled by two independent external cooling loops.

In a propellant-type excursion, it is expected that:

- 1) The reactor vessel integrity will hold.
- 2) The top shield will rise substantially, releasing sodium and plutonium and fission products to the containment-dome region, resulting in a pressure in that region of about 40 psig. More detailed analysis may show that it is desirable to vent the vessel or reactor floor to the equipment galleries to keep the top-hat pressure from exceeding 40 psig.

- 3) Leakage from the top hat to the outer containment will be nominally 10%/day as in the preceding case.
- 4) Leakage from the outer containment and postaccident core cooling will be the same as that in the TNT-explosion discussion.

A tentative evaluation of the resultant doses to the critical organs is given in the next section.

D. SITE EVALUATION

In evaluating the suitability of a proposed facility site, 10 CFR 100 (Reactor Site Criteria) requires the assumption of a fission-product release from a major accident hypothesized for this purpose. The resultant facility-environs radiation doses, which are calculated with the appropriate containment leakrates and meteorological conditions of the site, provide a basis for testing the adequacy of the site in this respect. More specifically three environmental boundaries or distances which are primarily based on population and land use considerations must be established. These are:

- 1) Exclusion area: an area surrounding the reactor which is completely under the control of the reactor operator or licensee.
- 2) Low-population zone: an area immediately surrounding the exclusion area with a population density and total number of people for which appropriate measures could be taken in their behalf in a serious accident.
- 3) Population center: a densely populated center of more than 25,000 people.

The radiation-dose reference values for these areas are discussed later in this section.

The following assumptions have been used in the radiological assessment of the DBA. These are conservative preliminary choices.

- 1) Fission-Product and Plutonium Core Release.
 - a) The reactor has operated continuously at full power (2500 Mwt) for 1 year and is operating at the time of the DBA.

- b) The entire reactor core melts releasing 100% of the noble gases and 100% of the halogens as aerosols. 19% of the core vaporizes releasing a like percentage of the remainder (i. e. , non-volatile) fission products and plutonium inventory as aerosols. All of this radioactivity is dispersed (along with considerable uranium and sodium) to the primary galleries as the vessel ruptures.

2) Containment Leakage Rates.

- a) The inner containment (primary galleries) atmosphere is assumed to leak immediately and continuously into the outer containment atmosphere at its design rate of 10%/24 hr (@ 20 psig and 100°F).
- b) The outer containment (reactor building) atmosphere is assumed to leak immediately and continuously to the environs at its design rate of 0.1%/24 hours (@ 10 psig and 100°F).

3) Radioactivity Depletion.

- a) Within the inner containment, particle agglomeration, settling, and plateout will be effective in reducing the radioactive aerosols which would leak. The assumed removal rates (atmosphere half-residence times) are: Pu and nonvolatile fission products, $T_{1/2} = 5$ hr; halogens, $T_{1/2} = 0.5$ hr; rare gases, $T_{1/2} = \infty$ (i. e. , no removal).
- b) Within the outer containment no activity removal by natural forces is assumed, but removal by the containment-atmosphere cleanup system occurs at one-building air volume per hour and at the following filtering efficiencies: Pu and non-volatile fission products, 99.9%; halogens, 90%; rare gases, 0%.

4) Radioactivity Transport to Environs.

- a) Volume source is based on uniform leakage over surface of reactor building.
- b) Pasquill Type F (strong-inversion) atmospheric diffusion conditions and constant wind direction and speed (1-m/sec) exist.

- c) No downwind reduction of the airborne radioactivity is assumed due to other mechanisms (e. g., fallout, wind direction or speed variance, weather changes).
- d) For direct building-dose calculations, 6 ft of concrete shielding over primary galleries and 2 ft in reactor building structure are assumed.

1. Radiological Evaluation of DBA

a. Dose Calculations.

The fission-product inventory was calculated with the CURIE portion of the CURIE-DOSE-THUNDERHEAD code (Appendix, Paragraph C). The Pu inventory was manually added. The total radioactivity inventory, modified by the appropriate release fractions, was used with the COMRADEX code to calculate the doses directly downwind. Internal doses due to inhalation of radioactive material were calculated for the lung, bone, and the thyroid as critical organs. A total whole-body dose was also derived, which is the sum of the direct dose from radioactivity held up in the building, external dose from the passing cloud, and the internal dose from the inhalation of isotopes having the total body as the critical organ. Two integrated exposure times were evaluated, 2 hr and 30 days after commencement of the accident. The significance of these two periods is associated with the reference exposure times stated in 10 CFR 100 for site evaluation (see b. below). The results are shown in Figures III-2 and -3.

b. Dose Criteria and Evaluation.

Specific dose-criteria guidance for site evaluation purposes is given only for the whole body and thyroid in 10 CFR 100. These specify that the thyroid dose should not exceed 300 rem and the whole-body 25 in the first 2 hr at any point on the exclusion-area boundary, and during the entire release at any point on the outer boundary of the low-population zone. The 30-day doses (Figure III-3) are intended to encompass the entire release period. However,

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Figure III-2. Design Basis Accident Two-Hour Dose

7950-5703

Figure III-3. Design Basis Accident Thirty-Day Dose

for this evaluation a bone dose of 150 rem and lung dose of 75 are assumed as the appropriate equivalences for the stated whole-body and thyroid doses*.

The calculated 2-hr doses at the given exclusion-area radius (2000 ft) are:

<u>Dosage (rem)</u>	<u>Dose Ratio (calculated/guideline)</u>
Bone - - - - - 15	0.10
Thyroid - - - - - 2	0.0067
Lung - - - - - 1	0.013
Whole Body - - - 0.05	0.0020

Similarly the calculated 30-day doses at the given low-population zone boundary (10 mi) are:

<u>Dosage (rem)</u>	<u>Dose Ratio (calculated/guideline)</u>
Bone - - - - - 3	0.020
Lung - - - - - 0.5	0.0067
Whole Body - - - 0.05	0.0020
Thyroid - - - - - 0.09	0.0003

These data show that even with the conservatism used in this preliminary site evaluation, a minimum safety factor of 10 for the exclusion-area-radius doses and about 50 for the low-population-zone doses exists with the given site.

* These dose guidelines are based on the approach recently taken by the Division of Materials Licensing of the AEC in evaluating the safety of a license application by the General Electric Co. (Docket No. 50-268), October 6, 1967. In that approach the intake of long-lived radionuclides shall not be greater than 50 times the annual intake permitted by 10 CFR 20.

APPENDIX

APPENDIX

A. JUSTIFICATION OF FUEL-SYSTEM PRESELECTION

The principal requirements that the fuel system must meet are:

- 1) Availability of a sufficient technological base by 1976-77 to permit design of a highly reliable core with satisfactory performance characteristics (i. e., hot-spot clad temperature of $>1200^{\circ}\text{F}$, a maximum nominal linear power rating of >15 kwt/ft, and an average core fuel exposure of >75 Mwd/kg).
- 2) The ability to take maximum advantage of the projected LWR fuel-cycle industry so as to minimize fabrication and reprocessing costs during the early LMFBR period. The target for combined core fabrication plus fuel recovery costs are: $\sim \$300/\text{kg}$ by 1985.

With respect to the availability of a sufficient technological base by 1976-77, mixed-oxides is the preferred fuel system. Considerable related operational and production experience is becoming available from LWR UO_2 use. Also, there will be a great deal of applicable information and experience from Pu-recycle in LWR's.

The first LMFBR ceramic-fuel experience on a large scale will be in the FFTF which will use mixed-oxide through the 1970's. The AI demonstration plant will provide additional experience starting about 1975. These two facilities plus the planned AEC development program will provide the production and technological bases necessary for the early commercial plant. In addition, all the prototypes currently planned and under construction by other countries will use mixed-oxide. These programs will add significant knowledge on mixed-oxide fuel.

Studies at AI have shown that mixed-oxide-fueled LMFBR's can achieve satisfactory economic performance if the above requirements are met.

The fuel-cycle costs for the base design 1000-Mwe LMFBR for various burnup levels are shown in Table A-1. Based on estimates of fuel-cycle costs

TABLE A-1

FBR FUEL-CYCLE COST VS CORE BURNUP (1986-1990 ECONOMY)

CORE BURNUP ITEM	COST (mills/kwh)		
	50 Mwd/KgH	75 Mwd/kgH	100 Mwd/kgH
Fabrication			
Core	0.40	0.28	0.24
Blanket	0.07	0.07	0.06
Fuel Recovery	0.25	0.18	0.16
Ex-Core Inventory	0.20	0.13	0.10
Net Pu Credit (after 2% loss)	-0.32	-0.29	-0.26
Fuel Expense Subtotal	0.60	0.37	0.30
In-Core Inventory	0.34	0.35	0.35
Total	0.94	0.72	0.65

for LWR's, achievement of a burnup of 75 Mwd/kgH will give the LMFBR a fuel-cost advantage of about 0.6 mill/kwh. This fuel-cycle cost advantage will permit the LMFBR to incur a capital cost disadvantage of up to about \$30/kwe above the LWR before reaching the overall breakeven point.

B. JUSTIFICATION OF CORE GEOMETRY

The potential safety implication of large positive sodium-void worth in FBR cores has led to several studies of high-leakage core geometries (modular, annular, pancake) as a means of reducing the void effect. Preliminary studies at AI were reported at the April 1967 ANS Fast Breeder Reactor Topical Meeting in San Francisco⁽¹¹⁾. These studies, in conjunction with the belief that a large (>\$1) positive sodium-void worth could not be allowed, formed the basis for the selection by AI of a modular core for the demonstration plant. Since the presentation in San Francisco, additional safety studies have shown that modular cores (and also annular) are subject to more severe containment DBA's than regular-geometry cores. (A regular-geometry core is one in which height

and diameter are constrained only by economic considerations.) These cores, in addition, are more economic, predictable, and require less physics and safety research and development. AI has, therefore, chosen a regular-core geometry for the 1000-Mwe Follow-On design.

Modular and annular cores are subject to more severe hypothetical accidents because of the potentially large ramprates associated with fuel melting and core compaction. This is due to the relatively large reduction in radial leakage which takes place upon axial core compaction. Unprotected loss-of-flow accidents lead to gross fuel melting and core compaction in low-void-worth cores. (Regular-geometry cores, however, are expected to be destroyed by voiding ramprates before meltdown occurs.) Analysis performed on the Company program has shown that, assuming coherent meltdown under one-g force, the maximum ramrate for a 500-Mwe modular core is about \$270/sec. A 500-Mwe regular-geometry core of the same 50-in. height has a corresponding maximum ramrate for meltdown of \$135/sec. The maximum ramrate associated with any voiding accident in the regular-geometry core is less than \$100/sec even under the most pessimistic assumptions. As another disadvantage the complex geometry and neutronic coupling of modular and annular cores causes a large uncertainty in the prediction of energy release during nuclear excursions.

A "pancake" core offers another way of reducing the sodium-void coefficient. However, for typical design studies where coolant velocity or coolant pressure-drop is held constant as height varies, the ramrate in accidents involving large gas bubbles increases as core height decreases, in spite of the reduced void effect. The reactivity worth per unit volume of void at the center of the core also increases as core height decreases. This means that the probability of a gas bubble of a given size causing a serious incident is higher for shorter cores. Reactivity ramprates for a large gas bubble going through the core at 30 ft/sec are shown in Table A-2 for the 500-Mwe demonstration plant. In addition, the shorter cores have smaller Doppler coefficients (as do modular and annular) and thus greater energy releases for a given ramrate. These gas-bubble ramprates can be reduced and Doppler coefficients increased by reducing the sodium-coolant velocity. However, this reduces the breeding ratio

TABLE A-2
GAS-BUBBLE REACTIVITY RAMPRATES

Core Height (in.)	Maximum Sodium-Void Effect (\$)	Height of Positive Void Region (in.)	Voiding Time, Positive Region (msec)	Maximum Ramprate (\$/sec)	Maximum Bubble Worth (¢/liter)
24	2.8	16	44	97	1.89
36	4.0	27	75	84	1.75
50	5.3	41.2	114	75	1.59

because of the larger sodium volume fraction, thus resulting in increased economic penalty. The effect of core height on loss-of-flow accidents has not been completely analyzed yet, but there may be a slight advantage for shorter cores.

In summary, modular and annular cores have more severe DBA's and were therefore rejected. The overall safety assessment of pancake vs regular-geometry cores is not clear at this time. Both appear to meet the AI safety guidelines, the regular-geometry core has been selected as the reference on the basis of better economic performance.

C. CODE DESCRIPTIONS

1. CAESAR-IV

CAESAR⁽¹²⁾ is a general multigroup one-dimensional diffusion-equation code programmed in the FORTRAN-IV language for an IBM 360 computer. The code is designed for use in conjunction with a builtin microscopic library, but macroscopic cross-sections and/or microscopic cross-section changes may be entered at execution time. Up to 35 groups, 20 regions, and 101 spacepoints may be used. Criticality searches based upon element concentrations, transverse buckling, homogeneous poison, dimension, fuel loading, or poison

boundary may be made. First-order perturbation calculations may also be made with CAESAR since the PERT code ⁽¹³⁾ is incorporated as a subroutine. Complete edits include median fission, absorption and flux energies, region-wise and total breeding ratios in the dimension calculated, neutron balances, isotope masses, flux and power profiles, and other pertinent reactor parameters. for our calculations, an 18-group library was used.

2. AILMOE

The AILMOE code ⁽¹⁴⁾, written in FORTRAN-IV for the IBM 360 computer, is the Atomics International version of the ELMOE Code ⁽¹⁵⁾. AILMOE performs a multigroup calculation of fast-reactor spectra by Fourier transform-techniques with particular emphasis on light element resonance scattering. 678 fine-energy groups encompass the energy range of 3.0 Mev to 22.6 ev. The output of major interest is the detailed flux spectrum and elastic-scattering removal cross-sections for the light elements. These removal cross-sections are calculated for any specified coarse multigroup scheme from the fine group data.

3. GRISM

GRISM ⁽¹⁶⁾ is an integrated nuclear data handling code consisting of several individual modules to be described below, which will generate multigroup neutron constants from a basic file of microscopic cross-section data. The basic modules are as follows:

a. GRAVE.

GRAVE (Group-Averaging) ⁽¹⁷⁾ is a program designed to compute multigroup cross-sections averaged over any given input flux spectrum from a master library of microscopic cross-sections. The master library in this case was developed by the Atomics International Cross-Section Analysis and Applications Project ⁽¹⁸⁾. The group cross-sections obtained are the total, elastic scattering, capture, fission, absorption, and inelastic scattering. Additional group parameters obtained are: ν , the average number of neutrons per fission; ξ , the average logarithmic energy decrement per elastic collision; and μ , the average cosine of the angle of scattering.

b. PRISM

PRISM⁽¹⁹⁾ calculates elastic, and total scattering matrices for any given group structure from a library of Legendre scattering coefficients derived from experimental data, from measured nuclear level data, and from optical model calculations where the measured data are not available.

c. TRIX-1

TRIX-1⁽²⁰⁾ calculates multigroup, Doppler-broadened, effective resonance integrals and cross-sections and their temperature coefficients for heterogeneous or homogeneous resonance absorbers. Using basic resonance parameters, geometry, and concentrations the code combines resolved resonance, unresolved resonance, $1/v$, and negative-energy resonance calculations. The theory is based on the single-level Breit-Wigner line shape with no overlapping from neighboring resonances. Analytical approximations based on the work of Chernick-Vernon and Goldstein-Cohen are used. For fissile isotopes, where the single-level calculation is inadequate, adjustments to the calculated integrals are made based upon the difference between the calculated and measured infinitely dilute integrals.

d. CONTRACT

CONTRACT⁽²¹⁾ takes the output from GRAVE and PRISM and prepares the data which can be transformed into CAESAR libraries.

4. COST-I

The COST-I code⁽²²⁾ developed on the Company program determines the equilibrium fuel-cycle economics for FBR cores, given the equilibrium nuclear characteristics of the core in question. Input items required are detailed fissile and fertile material neutron balances, masses of fissile and fertile isotopes, cost values for the various segments of the fuel cycle, viz., fabrication, reprocessing, shipping, interest on inventory and fabrication, and the time estimates for the various processes. Output includes regionwise and total fuel-cycle costs.

5. SIZZLE

SIZZLE⁽²³⁾ is a one-dimensional multigroup diffusion-theory burnup code which can calculate the burnup for both U-Pu and U-Th cycles. This code, like CAESAR, is designed for use in conjunction with a builtin microscopic cross-section library. The code functions as follows:

- 1) A one-dimensional diffusion-theory calculation is run on the specific reactor configuration to determine reactivity and spatial flux profiles;
- 2) The multigroup spectrum derived from this calculation is used to reduce the multigroup library to a few-group structure with up to six groups as specified by the user (all subsequent calculations to be made in this few-group structure);
- 3) A new spatial calculation is then run with the specified few-group structure;
- 4) The few-group fluxes thus obtained are then used to calculate the burnup over specified core areas (burnup calculations made region-wise) with time steps specified by the user; and
- 5) Alternate spatial and burnup calculations are then made until a reactivity or time limit, as desired by the user, is reached. Output of the code consists of reactivity, flux profiles, isotopic densities, and burnup by region as a function of time.

6. ANISN

The ANISN code⁽²⁴⁾ performs one-dimensional transport-theory calculations in the S_n and P_l approximations. A diffusion-theory calculation option is also available. A wide variety of boundary conditions, searches, and running options are available. (Reference 24 gives greater detail.)

7. PABUC

PABUC⁽²⁵⁾ is a generalized one-dimensional fuel-management and burnup extension of ANISN. Either diffusion-theory or discrete-ordinates (S_n) transport theory are available. In PABUC, flux and eigenvalue calculations are

made using a modified version of ANISN. The nuclide production and depletion chains are solved using the NUCY⁽²⁾ subroutine.

The nuclide chains are supplied as input data using a generalized format. Radioactive decay, neutron capture, and fissile capture can be specified in the chain descriptions. A single reaction (e. g., radioactive decay) can follow more than one path, with the branch probabilities supplied as input data. Direct yield from fission is allowed for convenience in treating fission-product buildup.

Fuel burnup is accomplished on a zone basis using fluxes from the ANISN calculation and burnup time steps specified by the user. The reactor power is renormalized to a specified constant value at each time step. The fluxes may be recalculated at any time step at the discretion of the user. At fuel-movement time at the end of any time step, any fraction of the fuel in a zone may be removed and replaced. Several batches of fuel can be intermingled within the same spatial zone without losing their individual identity. This feature of the program is particularly useful in describing the scatter-reloading fuel-management scheme.

8. FAITH

The FAITH (Fast AI Thermal Hydraulic) computer code sizes the core-region fuel elements. Developed by AI on a Company-funded program, the code assumes each fuel element to consist of a hexagonal array of fuel pins inside a hexagonally shaped fuel-element box. A fuel-pin spacing and fuel-element box size are determined by the requirement that the element flow-area pass sufficient sodium at a specified coolant velocity to obtain the required ΔT across the maximum-powered fuel element. The pressure drop across the fuel element is also determined by the element size and the coolant velocity. In this study fuel elements are sized for fixed coolant-velocity and ΔT .

In addition to determining the fuel-pin spacing and pressure drop the FAITH code calculates the thickness of the fuel-element box, the material volume fractions in the cell, the fuel clad hot-spot temperature, and the number of fuel elements required.

9. FTB-3

Maximum fuel temperature are calculated by the Fuel Thermal Behavior code developed on a Company-funded program. The FTB-3 code is a computer

program that described the thermal behavior of gas-bonded ceramic fuel rods. It calculates the expansion of the cladding, the thermal conductivity of the gas bond, the thickness of the fuel-clad gap, the temperature drop across the gap, the temperature distribution in the fuel, redistribution of the fuel due to sintering, and thermal expansion.

In the computer model the fuel is subdivided into three regions, each with a different densification and hence a different conductivity. Region I is that part of the fuel which is below 1600°C and has experienced no sintering or densification. The average conductivity in Region I (extending up to 1600°C) is 0.018 watts/cm-°C. Region II is an intermediate region between 1600 and 1800°C which has experienced equiaxial grain growth and partial densification. This region has a conductivity of 0.0236 watts/cm-°C. Region III is that part of the fuel above 1800°C. It has experienced columnar grain growth and essentially complete densification.

The above maximum temperature calculations are for beginning-of-life conditions when the 0.0025-in. radial gas gap is filled with pure helium. Two competing thermal effects occur during reactor operation:

- 1) Reduction in radial gap width due to fuel growth, and
- 2) Reduction of the effective gap conductivity due to fission-gas release.

For this study, it is assumed that the growth effect predominates; thus fuel temperatures decrease with lifetime.

10. CURIE-DOSE-THUNDERHEAD

The CURIE-DOSE-THUNDERHEAD program⁽²⁶⁾ is a combination of three basic computer programs used in succession. All are programmed in the FORTRAN-IV language for an IBM 360 computer. CURIE calculates fission-product buildup and decay as a function of reactor operating time, operating power, and decay time. DOSE uses the fission-product inventory obtained from CURIE to calculate the dose to 14 internal body organs due to inhalation of fission products released to the atmosphere and transported and diffused downwind. THUNDERHEAD also uses the fission-product data from CURIE to calculate

cloud gamma exposure from the released fission products. DOSE and THUNDERHEAD consider the fission products released from the reactor as a "puff" source, and the cloud diffusion is calculated from either the Sutton equation or the Convair modification.

11. COMRADEX

The COMRADEX program ⁽²⁷⁾ is a recent development improving on the DOSE-THUNDERHEAD portion of the CURIE-DOSE-THUNDERHEAD code. Programmed in the FORTRAN-IV language for an IBM 360 computer, it was specifically developed for calculating doses from postulated power-reactor accidents. With a radioactivity-source input obtained from CURIE or some similar code it permits the incorporation of as many as four successive levels of containment with time-varying leakrates. Filtration, cleanup, fallout, and plateout of activity in each containment shell can also be analyzed. The doses calculated include the direct gamma dose from the building (with shielding, if desired), the internal doses to as many as 14 organs including the thyroid, bone, lung, etc., from inhaling the contaminated air, and the external gamma doses from the cloud. The atmospheric diffusion model used is that of Pasquill.

12. System Optimization Code

This code is used to establish the entire plant economic trade-offs with system and design parameter variations such as reactor outlet temperature, primary-loop ΔT , steam generator sodium-inlet temperature, secondary-loop ΔT , reactor Δ_p or coolant velocity, steam conditions, reactor height, etc. The code calculates the variable capital costs, fuel-cycle costs and differential plant availability for each set of parameters and relates these to a total differential energy cost through an economic model.

The capital costs are calculated from equations relating individual component or subsystem costs to the various parameters and from calculated heat-exchanger areas. The fuel-cycle costs are included as curve fits of data obtained from other nuclear code calculations for various reactor heights, Δ_p 's, etc., and the unavailability is input for those parameters which have a significant contribution.

For a given set of specific prime parameters, i. e., core height, reactor outlet temperature, steam generator inlet temperature, and steam conditions

the code iterates on primary- and secondary-loop ΔT 's to achieve a minimum energy-cost condition. This is repeated for variations in any of the prime parameters to obtain overall optimum plant operating conditions and design trade-offs for the range of parameters desired.

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