LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:
A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.
As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

1000-MWe LMFBR FOLLOW-ON STUDY
— Task II and III Final Report —
Volume 4
Trade-Off Studies

Approved by B. J. Goulding
Project Engineer

M. W. Croft
Project Manager

November 1968

ANL Contract No. 31-109-38-1998
B&W Contract No. 59-3102

Prepared for
ARGONNE NATIONAL LABORATORY

by
BABCOCK & WILCOX
Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

DISCLAIMER

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

DISCLAIMER

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

PREFACE

Babcock & Wilcox (B&W) is participating in the AEC's 1000-MWe Fast Breeder Reactor Follow-On Study Program under Argonne National Laboratory (ANL) Contract No. 31-109-38-1998. As stated by ANL,

"The over-all objective of the current 1000-MWe LMFBR Follow-On Study Program is to provide the AEC with information needed to establish research and development programs which will lead to the achievement of economic and safe LMFBR power plants by a viable nuclear industry. This information is to be developed through studies of 1000-MWe LMFBR power plant reference conceptual designs favored by the nuclear industry for further development and eventual sale to the electric utilities in the period of 1975 to 1985."

The B&W Follow-On Study consists of four tasks:

Task I — The development of four basic plant concepts based on the established ground rules and the estimated state of technology in 1980, and the selection of a preliminary reference design that incorporates the best features of each concept.

Task II — Detailed engineering work on the preliminary reference design, and, in parallel, the performance of in-depth engineering trade-off studies.

Task III — Parametric studies to identify and evaluate basic problem areas and to provide a basis for the selection of an optimized reference design.

Task IV — Preparation of a document outlining the research, development, and component testing required for the successful design and construction of the reference 1000-MWe LMFBR design.

This five-volume report describes the work performed under Tasks II and III. Volume 4 describes the engineering trade-off studies carried out as part of the work under Task II. The titles of Volumes 1,

2, 3, and 5 are listed below, along with a brief description of the contents of each.

Volume 1 — Summary Description and Cost Estimate

Cost data and a description of the overall plant and the reference operating conditions and parameters.

Volume 2 — Conceptual System Design Descriptions (1 - 7)

Descriptions of land, structures, and conventional systems and the reactor and associated systems, such as shielding, control, and supports.

Volume 3 — Conceptual System Design Descriptions (8 - 14)

Descriptions of reactor auxiliary intermediate coolant, steam generator, energy conversion, instrumentation and control, electrical, and safety systems.

Volume 5 — Parametric Studies

Descriptions of the parametric studies carried out under Task III.

CONTENTS

1. INTRODUCTION
2. SPOILED VS UNSPOILED CORE TRADE-OFF STUDY
3. NUMBER OF PRIMARY AND INTERMEDIATE HEAT TRANSFER LOOPS TRADE-OFF STUDY
4. VENTED VS NON-VENTED FUEL PIN TRADE-OFF STUDY
5. STEAM CYCLE SELECTION TRADE-OFF STUDY
6. SHIPPING CASK TRADE-OFF STUDY
7. ON-SITE VS OFF-SITE REPROCESSING AND FABRICATION TRADE-OFF STUDY
8. PLANT ARRANGEMENT AND REFUELING METHOD TRADE-OFF STUDY

1. INTRODUCTION

CONTENTS

	Page
1. GENERAL.....	1
2. TRADE-OFF STUDIES	3
2.1. Spoiled Vs Unspoiled Core	3
2.2. Number of Primary and Intermediate Heat Transfer Loops	3
2.3. Vented Vs Non-Vented Fuel Pin.	4
2.4. Steam Cycle Selection	4
2.5. Shipping Cask.	4
2.6. On-Site Vs Off-Site Reprocessing and Fabrication ..	4
2.7. Plant Arrangement and Refueling Method	5
3. REFERENCE 1000-MWe LMFBR DESIGN	7

List of Tables

Table

1. Summary of Key Plant Parameters and Operating Conditions	12
--	----

List of Figures

Figure

1. Cutaway View of the Reactor Building	6
---	---

1. GENERAL

The seven trade-off studies covered a number of major areas of LMFBR technology as specified in the contract with ANL. Two or more alternate concepts, or approaches to the design of specific reactor features, were compared and evaluated with a view to selecting the superior alternate for inclusion in the reference design LMFBR. The studies also provided a means for qualitatively assessing the effects on power costs of partial or non-attainment of R&D accomplishments to be identified during Task IV of the Program.

The trade-off studies were performed over a period of about 6 months, in parallel with the engineering of the reference design plant. In certain instances, the trade-off studies enabled us to make design decisions as the work on the reference plant progressed. In other cases, the results indicated that certain features selected for the reference plant on a preliminary basis were not optimum choices, so alternate features were substituted. In still other instances, the trade-offs substantiated the choices originally selected for the reference plant.

For convenience, all the trade-off studies are included in this volume of the Task II and III report. Actually, each study was initiated and performed separately and independently of the other studies except for the normal liaison required in related studies. The reports appear in this volume in the sequence in which they were completed; there is no other significance to the order. Each report was written as a complete document, with its own table of contents, introduction, conclusions, and discussions, but each follows a similar overall format.

Since the trade-off studies were performed in parallel with the engineering work on the conceptual design, the reference plant is slightly different in each of the separate studies. In addition, the reference design changed as superior alternates were defined in the individual studies and incorporated into the plant, and it continued to evolve toward

the optimum after the trade-off studies were completed. The effects of these studies on the development of the conceptual design are summarized in section 2, and the final conceptual design is described briefly in section 3 to provide a frame of reference for the trade-off studies included in this volume. Additional information on the reference 1000-MWe LMFBR is given in Volume 1 of this report, a complete summary, and in Volumes 2 and 3, which describe each major system in the plant.

2. TRADE-OFF STUDIES

The role played by these trade-off studies in the development of the final reference design LMFBR is discussed in the following paragraphs.

2.1. Spoiled Vs Unspoiled Core

In the preliminary reference design, BeO was used to spoil the core to achieve a minimum Δk_{void} . This trade-off study showed that the fuel cost penalty for the degree of spoiling was prohibitively uneconomical and unattractive from the standpoint of fuel conservation. Furthermore, other evaluations performed as a result of this work indicated that extreme spoiling was unnecessary to achieve the required safety characteristics.

We therefore adopted the design criteria of partial voiding of the core, which permitted us to design the core for optimum economics and doubling time. Greater emphasis was also placed on incorporating engineered safeguards in the reference plant design to eliminate the possibility of total voiding.

2.2. Number of Primary and Intermediate Heat Transfer Loops

Of the six different configurations studied, one was at least 10% higher in capital cost than the others; however, there was only a 3% difference in capital cost between the least expensive and the most expensive of the other five arrangements. Although there was no clear economic incentive offered by any one system, other considerations, such as ease of maintenance, reliability, and potential for reducing component design problems, led to the decision to retain the system chosen for the preliminary reference plant.

2.3. Vented Vs Non-Vented Fuel Pin

This study concluded that a vented fuel pin was economically superior to a non-vented design, and that the porous plug vent had more advantages over the other types with respect to economics, nuclear performance, and safety; contingent upon a successful R&D program. The study therefore confirmed the preliminary selection of a vented fuel concept during Task I of the Program.

2.4. Steam Cycle Selection

In the Task I work, a 2400 psig/1000 F/1000 F steam cycle was selected for the preliminary reference plant. This cycle required bi-metallic tubes, i.e., stainless steel in the superheater section and Croloy 2-1/4 in the evaporator section. However, during the performance of the trade-off study, additional cost data on the bi-metallic tubes became available. As a result we recommended a 2400/950/950 steam cycle for the reference design, which permits the use of Croloy 2-1/4 in the steam generator tubes.

2.5. Shipping Cask

The preliminary reference design was based on using conventional fuel storage and shipment methods to the extent possible. This criterion resulted in a decision to store and ship the fuel in water.

The trade-off study showed that in order to ship in water, a storage period of about 1 year was required; this was prohibitively expensive owing to the high plutonium carrying charges. The study also showed that fuel could be shipped in a sodium-filled cask after only about 70 days of storage; it was therefore decided to store fuel in sodium at the site and thus eliminate the requirement for steam cleaning of the spent fuel assemblies. These features were then adopted for the reference design on the basis that there was sufficient economic incentive to perform the required R&D to permit shipment in sodium.

2.6. On-Site Vs Off-Site Reprocessing and Fabrication

In this study, we examined the economics of on-site and off-site reprocessing and fabrication to permit assessment of the most economical approach when the costs for the various parts of the fuel cycle,

particularly those associated with the shipping of spent LMFBR fuel, could be defined more accurately. Off-site fuel reprocessing and fabrication was selected for the reference design because it represented the expected near-term approach for the LMFBR industry.

2.7. Plant Arrangement and Refueling Method

This study confirmed the choice made in the preliminary reference design to use a pot arrangement with under-the-plug refueling. The study permitted us to make a more detailed and careful feasibility and economic comparison of the four alternate arrangement and refueling method combinations than was made for the preliminary design.

Figure 1. Sectional View of Reactor Building

3. REFERENCE 1000-MWe LMFBR DESIGN

The B&W LMFBR power plant is designed as a complete, central station, electrical generating facility. It includes all systems, components, structures, and services necessary for construction of the plant on the AEC's hypothetical site in New England (as described in TID-7025, "Guide to Nuclear Power Cost Evaluation"). Figure 1 is a sectional view of the reactor building showing the general location of major equipment in the building. The key plant parameters and operating conditions are summarized in Table 1.

The heat generating equipment, which includes all the primary reactor equipment, the fuel handling facilities and radioactive auxiliary systems, is located in the containment structure. The heat transfer equipment, including the intermediate sodium coolant system, is on the reactor island between the containment structure and the turbine island. The reactor system is an integral pot with an under-the-plug refueling system. All the primary system components, including the core, core pressure vessel, radial neutron shielding, primary pump, IHX, irradiated fuel decay storage, and cold traps, are housed inside the reactor vessel.

The nuclear heat generated in the reactor system is transferred to the energy conversion system by a separate, intermediate sodium system. The intermediate system transfers the reactor heat from the primary sodium through the six IHXs and conducts it to the three once-through steam generators and three reheaters via three sodium pumps. Two IHXs and one pump serve one steam generator and one reheater to form one intermediate circuit; hence, the total intermediate sodium system is equally divided among the three intermediate circuits. The steam generator and the reheater are connected in parallel, and a valved bypass is connected in parallel for use during partial-load operation.

The energy conversion system utilizes a tandem-compound, six-flow, reheat turbine with 30-inch-diameter last-stage buckets. The full-power turbine output at the operating exhaust pressure of 1.5 inches of Hg is 1,050,790 kW. The generator is a conventional 3600 rpm unit with a gross output of 1,033,926 kW and an average auxiliary load of 27,000 kWe; therefore, the net plant output is 1,006,926 kWe.

The nuclear steam system produces 2450 MWe, and the complex of nuclear and turbine islands produces 1007 MWe net at a thermal efficiency of 41.2%. Coolant enters the core at 800 F, flows up through the core at a rate of 92,700,000 lb/hr, and exits at a bulk temperature of 1100 F. The intermediate heat exchangers (IHXs) transfer reactor heat to the three-loop intermediate nonradioactive sodium system and increase the secondary sodium temperature from 700 to 1000 F. Steam at 2400 psig and 950 F is generated in three once-through steam generators and flows to the high-pressure cylinder of the tandem-compound, six-flow reheat turbine. Three sodium reheaters raise the temperature of the exhaust steam to 950 F before it enters the intermediate-pressure cylinder.

The cylindrical core utilizes a hexagonal, removable, solid can design to enable individual orificing with vented-to-sodium fuel pins. The fuel pin diameter is 0.280 inch, and the pins are spaced on a 0.337-inch triangular pitch. The cladding, 10-mil, 304 stainless steel, encloses vibratory compacted (85% dense), enriched $\text{UO}_2\text{-PuO}_2$. The core contains 288 fuel assemblies and is surrounded by a radial and an axial blanket. The L/D ratio is 0.29 and the active length is 34.7 inches. The fissile loading of 2770 kg results in an average core enrichment of 11.6%. The average fuel burnup is 100,000 MWD/T, and the breeding ratio of 1.36 provides a doubling time of 10.4 years.

In the B&W integral pot system, the reactor vessel contains the primary sodium, the core, and all primary system components. This 52-foot-diameter vessel is offset from the geometric plan center of the building to provide a more efficient and compact reactor building layout. A biological concrete shield of 6.5 feet radial thickness completely surrounds the reactor vessel and supports the vessel and its cover structure. The inside surface of the biological shield is steel lined, and the

reactor vessel is located near the biological shield to minimize a loss of primary coolant. Personnel access is provided between the radial biological shield and the containment building.

The cylindrical reactor vessel has a semi-elliptical bottom and a flat, shielded, vault-cover structure on top. The reactor vessel cover supports all the primary equipment and is directly attached to the cylindrical wall of the reactor vessel and forms the pressure vessel head. The head is a radial plate beam with a pressure membrane on its underside. The upper side of the head is at the operating floor level. Three eccentrically mounted plugs, supported by bearings to facilitate indexing, are located in the cover structure above the core. One large rotating plug is centrally located in the cover structure and contains two other plugs eccentrically mounted to each other to facilitate the refueling operation.

The refueling and reactor support system provides for safe and reliable handling of all reactor core assemblies from the receipt of new assemblies at the site, through the reactor loading and unloading stages, and then through cask loading in preparation for shipment to the reprocessing area. It includes a fuel transfer machine, mounted off center of the inner plug, which enables the fuel assemblies to be moved from the core and transferred under sodium to either of the two rotating storage racks in the reactor vessel by a combination of rotary plug movements. One third of the fuel assemblies are removed from the core annually while the reactor is in a hot shutdown condition.

After a prescribed decay period in the rotating storage racks, and with the reactor on-load, the irradiated fuel is removed through the reactor cover structure by a gas-cooled, fully shielded transfer machine which is permanently attached to a floor-mounted gantry straddling the reactor cover structure. This machine loads new core assemblies into the storage racks and installs the refueling mechanism and any other equipment that is required to penetrate the reactor vessel. The machine also transfers all core assemblies into and out of the reactor containment through a double-valved transfer tube.

The reactivity control system consists of conventional regulating assemblies with solid-absorber columns; each assembly is connected by a connecting rod to its top-mounted, hermetically sealed drive unit

located in the intermediate, eccentrically mounted plug. The drive rod actuators protrude above the charge face level and contain the electric motors and gears. The regulating, shim, and safety rods are identical in design.

The cylindrical core vessel is centrally located in the reactor vessel and is supported from brackets attached to the upper strake of the reactor vessel. The vessel has a semi-elliptical bottom and a flat top grid structure. This structure forms the grid plate which supports and holds the lower end of the fuel and blanket assemblies for the reactor core. The wall of the vessel extends up around the core to form a flow divider between the core and the lower neutron shields.

The primary shielding system comprises axial and radial neutron and biological shields. Combination radial thermal-neutron shields are placed around the core to protect the primary equipment and the intermediate sodium from neutron activation. These shields are placed in hexagonal cans supported by the core tank structure in the vertical position. The neutron shield is boronated graphite. The primary radial biological shield is concrete and the axial shield is concrete and steel. The radial neutron shield satisfies the neutron shielding requirements as well as the thermal shielding requirements. The axial and the radial biological shields protect against radiation from spent fuel and the core and coolant.

Located in the outer section containing the 800 F sodium between the radial thermal-neutron shields and the reactor vessel are the six primary sodium pumps, six IHXs, two cold traps, and two emergency cooling systems. The primary pumps are located in the cold leg of the circuit and force the sodium coolant from the sump into the lower core pressure chamber, up through the core to the upper header chamber, then by gravity through the shell side of the IHX, and back to the sump.

The six vertically mounted IHXs are gravity-flow, straight-through, shell-and-tube units. Intermediate sodium enters the IHXs in a header above the charge face floor and is conducted through a centrally located downcomer to a plenum at the lower end of the tube bundle. From this plenum, sodium flows up through the bundle tubes, passes into an upper plenum, and is conducted back to the charge face. Blocking valves are

included as an integral part of the IHX to prevent the gross dumping of hot sodium into the pot when flow to an intermediate loop is lost.

The three intermediate sodium pumps mounted in the cold leg are mechanical, vertical-shaft, mixed-flow units with a free-gas surface. The pumps are driven by fixed-speed electric motors through a variable-speed, eddy-current coupling; an oil-lubricated face seal provides a leak-tight barrier between the cover gas and the atmosphere.

Both the steam generator and the reheater have once-through designs on both the sodium and the water sides and have single-tube walls. The heat transfer surface is a helical coil, single-bundle design with counterflow of water and sodium. Intermediate sodium enters the shell through a torus at the top and flows over the tubes while the water and steam flow through the tubes. Steam from the steam generator is fed to the high-pressure turbine, and the exit steam taken from the last stage of the high-pressure turbine is recirculated through the sodium reheaters. From the reheater, the steam is passed back into the turbine house and connected to the intermediate-pressure turbine. Facilities are available for the storage, purification, and transfer (as needed) of sodium for the primary and intermediate systems.

Table 1. Summary of Key Plant Parameters and Operating Conditions

<u>Plant</u>	
System configuration	Primary tank
Plant life, yr	30
Refueling scheme	Under-rotary plug then through-plug shield cask
Fuel decay storage location	Primary tank
Primary loops	6 (open)
Intermediate loops	3
Control—shim and regulating-safety rods	Mechanical
Cylindrical containment building diameter, ft	100
<u>Core</u>	
Geometry	Cylindrical
L/D ratio	0.29
No. fuel assemblies	
Core	288
Radial blanket	138
Volume fractions	
Fuel	0.442
Steel	0.169
Sodium	0.353
Control	0.035
Fissile loading, kg	2770
Average enrichment, %	11.6
Average burnup, MWD/T	100,000
Core BR	0.93
Total BR	1.36
Average linear heat rate, kW/ft	8.7
Specific power, MWt/kg fissile material	0.885
k void (EOL), \$	7.5
Doppler effect, $-T(dk/dt)$	0.005
Doubling time, yr	10.4

Table 1. (Cont'd)

Operating Conditions

Primary circuit

Inlet temperature, F	800
Bulk outlet temperature, F	1100
Nominal hot channel cladding OD, F	1160
Nominal hot channel fuel central BOL/OEC, F	3896/3800
Plenum-to-plenum pressure drop, psi	34.4
Flow rates, lb/hr $\times 10^6$	
Core and axial blankets	83.78
Control rods	3.40
Radial blanket	3.97
Thermal shields	0.55
Leakage	1.00
Total	92.70

Intermediate circuit

Inlet temperature, F	700
Outlet temperature, F	1000
Flow rates, lb/hr $\times 10^6$	
Steam generator	66.69
Reheater	25.41

Energy conversion

Steam cycle, psig/F/F	2400/950/950
Reactor thermal power, MW	2450
Electrical power, MWe gross	1034
Electrical power, MWe net	1008
Net plant efficiency, %	41.2

2. SPOILED VS UNSPOILED CORE
TRADE-OFF STUDY

CONTENTS

	Page
1. INTRODUCTION	1
2. SUMMARY	2
3. CONCLUSIONS AND RECOMMENDATIONS	3
4. SYSTEM REQUIREMENTS	4
5. DISCUSSION	5
5.1. Safety Characteristics	5
5.2. Methods of Spoiling	6
5.3. Description of Trade-Off	7
6. RESULTS	12
APPENDIX — Methods and Tables	A-1

List of Tables

Table

1. Physical Parameters	11
2. Fuel Cost Summary and Miscellaneous Parameters	14
3. Summary of Results	17
4. Partial Voiding Results, Δk_{void}	20
A-1. CCD	A-6
A-2. CCD .14 BeO	A-9
A-3. CCD .20 BeO	A-12
A-4. CCD .32 BeO	A-15
A-5. ICI	A-18
A-6. ICD	A-21
A-7. PCW	A-24
A-8. PCD	A-27
A-9. ACW	A-30

List of Figures

Figure		Page
1.	Effect of Fuel Average Burnup on Δk_{void}	13
2.	Fuel Costs for Spoiled and Unspoiled LMFBR for Single-Batch Refueling	16
3.	Fuel Costs for Spoiled and Unspoiled LMFBR, Including Capital Cost Penalty	19
A-1.	Minimum Fuel Cost at Each Pitch for a 40-Inch LMFBR Core at 100 psi Core Pressure Drop	A-4

1. INTRODUCTION

The goal of the national LMFBR program is the development of sodium-cooled fast breeder reactors that are safe and economical. Safety can be divided into two major categories: The first and most important is the protection of plant personnel and the general public from the maximum hypothetical accident and its consequences. The second is the protection of the reactor owner's capital investment.

However, if the LMFBR concept is very safe but as a consequence has high power costs, then there is no incentive for further development work. A clearer picture of the relative position of safety and economics in the overall reactor design is therefore essential if the LMFBR is to be developed on a timely basis. The 1000-MWe LMFBR Follow-On Studies are an important part of that development effort.

This trade-off study is intended to demonstrate the economic penalties associated with spoiling an LMFBR core. Spoiling is defined as any design feature that is introduced specifically to improve safety characteristics, such as the coolant voiding and Doppler coefficients, without regard for economics. Invariably, spoiling increases fuel costs. Since safety considerations are of paramount importance, the economic penalties to be expected for various types and degrees of spoiling must be determined. In addition, the relative merits of the chief types of spoiling must be established. Thus, on the basis of this study, a type of spoiling can be chosen. The safety philosophy permitting the reactor designer to choose exactly the required degree of spoiling remains to be developed.

2. SUMMARY

Two safety characteristics are compared in this study: the coolant void coefficient, which is positive, and the Doppler coefficient, which is negative. The interplay of these characteristics determines the operating and safety behavior of the LMFBR core.

The effects of three types of spoiling in varying degrees on these safety characteristics are evaluated, and the fuel cost and doubling time penalties associated with the spoiling are calculated and compared. Geometric spoiling (with both small and large coolant fractions) and BeO spoiling are considered.

As expected, there is a large fuel cost penalty associated with any significant degree of spoiling. If a zero coefficient for voiding of the core and axial blanket is required, then the penalty for the most favorable spoiling method will be more than 0.9 to 1.0 mill/kWh as compared with an unspoiled core. The fuel cost penalty for spoiling is not strongly dependent on the type of spoiling. However, geometric spoiling with a low coolant fraction is the best method.

If capital cost penalties associated with spoiling are included, the extrapolated penalty for a zero void coefficient becomes approximately 1.6 mill/kWh, and BeO spoiling is the best method.

3. CONCLUSIONS AND RECOMMENDATIONS

1. Spoiling in any form is detrimental to fuel costs. However, of the three types of spoiling considered (BeO spoiling, geometric spoiling with low coolant fraction, and geometric spoiling with high coolant fraction) the first two are preferable in terms of safety and economics.

2. The safety criterion suggested by other manufacturers, that the reactivity insertion for voiding the core and axial blanket be zero or negative, appears to be unfeasible from the standpoint of economics and conservation of natural resources. Therefore, greater emphasis must be placed on the engineered safeguards to eliminate total voiding from the list of credible accidents.

3. With partial voiding as the design criterion, the lowest feasible coolant fraction is preferred. Cores with high coolant fractions are strongly dependent upon the increased leakage when sodium is removed. With partial voiding, the leakage component is reduced.

4. Void deficits should be calculated in the actual geometry at end-of-cycle conditions. Failure to show the plutonium buildup in the near blankets results in an optimistic Δk_{void} .

5. A comparison of the calculated Doppler coefficients clearly indicates that, of the three methods of spoiling considered, geometric spoiling with a low coolant fraction and BeO spoiling are preferable. This is especially evident when the Doppler coefficients with the core half voided are compared. Since the Doppler coefficient limits the DBA energy release, an adequate Doppler coefficient is very important.

4. SYSTEM REQUIREMENTS

1. The prompt power coefficient shall be negative everywhere in the core.
2. The core pressure drop shall be less than 100 psi.
3. The maximum linear heat rate is 17 kW/ft.
4. The fuel pins for all cores in the study shall have the same total active length.
5. The unspoiled core that serves as the basis for comparing the effects of spoiling is designed without regard to sodium voiding.
6. DBA considerations will not affect the choice of method or degree of spoiling in this study. Preliminary analyses during Task I indicated that the Bethe-Tait energy release is not strongly geometry-dependent.
7. The degree of spoiling required for safe operation is not specified. The intent of the study is to determine the cost penalties associated with spoiling. As a result, none of the cores are designed to a specific set of criteria.

5. DISCUSSION

5.1. Safety Characteristics

Two LMFBR safety characteristics are considered: sodium voiding, and the Doppler coefficient. Although fuel slumping may be important, it will not be treated here. Even though other characteristics exist, those just mentioned can occur most rapidly and be the largest.

The effects of the sodium void coefficient, which is potentially the most serious of the positive reactivity coefficients, are encountered as a change in coolant density, a local voiding, or a gross core voiding. It can be argued that total core voiding is not credible; and that if it is, a positive void coefficient is unimportant since the core will be destroyed in any event. Local voiding presents a different picture. In this case the maximum worth of the voided assemblies must be limited so that the control system can prevent the spread of damage. Finally, the reactivity change resulting from a sodium density change following a change in power must be kept small so that the power coefficient is negative.

The sodium void coefficient is the algebraic sum of three components, two of which are positive; i.e., as sodium is removed, reactivity is added to the core. These are the sodium capture and the spectral components. The first of these components, which is small, is simply the result of removing an absorber (sodium) from the core. The spectral component results from the shift in the flux spectrum as the sodium, which acts as a moderator, is removed. Fuel materials, notably ^{238}U , ^{239}Pu , and ^{240}Pu are more efficient in a harder spectrum.

The negative component of the sodium void coefficient is caused by a change in leakage. Removal of the sodium makes the core relatively more transparent to neutrons and hence tends to increase the leakage.

The Doppler coefficient is a measure of the reactivity change that is caused by a change in the absorption rates in fertile materials as the

fuel temperature changes with power. Since the Doppler coefficient is the only prompt negative power coefficient in a plutonium-fueled LMFBR, it is essential that the Doppler coefficient be large enough to ensure a negative power coefficient. In addition, a large Doppler coefficient can limit the energy release during the design basis accident and hence ease the containment design requirements.

5.2. Methods of Spoiling

Spoiling can act on either the coolant void coefficient or the Doppler coefficient, or both. However, as will be shown later, one method of spoiling improves voiding at the expense of the Doppler coefficient.

Two types of spoiling are considered: geometric and material spoiling. Geometric spoiling acts on the leakage component by increasing the total leakage. Hence, in a high-leakage core, the absolute change in reactivity due to a change in leakage is greater than for a lower-leakage core.

Geometric spoiling can enhance either radial or axial leakage. Annular and modular geometries increase the radial leakage, while the flattened cylinder or pancake geometry enhances the axial leakage. Previous studies have shown that modular geometry has no advantages over annular geometry from the sodium voiding standpoint, and the loosely coupled modules may result in control problems. In addition, two-dimensional calculations are necessary for meaningful results. Based on these factors, it was decided not to include modular geometry in the study.

Material spoiling can act on either the leakage or the spectral component, or both. For high-leakage cores, increasing the coolant fraction lowers the void coefficient. A side effect, which tends to make the void coefficient less positive, is the corresponding reduction in fuel volume fraction and the resultant increase in enrichment.

The means of material spoiling most frequently discussed in the literature is the addition of a moderating component such as BeO. The moderator degrades the spectrum to such an extent that the loss of the moderation of sodium changes the total moderation very little. An increase in enrichment is also required as BeO is added, so the voiding is again improved.

All means of spoiling except the use of BeO tend to reduce the Doppler coefficient. This is primarily because of the increased enrichment required by the higher leakage and/or the lower fuel fraction. Although the enrichment is increased when BeO is added, the degraded spectrum permits more resonance captures in the fertile material, and hence, the Doppler coefficient is enhanced.

5.3. Description of Trade-Off

All spoiled cores in this study are evolved from and compared with a compact cylinder ($L/D = 0.4$) which is designed for minimum fuel costs and short doubling time with less emphasis on the sodium voiding characteristics. The pin pitch is 0.340 inch and the pin diameter is 0.292 inch. This combination of pin diameter and pitch results in a core pressure drop of just under 100 psi, which was set as an upper limit. The pitch was chosen on the basis of a parametric study which indicated that fuel costs started to climb when the pitch was reduced below 0.340. This study is discussed briefly in the appendix. Revised fuel cost methods indicate that this combination of pitch and pin diameter might not truly be the optimum for lowest costs. However, the new information came after the trade-off study was started. The clad thickness is 0.010 inch.

The size of the core is determined by the total active length of the pin, 2.7×10^6 inches. This length is set by the following parameters:

$P_{\max}/\bar{P}$	1.60
Max kW/ft	17
Power in core, MWt	2390

The L/D of 0.4 represents a near-minimum leakage configuration for a zone-loaded core. As the degree of radial power flattening decreases, the minimum leakage L/D increases. For a non-zone-loaded core, the minimum leakage L/D is considerably higher. The combination of L/D , total active length, pins per assembly, and the requirement for a symmetric array of assemblies resulted in an active length of 40.1 inches and an equivalent diameter of 99.1 inches. The control fraction was assumed to occupy the equivalent of 21 pins per assembly.

Volume fractions of the compact cylinder, designated "CCD," are given in Table 1.

All cores have the same radial blanket assembly. The number of assemblies, of course, varies with the core geometry. The pin diameter is 0.506, the pitch is 0.546, and the clad thickness is 0.018 inch. In all cases the radial blanket length is the same as that of the core. As a result, the breeding ratio is somewhat conservative.

In the radial blankets the depleted UO_2 is assumed to have a smeared density of 90% of theoretical, while in the core and axial blankets, the smeared density is 85% of theoretical. The uranium is depleted to an initial enrichment of 0.3% ^{235}U . The plutonium isotopic composition is as follows:

<u>Isotope</u>	<u>%</u>
^{239}Pu	70
^{240}Pu	25
^{241}Pu	5

All of the spoiled cores are derived from the compact cylinder in that all have the same pitch and the same total length of pin. A range of compositions is achieved by varying the pin diameter. Three compositions, representing three coolant fractions, are included in the study. For convenience, each composition is represented by a code letter. The compositions are summarized below.

<u>Designation</u>	<u>Pin OD, in.</u>	<u>Sodium fraction</u>
D	0.292	0.346
I	0.275	0.430
W	0.250	0.502

Similarly, a number of geometrical shapes were investigated; these shapes and their code designations are given below.

<u>Designation</u>	<u>Description</u>	<u>Leakage</u>	<u>L/D</u>
CC	Compact cylinder	Low	0.40
IC	Intermediate cylinder	Intermediate	0.248
PC	Pancake cylinder	High	0.135
AC	Annular core	High	--

Thus, for example, the combination PCW indicates a high-leakage, pancake core with a high sodium fraction. In several cases, a given geometry is analyzed with more than one composition.

Spoiling by increased leakage is demonstrated with an annular core, "ACW," which is 60 inches tall and 17 inches thick, and two pancaked cylinders, "PCW" and "PCD," which are 19.3 inches tall and have an L/D ratio of 0.135. Cores ACW and PCW both have the same pin diameter, 0.250 inch and hence, the same composition. This pin diameter was chosen to give a sodium fraction of about 50%.

The core designated "PCD," pancaked cylinder dry, has the same dimensions as PCW but has a larger fuel pin, which results in a lower coolant fraction. The 0.292-inch pin is the same as that used in the compact cylinder. This core demonstrates the effects of sodium fraction on voiding in a high leakage core.

An intermediate cylinder with an L/D of 0.248 was chosen to fill the gap between extreme spoiling and no spoiling. This core was calculated first with a pin diameter of 0.270 inch, which results in a sodium volume fraction between those of the PCW and the CCD, and then with the same pin diameter used in the PCD and in the compact cylinder "CCD." The first core is designated ICI and the second ICD.

The effects of using BeO are demonstrated in a series of three cores based on CCD with BeO fractions of 0.14, 0.20, and 0.323 of the total fuel plus BeO fraction. These cores are all 40.1 inches tall. Since the cores must grow radially if the total length of the pin is to be preserved as BeO pins replace fuel pins, the L/D decreases from 0.40 for no BeO to 0.333 for the case with maximum BeO. If the BeO were assumed to be mixed with the fuel and no penalty in allowable kW/ft resulted, then the cores could remain the same size. However, it was felt that unknown effects on the fuel performance and reprocessing cost made this assumption less desirable.

In this study the BeO pins are assumed to be the same as fuel pins except for the substitution of BeO. In reality the pins would probably be somewhat larger to cut down the bypass flow past the cool BeO pins. The BeO pins are uniformly distributed throughout the core and do not extend into the axial blanket. Since the moderation component of the sodium worth in the axial blankets is smaller than that of the core, moderator was not added to the blanket.

All of the cylindrical cores are zone loaded for power flattening. The pancake and intermediate cylinders have three enrichment zones of approximately equal volumes, while the compact cylinder and BeO cores have two enrichment zones.

Since leakage varies greatly from core to core, the blanket thicknesses also differ. The axial blanket thickness ranges from 24 inches in the pancake cores to 6 inches in the annulus. Similarly, the radial blanket thickness ranges from 5.5 inches in the pancake to 17 inches in the compact cylinder. The physical parameters of all the cores are given in Table 1.

Table 1. Physical Parameters

	CCD	CCD .14 BeO	CCD .20 BeO	CCD .32 BeO	ICI	ICD	PCW	PCD	ACW
Core diameter, in.	99.1	107.0	110.8	120.4	116.8	116.8	142.9	142.9	110.8
Core height, in.	40.1	40.1	40.1	40.1	28.9	28.9	19.3	19.3	60.5
L/D	0.405	0.375	0.362	0.333	0.248	0.248	0.135	0.135	H/T = 3.45
Core volume, liters	5077	5918	6339	7485	5077	5077	5077	5077	5077
Axial blanket thickness, in.	15	15	15	15	18	18	24	24	6
Radial blanket thickness, in.	17	18	17	16	12	12	5.5	5.5	17.5 I. B. 16.9 O. B.
No. of fuel elements	217	252	271	320	301	301	451	451	144
Fuel pin OD	0.292	0.292	0.292	0.292	0.270	0.292	0.250	0.292	0.250
Volume fractions									
Fuel	0.505	0.433	0.404	0.342	0.427	0.505	0.362	0.505	0.362
Na	0.346	0.345	0.346	0.345	0.430	0.346	0.502	0.346	0.502
SS	0.127	0.128	0.127	0.127	0.121	0.127	0.111	0.127	0.111
BeO	0.0	0.072	0.101	0.163	0.0	0.0	0.0	0.0	0.0
B ₄ C	0.022	0.022	0.022	0.023	0.022	0.022	0.024	0.022	0.024
I. B. R.	1.39	1.28	1.18	1.04	1.36	1.40	1.28	1.40	1.39
Fissile Pu, kg	2279	2714	3012	3448	2377	2501	2797	2947	2868
Full power days	721	721	721	680	645	721	521	721	521
MWD/T $\frac{\text{MWD core}}{\text{kg core}}$	77,322	77,608	77,494	70,990	77,110	74,300	74,500	74,434	72,562
$\bar{E}$, BOL	0.1076	0.1280	0.1417	0.1578	0.1295	0.1156	0.1832	0.1388	0.1902

6. RESULTS

Table 1 is a summary of the physical characteristics of the cores analyzed in the trade-off study; it also includes some of the calculated results. These results are as calculated for the burnups given, which vary between 71,000 and 78,000 MWD/T. This burnup range is somewhat higher than the average in the core at the end of an equilibrium cycle in which one third of the core is discharged at an average burnup of 100,000 MWD/T. Thus, the average conditions in a partially refueled core are simulated in a batch refueling calculation. Studies have indicated that the safety characteristics are strongly burnup-dependent and should be calculated at the least favorable (high burnup) conditions.

Figure 1 shows effect of burnup on voiding characteristics for the PCD. As expected, the Δk_{void} increases monotonically with increasing MWD/T. In this case the increase is $0.6 \times 10^{-7} \Delta k/k/\text{MWD/T}$. An interesting side point is the change in beginning-of-life Δk_{void} with and without control. The Δk_{void} at $k_{\text{eff}} = 1.00$ is 36% higher than the Δk_{void} at $k_{\text{eff}} = 1.04$. Certainly, excess reactivity is required for burnup, and control is required to keep the $k_{\text{eff}} = 1.0$. Therefore, failure to factor this into voiding calculations will give optimistic results. Similarly, failure to include the effects of burnup can also result in an optimistic Δk_{void} .

Fuel costs and the individual cost components for all the cores (assuming that the discharge burnup of 100,000 MWD/T is achieved in a single batch refueling) are summarized in Table 2. The corrected beginning- and end-of-life fissile plutonium masses, the corrected integrated breeding ratios, and the effective full power days (EFPD) required for a core burnup of 100,000 MWD/T are also included. These corrections are discussed in the appendix. The fuel costs are corrected to 100,000 MWD/T for two reasons: (1) all fuel costs should be at the same burnup for a meaningful comparison; (2) the reference discharge burnup is 100,000 MWD/T.

Figure 1. Effect of Fuel Average Burnup on Δk_{void} (PCD Core)

Table 2. Fuel Cost Summary and Miscellaneous Parameters

	<u>CCD</u>	<u>CCD .14 BeO</u>	<u>CCD .20 BeO</u>	<u>CCD .32 BeO</u>	<u>ICI</u>	<u>ICD</u>	<u>PCW</u>	<u>PCD</u>	<u>ACW</u>
Fuel cost, mill/kWh									
Fabrication	0.300	0.348	0.365	0.418	0.355	0.331	0.476	0.387	0.393
Reprocessing & reversion	0.153	0.158	0.163	0.176	0.164	0.158	0.219	0.195	0.257
Shipping	0.042	0.046	0.047	0.052	0.045	0.047	0.068	0.064	0.055
Plutonium working capital	0.521	0.588	0.635	0.718	0.552	0.570	0.683	0.678	0.739
Fabrication working capital	0.051	0.058	0.062	0.071	0.053	0.058	0.062	0.068	0.056
Plutonium credit	-0.376	-0.287	-0.174	0.045	-0.360	-0.392	-0.238	-0.393	-0.329
Reprocessing & reversion working capital	-0.040	-0.041	-0.043	-0.047	-0.039	-0.043	-0.048	-0.053	-0.072
Total	0.651	0.870	1.055	1.433	0.770	0.729	1.222	0.946	1.099
Miscellaneous parameters ^(a)									
Initial Pu, kg ²³⁹ Pu + ²⁴¹ Pu	2279	2714	3087	3797	2389	2501	2951	3006	3112
Final Pu, kg ²³⁹ Pu + ²⁴¹ Pu	3184	3415	3541	3771	3175	3472	3409	3997	3754
I. B. R.	1.35	1.27	1.16	0.97	1.34	1.36	1.23	1.37	1.32
EFPD	941	930	930	930	833	970	690	975	725

(a) Discharge burnup of 100,000 MWD/T.

As expected, spoiling in any form or degree causes an increase in fuel cycle costs. The difference between CCD and PCW, for example, is 0.571 mill/kWh. Increased fabrication and plutonium working capital costs and decreased plutonium credit are the major contributors to the increased fuel costs.

The Δk_{void} and Doppler coefficients and the doubling times for all of the cores are given in Table 3. The dependence of fuel cycle costs on the Δk_{void} is shown graphically in Figure 2.

It is interesting to note that for a given Δk_{void} the fuel cost penalty is not a strong function of the method of spoiling. On the basis of the batch refueling fuel costs, high leakage coupled with small coolant fraction appears to be the most economic approach to spoiling. BeO spoiling is somewhat less favorable, and high leakage with a large coolant fraction is the least favorable.

The assumption of single-batch refueling tends to penalize cores with low internal breeding ratios, since a very large initial excess Δk_{eff} is required for the core to achieve long burnups. With partial refueling, the excess Δk_{eff} requirements are considerably less; as a result the fissile loading is reduced, and the breeding ratio is increased. On the other hand, a core with a high internal breeding ratio requires relatively little excess Δk_{eff} for long burnups, so the lower excess Δk_{eff} permitted by partial refueling has relatively less effect on fissile loading and breeding ratio.

To ensure that the batch refueling assumption did not distort the results of the trade-off study, the fuel cost improvement resulting from a three-batch partial refueling scheme was estimated for each of the cores. The procedures used to estimate the change in fuel costs are discussed in the appendix.

A further correction that should be included for completeness is the capital cost penalty associated with spoiling. Although the cores were chosen to minimize capital cost effects, some changes did occur. For easier comparison, the capital cost penalty is defined as the difference in capital costs (in mill/kWh) between the core in question and core CCD, which has the lowest capital costs.

Figure 2. Fuel Costs for Spoiled and Unspoiled LMFBR for Single-Batch Refueling

Table 3. Summary of Results

	Δk_{void}	mills/kWh					Doppler coefficient, $-T(dK/dT)$	
		Single-batch fuel costs	Three-batch fuel cost	Capital cost penalty ^(a)	Three-batch fuel cost + capital cost penalty	Single-batch doubling time, years	Full dense	50% void
CCD	0.0210	0.651	0.63	0.00	0.63	9.4	0.0063	0.0053
CCD 0.14 BeO	0.0146	0.870	0.81	0.06	0.87	14.1	0.0067	0.0058
CCD 0.20 BeO	0.0108	1.055	0.92	0.11	1.03	47.0	--	--
CCD 0.32 BeO	0.0050	1.433	1.19	0.27	1.46	∞	0.0119	0.0115
ICI	0.0162	0.770	0.74	0.07	0.81	10.3	--	--
ICD	0.0176	0.729	0.70	0.08	0.78	9.9	0.0054	0.0044
PCW	0.0094	1.22	1.05	0.20	1.25	19.8	0.0046	0.0033
PCD	0.0112	0.946	0.87	0.20	1.07	11.9	--	--
ACW	0.0117	1.099	0.91	0.14	1.05	15.3	0.0037	0.0028

^(a) Capital cost penalty relative to capital costs of CCD.

Key results are given in Table 3. Fuel costs based on single-batch and partial refueling schemes are compared, and the capital cost penalties associated with spoiling are tabulated. Fuel costs, assuming partial refueling, are presented graphically as a function of Δk_{void} in Figure 3. With partial refueling, the penalties for spoiling are even less sensitive to the type of spoiling; however, the relative positions are roughly the same. When the capital cost penalties are included, BeO spoiling is the most economic, while geometric spoiling (with either low or high coolant fraction) is less favorable.

The BeO cores fare somewhat better in terms of capital cost, because the core diameters increase less rapidly than those of the pancake cores. As a result, the BeO-spoiled core vessel, core internals, reactor vessel, and biological shielding are smaller and hence cheaper. The same is true of the annular core, which has a core diameter of 111 inches, as opposed to 143 inches for PCD, which has roughly the same Δk_{void} .

It is clear that the cost penalty in achieving a zero or negative Δk_{void} for the core plus the axial blanket is severe. If the curves in Figures 2 and 3 are extrapolated to zero Δk_{void} , then the incremental fuel cost increases by at least 0.9 to 1.0 mill/kWh.

If doubling time is a criterion, the choice of spoiling method is much more obvious. Clearly BeO spoiling degrades doubling time much more than does geometric spoiling. The degraded flux spectrum and the increased enrichment accompanying BeO spoiling result in less favorable competition for neutron capture between the fertile material and the remaining core materials. In addition, the blanket fertile material absorbs fewer neutrons. As a result, the breeding ratio and doubling time decrease sharply (see Tables 2 and 3).

Geometric spoiling, on the other hand, does not degrade the breeding ratio nearly as rapidly. Although the increased fissile enrichment in the core cuts the core breeding ratio, the increased leakage results in more fertile captures in the blankets.

Figure 3. Fuel Costs for Spoiled and Unspoiled LMFBR, Including Capital Cost Penalty (Estimated for One-Third-Core Refueling)

For example, the PCD and CCD 0.20 BeO cores have nearly the same Δk_{void} . The beginning-of-life core breeding ratios are 0.79 and 0.85 for the PCD and the CCD 0.20 BeO cores—yet the overall breeding ratios are 1.45 and 1.20. The difference is in the blanket contribution. The doubling time for a dry pancake core is lower than that for an annular core with a corresponding Δk_{void} . Partial refueling would tend to reduce the doubling times of all the spoiled cores. However, the BeO spoiled cores would still have longer doubling times than the geometrically spoiled cores.

For a given high-leakage geometry, increasing the sodium fraction reduces the Δk_{void} (PCW versus PCD and ICI versus ICD). However, the results given in Table 4 demonstrate that for partial voiding, the high sodium fraction or "wet" core is clearly worse. The maximum positive Δk_{void} is more than twice the core plus the axial blanket value in the PCW, whereas in the PCD the factor is 1.5. Furthermore, the maximum void in the PCW is greater than that in the PCD. This is to be expected, since the spectral component of PCW voiding is greater than that of the PCD. With partial voiding the leakage component is, of course, reduced and the wetter lattice is penalized more.

Table 4. Partial Voiding Results, Δk_{void}

	<u>CCD</u>	<u>PCD</u>	<u>PCW</u>
Core + axial blanket	0.021	0.0112	0.0094
Core	0.022	0.0154	0.0185
Maximum positive	--	0.0174	0.0217

Past studies have indicated that the calculated Δk_{void} is strongly dependent on the plutonium distribution in the blankets. If, in the calculation, the buildup with life of the plutonium immediately adjacent to the core is not shown, the Δk_{void} is low. Merely assuming a uniform plutonium concentration in the blankets is not adequate. In one case, making the blanket plutonium loading uniform reduced to Δk_{void} by some 25%. Thus, failure to properly show the high plutonium concentration

in the near blanket regions, and failure to consider that excess reactivity and control are needed at any time other than at the end of life will result in optimistic values for the calculated Δk_{void} .

A number of factors that may affect the fuel cost of these cores should be mentioned. The blanket thicknesses are not necessarily optimized for each of the cores. In some cases the maximum power peaking slightly exceeds the specified value of 1.6, and in other cases the peak is as low as 1.4. Designing to minimized peaking for all cores could change the required total length of pin and hence, the core size. The net result is that, although the overall picture is valid, more detailed optimization could shift the fuel cost picture slightly.

Intermediate results and interesting information for each core which are not directly applicable to the trade-off are given in the appendix. Included in the tables are mass balances, beginning- and end-of-life instantaneous breeding ratios, power fractions in the core and blanket, power peaking factors, and plutonium isotopic composition.

APPENDIX
Methods and Tables

All the results were generated with the DHRG-SIZLE¹ code package in use at B&W. In this system, criticality and reactivity coefficient calculations are in 18 groups, while depletion takes place in 6 groups collapsed from 18 groups with space and energy weighting. Separate 18-group libraries are generated for core and blanket materials from a 68-group basic library in a fundamental mode calculation by DHRG, a revision of GAM I. Important cross section sources may be summarized as follows:

1. ^{239}Pu — ENDF/B evaluation by Greebler.²
2. ^{238}U — ENDF/B evaluation by Wittkopf, Roy, and Livolsi.³
3. ^{240}Pu — ENDF/B evaluation by Pitterle.⁴
4. ^{241}Pu — evaluation by Drake.⁵
5. Fe, Ni, Cr — ANL MC² calculation; spectrum corresponds to that of sodium-cooled, oxide commercial breeder.
6. Na, Be, O — ORNL GAM-I Library Listing (1965).

Cross sections for ^{238}U and ^{239}Pu in the resolved and unresolved resonance regions were prepared by the ERIC-2 code.⁶ The inelastic matrix for ^{238}U was prepared by the DISC-2 code, using the ENDF/B discrete level and continuum data;⁷ matrices for the remaining materials were taken from an ORNL GAM-I library listing of 1965, normalized to more recent total inelastic cross-section data. Using these methods, key parameters of assembly 48 of ZPR III agree very well with the values given by Greebler.⁸

The computation model used combines one-dimensional radial and axial calculations in which the transverse bucklings have been converged. All cores reach an average burnup of between 70,000 and 30,000 MWD/T. This is done for convenience in a single batch cycle although actual cores will probably be partially refueled. This assumption removes the added complication of type and frequency of refueling. The assumed burnup range is somewhat greater than the average found in a core at the end of a cycle in which a discharge burnup of 100,000 MWD/T is achieved in three partial refueling cycles, i.e., one-third of the core at 33,000 MWD/T, one-third at 67,000 MWD/T, and the remaining third at 100,000

MWD/T. The core composition, especially the fission products, for a given average burnup should be approximately the same regardless of the path followed to reach the burnup.

Coolant void coefficients are computed at end-of-cycle conditions by the Δk_{eff} method. Calculations have shown that voiding tends to be least favorable at the end of life. The calculations are in 18 groups and make use of full dense and void 18-group libraries. Since these calculations are in one dimension, the change in transverse buckling with voiding is, of course, included. Void calculations are with the core and axial blankets voided. The reactivity insertion when the core only is voided, as well as the maximum positive Δk_{void} , were calculated for several of the cores.

The Doppler coefficient was computed for selected cores by the k_{eff} method for an isothermal change in fuel temperature. Both ^{238}U and ^{239}Pu are included, but the contribution of ^{240}Pu is not included. Since the ^{240}Pu contribution is negative, the calculated Doppler coefficient should be slightly conservative. In addition, the contribution of the blankets to the Doppler coefficient is neglected.

The choice of pin pitch and diameter was based on results generated with the NAPS code.⁹ One of the trade-off study ground rules limited the core pressure drop to 100 psi. Using this as a guide and the compact cylinder as a reference, the pin diameter at a number of pitches was varied until the core Δp reached 100 psi. The fuel costs for these pin-pitch combinations are presented in Figure A-1. Since these results were generated, the NAPS code has been revised and updated. Consequently, data presented elsewhere might seem inconsistent with the results given here. In any event, the trends established by the trade-off study should not be affected by the reference pitch.

With one exception, fuel costs are calculated using the methods described in the Task I report. The exception is the unit fabrication cost, which is obtained from data described in Volume I of this report. In addition, the following assumptions were made for the BeO cores:

1. The fabrication cost of a BeO pin is the same as that of a fuel pin.
2. The reprocessing cost is dependent only on the masses of fuel. The BeO pins are removed before processing, and the axial blankets are separated from the BeO.

Figure A-1. Minimum Fuel Cost at Each Pitch for a 40-Inch LMFBR Core at 100 psi Core Pressure Drop (0.100-Inch Fixed Can)

3. Shipping costs are increased by the mass of BeO.

Fuel costs are computed for an exposure of 100,000 MWD/T in the core. Since the average burnup was between 70,000 and 80,000 MWD/T, an extrapolation was necessary. The initial fissile loading required to produce the desired burnup was estimated. The integrated breeding ratio was then corrected for the higher initial fissile loading and the estimated change in breeding ratio with burnup. Using the corrected breeding ratio and initial loading, a corrected final fissile loading was calculated. The corrected plutonium credit and working capital fuel cost components were then computed.

The estimate of fuel cost improvements resulting from partial refueling was based on the following assumptions:

1. The excess k_{eff} for a three-batch refueling scheme is one third of that required for a single-batch scheme.
2. The plutonium credit fuel cost component is proportional to the breeding gain (BR-1).
3. The plutonium working capital component is proportional to the average fissile loading.

The doubling time is given by the expression

$$DT = \frac{t_c}{b} \left(\frac{0.693 + \ln b}{\ln B} + \frac{b-1}{2} \right)$$

which reduces to the following for a batch loaded core:

$$DT = \frac{0.693 (IP + OOP)}{\ln \frac{F}{I}}$$

where

DT = doubling time, years

IP = in-pile time, years

OOP = out-of-pile time including fabrication, reprocessing, and storage, years

F = final fissile Pu mass less 1% reprocessing loss, kg

I = initial fissile Pu mass plus 1% fabrication loss, kg

The derivation of this expression is given in Volume I.

Table A-1. CCD

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	99.1	133.1	99.1	156.7	99.1
Height, in.	40.1	40.1	15.0	40.1	11.8
Volume, liters	5076.2	4068.9	3793.0	3530.4	2983.8
No. of fuel elements	217	174	217	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.292	0.506	0.292	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	85	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	950	950	950	950	950
Volume fractions					
Fuel	0.505	0.580	0.505	0.0	0.0
Na	0.346	0.236	0.346	0.5	0.346
Steel	0.127	0.16	0.127	0.5	0.127
B ₄ C	0.022	0.0	0.022	0.0	0.022

Type of loading, 2-zone batch, 127 fuel elements in zone 1
90 fuel elements in zone 2

2. Enrichments (CCD)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.0924	0.1151
Radial core zone 2	0.1291	0.1308
Core average	0.1076	0.1209

3. Burnup (CCD)

Lifetime, EFPD	721
MWD/T	77,322

* Weight fraction ²³⁹Pu + ²⁴¹Pu.

Table A-1. (Cont'd)

4. Initial and Final Pu Isotopic Mix (CCD)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	68.7
240	25.0	25.9
241	5.0	4.8
242	0.0	0.6

5. Breeding Ratios (CCD)

Integrated Breeding Ratio (I. B. R. = 1.39)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	1.101	1.064	1.083
Axial blanket	0.232	0.224	0.228
Radial blanket	0.217	0.210	0.213
Total	1.550	1.498	1.524
<u>End of life</u>			
Core	0.891	0.885	0.888
Axial blanket	0.241	0.239	0.240
Radial blanket	0.181	0.180	0.181
Total	1.313	1.304	1.309

6. Peaking Factors and Power Fractions (CCD)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.100	1.445
Radial core zone 2	1.239	1.024
Axial	1.285	1.229
<u>Power fractions, %</u>		
Core	97.00	90.60
Radial blanket	1.50	3.60
Axial blanket	1.50	5.80

Table A-1. (Cont'd)

7. Mass Balance, kg (CCD)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	54.41	47.20	61.61	163.22
²³⁸ U	<u>18,083.62</u>	<u>15,683.81</u>	<u>20,457.21</u>	<u>54,224.64</u>
Total U	18,138.03	15,731.01	20,518.82	54,387.86
²³⁹ Pu	2,127.11	0.0	0.0	2,127.11
²⁴⁰ Pu	759.70	0.0	0.0	759.70
²⁴¹ Pu	151.93	0.0	0.0	151.93
²⁴² Pu	<u>0.0</u>	<u>0.0</u>	<u>0.0</u>	<u>0.0</u>
Total Pu	3,038.74	0.0	0.0	3,038.74
Total U + Pu	21,176.77	15,731.01	20,518.82	57,426.60
<u>End of life</u>				
²³⁵ U	22.70	36.01	52.00	110.71
²³⁸ U	<u>16,212.01</u>	<u>15,245.80</u>	<u>20,093.31</u>	<u>51,551.12</u>
Total U	16,234.71	15,281.81	20,145.31	51,661.83
²³⁹ Pu*	2,194.80	367.61	317.01	2,879.42
²⁴⁰ Pu	827.41	12.20	7.00	846.61
²⁴¹ Pu	154.40	0.42	0.22	155.04
²⁴² Pu	<u>17.92</u>	<u>0.01</u>	<u>0.00</u>	<u>17.92</u>
Total Pu*	3,194.53	380.24	324.23	3,898.99
Total U + Pu*	19,429.24	15,662.05	20,469.54	55,560.82

*Includes ²³⁹Np.

Table A-2. CCD .14 BeO

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	107.0	142.9	107.0	166.3	107.0
Height, in.	40.1	40.1	15.0	40.1	11.8
Volume, liters	5918.3	4631.7	4424.4	3726.2	1750.5
No. of fuel elements	253	198	152	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.292	0.506	0.292	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	85	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	950	950	950	950	950
Volume fractions					
Fuel	0.433	0.580	0.505	0.0	0.0
Na	0.345	0.236	0.345	0.500	0.345
Steel	0.128	0.160	0.128	0.500	0.128
B ₄ C	0.022	0.0	0.022	0.0	0.023
BeO	0.072	0.0	0.0	0.0	0.0

Type of loading, 2-zone batch, 152 fuel elements in zone 1
101 fuel elements in zone 2

2. Enrichments (CCD .14 BeO)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.1124	0.1260
Radial core zone 2	0.1514	0.1470
Core average	0.1280	0.1345

3. Burnup (CCD .14 BeO)

Lifetime, EFPD	721
MWD/T	77,608

* Weight fraction ²³⁹Pu + ²⁴¹Pu.

Table A-2. (Cont'd)

4. Initial and Final Pu Isotopic Mix (CCD .14 BeO)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	66.9
240	25.0	27.2
241	5.0	5.3
242	0.0	0.6

5. Breeding Ratios (CCD .14 BeO)

Integrated Breeding Ratio (I. B. R. = 1.28)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	0.947	0.915	0.931
Axial blanket	0.213	0.206	0.209
Radial blanket	0.174	0.168	0.171
Total	1.334	1.289	1.312
<u>End of life</u>			
Core	0.841	0.835	0.838
Axial blanket	0.237	0.235	0.236
Radial blanket	0.171	0.170	0.170
Total	1.249	1.240	1.244

6. Peaking Factors and Power Fractions (CCD .14 BeO)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.191	1.383
Radial core zone 2	1.188	1.041
Axial	1.283	1.217
<u>Power fractions, %</u>		
Core	96.10	91.51
Radial blanket	1.27	3.18
Axial blanket	2.63	5.30

Table A-2. (Cont'd)

7. Mass Balance, kg (CCD .14 BeO)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	52.77	55.02	70.05	177.84
²³⁸ U	<u>17,539.55</u>	<u>18,285.73</u>	<u>23,278.91</u>	<u>59,104.19</u>
Total U	17,592.32	18,340.75	23,348.96	59,282.03
²³⁹ Pu	2,533.25	0.0	0.0	2,533.25
²⁴⁰ Pu	904.74	0.0	0.0	904.74
²⁴¹ Pu	180.95	0.0	0.0	180.95
²⁴² Pu	<u>0.0</u>	<u>0.0</u>	<u>0.0</u>	<u>0.0</u>
Total Pu	3,618.94	0.0	0.0	3,618.94
Total U + Pu	21,211.26	18,340.75	23,348.96	62,900.97
<u>End of life</u>				
²³⁵ U	22.95	43.73	60.90	127.58
²³⁸ U	<u>15,794.31</u>	<u>17,854.42</u>	<u>22,938.91</u>	<u>56,587.64</u>
Total U	15,817.26	17,898.15	22,999.81	56,715.22
²³⁹ Pu*	2,420.88	365.97	296.90	3,083.75
²⁴⁰ Pu	987.66	10.52	5.94	1,004.12
²⁴¹ Pu	191.68	0.29	0.12	192.09
²⁴² Pu	<u>21.75</u>	<u>0.0</u>	<u>0.0</u>	<u>21.75</u>
Total Pu*	3,621.97	376.78	302.96	4,301.71
Total U + Pu*	19,439.23	18,274.93	23,302.77	61,016.93

*Includes ²³⁹Np.

Table A-3. CCD .20 BeO

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	110.8	144.8	110.8	168.4	110.8
Height, in.	40.1	40.1	15.0	40.1	11.8
Volume, liters	6339.4	4491.4	4539.1	--	--
No. of fuel elements	271	192	271	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.292	0.506	0.292	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	90	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	950	950	950	950	950
Volume fractions					
Fuel	0.404	0.580	0.505	0.0	0.0
Na	0.346	0.236	0.346	0.500	0.346
Steel	0.127	0.160	0.127	0.500	0.127
B ₄ C	0.022	0.0	0.0	0.0	0.0
BeO	0.101	0.0	0.0	0.0	0.0

Type of loading, 2-zone batch, 169 fuel assemblies in zone 1
102 fuel assemblies in zone 2

BeO at 80% of its theoretical density

2. Enrichments (CCD .20 BeO)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.1223	0.1304
Radial core zone 2	0.1739	0.1623
Core average	0.1417	0.1424

3. Burnup (CCD .20 BeO)

Lifetime, EFPD	721
MWD/T	77,494

* Weight fraction of ²³⁹Pu + ²⁴¹Pu.

Table A-3. (Cont'd)

4. Initial and Final Pu Isotopic Mix (CCD .20 BeO)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	65.7
240	25.0	28.1
241	5.0	5.6
242	0.0	0.6

5. Breeding Ratios (CCD .20 BeO)

Integrated Breeding Ratio (I. B. R. = 1.18)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	0.849	0.819	0.834
Axial blanket	0.190	0.183	0.187
Radial blanket	<u>0.166</u>	<u>0.160</u>	<u>0.163</u>
Total	1.205	1.162	1.184
<u>End of life</u>			
Core	0.795	0.786	0.791
Axial blanket	0.219	0.216	0.218
Radial blanket	<u>0.168</u>	<u>0.166</u>	<u>0.167</u>
Total	1.183	1.168	1.176

6. Peaking Factors and Power Fractions (CCD .20 BeO)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.086	1.287
Radial core zone 2	1.248	1.077
Axial	1.280	1.205
<u>Power fractions, %</u>		
Core	97.2	91.8
Radial blanket	1.3	3.2
Axial blanket	1.5	5.1

Table A-3. (Cont'd)

7. Mass Balance, kg (CCD .20 BeO)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	51.55	58.94	67.92	178.41
²³⁸ U	<u>17,182.47</u>	<u>19,586.62</u>	<u>22,573.57</u>	<u>59,342.66</u>
Total U	17,234.02	19,645.56	22,641.50	59,521.08
²³⁹ Pu	2,811.42	0.0	0.0	2,811.42
²⁴⁰ Pu	1,004.08	0.0	0.0	1,004.08
²⁴¹ Pu	200.82	0.0	0.0	200.82
²⁴² Pu	<u>0.0</u>	<u>0.0</u>	<u>0.0</u>	<u>0.0</u>
Total Pu	4,016.32	0.0	0.0	4,016.32
Total U + Pu	21,250.34	19,645.56	22,641.50	63,537.40
<u>End of life</u>				
²³⁵ U	23.26	46.02	58.88	128.15
²³⁸ U	<u>15,566.72</u>	<u>19,177.92</u>	<u>22,233.29</u>	<u>56,977.93</u>
Total U	15,589.98	19,223.93	22,292.17	57,106.08
²³⁹ Pu*	2,560.94	345.49	297.18	3,203.61
²⁴⁰ Pu	1,095.41	10.08	5.87	1,111.35
²⁴¹ Pu	216.57	.31	.13	217.01
²⁴² Pu	<u>23.96</u>	<u>.01</u>	<u>.00</u>	<u>23.97</u>
Total Pu*	3,896.88	355.89	303.17	4,555.94
Total U + Pu*	19,486.86	19,579.82	22,595.34	61,662.02

*Includes ²³⁹Np.

Table A-4. CCD .32 BeO

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	120.4	153.2	120.4	176.6	120.4
Height, in.	40.1	40.1	15.0	40.1	11.8
Volume, liters	7484.7	4632.5	5596.0	3985.9	4405.9
No. of fuel elements	320	198	320	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.292	0.506	0.292	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	90	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	950	950	950	--	--
Volume fractions					
Fuel	0.342	0.580	0.505	0.0	0.0
Na	0.345	0.236	0.345	0.345	0.345
Steel	0.127	0.160	0.127	0.127	0.127
B ₄ C	0.022	0.0	0.0	0.0	0.0
BeO	0.163	0.0	0.0	0.0	0.0

Type of loading, 2-zone batch, 158 fuel assemblies in zone 1
162 fuel assemblies in zone 2

BeO at 80% of its theoretical density

2. Enrichments (CCD .32 BeO)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.1400	0.1398
Radial core zone 2	0.1749	0.1639
Core average	0.1578	0.1521

3. Burnup (CCD .32 BeO)

Lifetime, EFPD	680
MWD/T	70,990

* Weight fraction of ²³⁹Pu + ²⁴¹Pu.

Table A-4. (Cont'd)

4. Initial and Final Pu Isotopic Mix (CCD .32 BeO)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	64.7
240	25.0	28.8
241	5.0	5.9
242	0.0	0.6

5. Breeding Ratios (CCD .32 BeO)

Integrated Breeding Ratio (I. B. R. = 1.05)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	0.747	0.736	0.742
Axial blanket	0.162	0.160	0.161
Radial blanket	<u>0.104</u>	<u>0.103</u>	<u>0.103</u>
Total	1.013	0.999	1.006
<u>End of life</u>			
Core	0.750	0.751	0.750
Axial blanket	0.192	0.192	0.192
Radial blanket	<u>0.120</u>	<u>0.120</u>	<u>0.120</u>
Total	1.062	1.063	1.062

6. Peaking Factors and Power Fractions (CCD .32 BeO)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.068	1.142
Radial core zone 2	1.270	1.251
Axial	1.290	1.220
<u>Power fractions, %</u>		
Core	96.93	93.12
Radial blanket	1.44	2.66
Axial blanket	1.63	4.22

Table A-4. (Cont'd)

7. Mass Balance, kg (CCD .32 BeO)

	Core	Axial blanket	Radial blanket	Total
<u>Beginning of life</u>				
²³⁵ U	51.73	69.66	70.10	192.49
²³⁸ U	17,206.53	23,128.06	23,282.62	63,617.21
Total U	17,258.26	23,197.72	23,353.72	63,809.70
²³⁹ Pu	3,218.56	0.00	0.00	3,218.56
²⁴⁰ Pu	1,149.52	0.00	0.00	1,149.52
²⁴¹ Pu	229.91	0.00	0.00	229.91
²⁴² Pu	0.00	0.00	0.00	0.00
Total Pu	4,597.99	0.00	0.00	4,597.99
Total U + Pu	21,856.25	23,197.72	23,353.72	68,407.69
<u>End of life</u>				
²³⁵ U	27.70	57.16	56.32	141.18
²³⁸ U	15,873.30	22,688.66	23,057.62	61,619.58
Total U	15,801.00	22,745.82	23,113.94	61,660.76
²³⁹ Pu*	2,841.86	313.97	193.62	3,349.45
²⁴⁰ Pu	1,266.05	7.26	4.47	1,277.78
²⁴¹ Pu	259.12	0.19	0.53	259.84
²⁴² Pu	27.05	0.00	0.01	27.06
Total Pu*	4,394.08	321.42	198.63	4,914.13
Total U + Pu*	20,195.08	23,077.24	23,312.57	66,574.89

*Includes ²³⁹Np.

Table A-5. ICI

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	116.8	140.6	116.8	164.2	116.8
Height, in.	28.9	28.9	18.0	28.9	11.8
Volume, liters	5077.3	2288.5	3145.7	2666.0	2072.2
No. of fuel elements	301	138	301	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.270	0.506	0.270	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	90	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	975	975	975	975	975
Volume fractions					
Fuel	0.427	0.580	0.427	0.0	0.0
Na	0.430	0.236	0.430	0.500	0.0
Steel	0.121	0.160	0.121	0.500	0.0
B ₄ C	0.022	0.0	0.022	0.0	0.0

Type of loading, 3-zone batch, 109 fuel assemblies in zone 1
102 fuel assemblies in zone 2
90 fuel assemblies in zone 3

2. Enrichments (ICI)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.1112	0.1233
Radial core zone 2	0.1197	0.1263
Radial core zone 3	0.1622	0.1532
Core average	0.1295	0.1318

3. Burnup (ICI)

Lifetime, EFPD	645
MWD/T	77,110

*Weight fraction at ²³⁹Pu + ²⁴¹Pu.

Table A-5. (Cont'd)

4. Initial and Final Pu Isotopic Mix (ICI)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	67.3
240	25.0	27.0
241	5.0	5.1
242	0.0	0.5

5. Breeding Ratios (ICI)

Integrated Breeding Ratio (I. B. R. = 1.36)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	0.895	0.868	0.882
Axial blanket	0.384	0.384	0.384
Radial blanket	<u>0.162</u>	<u>0.157</u>	<u>0.160</u>
Total	1.442	1.409	1.425
<u>End of life</u>			
Core	0.762	0.757	0.760
Axial blanket	0.420	0.417	0.419
Radial blanket	<u>0.147</u>	<u>0.146</u>	<u>0.147</u>
Total	1.330	1.320	1.325

6. Peaking Factors and Power Fractions (ICI)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.025	1.216
Radial core zone 2	1.083	1.105
Radial core zone 3	1.191	1.055
Axial	1.213	1.156
<u>Power fractions, %</u>		
Core	96.20	86.40
Radial blanket	1.27	2.75
Axial blanket	2.53	10.85

Table A-5. (Cont'd)

7. Mass Balance, kg (ICI)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	45.57	70.65	34.64	150.86
²³⁸ U	<u>15,146.70</u>	<u>23,362.31</u>	<u>11,529.06</u>	<u>50,038.07</u>
Total U	15,192.27	23,432.96	11,563.70	50,188.93
²³⁹ Pu	2,218.77	0.00	0.00	2,218.77
²⁴⁰ Pu	792.38	0.00	0.00	792.38
²⁴¹ Pu	158.48	0.00	0.00	158.48
²⁴² Pu	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>
Total Pu	3,169.63	0.00	0.00	3,169.63
Total U + Pu	18,361.90	23,432.96	11,563.7	53,358.56
<u>End of life</u>				
²³⁵ U	20.75	52.57	27.86	101.18
²³⁸ U	<u>13,725.62</u>	<u>22,674.94</u>	<u>11,264.44</u>	<u>47,665.00</u>
Total U	13,746.37	22,727.51	11,292.30	47,766.18
²³⁹ Pu*	2,061.93	569.90	227.36	2,859.09
²⁴⁰ Pu	837.71	20.83	5.16	863.70
²⁴¹ Pu	158.44	0.79	0.13	159.36
²⁴² Pu	<u>16.32</u>	<u>0.02</u>	<u>0.00</u>	<u>16.34</u>
Total Pu*	3,074.40	591.44	232.65	3,898.49
Total U + Pu*	16,830.77	23,318.94	11,524.95	51,664.67

*Includes ²³⁹Np.

Table A-6. ICD

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	116.8	140.6	116.8	164.2	116.8
Height, in.	28.9	28.9	18.0	28.9	11.8
Volume, liters	5077.3	2288.51	3145.7	2744.4	2072.2
No. of fuel elements	301	138	301	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.292	0.506	0.292	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	90	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	950	950	950	950	950
Volume fractions					
Fuel	0.505	0.580	0.505	0.0	0.0
Na	0.346	0.236	0.346	0.346	0.500
Steel	0.127	0.160	0.127	0.127	0.500
B ₄ C	0.022	0.0	0.022	0.022	0.0

Type of loading, 3-zone batch, 109 fuel assemblies in zone 1
102 fuel assemblies in zone 2
90 fuel assemblies in zone 3

2. Enrichments (ICD)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.0997	0.1174
Radial core zone 2	0.1074	0.1194
Radial core zone 3	0.1438	0.1403
Core average	0.1156	0.1251

3. Burnup (ICD)

Lifetime, EFPD	721
MWD/T	74,300

*Weight fraction of ²³⁹Pu + ²⁴¹Pu.

Table A-6. (Cont'd)

4. Initial and Final Pu Isotopic Mix (ICD)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	68.4
240	25.0	26.2
241	5.0	4.9
242	0.0	0.5

5. Breeding Ratios (ICD)

Integrated Breeding Ratio (I. B. R. = 1.40)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	1.002	0.996	0.999
Axial blanket	0.374	0.373	0.374
Radial blanket	0.155	0.154	0.154
Total	1.531	1.524	1.527
<u>End of life</u>			
Core	0.825	0.821	0.823
Axial blanket	0.390	0.388	0.389
Radial blanket	0.126	0.125	0.126
Total	1.341	1.333	1.337

6. Peaking Factors and Power Fractions (ICD)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.015	1.303
Radial core zone 2	1.086	1.124
Radial core zone 3	1.230	0.990
Axial	1.2311	1.1765
<u>Power fractions, %</u>		
Core	96.41	87.90
Radial blanket	1.18	2.65
Axial blanket	2.41	9.45

Table A-6. (Cont'd)

7. Mass Balance, kg (ICD)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	54.91	83.16	34.68	172.75
²³⁸ U	<u>18,249.83</u>	<u>27,629.89</u>	<u>11,529.60</u>	<u>57,409.32</u>
Total U	18,304.74	27,713.05	11,564.28	57,582.07
²³⁹ Pu	2,334.25	0.00	0.00	2,334.25
²⁴⁰ Pu	833.86	0.00	0.00	833.86
²⁴¹ Pu	166.74	0.00	0.00	166.74
²⁴² Pu	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>
Total Pu	3,334.85	0.00	0.00	3,334.85
Total U + Pu	21,639.59	27,713.05	11,564.28	60,916.92
<u>End of life</u>				
²³⁵ U	24.56	65.09	27.87	117.52
²³⁸ U	<u>16,520.34</u>	<u>26,916.14</u>	<u>11,267.43</u>	<u>54,703.91</u>
Total U	16,544.90	26,981.23	11,295.30	54,821.43
²³⁹ Pu *	2,300.59	595.84	224.03	3,121.46
²⁴⁰ Pu	888.85	20.45	5.25	914.55
²⁴¹ Pu	166.82	0.69	0.14	167.65
²⁴² Pu	<u>18.05</u>	<u>0.01</u>	<u>0.00</u>	<u>18.06</u>
Total Pu *	3,374.31	616.99	230.42	4,221.72
Total U + Pu *	19,919.21	27,598.22	11,525.72	59,043.15

* Includes ²³⁹Np.

Table A-7. PCW

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	142.9	153.9	142.9	177.5	142.9
Height, in.	19.3	19.3	24.0	19.3	11.8
Volume, liters	5077.4	810.5	12619.1	1946.1	6309.5
No. of fuel elements	451	78	451	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.250	0.506	0.250	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	90	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	950	950	950	950	950
Volume fractions					
Fuel	0.362	0.580	0.362	0.0	0.0
Na	0.503	0.236	0.503	0.500	0.503
Steel	0.111	0.160	0.111	0.500	0.111
B ₄ C	0.024	0.0	0.024	0.0	0.0
Type of loading, 3-zone batch,	151 fuel elements in zone 1 162 fuel elements in zone 2 138 fuel elements in zone 3				

2. Enrichments (PCW)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.1603	0.1514
Radial core zone 2	0.1763	0.1616
Radial core zone 3	0.2162	0.1954
Core average	0.1832	0.1680

3. Burnup (PCW)

Lifetime, EFPD	521
MWD/T	74,500

*Weight fraction of ²³⁹Pu + ²⁴¹Pu.

Table A-7. (Cont'd)

4. Initial and Final Pu Isotopic Mix (PCW)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	65.7
240	25.0	28.4
241	5.0	5.5
242	0.0	0.4

5. Breeding Ratios (PCW)

Integrated Breeding Ratio (I. B. R. = 1.28)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	0.570	0.566	0.568
Axial blanket	0.585	0.582	0.584
Radial blanket	<u>0.074</u>	<u>0.073</u>	<u>0.073</u>
Total	1.229	1.221	1.225
<u>End of life</u>			
Core	0.565	0.574	0.570
Axial blanket	0.665	0.676	0.670
Radial blanket	<u>0.078</u>	<u>0.079</u>	<u>0.079</u>
Total	1.308	1.329	1.318

6. Peaking Factors and Power Fractions (PCW)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.038	1.049
Radial core zone 2	1.139	1.105
Radial core zone 3	1.186	1.133
Axial	1.135	1.096
<u>Power fractions, %</u>		
Core	95.04	83.16
Radial blanket	0.81	2.22
Axial blanket	4.15	14.62

Table A-7. (Cont'd)

7. Mass Balance, kg (PCW)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	34.63	119.28	12.26	166.17
²³⁸ U	<u>11,508.50</u>	<u>39,639.41</u>	<u>4,073.52</u>	<u>55,221.43</u>
Total U	11,543.13	39,758.69	4,085.78	55,387.60
²³⁹ Pu	2,610.86	0.00	0.00	2,610.86
²⁴⁰ Pu	932.46	0.00	0.00	932.46
²⁴¹ Pu	186.49	0.00	0.00	186.49
²⁴² Pu	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>
Total Pu	3,729.81	0.00	0.00	3,729.81
Total U + Pu	15,272.94	39,758.69	4,085.78	59,117.41
<u>End of life</u>				
²³⁵ U	19.64	95.52	9.17	124.33
²³⁸ U	<u>10,714.21</u>	<u>38,752.91</u>	<u>3,964.87</u>	<u>53,431.99</u>
Total U	10,733.85	38,848.43	3,974.04	53,556.32
²³⁹ Pu*	2,180.60	750.47	89.02	3,020.09
²⁴⁰ Pu	955.54	23.17	2.41	981.12
²⁴¹ Pu	185.77	0.78	0.12	186.67
²⁴² Pu	<u>14.78</u>	<u>0.02</u>	<u>0.00</u>	<u>14.80</u>
Total Pu*	3,336.69	774.44	91.55	4,202.68
Total U + Pu*	14,070.54	39,622.87	4,065.59	57,759.00

*Includes ²³⁹Np.

Table A-8. PCD

1. Physical Description

	<u>Core</u>	<u>Radial blanket</u>	<u>Axial blanket</u>	<u>Radial reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	142.9	153.9	142.9	177.5	142.9
Height, in.	19.3	19.3	24.0	19.3	11.8
Volume, liters	5077.4	810.5	12619.1	1946.1	6309.5
No. of fuel elements	451	78	451	--	--
Fuel pin pitch, in.	0.340	0.546	0.340	--	--
Fuel pin OD, in.	0.250	0.506	0.250	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Area/assembly, in. ²	35.572	35.572	35.572	--	--
Fuel percent dense	85	90	90	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Na	950	950	950	950	950
Volume fractions					
Fuel	0.505	0.580	0.505	0.0	0.0
Na	0.346	0.236	0.346	0.500	0.346
Steel	0.127	0.160	0.127	0.500	0.127
B ₄ C	0.022	0.0	0.022	0.0	0.0
Type of loading, 3-zone batch,	151 fuel elements in zone 1				
	162 fuel elements in zone 2				
	138 fuel elements in zone 3				

2. Enrichments (PCD)*

	<u>Beginning of life</u>	<u>End of life</u>
Radial core zone 1	0.1217	0.1268
Radial core zone 2	0.1337	0.1334
Radial core zone 3	0.1636	0.1535
Core average	0.1388	0.1373

3. Burnup (PCD)

Lifetime, EFPD	721
MWD/T	74,434

*Weight fraction of ²³⁹Pu + ²⁴¹Pu.

Table A-8. (Cont'd)

4. Initial and Final Pu Isotopic Mix (PCD)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	66.7
240	25.0	27.3
241	5.0	5.2
242	0.0	0.5

5. Breeding Ratios (PCD)

Integrated Breeding Ratio (I. B. R. = 1.40)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	0.791	0.794	0.792
Axial blanket	0.586	0.588	0.587
Radial blanket	<u>0.074</u>	<u>0.074</u>	<u>0.074</u>
Total	1.451	1.456	1.453
<u>End of life</u>			
Core	0.710	0.706	0.708
Axial blanket	0.639	0.635	0.637
Radial blanket	<u>0.061</u>	<u>0.061</u>	<u>0.061</u>
Total	1.409	1.401	1.405

6. Peaking Factors and Power Fractions (PCD)

	<u>Beginning of life</u>	<u>End of life</u>
<u>Peaking factors</u>		
Radial core zone 1	1.002	1.121
Radial core zone 2	1.163	1.128
Radial core zone 3	1.308	1.076
Axial	1.164	1.117
<u>Power fractions, %</u>		
Core	96.07	83.64
Radial blanket	0.70	1.82
Axial blanket	3.93	14.54

Table A-8. (Cont'd)

7. Mass Balance, kg (PCD)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	51.87	156.93	12.26	221.06
²³⁸ U	<u>17,239.65</u>	<u>52,154.12</u>	<u>4,073.52</u>	<u>73,467.29</u>
Total U	17,291.52	52,311.05	4,085.78	73,688.35
²³⁹ Pu	2,750.55	0.00	0.00	2,750.55
²⁴⁰ Pu	982.35	0.00	0.00	982.35
²⁴¹ Pu	196.47	0.00	0.00	196.47
²⁴² Pu	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>
Total Pu	3,929.37	0.00	0.00	3,929.37
Total U + Pu	21,220.89	52,311.05	4,095.78	77,617.72
<u>End of life</u>				
²³⁵ U	25.97	127.51	8.89	162.37
²³⁸ U	<u>15,804.11</u>	<u>51,004.58</u>	<u>3,941.74</u>	<u>70,750.43</u>
Total U	15,830.08	51,132.09	3,950.63	70,912.80
²³⁹ Pu*	2,490.73	964.75	107.60	3,563.08
²⁴⁰ Pu	1,015.09	29.75	3.23	1,048.07
²⁴¹ Pu	192.43	0.98	0.15	193.56
²⁴² Pu	<u>17.97</u>	<u>0.02</u>	<u>0.00</u>	<u>17.99</u>
Total Pu*	3,716.22	995.50	110.98	4,822.70
Total U + Pu*	19,546.30	52,127.59	4,061.61	75,735.50

*Includes ²³⁹Np.

Table A-9. ACW

1. Physical Description

	<u>Core</u>	<u>Inner blanket</u>	<u>Outer blanket</u>	<u>Axial blanket</u>	<u>Inner reflector</u>	<u>Outer reflector</u>	<u>Axial reflector</u>
Outer diameter, in.	110.8	75.8	144.4	110.8	40.9	168.5	110.8
Height, in.	60.5	60.5	60.5	6.0	60.5	60.5	11.8
Volume, liters	5077.2	3173.3	4652.3	1007.3	13046.0	5760.0	1928.8
No. of fuel elements	144	90	126	144	--	--	--
Fuel pin pitch, in.	0.340	0.546	0.546	0.340	--	--	--
Fuel pin OD, in.	0.250	0.506	0.506	0.250	--	--	--
Clad thickness, in.	0.010	0.018	0.018	0.010	--	--	--
Area/assembly, in. ²	35.572	35.572	35.572	35.572	--	--	--
Fuel percent dense	85	90	90	90	--	--	--
Steel type	304 SS	304 SS	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F							
Fuel	2500	2000	2000	2000	--	--	--
Na	950	950	950	950	950	950	950
Volume fractions							
Fuel	0.362	0.580	0.580	0.362	--	--	--
Na	0.503	0.236	0.236	0.503	0.900	0.500	0.503
Steel	0.111	0.160	0.160	0.111	0.100	0.500	0.111
B ₄ C	0.024	--	--	0.024	--	--	--

Type of loading, single-zone batch

2. Enrichments (ACW)*

	<u>Beginning of life</u>	<u>End of life</u>
Core	0.1902	0.1719

3. Burnup (ACW)

Lifetime, EFPD	521
MWD/T	72,562

*Weight fraction of ²³⁹Pu + ²⁴¹Pu.

Table A-9. (Cont'd)

4. Initial and Final Pu Isotopic Mix (ACW)

<u>Pu isotope, %</u>	<u>Beginning of life</u>	<u>End of life</u>
239	70.0	65.6
240	25.0	28.5
241	5.0	5.5
242	0.0	0.4

5. Breeding Ratios (ACW)

Integrated Breeding Ratio (I. B. R. = 1.39)

Instantaneous Breeding Ratios

	<u>Radial</u>	<u>Axial</u>	<u>Average</u>
<u>Beginning of life</u>			
Core	0.568	0.554	0.561
Axial blanket	0.045	0.044	0.044
Radial blanket	<u>0.765</u>	<u>0.746</u>	<u>0.755</u>
Total	1.377	1.343	1.360
<u>End of life</u>			
Core	0.542	0.535	0.539
Axial blanket	0.050	0.050	0.050
Radial blanket	<u>0.845</u>	<u>0.835</u>	<u>0.840</u>
Total	1.436	1.420	1.428

6. Peaking Factors and Power Fractions (ACW)

	<u>Beginning of life</u>	<u>End of life</u>
Peaking factors		
Radial core zone 1	1.128	1.105
Axial	1.342	1.223
Power fractions, %		
Core	0.9395	0.8037
Radial blanket	0.557	0.1824
Axial blanket	0.0048	0.0139

Table A-9. (Cont'd)

7. Mass Balance, kg (ACW)

	<u>Core</u>	<u>Axial blanket</u>	<u>Radial blanket</u>	<u>Total</u>
<u>Beginning of life</u>				
²³⁵ U	34.37	9.52	156.28	200.17
²³⁸ U	<u>11,421.14</u>	<u>3,164.08</u>	<u>57,693.80</u>	<u>72,279.02</u>
Total U	11,455.51	3,173.60	57,893.80	72,479.19
²³⁹ Pu	2,676.94	0.00	0.00	2,676.94
²⁴⁰ Pu	956.06	0.00	0.00	956.06
²⁴¹ Pu	191.21	0.00	0.00	191.21
²⁴² Pu	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>	<u>0.00</u>
Total Pu	3,824.21	0.00	0.00	3,824.21
Total U + Pu	15,279.72	3,173.60	57,849.08	76,303.30
<u>End of life</u>				
²³⁵ U	20.15	7.87	130.88	158.90
²³⁸ U	<u>10,670.97</u>	<u>3,099.42</u>	<u>56,478.30</u>	<u>70,348.69</u>
Total U	10,691.12	3,107.29	56,709.18	70,507.59
²³⁹ Pu *	2,240.38	54.83	956.23	3,251.44
²⁴⁰ Pu	971.49	1.03	22.62	995.14
²⁴¹ Pu	187.49	0.02	0.55	188.06
²⁴² Pu	<u>13.87</u>	<u>0.00</u>	<u>0.01</u>	<u>13.88</u>
Total Pu *	3,413.23	55.88	979.41	4,448.52
Total U + Pu *	14,104.35	3,163.17	57,688.59	74,956.11

*Includes ²³⁹Np.

REFERENCES

- ¹ Neuhold, R. J., Roy, D. H., and Thomas, N. C., Fast Reactor Code Report No. 2, A Computer Package for Fast Reactor Design Calculations, Babcock & Wilcox, BAW-TM-187, Lynchburg, Virginia, May 1967.
- ² Proceedings of the Data Testing Subcommittee of CSEWG, August 1967.
- ³ Wittkopf, W. A., Roy, D. H., and Livolsi, A. Z., U-238 Neutron Cross-Section Data for the ENDF/B, Babcock & Wilcox, BAW-316, Lynchburg, Virginia, August 1967.
- ⁴ Pitterle, T. A. and Yamamoto, M., Evaluated Neutron Cross Sections of Pu-240 for the ENDF/B File, APDA-TM-43, January 1967.
- ⁵ Drake, M. K. and Dyos, M. W., A Compilation and Evaluation of the Nuclear Data Available for the Major Plutonium Isotope, GA-6576, July 1965.
- ⁶ Sumner, H. M., ERIC-2, A Fortran Program to Calculate Resonance Integrals and From Them Effective Capture and Fission Cross Section, AEEW-R-323, 1964.
- ⁷ Roy, D. H., DISC-2, A Program for Computing Inelastic Scattering Distribution Matrices for the DHRG Library, Babcock & Wilcox, BAW-TM-196, Lynchburg, Virginia, July 1967.
- ⁸ Breebler, P., Memo to CSEWG (ENDF/B), January 16, 1967.
- ⁹ Crane, R. A., et al., Sodium Parameter Study Code "NAPS," Babcock & Wilcox, BAW-1326, Lynchburg, Virginia, April 1968.

3. NUMBER OF PRIMARY AND INTERMEDIATE
HEAT TRANSPORT LOOPS
TRADE-OFF STUDY

CONTENTS

	Page
1. INTRODUCTION	1
1.1. Purpose	1
1.2. Summary	1
2. CONCLUSIONS AND RECOMMENDATIONS	3
3. DISCUSSION	5
3.1. Ground Rules Affecting Loop Selection	5
3.2. Power Plant Experience Affecting Loop Selection	8
3.3. Configuration Descriptions	9
3.4. Plant Parameters	12
3.5. Component Capital Costs	12
3.6. Reliability	19
3.7. Safety	20
3.8. Plant Performance	20
3.9. Maintenance	22
3.10. Operating Costs	22
4. REFERENCE CONFIGURATION SELECTION	23

List of Tables

Table

1. Boiler Feedwater Temperatures	7
2. Configuration Matrix	11
3. Component Capital Cost Data	14
4. Basic Steam Generator Data	15
5. Configuration Data Affecting Performance	21

List of Figures

Figure	Page
1. Variation of Pump Flow Rate With Core ΔT and Overall Plant Efficiency.	24
2. Schematic Diagram of Candidate Loop Configurations.	25
3. Configuration 1 Plan View.	26
4. Configuration 1 Elevation View.	27
5. Configuration 2 Plan View.	28
6. Configuration 2 Elevation View.	29
7. Configuration 3 Plan View.	30
8. Configuration 3 Elevation View.	31
9. Configuration 4 Plan View.	32
10. Configuration 4 Elevation View.	33
11. Configuration 5 Plan View.	34
12. Configuration 5 Elevation View.	35
13. Configuration 6 Plan View.	36
14. Configuration 6 Elevation View.	37

1. INTRODUCTION

The heat generated in the core of an LMFBR is transported to the steam-raising equipment via primary and intermediate loops of flowing sodium. The extensive matrix of technically feasible primary and intermediate loop configurations dictates a trade-off study to provide supplementary data for design decisions.

1.1. Purpose

This study was performed to establish the best number of primary and intermediate loops for a 1000-MWe pot-type LMFBR. The study covered various arrangements and combinations of primary and intermediate components and piping.

1.2. Summary

Even though many loop configurations are possible, National LMFBR Program ground rules, power plant economic experience, and projected test facility capabilities permitted a reduction in the number of configurations receiving detailed attention. These factors and a decision to stay with loops and components that are multiples of three permitted a reduction of the number of configurations to six. Including loops and components that are multiples of two would have more than doubled the scope of work; also, designs and costs for the major components (pumps, IHXs, steam generators, and reheaters) were not available, and considerable effort would have been required to obtain them.

Realizing that power generation cost is the ultimate parameter in selecting the best configuration, the major emphasis in this study was placed on an evaluation of the capital cost differential between the configurations, since it is the most tangible of all the factors that affect plant economics. Other factors of a qualitative nature considered in this trade-off were the effects of safety requirements, reliability, performance, operating costs, and maintenance.

For each of the six configurations, component and piping layouts were prepared. Using the best information and methods available, costs were developed for primary pumps and drivers, intermediate pumps and drivers, IHXs, steam generators, reheaters, intermediate piping and valves, primary and intermediate sodium inventory, feedwater and steam piping, reactor vessel and cover plug, biological shielding, and steam generator and reheater buildings.

Based on the results of this study, configuration 4, having six primary loops and three intermediate loops, was selected as the best for the 1000-MWe LMFBR reference plant.

2. CONCLUSIONS AND RECOMMENDATIONS

1. Loop configuration 4, having six primary loops and three intermediate loops, has been selected as the most desirable configuration for the reference concept LMFBR. This trade-off demonstrated that no conclusively superior heat transport system configuration can be identified by the usual evaluation methods. Configuration 4 was selected because of its potential for alleviating some of the design problems associated with the head and diameter of the reactor vessel. In addition, this configuration results in a minimum reactor pot and cover plug size with accompanying capital cost savings.

2. Loop configuration 6, which has six completely separate but identical heat transport systems, is not an attractive design because its capital costs are at least 10% higher than those of the other configurations considered. The large number of major components in this configuration violates a generally accepted rule that fewer components accomplishing the same function result in a lower total component cost.

3. Among the five remaining configurations evaluated, configuration 2, having three primary pumps, six intermediate heat exchangers, three intermediate pumps, three steam generators, and three reheaters, has the lowest capital cost. However, a maximum of only 3.2% exists between the least expensive and the most expensive heat transport system. Clearly, this difference approaches the range of accuracy of the cost analysis procedures. The optimum heat transport system configuration cannot be selected solely on a capital cost basis.

4. Plant safety is not materially affected by the choice of a loop configuration.

5. Plant performance is not noticeably compromised by operational or design characteristics peculiar to any of the heat transport systems studied except configuration 6.

6. A brief review of the effect of the number of primary and secondary loops on the overall plant operating costs indicates little correlation. Indeed, it might be necessary to await data on several operational LMFBR plants before definite conclusions can be drawn.

7. Weighing the advantages of one configuration over another, from a reliability standpoint, will depend upon contract negotiation with the utility buying the plant. The utility must decide whether or not the added plant availability of configurations having more loops is worth the capital cost increase.

8. Factoring the maintenance aspect into the configuration selection will also be primarily dependent upon the policies and preferences of the utility buying a 1000-MWe plant.

3. DISCUSSION

3.1. Ground Rules Affecting Loop Selection

Plant size is important in selecting the number of coolant loops based on physical limitations on materials, manufacturing processes, and testing facilities. Ground rules state that a net electrical output of 1000 MW should be obtained from the proposed LMFBR design. Examination of the seven steam cycles studied by United Engineers indicates a wide variation in plant efficiency (33 to 43%). Immediately, the range of the primary flow rate can be defined based on the net plant output, cycle efficiency, average specific heat of sodium, and limiting core temperature difference:

$$W_{vol} = \frac{(0.1247)Q}{(h_{Na_{out}} - h_{Na_{in}}) \rho_{Na}} = (W_{pump})(N_{pump}) \quad (1)$$

where

Q = core thermal output, Btu/hr

$h_{Na_{in}}$ = core inlet sodium enthalpy, Btu/lb

$h_{Na_{out}}$ = core outlet sodium enthalpy, Btu/lb

ρ_{Na} = sodium density at the point of interest,
lb/ft³

W_{vol} = volumetric sodium flow rate at the point of
interest, gpm

W_{pump} = volumetric sodium flow rate per pump, gpm

N_{pump} = No. of pumps in system

Equation 1 may be more closely related to typical plant parameters by the following equation:

$$W_{pump} = \frac{(4.26 \times 10^5)P}{e \bar{C}_p \Delta T \rho_{Na} N_{pump}} \quad (2)$$

where

P = net plant electrical output, MWe

e = overall plant efficiency, dimensionless

$\bar{C}_p$ = average sodium specific heat, Btu/lb-°F

ΔT = core sodium temperature difference, °F

ρ_{Na} = sodium density, lb/ft³

Figure 1 is a graphical representation of equation 2 over the range of interest for a 1000-MWe LMFBR. A lower limit on the core coolant temperature difference has been established by considerations of core physics, safety, economics, and structural capabilities. There is a complex functional relationship between the minimum core ΔT and the factors just mentioned; however, there is a relatively simple qualitative explanation. Minimum fuel inventory and maximum breeding ratio are goals directly related to fuel costs that affect the bulk sodium temperature rise across the core. A minimum coolant ΔT and hence a maximum core flow rate are primary contributors to core Δp and are thereby influential in setting structural requirements. The fuel inventory is reduced by eliminating all unnecessary parasitic neutron absorbers and geometrically arranging fuel for maximum neutron economy. Having established coolant channel geometry and size, a minimum core temperature difference may be computed assuming the maximum core pressure drop considered in this study (100 psi). By extrapolating the reference design core hydraulic data, the minimum ΔT is 250 F. Limiting the core pressure loss to 100 psi is based on an assessment of the bundle structural capabilities and the practicality of building large high-pressure sodium pumps. Apparently others in industry and government have reached the same conclusion because the national LMFBR Program Plan has implied that pumps to be developed in the foreseeable future will be limited to somewhere between 350 and 450 feet of head.

The upper limit on the core coolant temperature difference is based on the maximum core outlet temperature that is possible with materials expected to be available for 1000-MWe LMFBR core construction and the minimum boiler feedwater temperature. Estimates place the maximum bulk core outlet temperature at 1175 F. The scope of the present LMFBR Study has been restricted to the seven steam cycles

evaluated by United Engineers under a previous contract with ANL. Boiler feedwater temperature and corresponding steam cycles are listed in Table 1.

Table 1. Boiler Feedwater Temperatures

Pressure, psia	Cycle		Feedwater temperature, F
	Steam temperature, F	Reheat temperature, F	
3500	1000	1000	485
2400	1000	1000	478
1800	900	900	465
1000	Dry and sat.	--	419
1450	1000	No RH	434
2400	950	950	484
2400	950	No RH	484

Subtracting the minimum boiler feedwater temperature from the projected maximum core outlet sodium temperature gives a value of 756 F for the upper limit on the core coolant temperature difference. Surely this number is far in excess of practical core temperature differences by virtue of the exorbitant heat transfer surfaces required in the IHX and steam generator when the feedwater temperature approaches the core inlet temperature. The core ΔT is not limited by thermal gradients in the fuel assemblies, since the fuel pins are permitted to expand freely. Figure 1 shows that the core temperature difference and plant efficiency set the minimum number of primary sodium loops within the framework of the projected sodium pump development program.

Safety standards,¹ as applied by B&W for emergency core cooling, dictate the minimum number of steam generators and intermediate heat transport loops to be incorporated in a 1000-MWe LMFBR. At least one of the normal reactor heat removal paths must be operational at all times and be backed up by an independent emergency core cooling system fully

capable of withstanding a maximum hypothetical accident (MHA). A minimum of two steam generators and two intermediate loops are required to satisfy this condition when the plant is down for steam generator maintenance. Physical size is another extremely important factor affecting the selection of the number of steam generators. According to reference 2, LMFBR steam generators of one-third capacity approach the limits for shop fabrication, thus indicating the likelihood of field fabrication and erection for less than three units with all the accompanying problems of quality control and added expense.

3.2. Power Plant Experience Affecting Loop Selection

From the previous section it is obvious that physical limitations and safety criteria do not appreciably narrow the matrix of candidate primary and secondary loop configurations. However, large central station power plant experience has shown that the number of heat transport loops should be minimized without adversely reducing reliability and plant capacity if one loop fails. Pressurized water reactors (PWR's) in the 1000-MWe range now contemplated have at least three heat transport loops. It is highly unlikely that more than six primary or secondary loops will be designed into a 1000-MWe LMFBR because the increased capital cost with smaller components buys only a marginal increase in operating flexibility.

Some consideration was given to including a four-loop plant in the trade-off study; however, this configuration was eliminated because all previous design work had been concentrated on systems that are a multiple of three. To evaluate a four-loop configuration properly, intermediate heat exchangers, pumps, pipework, steam generators and reheaters would all have to be sized for the new duty, and layout drawings would have to be prepared to investigate the effect on reactor symmetry and vessel size. If a four-loop plant were included, two- or eight-loop plants would probably have to be evaluated to obtain cost trends, thus opening a new series of plant configurations to be studied. Evaluating a four-loop plant appears to be a worthy project requiring future consideration, but such an undertaking is not compatible with the work scope of the present trade-off study. All factors considered, the detailed loop trade-off investigation has been restricted to configurations having

between three and six primary loops and between three and six secondary loops.

3.3. Configuration Descriptions

3.3.1. Reference Design (Configuration 4)

The primary loop of the selected reference design has six pumps and six heat exchangers. These components are physically independent of each other as a result of unfavorable cost and technical problems unique to the previously proposed combined unit.

The intermediate sodium system has three identical one-third-capacity loops. Each loop has a pump, two intermediate heat exchangers, a steam generator, a reheater, and all the necessary piping and valves required to direct and control the sodium flow (see Figure 2). The advantages attributed to the reference design heat transport system are as follows:

1. The reactor vessel diameter is minimized because the intermediate heat exchanger diameter is the smallest for the six-IHX design.
2. The primary sodium inventory is minimized because of the small reactor vessel diameter made possible by the choice of six IHX's.
3. Since the failure of a primary pump minimizes the reduction in plant capacity, only a small loss in saleable power results.
4. Control of the intermediate system is simplified with the smaller number pumps.
5. The capability for full plant capacity is improved with only one pump per intermediate loop because the control scheme does not permit load variations between loops. One large pump is more reliable than two half-capacity pumps.
6. Less preventive maintenance is required with fewer intermediate pumps.

3.3.2. Configuration 1

Configuration 1 represents the greatest variation from the reference design with respect to the primary sodium system (see Figure 2). Major components in both the primary and the intermediate loops are minimized consistent with the previously discussed limits. The three primary pumps, the three intermediate pumps, and the three intermediate heat exchangers of this configuration suggest the following advantages:

1. Reduced instrumentation and control equipment costs.
2. Reduced preventive maintenance.
3. Reduced total capital cost associated with the fewer components for the same.

The disadvantages of the triple-component design are as follows:

1. The reactor vessel size is increased from 52 to 57 feet when the number of intermediate heat exchangers is reduced from six to three. Corresponding increases in the cover plug diameter and biological shield volume are additional undesirable features accompanying the reduction in the number of intermediate heat exchangers.
2. The sodium inventory is maximized.
3. The operational flexibility is lower than that of the other configurations studied.
4. Handling larger components may be more difficult during fabrication and maintenance.

3.3.3. Configuration 2

Configuration 2 differs from the reference design only in the number of primary pumps (see Figure 2). Three primary pumps and six intermediate heat exchangers offer the advantage of reduced primary component costs without an increase in reactor vessel size, component size, and biological shield volume.

3.3.4. Configuration 3

The primary system for configuration 3 is the same as in configuration 2. Configuration differences occur in the secondary system where the flow control valves of configuration 2 are replaced by three additional variable-speed pumps for configuration 3 (see Figure 2). Six variable-speed secondary pumps ease the problem of balancing the intermediate heat exchanger outlet temperature.

3.3.5. Configuration 5

Configuration 5, which has six primary pumps, six intermediate heat exchangers, six intermediate pumps, and three steam generators (see Figure 2) was selected for evaluation to show the relative cost incentives for using three steam generators in a six-loop system instead of making the loops entirely independent as in the configuration to be discussed next.

3.3.6. Configuration 6

Although studies by B&W's Boiler Division favor one-third-capacity steam generators for a 1000-MWe LMFBR, six completely independent loops were chosen for this configuration so that at least one datum point at each extremity of the configuration matrix would be available to show cost trends (see Figure 2).

Candidate configurations are summarized in Table 2.

Table 2. Configuration Matrix

<u>Configuration</u>	<u>No. of primary pumps</u>	<u>No. of IHX's</u>	<u>No. of intermediate pumps</u>	<u>No. of steam generators and reheaters</u>
1	3	3	3	3
2	3	6	3	3
3	3	6	6	3
4	6	6	3	3
5	6	6	6	3
6	6	6	6	6

3.4. Plant Parameters

To isolate the effect of the number of primary and intermediate loops on the plant economics, the following design features and performance parameters were held constant:

Reactor outlet temperature, F	1150
Reactor inlet temperature, F	800
IHX hot end ΔT	75
IHX cold end ΔT	125
Steam cycle	2400/1000/1000
Core geometry	Reference design
Maximum velocity, intermediate and primary piping, ft/sec	20
Isolation valves on all lines leaving or entering the containment building	
Isolation valves on sodium outlet of SG and RH.	
All SG and RH to be separated by blast shielding and protected from fire propagation	

All data reflect the reference design philosophy as of September 5, 1967.

3.5. Component Capital Costs

Table 3 lists all the major items associated with the heat transport loops affecting plant capital costs. The dollar value for all items is included as part of the table, with grand totals appearing at the bottom of the columns.

Because sodium-cooled reactor technology is still in its infancy little or no component cost data are available in many areas. Component pricing methods are discussed in the following paragraphs.

3.5.1. Pumps

Original plans for computing pumping equipment costs were to utilize data generated by the Byron Jackson Company in the AEC-sponsored pump study and prices quoted by several motor manufacturers to establish an accurate estimation procedure. Because pump data were not available when needed, the total cost of the motor and

pump was estimated to be \$150.00 per kilowatt based on the electrical power consumption of the motor. Whenever the number of pumps was reduced by a factor of two for a particular service, a corresponding one-third reduction in the total pump capital cost was applied.

3.5.2. Intermediate Heat Exchangers

Early in the current study the combined intermediate heat exchanger (IHX) and primary pump idea was favored as a means of keeping the reactor vessel size within reason. Although small units working in a gas environment have been successfully built and operated in the United Kingdom, experience with large units is nonexistent. Without a precedent, cost figures on such a unit would be meaningless. To get a first-order approximation on the cost of the IHXs for a 1000-MWe plant, B&W Boiler Division engineers made a detailed analysis of three scaled-up Fermi-type designs. To do this, the Fermi-type heat exchanger was divided into its component parts. Each part was priced using the latest manufacturing data and currently accepted techniques. Total cost figures on the three designs were divided by their total physical weights to obtain cost factors for interpolation and extrapolation. The cost factors for all three designs were identical at 4.56 dollars per pound of dry weight. Because of the consistency of this figure and the relative ease of application, it was adopted for use in all trade-off studies associated with the present contract.

3.5.3. Steam Generators

For a number of years B&W's Boiler Division has been designing and developing sodium-heated steam generators. Details of the 1000-MWe design may be found in reference 2. To extend the range of sodium inlet temperatures, the superheater bundle was split into two segments with the interface occurring where the sodium temperature is 1000 F. Bundle tubes sensing temperatures above 1000 F are made of 316 stainless steel, and those experiencing temperatures below 1000 F are made of Croloy 2-1/4.

Table 3. Component Capital Cost Data

Item	Unit cost, \$					
	Configuration 1	Configuration 2	Configuration 3	Configuration 4	Configuration 5	Configuration 6
Primary pumps and drivers	1,736,000	1,736,000	1,736,000	2,606,000	2,606,000	2,606,000
Intermediate pumps and drivers	460,000	460,000	690,000	460,000	690,000	690,000
Intermediate HX	2,296,000	3,079,000	3,079,000	3,079,000	3,079,000	3,075,000
Steam generator	11,300,000	11,300,000	11,300,000	11,300,000	11,300,000	13,108,000
Reheater	3,971,000	3,971,000	3,971,000	3,971,000	3,971,000	4,606,000
Intermediate piping	118,000	130,000	137,000	130,000	137,000	130,000
Valves	440,000	683,000	558,000	683,000	558,000	713,000
Primary sodium inventory	509,000	405,000	405,000	405,000	405,000	405,000
Intermediate sodium inventory	325,000	324,000	322,000	324,000	322,000	313,000
Feedwater piping	795,000	795,000	795,000	795,000	795,000	1,186,000
Steam piping	817,000	817,000	817,000	817,000	817,000	1,218,000
Reactor vessel	1,950,000	1,553,000	1,553,000	1,553,000	1,553,000	1,553,000
Cover plug	5,691,000	4,836,000	4,836,000	4,836,000	4,836,000	4,836,000
Biological shielding	211,000	195,000	195,000	195,000	195,000	195,000
SG and RH buildings	122,000	116,000	116,000	116,000	116,000	89,000
Grand totals	30,741,000	30,400,000	30,510,000	31,270,000	31,380,000	34,723,000

The laborious task of making a detailed cost analysis every time a slight variation in duty occurred was circumvented by utilizing accurate data compiled on the one-third-capacity unit. Important descriptive parameters for this unit are listed in Table 4.

Table 4. Basic Steam Generator Data

Item	Economizer section	Boiler section	Superheater section 1	Superheater section 2
Number of tubes	676	676	676	1220
Tube OD, in.	1.0	1.0	1.0	0.875
Shell OD, in.	156.0	156.0	156.0	156.0
Tube thickness, in.				
3500 psi	0.247	0.247	0.247	*
2400 psi	0.180	0.180	0.180	*
1800 psi	0.145	0.145	0.145	*
1450 psi	0.120	0.120	0.120	*
Heat transfer coefficient, Btu/ft ² -°F	510.0	525.0	473.0	361.0

* Tube thickness $\frac{PD}{2SE + 0.8P}$

where
S : allowable stress, psi
D : diameter, in.
P : pressure, psi
E : joint efficiency

The 1964 selling price for a one-third-capacity steam generator was escalated to 1967. Scaling this cost up and down with thermal duty was accomplished in the NAPS computer code by:

1. Computing the required heat transfer area for the Croloy bundle and stainless steel bundle and adding 10% excess area.
2. Computing the weight of the bundle sections and the shell for the steam generator being priced.
3. Adjusting the baseline cost figures by physical weight factors.

In equation form the scaled price becomes

$$\begin{aligned} \text{steam gen scaled price} &= [\$(\text{baseline boiler bundle})] \left[\frac{W(\text{scaled steam gen boiler bundle})}{W(\text{baseline boiler bundle})} \right] \\ &+ [\$(\text{baseline superheater sect 2})] \left[\frac{W(\text{scaled superheater sect 2})}{W(\text{baseline superheater sect 2})} \right] \\ &+ [\$(\text{baseline shell})] \left[\frac{W(\text{scaled shell})}{W(\text{baseline shell})} \right] \end{aligned}$$

where \$ = dollar values and W = physical weight.

3.5.4. Reheaters

Reheater costs were figured as a function of surface area only. A scaling factor of \$67.40/ft² has been used throughout this study.

3.5.5. Intermediate Loop Piping

The intermediate loop piping for all six configurations met the following specifications:

Material	AISI 304SS
Design pressure, psi	150
	<u>Design temp, F</u> <u>Allowable stress, psi</u>
Hot leg	1,125 5,650
Cold leg	775 13,575
Reheater cold leg	900 5,650

Piping costs were derived by:

1. Sizing all pipes for a maximum sodium velocity of 20 ft/sec.
2. Computing wall thickness as a function of pipe size and allowable stress.*
3. Determining the quantity of each size of pipe from the layout drawings shown in Figure 3-14. These layouts of the various combinations were used to locate and position the steam generators, reheaters, intermediate pumps, and valves in order to minimize the length of

* Arbitrary minimum wall thickness was set at 3/8 inch.

the intermediate sodium piping required to complete the loop and still provide for thermal expansion during reactor operation.

4. Computing the total weight of all intermediate piping.
5. Computing the cost of the pipe alone using a figure of \$0.42/lb.
6. Computing the total cost of the pipework including fabrication and erection by multiplying the materials cost by a factor of 1.79.*
7. Mixing sections were not included in the tabulated costs, because costing information was not available. Configuration 6 is the only one in which the total cost would have changed materially relative to the others if the cost information had been available, but configuration 6 was already the most expensive, and the addition would have made it more so.

3.5.6. Sodium Valves

Cost data on large sodium valves are almost nonexistent at the present time. With the assistance of the Liquid Metals Engineering Center, a 1966 study report on a 30-inch butterfly control valve issued by the Allis Chalmers Company was located⁴. Two important conclusions found in the report are as follows:

1. The valve designs analyzed ranged in cost from \$16,000 to \$125,000 depending on the type of service.
2. The cost of a 30-inch butterfly valve not completely suitable for sodium service was listed at \$16,000.

In the absence of adequate data the results of the AC report were adjusted as follows so that some account could be made for valve cost in trading off loop configurations:

1. The cost of the 30-inch butterfly valve was set at \$25,000.00 assuming that complete sodium serviceability could be bought for an additional \$9,000.00 per valve.
2. The cost of an isolation valve was estimated to be 1.25 times as much as that for a butterfly control valve.

- - - - -
* Derived from data shown in reference 3.

3. Valve costs for diameters other than 30 inches were scaled from the reference using the equation:

$$\text{\$ scaled valve} = \text{\$ reference valve} \left(\frac{\text{Diam scaled valve}}{\text{Diam reference valve}} \right)^{0.6}$$

3.5.7. Sodium Inventory

Sodium volume figures for the primary and secondary loops were converted into weights by assuming an average density of 51 lb/ft³. Inventory costs were based on a price of \$0.20 per pound.

3.5.8. Steam System Pipework

Steam piping, feedwater piping and associated valve costs were determined by adjusting the figures published by United Engineers in Volume 3 of their study on Large Steam-Electric Generating Plants. A cost of \$1,855,000 for their "B" unit designed for four sodium-heated steam generators was adjusted for three and six steam generators by setting a ratio and raising it to the 0.6 power. In equation form this may be expressed as

$$\begin{array}{l} \text{\$ steam pipework} \\ \text{for N steam generators} \end{array} = \left(\begin{array}{l} \text{\$ steam pipework} \\ \text{for 4 steam generators} \end{array} \right) \left(\frac{N \text{ steam generators}}{4 \text{ steam generators}} \right)^{0.6}$$

3.5.9. Reactor Vessel

Reactor vessel costs have been computed on the basis of \$3.00 per pound. This figure has been arbitrarily applied because stainless steel vessels of this size, quality, and character have not been constructed to date.

3.5.10. Cover Plug

Because of insufficient design detail an overall price of \$5,000,000.00 was estimated for a cover plug 60 feet in diameter. This price was scaled to larger and smaller cover plugs using an area weighting factor based on the price given above (\$1,760/ft²).

3.5.11. Biological Shielding

Changes in the volume of the biological shielding follow changes in the diameter of the reactor vessel because shield thickness is a function only of the source intensity, source character, and permissible dose rate in the areas being protected. A cost factor of \$100.00 per cubic yard was used to establish shielding costs for all the configurations. This factor is for ordinary concrete and includes materials, labor, and reinforcing steel.

3.5.12. Buildings

Buildings to enclose the steam-raising equipment were priced in accordance with the accepted estimation practice of relating cost to floor space. The factor chosen for this study was \$10.00 per square foot. Since the buildings being priced will require special items, such as an overhead crane to remove boiler bundles, the \$10.00/ft² cost factor could be significantly affected; however, cost differentials should be sufficiently accurate for the scope of this trade off.

3.6. Reliability

The basic question to be considered here is whether or not the number of primary and secondary heat transport loops affects the overall plant reliability. A quantitative analysis of this factor has not been attempted because of the limited contract scope; however, certain general observations are presented in the following paragraphs.

Reliability in a nuclear plant influences two very important areas, safety and plant availability. Failure of the main heat transport systems will not adversely affect plant safety in the reference design because adequate emergency cooling has been provided by an independent system. This fact reduces the problem to evaluating the effect of the heat transport system's reliability on plant availability. The following capabilities and limitations affecting plant availability are a result of the plant design and control philosophy:

1. Malfunction of any component in a secondary loop results in a complete shutdown of that loop because an unbalanced output from the steam generators cannot be handled. If a failure occurred in the intermediate sodium system, the plant output would drop in larger

increments for a plant with three steam generators than for a plant with six generators.

2. Malfunction of a primary system pump does not affect the plant output nearly as much as an intermediate system component does because the open primary loop design permits the operation of any number of the primary pumps without special equipment.

Selecting the optimum number of primary and secondary loops from a plant reliability (availability) standpoint will actually be up to the utility buying the plant. Generally speaking, a utility is willing to endure added capital costs, within reason, for an increase in plant availability.

3.7. Safety

Safety was eliminated as a trade-off factor by stipulating that all configurations would have to meet the minimum safety requirements as discussed in 3.1.

3.8. Plant Performance

Differences in the plant performance of the six loop configurations could occur only because of the differences in heat loss from the pipe-work and differences in the pressure loss of the flowing sodium. Table 5 summarizes the pipework surface area and the components causing pressure losses.

Assuming equal thicknesses of insulation, only configuration 6 would be expected to show significantly lower performance due to heat losses. Since all sodium pipework was sized for a maximum velocity of 20 ft/sec, pumping power can be discussed on a relative basis in terms of the number of valves and bends. Again only configuration 6 can be downgraded because of substantial performance differences without a detailed analysis of pumping power requirements.

Table 5. Configuration Data Affecting Performance

Configuration number	Pipework surface area, ft ²	Steam generator and reheater surface area, ft ²	Total surface area, ft ²	Total No. of valves	Total No. of 90-degree bends	Total No. of 45-degree bends	Total No. of 180-degree bends
1	9,665	4910	14,575	18	52	0	3
2 and 4	11,190	4910	16,101	30	57	4	3
3 and 5	11,803	4910	16,713	24	65	9	3
6	11,452	3640	15,092	36	84	24	6

3.9. Maintenance

Although the cost of routine maintenance increases with the number of components, the size of the components in question can be very influential on the ease with which maintenance can be performed and on the expenditures required for spares. Without going into a detailed analysis, there appears to be at least some minor advantages related to the maintenance aspect that would favor configurations with a larger number of loops. The degree of importance of maintenance on the loop configuration selection is another item that will ultimately depend on the experience and preference of the utility buying the plant.

3.10. Operating Costs

Operating cost variations in the number of heat transport loops are interrelated to the previously discussed topics, maintenance and reliability. If the assumption is made that there is no change in component reliability with size, then configurations having the largest number of loops will require more attention to keep the plant operational. Considering the absolute reliability of the components expected by 1980 and the trends in labor versus management relationships, no discernable differences in operating costs are expected as a result of variation in the number of loops.

4. REFERENCE CONFIGURATION SELECTION

Having considered the philosophy and the data assembled for this trade-off study, the following statements explain the selection of Configuration 4 as the reference design:

1. Configuration 6 was eliminated because capital costs were distinctively higher than for any other configuration. The gain in plant availability would not be worth the added costs.
2. Configuration 5 was eliminated because the additional pumps in the secondary loop actually reduce the reliability and hence the plant availability. Any maintenance advantage in the smaller pumps would not overshadow the availability factor.
3. Configuration 1 was eliminated because the three large intermediate heat exchangers exert quite an influence on the reactor vessel size. A difference of 5 feet occurs between configurations having three and six intermediate heat exchangers.
4. Configurations 2 and 3 were eliminated because plant availability is compromised when only three primary pumps are used with six intermediate heat exchangers.

Figure 1. Variation of Pump Flow Rate With Core ΔT and Overall Plant Efficiency

Figure 2. Schematic Diagram
of Candidate Loop
Configurations
(B&W Dwg A31094-E)

Figure 3. Configuration 1 Plan View
(B&W Dwg A31082-E)

Figure 4. Configuration 1 Elevation View (B&W Dwg A31083-E)

- ① IHX
- ② PRIMARY PUMP
- ③ INTERMEDIATE PUMP
- ④ STEAM GENERATOR
- ⑤ REHEATER

Figure 5. Configuration 2 Plan View
(B&W Dwg A31084-E)

Figure 6. Configuration 2 Elevation
View (B&W Dwg A31085-E)

Figure 7. Configuration 3 Plan View
(B&W Dwg A31086-E)

Figure 8. Configuration 3 Elevation
View (B&W Dwg A31087-E)

Figure 9. Configuration 4 Plan View
(B&W Dwg A31088-E)

Figure 10. Configuration 4 Elevation
View (B&W Dwg A31089-E)

Figure 11. Configuration 5 Plan View
(B&W Dwg A31090-E)

Figure 12. Configuration 5 Elevation
View (B&W Dwg A31091-E)

Figure 13. Configuration 6 Plan View
(B&W Dwg A31092-E)

Figure 14. Configuration 6 Elevation
View (B&W Dwg A31093-E)

REFERENCES

- ¹ Delicate, W. S. , "General Design Criteria for a Liquid Metal Cooled Fast Breeder Reactor," Babcock & Wilcox, AED, September 29, 1967.
- ² "Preliminary Design Report, Full Size Steam Generator," Babcock & Wilcox, Boiler Division, Barberton, Ohio, AEC Contract No. AT(11-1)-1280, Vol 1, July 1, 1964.
- ³ "Engineering Study for Large Steam-Electric Generating Units and Sodium Heated Steam Generating Islands, "United Engineers & Constructors, Inc. , Philadelphia, ANL Contract No. 31-109-38-1717, May 1, 1966.
- ⁴ "Conceptual Design of Large Sodium Valves," Allis Chalmers Company, Bethesda, Maryland, AEC Contract No. AT(11-1)-1357, March 1966.

4. VENTED VS NON-VENTED FUEL PIN
TRADE-OFF STUDY

CONTENTS

	Page
1. INTRODUCTION	1
1.1. Purpose	1
1.2. Objective	2
1.3. Scope	2
1.4. Summary	2
2. CONCLUSIONS AND RECOMMENDATIONS	3
3. DISCUSSION	4
3.1. Fuel Pin Designs	4
3.2. Effects on Reactor Systems	8
4. RESULTS	17
4.1. Calculational Methods	17
4.2. Basic Design Parameters	18
4.3. Economic Comparison	20
4.4. Effects of Economic Uncertainties	22
APPENDIX — Gaseous Fission Product Release and Deposition	A-1

List of Tables

Table

4-1. Reference Core Parameters	23
4-2. Economic Comparison of Vented and Non-Vented Concepts	24
A-1. Fission Gas Release Fractions and Leak Rates	A-9
A-2. Equilibrium Cover Gas Activities	A-10
A-3. LMFBR Gaseous Fission Product Study - Maximum Permissible Activity Concentrations in Air	A-11
A-4. Direct Dose Rates From Inert Gases in the Reactor Building	A-12
A-5. Reactor Building Access With No Protection With Leaking Cover Gas	A-13

List of Figures

Figure	Page
3-1. Bottom Venting Using Mechanical Connections	13
3-2. Bottom Venting Using Diving Bell Connectors	14
3-3. Bottom Venting Using Porous Plug Connectors	14
3-4. Venting to Sodium Using Porous Plug	15
3-5. Venting to Sodium Using Diving Bell	15
3-6. Venting to Sodium Using Combination Porous Plug and Diving Bell	15
3-7. Venting to Gas Blanket	16
4-1. Effects of Venting on Unit Power Generation Costs Showing Perturbations of Fuel OD and Clad Thickness . . .	26
4-2. Effects of Venting on Unit Power Generation Costs Showing Perturbations of Cladding OD and Clad Thickness	27
4-3. Effects of Venting on Unit Power Generation Costs Showing Perturbations of Cladding OD and Clad Thickness With BeO Addition	28
A-1. Equilibrium Cover Gas Activity as a Function of Cover Gas Purge Rate (No Leaks)	A-14
A-2. Decay Tank Release Rate of Activity After Holdup as a Function of Cover Gas Purge Rate	A-15
A-3. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 0.0 cfm, Fuel Leak Only)	A-16
A-4. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^4 cfm, Fuel Leak Only)	A-17
A-5. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^5 cfm, Fuel Leak Only)	A-18
A-6. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^6 cfm, Fuel Leak Only)	A-19
A-7. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 0.0 cfm, Vent + Leak)	A-20
A-8. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^4 cfm, Vent + Leak) . . .	A-21

List of Figures

Figure	Page
A-9. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^5 cfm, Vent + Leak) . .	A-22
A-10. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^6 cfm, Vent + Leak) . .	A-23
A-11. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 0.0 cfm, Vent Only)	A-24
A-12. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^4 cfm, Vent Only) . . .	A-25
A-13. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^5 cfm, Vent Only) . . .	A-26
A-14. LMFBR Gaseous Fission Product Study - Fraction of MPC of Equilibrium Cover Gas in Reactor Building as a Function of Cover Gas Leak Rate (Containment Atmosphere Purge Rate = 2.4×10^6 cfm, Vent Only) . . .	A-27

1. INTRODUCTION

The release of considerable quantities of fission gas from high-burnup fast ceramic fuels is a well known phenomenon which is a major factor in reactor design. The problem arises as how to contain or dispose of these gases safely and economically without destroying the integrity of the clad and thus subjecting the fuel to possible chemical attack from the coolant. The containment of the fission gas products within the fuel pin requires either a relatively large gas plenum within individual fuel pins or a relatively thick cladding. These alternatives prove detrimental to thermal and nuclear performance. At this time there is no gas absorber available to reduce the fission gas pressure within the clad to an acceptable level. As an alternative a vented fuel element would reduce the cladding stress caused by released fission product gas pressure, thereby increasing the potential for extended fuel burnup. The development of a successful LMFBR design is dependent upon achieving a low fuel cycle cost. However, fabrication and reprocessing costs are invariably high owing to the problems associated with handling plutonium fuels. To minimize the effects of these costs it becomes necessary to achieve high fuel burnup rates while maintaining good nuclear economy. These requirements for fuel burnup and nuclear economy tend to place certain restraints on fuel pin design in terms of fission gas containment. For this reason the question of vented versus non-vented fuel pins becomes very important to the overall economics of the reactor.

1.1. Purpose

The successful development of a 1000-MWe LMFBR by the early 1980's will require the early implementation of an efficient development program. In the preliminary stages of this program it is necessary to outline both the developmental requirements and the priorities. This

study will establish the basis for such a priority in regard to vented fuel development.

1.2. Objective

In order to establish the requirements of the 1000-MWe LMFBR development program, it becomes necessary to determine the economic and safety advantages of certain design concepts. This study will provide sufficient information to allow an effective evaluation of both vented and non-vented fuel pins.

1.3. Scope

Theoretically, a trade-off study of this type should compare the best vented-fuel design against the optimum non-vented fuel design. This approach does not, however, lend itself to developing a clear understanding of the effects of basic reactor parameters, because of the large number of design changes required to optimize a particular fuel pin design. Consequently, the approach in this study has been to study perturbations around a reference design. It is believed that such a study provides an adequate basis for venting concept selection and allows a clear understanding of basic trends.

1.4. Summary

The overall economic effects and design features of non-vented and various vented arrangements are determined and compared in this study. Included are the economic effects on the fuel cost, the reactor primary and intermediate systems, and the turbine island. Because of present design uncertainties, the costs of the reactor auxiliary systems cannot be determined accurately. Hence, these costs are assumed to be constant in the analysis and only economic trends are discussed. However, this assumption does not appear to detract from the final conclusions that are reached. As will be shown, the overall effect of these trends does not appear to alter the advantages of the porous-plug, vented-to-sodium arrangements.

2. CONCLUSIONS AND RECOMMENDATIONS

1. The porous plug, vented-to-sodium concept offers the most potential in economic savings, nuclear performance, and safety, subject to a successful research and development program (see Figure 3-4). Consequently, this design is recommended as the first choice in the design of the reference core.

2. The diving bell concept combines the nuclear and safety advantages of the porous plug design, but requires no new major research and development program. Economically, this design is somewhat inferior to the porous plug concept. Nevertheless, it may be an attractive backup for the porous plug design.

3. Concepts requiring a manifold to each individual fuel pin are considered unattractive because of high fabrication costs. In addition, several of these designs present a major design problem by requiring gas-tight couplings that must be connected and disconnected remotely.

4. Non-vented fuel pins do not appear to be as economically attractive as vented-to-coolant concepts within the limitations of currently available materials. Practical limitations on the gas-void length dictate a clad thickness that would prove to be less desirable in terms of nuclear performance.

5. All of the venting schemes offer a safety advantage in that the cladding need not act as a fission gas pressure containment vessel, thereby reducing the demands on the clad material. Therefore, the probability of fuel pin failure might be expected to be reduced. Should cladding failure occur, the consequences would be less severe because of the reduced pressures and the quantity of fission gases that might be released.

3. DISCUSSION

3. 1. Fuel Pin Designs

Various approaches to venting or retention of fission gases have been considered. The relative merits of each approach must be weighed in terms of both economics and safety. Upon this basis several designs may be selected for further evaluation.

3. 1. 1. Non-Vented

It is generally agreed that the non-vented concept is the most conservative in terms of current technology. It provides a positive barrier to the release of fission products to the coolant or to the ingress of sodium into the fuel.

However, failure of the fuel element cladding appears to be a much more serious problem with the non-vented design because of the increased probability of failure propagation. Fuel element failure propagation is a problem that has been discussed, but data are currently lacking. The primary modes of propagation that have been considered are as follows:

1. The release of relatively large volumes of fission gas into the coolant flow channel may restrict coolant flow around adjacent fuel pins so as to cause failure of these pins.
2. The release of gaseous fission products may impinge hot fuel particles against adjacent fuel elements so as to produce failures in these pins.
3. The release of gaseous fission products may spray hot fuel particles into the coolant causing vaporization and, hence, resulting in a positive reactivity insertion.

Research should be undertaken to provide a better understanding of this problem. However, the potential problem can be

expected to be greatly reduced with vented fuel elements because of the lower fission gas pressures and volumes retained in these elements.

3.1.2. Separate Vent System

Venting of the fission products into a plenum chamber located at the bottom of the fuel elements, as is done in the Peach Bottom and Dragon (U.K.) HTGR, appears to be more attractive, but the problems associated with designing a positive connect-disconnect mechanism may prove cumbersome.

Considering refueling via the top of the reactor vessel, the collector plenum can be permanently mounted below the fuel in the core supporting structure or mounted in detachable collectors at the top of the assembly. Considering refueling via the bottom of the reactor vessel, the positions of the permanent and detachable connectors are reversed. The separate vent system creates a sealing problem during the refueling operation since the attachment connection between the fuel bundle and the vent system must be broken when removing fuel. During this period it is desirable for the fission gases in the fuel pin to remain entrapped so that the vent system does not contaminate the reactor coolant sodium and large volumes of the reactor coolant are not lost into the cleanup system, thereby releasing gaseous fission products to the primary coolant system or appreciably increasing the pressure of the vent system.

The device shown in Figure 3-1 presents a possible solution to this problem, but a device of this type has a disadvantage in that an operating mechanism of relatively small proportions is built into the permanent structure of the reactor. Such a device could be simply adapted into a replacement assembly although some difficulty might be experienced in identifying a faulty installed assembly.

The device shown in Figure 3-2 presents another possible solution to the problem. Commonly referred to as the diving bell seal, it has the advantage of having no moving parts or accurate radial clearance requirements. The device is self-centering, and the accuracy of its radial location is not critical. The disadvantage of such a seal is that it is directly affected by pressure changes. Hence, the height of

the seal depends on the requirement that the seal be able to control the fission gas contraction that occurs upon core shutdown.

A third alternative uses a double filter plug as shown in Figure 3-3. One plug is permanently attached to the fuel bundle and the other is mounted on the vent system. With this solution, locating tolerances are not critical. This system inherently includes many of the disadvantages of the vent-to-coolant arrangement. Fission gas can be dispersed into the coolant, and some sodium may pass through the filter plug into the fuel element.

Any scheme that incorporates venting connections near the inlet of the fuel bundle presents a potential safety problem. A failure of any of these connections could potentially void the adjacent fuel bundles. Aside from the safety problem, all vented-to-cover-gas designs appear to have somewhat higher power generation costs than the porous-plug, vented-to-coolant concept. The major cost differential appears to be due to the increased fabrication problems associated with building a manifold-type assembly. The combination of safety and economic factors make each of the discussed vented-to-cover-gas arrangements appear relatively unattractive, and hence, no detailed cost studies will be performed.

3.1.3. Venting Directly to Reactor Coolant

The possibility of venting directly to the reactor coolant has been successfully tested and appears to offer numerous advantages. Because fission gas pressure is relieved, fuel cladding stress is reduced. Consequently, the incentive to use exotic materials is reduced, and the use of thinner cladding is made possible. As a result, improved nuclear performance is achievable, but is gained at the expense of greater fission gas release in the primary system.

If the fission products are vented directly to the sodium, then the particle retention vent must be of high integrity. Experimentation shows that the expected fission product hold-up time is about 5 days,² although this time may be significantly reduced during a power excursion. All vents must be able to pass the fission products through the vent under relatively small differential pressures and at the same time must prevent liquid sodium from passing through the vent when it is wetted by

sodium and under back pressure. Small quantities of sodium may be expected to pass through the vents and to accumulate inside the fuel pin.⁷ Present designs appear incapable of preventing such a process. It is essential that this sodium does not clog the venting medium and result in over-pressurization of the clad.

Initially, concern had been expressed that fission gas bubbles might increase the possibility of sodium voiding in the core. This does not appear to be a problem with the present reactor design. The reference plant utilizes a pot-type reactor vessel in which coolant velocities are sufficiently low to insure adequate de-entrainment of fission gas products during circulation through the pot.

The simplest form of venting is the porous plug shown in Figure 3-4. This device consists of a porous material located in the top end of an open fuel pin and sealed into position. The porous plug could be manufactured from a sintered metallic material. The material to be used, grade, bond, method of manufacture, diffusion length, and the like would require a development and test program to verify the operational requirements.

Another solution is the fluid diving bell seal shown in Figure 3-5. The primary advantage of this design is that it does not rely upon an extensive development program.

Still another solution is to combine the filter plug and the fluid seal as shown in Figure 3-6. The advantage of this device is that after the pressure equilibrium has been established between the fission product gas and the coolant, the filter plug would not be subjected to continuous wetting by the sodium coolant and hence should have a reduced tendency to plug up.

3. 1. 4. Venting Directly to the Reactor Gas Blanket

This solution involves the continuation of the fuel pin cladding containment to a level above the reactor sodium level, thereby enabling the fission product gas to vent directly into the cleanup gas blanket as shown in Figure 3-7. In order to offer a more attractive engineering and cost solution, the fission products of the fuel bundle are collected in a chamber and vented through a single riser tube. As a safety feature, a simple non-return valve would be located at the outlet of the riser tube

to prevent sodium entry to the fuel pins under transient behavior of the reactor sodium level.

The advantage of this scheme is that the fission product gases are released directly to a gas environment, thereby reducing the fission product levels in the reactor coolant system. In addition, more information is available regarding gas cleanup systems (HTGR) as compared to sodium. Fabrication costs and problems associated with handling the long fuel elements appear to be the major disadvantage of this approach.

3.1.5. Periodic Venting

Removing the fission gas periodically is possible, but the advantages are limited and the engineering considerations tend to make this approach impractical. Among the designs suggested are remotely operated valves, relief valves, and melting plugs. The problems associated with the reliable design of these remotely controlled mechanisms make this approach difficult to engineer. Among these problems are the release and efficient resealing of every pin or even each fuel bundle assembly (bearing in mind the sodium coolant decay heat) and the shutdown time investment.

3.2. Effects on Reactor Systems

Selection of a fuel pin design will cause perturbations throughout the reactor plant design. While a detailed economic analysis of such design features does not readily lend itself to calculation, these factors must be weighed in selection of a fuel pin design.

3.2.1. Fuel Handling

Fuel handling definitely favors the vented-to-sodium concept. The length of the cladding extension for the non-vented fuel will be reflected in a size (and cost) penalty to the entire fuel handling system. Since provision must be made to handle failed elements, no additional provisions need be made for the release of fission gases from the vented pins. Because every fuel pin contains a potentially lethal quantity of fission gas under relatively high pressure, a potential clad failure becomes a much more serious hazard for the non-vented pin.

3.2.2. Primary System Component Maintenance

Radioactive products are released to the primary system from activated stainless steel (^{58}Co , ^{54}Mn), failed fuel elements, and, if applicable, fission gas vents. It appears probable, based on experience with other sodium reactors, that most of the contamination will eventually either plate out or alloy with the primary system internals. It is presently believed that contamination due to both primary sodium activation and fission gas release may be removed by steam cleaning, as demonstrated in the operation of BR-5.⁴ However, failed fuel pins will most probably dictate chemical removal from the stainless steel surface containing these components.

Concern for the effects of fission gas venting on maintenance procedures led to an analysis that indicated one possible problem area: the buildup of ^{134}Cs on the primary system components may increase the shielding requirements during cleanup and maintenance. This buildup is the result of the release of the parent ^{133}Xe through the fission gas vent, subsequent decay to ^{133}Cs , and transmutation to ^{134}Cs by neutron capture. Since shielding, in either case, must be designed to accommodate certain accident situations in which relatively large quantities of fuel and fission products are released through fuel failure, the additional shielding requirements for the vented-to-sodium concept are not considered severe.

3.2.3. Fission Gas Control

Concern had been expressed as to how the release of fission gas products from vented fuel pins would affect the design requirements of the fission gas control system. Consequently a study was undertaken, and the results (listed in Table A-2) indicate that vented fuel elements do substantially increase the cover gas activity.⁸ However, this difference is due primarily to isotopes with relatively short half-lives, so that this additional activity will not affect the feasibility of one design as compared with the other in regard to total release rate. As a result of restrictions on releasing activity to the atmosphere, the cover gas purge rate must be reduced, and the decay tank holdup time must be increased for the vented-to-sodium design (see Table A-3). However, this does not increase the shielding requirements, as is

discussed in section 3.2.4. This increased holdup time will be reflected in the design requirements and hence in the cost of the decay tanks.

Activity in the containment atmosphere will be governed by leakage of the cover gas through the penetration seals and, if purging and stacking of waste gas are considered, by the allowable concentration external to the reactor building. The relatively large increase in cover gas activity level associated with vented pins will tend to restrict entrance into this region if seal leakage occurs. Sufficiently high leakage rates would dictate the incorporation of a hot containment building design adding substantially to the total capital cost.

3.2.4. Shielding

Stable and long-lived isotopes of xenon and krypton are of primary concern in the release of fission gas.³ It has been found that at steady state operation the 2.76 Mev ^{24}Na gamma dictates the shielding requirements for the coolant during normal operation. However, after long periods of operation the ^{134}Cs (indirect daughter of ^{133}Xe) buildup in the sodium coolant could pose a problem in a vented system. Although the ^{134}Cs gammas are soft (0.6 and 0.8 Mev), this isotope has a tendency to plate out on piping and containment walls, and such concentration could conceivably result in large specific activities. The problem is further complicated by the inability of sodium purification systems to separate this isotope from the coolant because of the chemical similarity to sodium.

It should be noted from Table A-4 that the decay tank shielding requirements for leaking or defective fuel are approximately 1.5 times those required for the vented fuel even though the fission gas activity is less by a factor of 50 and the volume is less by a factor of 100. The reason for this is due to the short-lived ^{87}Kr (76m half-life) and ^{88}Kr (2.8h half-life) isotopes which emit 1.5 Mev and 2.57 Mev gammas respectively. Both isotopes are present in the vent-to-coolant and defective fuel fission gas; however, because of the 5-day hold-up time for the vent-to-coolant fission gas, the ^{87}Kr and ^{88}Kr activity is negligible. In fact, for the vented fuel to dictate the requirements, only 0.016% of the fuel pins could be defective. This appears to be an overly optimistic estimate of pin failures.

3.2.5. On-Site Versus Off-Site Reprocessing

The fuel element containment must be at least as stringent for the nonvented as for the vented design, since clad failure must be postulated during shipping, and detection of these failures appears difficult if at all possible. The consequences of clad failure appear much more severe for nonvented pins because of the large volumes of high pressure gas involved. Unless fission gas is released from non-vented pins before shipment, shipping cask design requirements become much more severe.

The diving bell concept presents a shipping problem in that the fuel pin must remain vertical to prevent the ingress of sodium or other liquid transfer coolant. Apparently this design would require that fuel elements be disassembled on site and that the vents be plugged before shipment, although such a problem would not exist with gas coolants. A similar problem exists for venting arrangements utilizing a manifold if bundles need to be disassembled before shipment. On-site reprocessing would tend to nullify many of these disadvantages making these concepts more attractive in comparison.

3.2.6. Instrumentation

There are two main methods of detecting cladding failure in unvented fuel: (1) monitoring the cover gas for noble gas fission products, and (2) monitoring the coolant downstream of the core for delayed neutrons from halogen fission products. The latter method is presently preferred by B&W, and since venting releases only gaseous fission products, it will not affect the applicability of the delayed neutron method.¹ The expected 5-day holdup time will be sufficient to effectively eliminate the principal sources of monitored delayed neutrons (^{137}I , 22-second half-life; ^{87}Br , 56-second half-life) prior to venting. Other delayed neutron precursors have even shorter half-lives and also may not escape the porous filter. Similarly, vented fuel may also be compatible with the cover gas monitoring method. A holdup time of 5 days will effectively eliminate discharge through the vent of short-lived ^{87}Kr and ^{88}Kr , which are the major noble gas source of high-energy gammas. Thus an energy-selective cover gas gamma monitor can detect cladding failure in vented fuel. On this basis the use of vented fuel is compatible with failed fuel

detection, although it is expected that the ability to detect relatively minor failures may be impaired.

3.2.7. Safety

It appears that the relatively high steel fractions associated with non-vented designs will dictate an increased fuel enrichment. Higher fuel enrichments will tend to decrease the sodium void coefficient, thereby indicating a safer core. However, the increased coolant fraction in a low-leakage core like the B&W reference design would have the opposite effect and would increase the void coefficient. Furthermore, the increased fissile enrichment would make the Doppler coefficient more positive, which, in the B&W reference design, has more effect on the consequences of accidents than does the sodium void coefficient.

In one sense, the probability of local voiding, which is potentially more serious than any overall core effects, is reduced. Should a non-vented pin fail, the high-pressure stream of fission gases could cause vapor blanketing and subsequent failure of adjacent fuel pins. The high internal pressure could, under accident conditions, force a stream of molten fuel into the sodium with disastrous results.

Taking all things into account, one might expect that the overall safety characteristics of a vented design might well be more favorable than those of a non-vented design.

Figure 3-1. Bottom Venting Using Mechanical Connections

Figure 3-2.
Diving Bell Connectors

Figure 3-3.
Porous Plug Connectors

BOTTOM VENTING

Figure 3-5.
Diving Bell

VENTING TO SODIUM

Figure 3-7. Venting to Gas Blanket

4. RESULTS

This section contains the results of a parameter study on the preliminary reference core (Table 4-1 lists basic data for the preliminary reference design). The effects of various venting schemes on power generation costs and nuclear performance are determined and compared. Since the effects of these schemes are reflected throughout the reactor design, an attempt has been made to allow optimization of each venting arrangement by varying the fuel pin clad thickness and outside fuel diameter or outside clad diameter.

Key results are listed in Table 4-2 showing detailed economic comparisons of various venting arrangements and core designs. The designs selected for the study have been chosen on the bases of economic and other practical considerations.

4.1. Calculational Methods

Calculations have been performed using the NAPS (Sodium Parameter Study) computer code.⁸ Given the basic reactor plant parameters, the code is designed to calculate thermal, hydraulic, mechanical, steam plant, and nuclear performance and, in addition, to estimate the cost of both fuel and major components. Numerous uncertainties exist in any such analysis, so that it has been found impractical to incorporate the following areas of reactor design:

1. Fabrication and materials costs for remote tanks, connectors, or manifolds associated with various venting arrangements.
2. Economic effects of on-site and off-site reprocessing. Off-site reprocessing has been assumed in all cases, and no cost penalty has been incorporated for disassembling or capping fuel pins.
3. Problems associated with instrumentation have been discussed, but no economic distinctions have been made between different venting

designs. It is not thought that the difference in the instrumentation costs will be significant in terms of the overall reactor costs, and in this regard such an assumption appears justified.

4. No economic distinctions have been made for variations in containment, decay tanks, primary clean-up systems, etc.

5. The effects of venting on fission product cross section are neglected.⁶

4.2. Basic Design Parameters

Calculations have been made on the basis of a preliminary reference concept. The basic design and operating characteristics of this concept are shown in Table 4-1. During this study the effects on the reference core of perturbing fuel pin design, fission gas release rate, and clad thickness have been determined. The overall economic effects of these parameters are shown in Figures 4-1 through 4-3.

Available data on fission gas release rates are quite scattered and have ranged from 40 to 95%.⁵ Therefore, this parameter was used as a key variable. A 100% release rate was assumed for clad design requirements for vibratory compacted fuel, but the use of pellets may reduce this release rate to around 50%. Improved data may allow a less stringent design criterion. However, it should be noted that the results cannot be used for a direct comparison of vibratory compacted versus pellet fuel. Allowable fuel burnup, a major parameter in determining fuel costs, will be different for the two types of fuel. Consequently, the assumptions made for the purposes of this trade-off study would not make such a comparison meaningful.

A second factor, clad thickness, in the power generation costs comparison introduces an entirely new scope of problems in fuel clad design. The development of an adequate cladding model is beyond the scope of this study, so that the following assumptions have been made:

1. A fuel burnup rate of 100,000 MWD/tonne is achievable with 0.010-inch clad in a vented design and 0.015 in a non-vented design.
2. Erosion and corrosion will not cause a reduction in clad thickness of greater than 0.0015 inch.

3. The fuel cladding design limit for the containment of fission gas will be 1% creep at 30,000 hours of operation. Material properties are considered to be temperature-dependent.
4. Fuel swelling will not limit fuel burnup to below 100,000 MWD/tonne for any specified cladding thickness.

These assumptions appear to be reasonable at this time, although further research is needed to determine their validity. Based on the assumed erosion and corrosion allowance and the expected manufacturing problem, a clad thickness of as little as 0.010 inch appears to be practical with vented systems. Since most fission gas products are quickly released, the major potential stress on the clad will be due to fuel swelling. However, should fuel swelling prove to be a major problem it will limit both vented and non-vented designs at comparable levels. Additional cladding thickness will not significantly aid in accommodating the resulting strain. If fuel burnup is limited at some level lower than 100,000 MWD/tonne, then the economic advantages of the vented fuel elements will be reduced. Similarly, if higher burnups are ultimately achieved, then the vented designs will offer additional economic incentives.

Referring to Figure 4-1, the economic effects of varying the clad thickness while maintaining a constant pin size and pitch may be seen. Estimates of the overall power generation costs for various fuel pin designs are indicated on the curves. The importance of maintaining a minimum clad thickness becomes apparent. The core breeding ratio, and hence the plutonium credit, is dependent upon this variable, producing the relatively large variations upon power generation costs. The porous plug concept obviously offers a large economic advantage over the non-vented design, primarily due to its ability to operate with thinner cladding.

The diving bell design appears to be quite competitive, but in this case judgment must be used in the interpretation of results. If off-site reprocessing is considered to be desirable, some additional problems (costs) must be associated with its use. Because a diving bell will operate properly only in the vertical position, provisions must be made for safe shipment. Obviously these limitations do not apply to the combination diving bell-porous plug concept or shipment in gas coolants.

The non-vented concept is, in this case, judged to be uneconomical because of the thicker cladding and the long gas void length required to contain fission gas products. (Fuel pin cladding thickness and void length are determined from basic stress considerations.) In judging non-vented designs it should be noted that present costing methods within the code do not allow detailed analysis of power generation costs for cores with gas void lengths in excess of about 30 inches. The curves used for estimating the fabrication costs were not designed to operate for extremely long gas voids. In addition, other factors that will affect the capital cost have not been included, i.e., larger containment buildings, increased handling problems, etc. For this reason, a dashed line is shown in this portion of the curves indicating estimates of costing trends.

An examination of the detailed cost analysis indicates that a significant cost penalty is associated with the non-vented concept owing to the lower fuel mass used in the analysis (clad was thickened by displacing fuel). In addition to lowering the breeding ratio, the lower fuel mass tends to increase fuel costs by decreasing the core cycle time and hence the total power production. Spreading the fixed costs over less time increases unit fuel costs. Simultaneously, the lower fuel mass reduces the inventory costs so that the overall trends are indeterminate without a parallel analysis to determine the effects of fuel inventory. For this reason, a parametric study has been included in which both fuel pin outside diameter and cladding thickness have been varied so that the fuel inventory remains constant. The results, shown in Figure 4-2, indicate the same trends as previously discussed in Figure 4-1.

It is apparent from these graphs that the vented arrangements offer significant cost incentives because of the thinner cladding and the shorter total core length. The variations in clad thickness appear to be the predominant factor, producing wide variations in doubling time and plutonium credit. A detailed economic comparison of the porous plug, the diving bell, and the non-vented concepts is shown in Table 4-2.

4.3. Economic Comparison

Based on the economic trends shown in the parametric study and on other practical considerations, five perturbations of the preliminary reference design have been selected for detailed comparison in Table

4-2. The preliminary cost analysis for the reference core is listed in the first column. In comparison, the effects of altering the design to incorporate a diving bell seal are shown in the adjacent column where the height of the seal is selected as 20 inches based on the expected cool-down requirements. The longer fuel bundle length is reflected in the larger, and consequently more expensive, core vessel, biological shielding, and sodium inventory. In addition, a larger pressure drop across the core will require larger pumps and thicker cans. The thicker cans result in larger core steel fractions and, subsequently, lower breeding ratios. Consequently, the plutonium credit will be reduced and fuel costs will be increased.

Non-vented fuel elements shown in the third column have both of the above-mentioned disadvantages. The longer fuel elements will be accompanied by a higher core pressure drop, and in addition, the thicker cladding required by this design will further increase the core steel fractions.

It should be noted that core design plays an important part in determining the economic differential between vented and non-vented pins. Consideration has been given to including BeO in the reference design as shown in Figure 4-3. The addition of beryllium oxide pins will result in more coolant flow channels and consequently in higher fuel clad temperatures at constant reactor coolant flow rates. The material properties of all applicable alloys tend to degrade rapidly with increasing temperature in the range of interest. As a result, excessive gas void lengths and cladding thickness will be required for the non-vented design. As an alternative, coolant mixed mean outlet temperatures could be lowered, but this solution appears unattractive economically because of its overall effect on cycle efficiency. In addition, larger and more expensive intermediate heat exchangers, reheaters, and steam generators will be required to maintain proper steam temperatures.

A detailed economic comparison of the porous plug, the diving bell, and the non-vented designs for a core containing BeO is also listed in columns 4, 5, and 6, respectively. The same basic trends are shown as for the core without BeO, but economically, the non-vented concept appears even less attractive. Increased clad temperatures have degraded the clad properties sufficiently to require 0.020-inch cladding thickness.

Probably the most significant factor affecting the economic comparison is the doubling time reported for the various concepts. The thicker cladding required for a non-vented pin has produced a major increase in doubling time which is reflected in increased fuel costs through the plutonium credit. The thicker cladding is again reflected in increased enrichments, resulting in higher carrying charges for inpile plutonium.

4.4. Effects of Economic Uncertainties

It has been shown that a significant economic advantage is offered by vented fuel pins. As previously mentioned, however, certain unknowns do exist in this analysis and these factors should be accounted for. Fuel handling may offer an additional economic advantage to the vented arrangement. Problems associated with a fuel pin failure during handling would tend to dictate additional design problems for non-vented pins. The additional length of all handling components, which non-vented pins would dictate, will also tend to increase costs. As stated previously, decay tank shielding requirements are dictated primarily by fuel element failure so that if fewer failures do indeed occur in vented systems, an economic advantage may be derived.

Decontamination of components is a major concern in performing maintenance on the primary system. Present data indicate that contamination on the surface of primary system components due to both activated sodium and fission gas product plate-out may be adequately removed by steam cleaning. However, the removal of contaminants released from failed fuel elements may require chemical cleaning. If such is proved to be the case, and venting does reduce fuel element failures as expected, then an additional incentive is provided for venting.

Conversely, the fission gas control system tends to favor the non-vented arrangement, since venting may dictate a different design approach. Vented elements produce a sustained higher activity level in the upper containment area; this will tend to increase fission gas control requirements and to restrict access to this region. Should access be severely restricted, it may become necessary to include a hot containment area within the design, requiring a rather large capital investment.

Because of the high cover gas activity associated with fission gas venting this design may require monitoring of the primary coolant for delayed neutrons. It is presently thought that this method will be used in the reference core regardless of the fuel pin design. However, if monitoring of the gamma activity should appear attractive, it is expected that any resulting cost differential will be insignificant when prorated over the life of the plant. In addition, the sensitivity of this device may be reduced, making detection of defective fuel elements more difficult.

Considering the overall effects of these factors, it does not appear that the resulting economic trends will basically change the results reported in the detailed analysis. In the light of the economic, safety, and nuclear advantages indicated, there are significant incentives for developing and utilizing vented fuel elements.

Table 4-1. Reference Core Parameters

Fuel Bundle Design

Pin pitch	0.34
Pin OD, in.	0.28
Clad thickness, in.	0.010
Active fuel length, in.	34.7
Axial blanket length, in.	15.9
Gas void length, in.	Vented (2 inches)
Bundle pitch, in.	6.585
Can thickness, in.	0.129
Pins per bundle	331
Beryllium fraction	0.0
Total bundles	288
Total core pins	95,328
Blanket pin OD, in.	0.464
Total blanket pins	19,050
Average fuel burnup, MWD/tonne	100,000

Thermal and Hydraulic Design

Reactor thermal output, MW	2448
Maximum linear heat rate, kw/ft	17.0
Maximum core heat flux, Btu/hr-ft ²	713,000
Hot channel flow rate, lb/hr	1021
Total flow rate, lb/hr	92,373,000
Core pressure drop, psi	28.0

Steam Plant Design

Operating pressure, psi	2400
Turbine inlet temperature, F	950
Reheat temperature, F	950

Table 4-2. Economic Comparison of Vented and Non-Vented Concepts

	Without BeO			With 10% BeO		
	Porous Plug ^(a)	Diving Bell ^(a)	Nonvented ^(a)	Porous Plug ^(b)	Diving Bell ^(b)	Nonvented ^(b)
<u>Fuel Bundle Design</u>						
Pin OD, in.	0.28	0.28	0.28	0.28	0.28	0.30
Pin pitch, in.	0.34	0.34	0.34	0.34	0.34	0.34
Clad thickness, in.	0.01	0.01	0.015	0.01	0.01	0.020
Equivalent void length, in.	2.0	20.0	40.0	2.0	20.0	40.0
Maximum clad surface temperature, F	1204.0	1204.0	1204.0	1,243	1,243	1,238
Primary pumping power, MW	7.68	8.73	9.88	7.14	8.05	14.72
<u>Nuclear Performance</u>						
Internal breeding ratio	0.97	0.96	0.89	0.92	0.91	0.83
Total breeding ratio	1.33	1.32	1.24	1.25	1.24	1.13
Doubling time, years	14.6	15.3	20.8	20.7	21.8	49.9
Enrichment, %	12.8	12.9	14.0	13.7	13.8	15.4
Initial mass of metal in core, kg	23,800	23,800	22,100	23,900	23,900	23,800
Electrical output, MW	1,038	1,037	1,036	1,039	1,038	1,031
<u>Fuel Costs (mills/kwhr)</u>						
Fabrication	0.3725	0.3726	0.3860	0.3996	0.3996	0.4006
Reprocessing	0.1710	0.1714	0.1753	0.1732	0.1739	0.1822
Shipping	0.0423	0.0533	0.0665	0.0426	0.05379	0.06608
Carrying charges						
Plutonium in pile	0.4243	0.4268	0.4149	0.4397	0.4425	0.4736
Plutonium out of pile	0.2052	0.2063	0.2120	0.2121	0.2134	0.2292
Fabrication	0.0657	0.0657	0.0637	0.0705	0.0705	0.0707
Reprocessing	-0.0460	-0.0461	-0.0445	-0.0466	-0.0468	-0.0488
Plutonium credit	-0.2950	-0.2836	-0.2071	-0.2164	-0.2067	-0.0973
Total	0.9399	0.9665	1.0669	1.0746	1.1002	1.2764

(a) Perturbations based on variations in clad thickness and fuel OD.

(b) Perturbations based on variations in clad thickness and cladding OD, with 10% BeO addition.

Table 4-2 (Cont'd)

	Without BeO			With 10% BeO		
	Porous Plug ^(a)	Diving Bell ^(a)	Nonvented ^(a)	Porous Plug ^(b)	Diving Bell ^(b)	Nonvented ^(b)
<u>Capital Costs (\$)</u>						
Core vessel	496,000	512,000	529,000	495,000	510,000	537,000
Reactor vessel	1,876,000	1,918,000	1,965,000	1,873,000	1,916,000	1,981,000
Internals	1,411,000	1,417,000	1,424,000	1,427,000	1,434,000	1,458,000
Biological shielding	4,286,000	4,308,000	4,330,000	4,274,000	4,294,000	4,410,000
Sodium inventory	729,000	739,000	751,000	728,000	738,000	755,000
Reheater	4,058,000	4,058,000	4,058,000	4,058,000	4,058,000	4,058,000
Steam generators	7,883,000	7,883,000	7,886,000	7,883,000	7,883,000	7,884,000
Intermediate heat exchangers	3,584,000	3,585,000	3,585,000	3,583,000	3,584,000	3,590,000
Primary & secondary pumps	2,214,000	2,371,000	2,543,000	2,133,000	2,269,000	3,270,000
Other direct cost	29,138,000	29,138,000	29,138,000	29,138,000	29,138,000	29,138,000
Total nuclear island direct costs	55,674,000	55,929,000	56,209,000	55,592,000	55,824,000	57,079,000
Turbine island direct costs	43,342,000	43,299,000	43,251,000	43,365,000	43,327,000	43,049,000
Total	98,998,000	99,228,000	99,460,000	98,957,000	99,151,000	100,127,000
Land and land rights	660,000	660,000	660,000	660,000	660,000	660,000
Indirect costs	39,599,000	39,691,000	39,784,000	39,583,000	39,660,000	40,508,000
Total capital costs	139,257,000	139,579,000	139,904,000	139,200,000	139,471,000	140,837,000
<u>Power Generation Costs (mills/kwhr)</u>						
Fixed charges	2.232	2.239	2.248	2.229	2.236	2.272
Fuel costs	0.940	0.966	1.067	1.075	1.100	1.276
Operating, maintenance, & insurance	0.265	0.266	0.266	0.265	0.266	0.267
Total power generation	3.437	3.471	3.581	3.569	3.601	3.815

(a) Perturbations based upon variations in clad thickness and fuel OD.

(b) Perturbations based upon variations in clad thickness and cladding OD, with 10% BeO addition.

Figure 4-1. Effects of Venting on Unit Power Generation Costs Showing Perturbations of Fuel OD and Clad Thickness

Figure 4-2. Effects of Venting on Unit Power Generation Costs Showing Perturbations of Cladding OD and Clad Thickness

Figure 4-3. Effects of Venting on Unit Power Generation Costs Showing Perturbations of Cladding OD and Clad Thickness With BeO Addition

REFERENCES

- ¹ O'Neill, G. L., et al., Safety and Economic Aspects of Vented Fuel in Sodium Cooled Reactors, Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, ANL-7120.
- ² Smidt, D. and Sommer, W., Application of the Vented Fuel Concept to a Sodium Cooled Power Breeder With 1000 MWe, Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, ANL-7120.
- ³ Kunkel, W. and Berger, S., Fission Product Retention in Sodium and Applications to Vented Fuel Design, ANL-7120.
- ⁴ Leipanskii, O. D., et al., Operational Experience With the BR-5 Reactor, AEC-TR-5890.
- ⁵ Design and Development of Fuel and Fuel Elements for Prototype Fast Reactors, London Conference on Fast Breeder Reactors, May 1966.
- ⁶ The Effects of Vented Fuel on Fission Product Cross Section, GEAP-4418.
- ⁷ A Technical and Economic Evaluation of Vented Fuel for Sodium Cooled Fast Ceramic Reactors, GEAP-4770.
- ⁸ Crane, R. A., et al., Sodium Parameter Study Code NAPS - A Computer Program for Parametric Studies of Sodium-Cooled Fast Breeder Reactor Plants, Babcock & Wilcox, BAW-1326, Lynchburg, Virginia, April 1968.

APPENDIX
Gaseous Fission Product Release and Deposition

1. Introduction

This appendix describes a parametric study of the gaseous fission products generated and deposited in the current reference LMFBR design, considering vented-to-coolant and defective fuel. Prime emphasis has been placed on determining and comparing the safety and shielding characteristics of vented and non-vented fuel.

There were, unavoidably, many assumptions in this study, because, for the most part, key parameters require extensive research and development substantiation, which at this point is almost nonexistent. Many required assumptions have been taken from reference 1, in which useful experimental data are available. Varying key parameters, such as gas holdup time, could alter the results considerably. However, it is felt that the assumed values for key parameters represent conservative bounds for the results. The criteria and primary assumptions employed in the study are presented in the following sections.

2. Criteria

It is assumed that defective fuel must be considered as a design basis, whether considering vented fuel or not. It is believed that there is an increase in probability of fuel pin failure in non-vented fuel because of increased thermal stresses and increased internal gas pressure. Since this effect is largely unexplored at this point, the percentage of failed fuel is assumed to be the same for both vented and non-vented fuel.

The calculation compares vented with non-vented fuel, considering the worst possible case as pertaining to total cover gas activity and the resulting repercussions, since in both cases the fuel is assumed defective, and also considering that operation continues for a significant length of time after failure. In this light, this study differs from a similar study in reference 1, since in that study vented fuel is evaluated without consideration of defective fuel. In this instance, the hazards of fission gas generated by venting are increased by the hazards of the energetic, shorter-lived isotopes released by fuel failure. For comparison purposes, cases were also run considering vented fuel only.

A continuous cover gas purge system is considered in the study. The periodic purge of the cover gas system has not been considered. Also, a continuous containment purge system only is considered.

In the vented case, the fuel is assumed to be vented directly to the coolant. Iodine and bromine isotopes are assumed to combine chemically and to remain entrained in the primary coolant. All solid fission products are assumed to remain in the fuel in the vented case and to either become entrained in the coolant or plate out on components in the case of defective fuel.

Isotopes of xenon and krypton were considered as gaseous fission products.

3. Release to Coolant

3.1. Vented Fuel

Half (50%) of the generated fission gas is assumed released to the coolant. The diffusion time, or holdup time, is assumed to be five days. This automatically limits attention to the long-lived fission gas isotopes, i.e., ^{135}Xe , $^{133}\text{Xe}^m$, ^{133}Xe , $^{131}\text{Xe}^m$, ^{85}Kr , in the vented-to-coolant case.

These assumptions are essentially the same as those listed in reference 1 where pellet fuel is considered. The use of vibro-compacted fuel is not expected to alter the vented release fraction or holdup time appreciably.

3.2. Defective Fuel

The assumed percentage of defective fuel corresponds to a failed fuel percentage of 0.35% of the total pins. 100% of the generated fission gas in the failed pins is assumed released to the coolant at a holdup time of 10 minutes. In the absence of data on release fractions for vibro-compacted fuel, the 100% release figure was chosen. The number of leaking fuel pins assumed is believed to be conservative. The resultant release fractions and leak rates for the fission gas isotopes, considering vented, defective, and vented plus defective fuel are given in Table A-1.

4. Release to Cover Gas

We assume that 75% of the entrained fission gas is released from the coolant to the cover gas system with each coolant circulation in both the vented and defective fuel cases. This figure is taken from reference 1 as there is expected to be no difference in fission gas bubble size and coolant velocity in the present case as compared to the case presented in reference 1. These are the two most important factors in determining fission gas separation from the primary coolant.

5. Time-Dependent Model

The calculation of the gaseous fission product inventory in the core, coolant, and cover gas was made using BURP1.² The model chosen was meant to describe conditions during one 300-day fuel cycle, considering vented, defective, and vented plus defective fuel. The three cases can be described by the following tabulation:

<u>Full power operating time, days</u>	<u>Case 1 - vent only</u>	<u>Case 2 - leak only</u>	<u>Case 3 - vent + leak</u>
0-100	Vent	--	Vent
100-200	Vent	--	Vent
200-299	Vent	Leak	Vent + leak
300	Shutdown	Shutdown	Shutdown

The defective fuel is assumed to fail late in the fuel cycle and to remain in the core until shutdown for refueling; i.e., it is assumed that operation continues even if failed fuel detection has occurred. In the vented case the fuel is assumed vented throughout the entire fuel cycle. All of the results presented correspond to the end of the fuel cycle, at which time all the isotopes reach saturation or equilibrium except ⁸⁵Kr. For each of the three cases above, three cover gas purge rates were considered: 10, 1, and 0.01 cfm.

Table A-2 shows the equilibrium cover gas activity of each isotope for each case and each cover gas purge rate. Figure A-1 shows total saturation activity for each case as a function of cover gas purge rate. These values compare quite well with those presented in reference 1.

6. Decay Tanks

Calculations were performed to determine the relative advantages and disadvantages of vented and non-vented fuel with regard to decay tank activity levels and required holdup times.

The decay tank holdup time is defined as the time since removal from the cover gas system for the ^{133}Xe activity to be reduced by decay to the activity level of ^{85}Kr . This definition is made because ^{133}Xe is by far the predominant activity source of all the gaseous fission product isotopes a few days after removal from the cover gas system, for both vented and non-vented fuel. The activity level of ^{85}Kr is taken as the minimum possible in the decay tank because of its 10.76-year half-life. It is pointless to hold up the decay tank gas any longer because almost nothing is gained in activity reduction of the ^{85}Kr . The required holdup times for the three cases and the three cover gas purge rates are tabulated below:

Cover gas purge rate, cfm	Holdup time, days		
	Leak only	Vent only	Vent + leak
10.0	280	375	380
1.0	190	315	318
0.01	78	168	170

Since there is a practical maximum activity release rate to the atmosphere from the decay tanks, and since the rate of release of activity is roughly equivalent to the release rate of ^{85}Kr after holdup, the rate of purge of the cover gas into the decay tanks will be limited or require longer holdup capacity. An activity release of 5000 microcuries per second to the atmosphere caused by the ^{85}Kr represents a practical maximum for the concept of controlled release. Figure A-2 shows the activity release rate as a function of the cover gas purge rate. As can be seen from the figure, the allowable purge rate in the vented case is only 0.1 cfm, while for failed fuel the allowable purge rate is 0.6 cfm.

7. Decay Tank Shielding Requirements

In the current reference design the cover gas purge system contains three 3000-ft³ decay tanks, each tank having a radius of 8 feet and

a height of 15 feet. The tank size could change drastically, depending on the required holdup time, cover gas purge rate, and decay tank pressure; however, these dimensions were used for the shielding requirement comparison. The three cases were considered, and three cover gas purge rates were considered. The results appear below.

Cover gas purge rate, cfm	Ordinary concrete required (in.) for a dose rate of 1 mRem/hr		
	Leak only	Vent only	Vent + leak
10.0	64	43	64
1.0	56	39	56
0.01	40	28	40

Decay tank shielding requirements considering failed fuel are greater because of the energetic gammas emitted by ^{87}Kr and ^{88}Kr , which have short half-lives (1.3 and 2.8 hours) and are emitted in significant quantities only by the failed fuel, since in the vented case the holdup time before release is about five days.

8. Leakage of Cover Gas

Because of the many perforations in the reactor top cover plug, and their associated seals, the problem of cover gas leakage through faulty seals into the containment atmosphere was investigated.

The leaking cover gas was assumed to be transferred immediately to the reactor building atmosphere and to become evenly dispersed throughout that atmosphere. Calculations were performed with and without a containment purge system. The containment building purge system is assumed to remove fission products completely (no recirculation).

The restrictions on the allowable activity concentrations for ingestion of the xenon and krypton isotopes were taken from 10 CFR 20 and are shown in Table A-3. The values given are for personnel working in an unrestricted area during a 40-hour work week and for unlimited exposure considering the general public.

The leakage of the cover gas into the containment atmosphere will be governed by the allowable concentration in the reactor building, and

if purging is to be considered, by the allowable concentration external to the reactor building. Figures A-3 through A-14 show the fraction (actually multiple) of the MPC* of the cover gas in the reactor building atmosphere as a function of the cover gas purge rate, for the three cases with various cover gas and containment purge rates. The values of the leak rate that result in an MPC = 1, then, represent the condition for unlimited access to the reactor building considering ingestion alone. This is the limiting criterion, since the allowable direct dose criterion is much less stringent. The broken lines on the figures in which containment purge is considered represent the restrictions imposed on activity released to the general public. The dispersion factor χ/Q , which is a measure of the reduction of activity by atmospheric conditions, stack height, exclusion radius, etc., was taken to be 10^{-6} sec/M³. This is a representative value used in the design of current commercial plants. As can be seen from the figures, the allowable leak rates are extremely small even with quite sizable containment purge rates.

For illustrative purposes, a calculation was performed to determine the severity of conditions resulting from a leak of cover gas to the containment atmosphere which approximately represents normal conditions. The assumed conditions, which are believed to be representative, are as follows:

Personnel protection	None
Cover gas purge rate, cfm	1.0
Cover gas leak rate, cc/day	25
Containment purge rate, cfm	34,000

Table A-4 shows the resultant γ and β direct dose rates in the upper containment atmosphere. BONG2 was used for the γ -dose calculation. The γ dose is the maximum that would be experienced inside the containment building. An infinite cloud model was selected for the β -dose calculation since a volume as large as the containment volume under consideration is essentially infinite for β energies up to 3 Mev.³ The β dose is essentially the same throughout the reactor building.

*MPC: maximum permissible concentration.

Table A-5 shows the reactor building accessibility considering no protection of plant personnel, subject to the imposed conditions.

9. Summary of Hazard Potential Considerations Due to Fission Gas Release in the Current Design

Cover gas is permitted to leak into the containment atmosphere only in minute quantities even under the best conditions. Personnel working in areas of the reactor building where the MPC is greater than unity must be protected by special clothing, breathing apparatus, shielding, or some combination thereof. A further restriction on the allowable leak rate is the specified allowable release of radioactive contamination to the general public.

10. Summary of Safety and Shielding Comparisons of Vented and Non-Vented Fuel

10.1. Shielding

The energetic gamma emitters ^{87}Kr and ^{88}Kr , which are released in significant quantities only by defective fuel because of the short half-lives of these isotopes, dictate the decay tank shielding requirements. The shields in the top cover structure are sized by core and primary coolant radioactive sources and not by the cover gas. Therefore, the shielding requirements will be the same for vented and non-vented fuel, i.e., assuming that defective fuel must be postulated as a design basis in both the vented and non-vented cases.

10.2. Safety

The leakage requirements of vented fuel are more stringent than non-vented requirements, as can be seen from Figures A-2 through A-14. However, the requirements are so stringent in both cases that this is not a serious disadvantage of vented fuel. In other words, if the seal leakage requirements for the case of defective fuel can be met, then the extra requirements of the vented fuel would be essentially solved also. However, defective fuel is not a normal operating condition; therefore, under normal operating conditions, the seal leakage requirements would be much less severe. In the case of vented fuel, the seal integrity must be maintained throughout the operating history of the plant.

Table A-1. Fission Gas Release Fractions and Leak Rates

Isotope	T _{1/2}	Vented	Leak rate, sec ⁻¹	0.35% defective	Leak rate, sec ⁻¹	Total	Leak rate, sec ⁻¹
⁸⁹ Kr	3.2 m	--	0	0.00040	0.145-5	0.00040	0.145-5
¹³⁷ Xe	3.8 m	--	0	0.00056	0.17-5	0.00056	0.17-5
¹³⁵ Xe ^m	15 m	--	0	0.00224	0.17-5	0.00224	0.17-5
¹³⁸ Xe	17 m	--	0	0.00234	0.16-5	0.00234	0.16-5
⁸⁷ Kr	1.3 h	--	0	0.00322	0.47-6	0.00322	0.47-6
⁸³ Kr ^m	1.86 h	--	0	0.00329	0.33-6	0.00329	0.33-6
⁸⁸ Kr	2.8 h	--	0	0.00336	0.23-6	0.00336	0.23-6
⁸⁵ Kr ^m	4.4 h	--	0	0.00343	0.15-6	0.00343	0.15-6
¹³⁵ Xe	9.2 h	0.000061	0.13-8	0.00350	0.73-7	0.003561	0.75-7
¹³³ Xe ^m	2.3 d	0.110	0.43-6	0.00350	0.12-7	0.1135	0.45-6
¹³³ Xe	5.27 d	0.260	0.53-6	0.00350	0.53-8	0.2635	0.54-6
¹³¹ Xe ^m	12 d	0.3750	0.40-6	0.00350	0.23-8	0.3785	0.41-6
⁸⁵ Kr	10.4 y	0.50	0.21-8	0.00350	0.73-11	0.5035	0.21-8

Table A-2. Equilibrium Cover Gas Activities

Cover gas purge rate, ft ³ /min:	Vent + leak, curies $\times 10^{-3}$				Leak only, curies $\times 10^{-3}$				Vent only, curies $\times 10^{-3}$			
	10.0	1.0	0.01	0.0	10.0	1.0	0.01	0.0	10.0	1.0	0.01	0.0
Isotope:												
⁸³ Kr ^m	9.8	10.6	10.7	10.7	9.8	10.6	10.7	10.7	--	--	--	--
⁸⁵ Kr ^m	39.8	46.5	47.3	47.4	39.8	46.5	47.3	47.4	--	--	--	--
⁸⁵ Kr	--	0.4	4.3	4.6	--	--	0.1	0.2	--	0.3	4.1	4.4
⁸⁷ Kr	70.0	73.6	74.0	74.0	70.0	73.6	74.0	74.0	--	--	--	--
⁸⁸ Kr	108.0	119.0	121.0	121.0	108.0	119.0	121.0	121.0	--	--	--	--
⁸⁹ Kr	9.6	9.6	9.6	9.6	9.6	9.6	9.6	9.6	--	--	--	--
¹³¹ Xe ^m	16.3	97.5	216.0	219.0	0.1	0.8	1.9	1.9	16.0	96.0	213.0	216.0
¹³³ Xe ^m	89.4	245.0	302.0	303.0	2.6	7.3	9.0	9.1	86.0	235.0	290.0	291.0
¹³³ Xe	4,470.0	18,800.0	29,000.0	29,200.0	59.7	253.0	390.0	393.0	4,410.0	18,600.0	28,600.0	28,800.0
¹³⁵ Xe ^m	66.2	66.8	66.9	66.9	66.2	66.8	66.9	66.9	--	--	--	--
¹³⁵ Xe	356.0	479.0	498.0	498.0	348.0	468.0	487.0	487.0	5.2	7.0	7.2	7.7
¹³⁷ Xe	33.9	34.0	34.0	34.0	33.9	34.0	34.0	34.0	--	--	--	--
¹³⁸ Xe	258.0	261.0	261.0	261.0	258.0	261.0	261.0	261.0	--	--	--	--
⁴¹ Ar	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2
	5,530.0	20,000.0	30,700.0	30,800.0	1,010.0	1,350.0	1,510.0	1,520.0	4,520.0	18,900.0	29,100.0	29,300.0

Table A-3. LMFBR Gaseous Fission Product Study -
Maximum Permissible Activity
Concentrations in Air

Isotope	MPC*, $\mu\text{ci}/\text{cm}^3$	
	Plant personnel (40-hr work week)	General public
$^{83}\text{Kr}^{\text{m}}$	1×10^{-6}	3×10^{-8}
$^{85}\text{Kr}^{\text{m}}$	6×10^{-6}	1×10^{-7}
^{85}Kr	1×10^{-5}	3×10^{-7}
^{87}Kr	1×10^{-6}	2×10^{-8}
^{88}Kr	1×10^{-6}	2×10^{-8}
^{89}Kr	1×10^{-6}	3×10^{-8}
$^{131}\text{Xe}^{\text{m}}$	2×10^{-5}	4×10^{-7}
$^{133}\text{Xe}^{\text{m}}$	1×10^{-5}	3×10^{-7}
^{133}Xe	1×10^{-5}	3×10^{-7}
$^{135}\text{Xe}^{\text{m}}$	1×10^{-6}	3×10^{-8}
^{135}Xe	4×10^{-6}	1×10^{-7}
^{137}Xe	1×10^{-6}	3×10^{-8}
^{138}Xe	1×10^{-6}	3×10^{-8}
^{41}Ar	2×10^{-6}	4×10^{-8}

* For a mixture of isotopes: $\Sigma = \frac{C_i}{\text{MPC}_i} \leq 1$.

Table A-4. Direct Dose Rates From Inert Gases in the Reactor Building

Cover gas purge rate, cfm	1					
Cover gas leak rate, cc/day	25					
Containment purge rate, cfm	2.4×10^4					
	<u>Vent + leak, mrem/hr</u>		<u>Vent only, mrem/hr</u>		<u>Leak only, mrem/hr</u>	
Isotope	<u>γ</u>	<u>β</u>	<u>γ</u>	<u>β</u>	<u>γ</u>	<u>β</u>
$^{83}\text{Kr}^m$	0.23(-4)	--	--	--	0.23(-4)	--
$^{85}\text{Kr}^m$	0.11(-3)	0.95(-3)	--	--	0.11(-3)	0.95(-3)
^{85}Kr	0.0	0.0	0.0	0.01	0.0	0.0
^{87}Kr	0.47(-3)	0.48(-2)	--	--	0.47(-3)	0.48(-2)
^{88}Kr	0.80(-3)	0.28(-2)	--	--	0.63(-3)	0.28(-2)
^{89}Kr	--	0.12(-3)	--	--	--	0.12(-3)
$^{131}\text{Xe}^m$	0.24(-3)	--	3.93	--	0.25(-5)	--
$^{133}\text{Xe}^m$	0.63(-3)	--	9.82	--	0.19(-4)	--
^{133}Xe	0.10(-1)	0.18	166.9	2846	0.14(-3)	0.24(-2)
$^{135}\text{Xe}^m$	--	--	--	--	0.15(-3)	--
^{135}Xe	0.10(-4)	0.12(-3)	0.0	0.18(-5)	0.10(-4)	0.12(-3)
^{137}Xe	--	0.47(-3)	--	--	--	7.60
^{138}Xe	0.61(-3)	0.71(-2)	--	--	0.61(-3)	0.71(-2)
^{41}Ar	<u>0.18(-5)</u>	<u>0.92(-5)</u>	<u>0.18(-5)</u>	<u>0.92(-5)</u>	<u>0.18(-5)</u>	<u>0.92(-5)</u>
Total γ	0.13(-1)		0.11(-1)		0.22(-2)	
Total β		0.19		0.17		0.19(-1)
Total	0.203		0.186		0.021	

Table A-5. Reactor Building Access With No Protection
With Leaking Cover Gas

Cover gas purge rate, cfm	1		
Cover gas leak rate, cc/day	25		
Containment purge rate, cfm	2.4×10^4		
	Hourly dose, mrem		
	<u>Vent + leak</u>	<u>Vent only</u>	<u>Leak only</u>
From gases	0.20	0.19	0.02
From primary and secondary gammas and neutrons	1	1	1
From ingestion (assuming MPC = 100 mrem/week)	<u>2.53</u>	<u>2.13</u>	<u>0.40</u>
Total hourly dose	3.73	3.32	1.42
Maximum weekly entry time, hr	29.5	30.0	Unlimited
Dose limit for remainder of week	None	None	None

Figure A-1. Equilibrium Cover Gas Activity as a
Function of Cover Gas Purge Rate
(No Leaks)

Figure A-2. Decay Tank Release Rate of Activity After Holdup as a Function of Cover Gas Purge Rate

Figure A-3. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 0.0 cfm, Fuel Leak Only)

Figure A-4. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^4 cfm, Fuel Leak Only)

Figure A-5. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^5 cfm, Fuel Leak Only)

Figure A-6. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^6 cfm, Fuel Leak Only)

Figure A-7. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 0.0 cfm, Vent + Leak)

Figure A-8. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^4 cfm, Vent + Leak)

Figure A-9. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^5 cfm, Vent + Leak)

Figure A-10. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^6 cfm, Vent + Leak)

Figure A-11. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 0.0 cfm, Vent Only)

Figure A-12. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^4 cfm, Vent Only)

Figure A-13. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^5 cfm, Vent Only)

Figure A-14. LMFBR Gaseous Fission Product Study -
 Fraction of MPC of Equilibrium Cover Gas in
 Reactor Building as a Function of Cover Gas
 Leak Rate (Containment Atmosphere Purge
 Rate = 2.4×10^6 cfm, Vent Only)

REFERENCES

- ¹ O'Neill, G. L., et al., A Technical and Economic Evaluation of Vented Fuel for Sodium-Cooled Fast Ceramic Reactors, GEAP-4770, May 1965.
- ² Perry, J. B. and Alcorn, J. M., BURP - A Computer Program for Calculating Buildup and Decay of Radioactive Fission Products, Babcock & Wilcox, BAW-TM-444, Lynchburg, Virginia, November 1966.
- ³ Perry, J. B., Reactor Building Accessibility With Failed Fuel, Babcock & Wilcox, NPGD-TM-3, Lynchburg, Virginia, June 1967.

5. STEAM CYCLE SELECTION
TRADE-OFF STUDY

CONTENTS

	Page
1. INTRODUCTION	1
1.1. Purpose	1
1.2. Summary	2
2. CONCLUSIONS AND RECOMMENDATIONS	4
3. DISCUSSION	6
3.1. General	6
3.2. General Approach to the Trade-Off	7
3.3. Ground Rules	7
4. RESULTS	13
4.1. Detailed Procedure	13
4.2. IHX Temperature Optimization	16
4.3. IHX Configuration Optimization	18
APPENDIX — Study — Sodium-Heated Vs Steam-Heated Reheater	A-1

List of Tables

Table

1. Steam Cycle Data	3
2. Steam Cycles and Associated Inlet and Outlet Temperatures	11
3. Reheater-Steam Generator Optimization Results	20
4. Summary — IHX Temperature Optimization	26
5. Principal Costs Associated With Figure 3	27
A-1. Economic Evaluation	A-5

List of Figures

Figure	Page
1. Effect of Reactor ΔT on Total Power Generation Cost, 3500 psi/1000 F/1000 F Steam Cycle	28
2. Effect of Reactor ΔT on Total Power Generation Cost, 2400 psi/1000 F/1000 F Steam Cycle	29
3. Effect of Reactor ΔT on Total Power Generation Cost, 2400 psi/950 F/950 F Steam Cycle	30
4. Effect of Reactor ΔT on Total Power Generation Cost, 1800 psi/900 F/900 F Steam Cycle	31
5. Effect of Reactor ΔT on Total Power Generation Cost, 2400 psi/950 F/NRH Steam Cycle	32
6. Effect of Reactor ΔT on Total Power Generation Cost, 1450 psi/1000 F/NRH Steam Cycle	33
7. Effect of Reactor ΔT on Total Power Generation Cost, 1000 psi/D&S Steam Cycle	34
8. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 250$ F	35
9. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 300$ F	36
10. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 350$ F	37
11. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 375$ F	38
12. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1100$ F, $\Delta T_R = 275$ F	39
13. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1100$ F, $\Delta T_R = 300$ F	40
14. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1100$ F, $\Delta T_R = 325$ F	41
15. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1050$ F, $\Delta T_R = 225$ F	42
16. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1050$ F, $\Delta T_R = 250$ F	43
17. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1050$ F, $\Delta T_R = 275$ F	44
18. Effect of IHX Tube Pitch on Shroud ID and Minimum Reactor Pot Height	45
A-1. Heat Balance for Live Steam Reheater ($T_{HRH} = 850$ F)	A-6

1. INTRODUCTION

One of the main incentives for the development of an LMFBR plant is the production of economically competitive electrical energy. It therefore becomes necessary to analyze the factors that are most influential to achieving this end. The selection of the proper steam cycle for the turbine island is of prime importance to the total capital cost and the total power generation costs.

The major thermodynamic losses for the entire plant occur in the main condenser: Approximately two thirds of the power produced in the core is rejected to the condenser cooling water. The net plant efficiencies for the candidate steam cycles range from 33 to 43%, and the corresponding core power requirements range from 3000 to 2300 MWt. The steam cycles that produce the lowest plant efficiencies result in the highest unit cost for the turbine island.¹ With a constant plant output of 1000 MWe for all of the steam cycles, the core power is greatest when the net plant efficiency is lowest; consequently, the fuel cycle costs are increased.

It is important to consider not only these factors, but also the capital costs of the entire plant with respect to the functional parameters that are compatible with the steam cycle. Since it is quite clear that for given steam conditions to the turbine there is an infinite choice of reactor plant operating conditions, we must select the functional plant parameters that will yield the minimum power generation cost when combined with the appropriate steam cycle and fuel cost. All these considerations were included in an intensive analysis to arrive at the objectives of this study.

1.1. Purpose

This study was undertaken to select a steam cycle and to establish the important plant functional requirements of temperature, pressure, and flow for the energy source and the main heat transport system that would yield the most economical power-producing capabilities for a 1000-

MWe plant, within the framework of the industrial and material technology anticipated in the 1980's.

1.2. Summary

Seven steam cycles were proposed by United Engineers and Contractors, Inc.¹ On the basis of the ground rules established for the 1000-MWe LMFBR Follow-On Study, these steam cycles were investigated to determine which one had the most attractive plant capital and fuel costs. (Table 1 is a summary of data for all seven cycles.) Reactor inlet sodium temperatures ranged from 600 to 900 F, reactor outlet temperatures ranged from 1050 to 1150 F, and hot-end temperature differentials for the intermediate heat exchangers ranged from 20 to 100 F for the selected steam cycle. For the sodium reheat steam cycles, the sodium reheater outlet temperatures were optimized to yield the minimum combined capital cost for the steam generator and reheater. The steam-heated reheat cycle was examined by Sargent & Lundy and is discussed in the appendix. Their results show that the steam-heated reheat cycle is less attractive than the sodium-heated reheat cycle. The NAPS program and the Philco 2000 computer were used to calculate capital costs for each of the 400 cases that were examined. The 2400 psi/950 F/950 F steam cycle was selected.

Table 1. Steam Cycle Data

Cycle designation, psi/F/F	H-P turbine inlet temp, F	Boiler feedwater temp, F	Economizer duty, Btu/hr $\times 10^9$	Boiler duty, Btu/hr $\times 10^9$	Superheater duty, Btu/hr $\times 10^9$
3500/1000/1000	1000	485	--	6.529 ^(a)	--
2400/1000/1000	1000	478	1.786	2.676	2.517
2400/950/950	950	484	1.833	2.813	2.398
1800/900/900	900	465	1.579	3.932	2.098
2400/950/NRH	950	484	2.254	3.458	2.948
1450/1000/NRH	1000	434	1.631	4.778	2.708
1000/dry & sat.	546	419	2.005	8.721	None

Cycle designation, psi/F/F	L-P turbine inlet steam pressure, psia	Reheater inlet temp, F	Reheater duty, Btu/hr $\times 10^9$	Boiler feed-water flow rate, lbm/hr $\times 10^6$	Reheater flow rate, lbm/hr $\times 10^6$
3500/1000/1000	553	560	1.632	6.88	6.350
2400/1000/1000	515	635	1.343	6.993	6.528
2400/950/950	545	610	1.362	7.350	6.837
1800/900/900	455	605	1.255	7.846	7.242
2400/950/NRH	--	--	--	9.035	--
1450/1000/NRH	--	--	--	8.450	--
1000/dry & sat.	--	--	--	13.496	--

^(a)Steam generator total.

2. CONCLUSIONS AND RECOMMENDATIONS

1. In general, total power generation costs decrease with an increase in reactor outlet temperature.
2. For any reactor outlet temperature, there is a reactor ΔT that corresponds to a minimum power generation cost.
3. An examination of the four reheat cycles for the optimum sodium outlet temperature indicated that this point occurred when the value $\Delta T''_{\text{IHX}} - \Delta T_{\text{RH}}$ was approximately 125 to 150 F.
4. An examination of the effect of varying the ΔT_M of the IHX indicated that there is an optimum ΔT_M that corresponds to a minimum power generation cost.
5. In general, a ΔT_M that consists of an equal hot- and cold-end temperature difference has a slight cost advantage over a ΔT_M that consists of an unequal hot- and cold-end temperature difference.

A comprehensive treatment of the reasoning behind these conclusions is presented in Volume 4, section 6 of this report, "Heat Transport System Technology - Optimization Relationships."

Within the limits of the accuracy of this study, the important data for the following four steam cycles differed very little.

2400 psi/1000 F/1000 F*
2400 psi/950 F/950 F
1800 psi/900 F/900 F
1450 psi/1000 F/NRH

The 3500 psi/1000 F/1000 F and the 1000 psi dry and saturated cycles were the least attractive economically. The 2400 psi/950 F/NRH cycle results indicated a definite incentive for reheat at this pressure.

*Selected before this study was performed.

A re-evaluation of steam generator and sodium reheater costs for the helical tube concept indicates that a single-material bundle is considerably cheaper than a two-material bundle.³ Boiler feedwater chemistry requirements make Croloy 2-1/4 an attractive tube bundle material. An upper temperature limit of 1025 F does not unduly penalize the cost of power produced under these conditions.

The primary factors that influenced the choice of the 2400 psi/950 F/950 F steam cycle were as follows:

1. Considerations of maximum fuel cladding temperature limits and minimum bulk outlet temperatures from the reactor.
2. Minimum core power requirements (a factor that tends to reduce hardware size as well as thermal pollution).
3. Maximum temperature limits on Croloy 2-1/4 in the steam generators.
4. Power production costs.

The following plant thermal and hydraulic values were selected to coincide with the 2400/950/950 steam cycle:

Reactor inlet temperature, F	800
Reactor outlet temperature, F	1100
Intermediate heat exchanger	
Hot end temperature differential, F	100
Cold end temperature differential, F	100
Tube OD, in.	0.625
Tube pitch, in.	0.875
Tube side ΔP , psi	13.5
Sodium temperature differential-reheater, F	175
Primary sodium flow rate, lb/hr $\times 10^6$	92.4
Intermediate sodium flow rate, lb/hr $\times 10^6$	92.4

3. DISCUSSION

3.1. General

A steam cycle selection study was performed to determine the important plant functional requirements that would yield a minimum power generation cost. The study was kept within the framework of the industrial and material technology anticipated in the 1980's.

The primary instrument used for evaluating the effects of varying thermal, hydraulic, and economic parameters was the NAPS Computer Code, which was developed specifically for this purpose. In general, the code computes the thermal, hydraulic, and cost characteristics of the reactor, the IHXs, the primary and intermediate pumps, the steam generators and reheaters, and the turbine island. It then makes a summation of the capital, fuel, and operating costs. It is important to note that it is the difference in total costs that determine the desirability of one condition over the other, and that the absolute values are not to be considered as the actual cost of the power produced. The principal reason for this is that, in order to provide a useful working tool within a reasonable time, it was necessary to limit the variables that could be examined. The choice of which variables would be entered into the code as constants was based on whether an item was considered to be very nearly constant for all plant configurations, such as costs for auxiliary systems, instrumentation, etc., or whether the item involved individual design evaluations, such as reactor building and fuel handling costs. Engineering judgment was used in applying economic values to these items. The principal independent variables that were investigated for this study were the reactor inlet and outlet temperature, the IHX hot and cold end temperature differentials, the reheater sodium outlet temperature, and the steam cycles shown in Table 2. The principal dependent variables used in the evaluation were the total power generation costs, maximum fuel cladding temperatures, sodium inlet temperature

to the steam raising equipment, and component sizes affecting the reactor vessel size.

3.2. General Approach to the Trade-Off

Determine how the total power generation costs are affected by varying the reactor inlet and outlet sodium temperatures for each of the seven steam cycles shown in Table 1.

Since the cost of the reheaters on the four reheat cycles is a strong function of the sodium outlet temperatures, determine the optimum sodium outlet temperature of these cycles.

Select a steam cycle based on total power generation cost, maximum fuel cladding temperatures, and other material considerations.

Optimize the IHX temperatures for the selected steam cycle as a function of the total power generation costs.

Determine the most acceptable IHX tube pitch, primary and intermediate side pressure drop, and shroud diameter for the selected steam cycle conditions and the optimized IHX temperatures.

3.3. Ground Rules

In order to perform this study in a manner that treats the entire investigation equitably, certain ground rules and assumptions were established. These items are discussed below. It should be noted that the data presented here may differ from the data given for the reference core in Volumes 1, 2, and 3 of BAW-1316; this is because this trade-off study was carried out before the reference core was definitely established.

3.3.1. Plant Configuration

The plant configuration, established through this study, consists of a reactor, six primary sodium pumps, six IHXs, three intermediate sodium pumps, three sodium steam generators, and three sodium reheaters (when required).

3.3.2. Core

In this study the basic core configuration and lattice arrangement were held constant. Core power variations resulting from changes in the steam cycle were accommodated by additional bundles added to the core circumference. No attempt was made to ensure core

symmetry. Since the active length remains constant, the axial peaking factor is essentially unchanged. It is assumed that suitable changes in zone loading will be made to maintain a constant radial peaking factor as the number of fuel bundles is changed. The only remaining uncertainty is the effect of the changes in zone enrichments on core performance. Over the range in question, the effect should be small.

The fuel assemblies were solid, hexagonal cans having an inside diameter of 6.512 inches across the flats. The NAPS computer program allows the can wall thickness to vary with core pressure drop in order to maintain a capable structure for the subassembly support and to minimize the steel fraction in the core. The core burnup is taken as 100,000 MWD/tonne. Each active core fuel assembly contains 298 vented fuel pins, each of which has an OD of 0.270 inch, arranged on a 0.340-inch triangular pitch. The fuel pin length includes 37.42 inches of active fuel bounded on either end by 15.5 inches of axial blanket fuel. An additional 2 inches is assumed to accommodate the porous plug for venting the fuel pin to the sodium. The fuel pin cladding is maintained at a constant thickness of 0.010 inch. Fuel pin spacing is maintained by the application of a twisted ribbon that extends over the full length of the fuel pins. Ten percent of each bundle is composed of beryllium pins to obtain the desired safety characteristics for this core design. The final reference plant core contains no beryllium.

The effects of the core functional parameters on the core physical parameters are of primary interest in this study, because the selection of the entire power generating system must not only produce minimum power costs, but it must also operate within the physical limitations of the material of construction.

3.3.3. Intermediate Heat Exchangers (IHXs)

The intermediate heat exchangers included in this study are of the Fermi type and are sized and costed on this basis. It is recognized that there are design differences between this type of heat exchanger and one that would eventually be utilized in a pot-type design. However, it is believed that such design differences, as related to costs, would cancel each other out.

Primary sodium is located on the shell side, and intermediate sodium flows in the tubes. Of principal concern to the physical

size of the reactor vessel is the total IHX length, the bundle OD, and the head of primary sodium required to drive the primary sodium through the shell side of the heat exchanger. Since the NAPS code does not calculate the physical envelope required to hold the primary pump, the heat exchangers, the thermal shields and the reactor internals, it was necessary to investigate (independently of this study) a number of heat exchangers in order to arrive at a reasonable configuration that could be used in this study.⁴

Of the 54 different heat exchangers that were examined at a constant log mean temperature differential, constant heat load, and constant primary and intermediate flow, one was selected: it uses 5/8-inch-OD tubes on a 7/8-inch pitch. This configuration yielded a minimum tube bundle diameter, while the sum of the static head of sodium and the height of the heat exchanger did not unduly penalize the depth of the reactor vessel.

3.3.4. Primary and Intermediate Sodium Pumps

As pointed out above, the plant configuration chosen for this study contains six primary and three intermediate sodium pumps. The number of primary pumps was chosen as a result of mechanical layout studies made to yield a compact arrangement (see Loop Trade-Off Study). The pump design information used in this study was obtained from the Byron Jackson Pump Division of the Borg Warner Co. under their study contract to the AEC.⁵ Unfortunately, costing information was not available for large sodium pumps in time to be incorporated in the steam cycle selection study. In lieu of this information, a figure of \$150 per kW of electrical energy to the pump-driver was used to compute the cost of the pumps and drivers. This figure is approximately twice that for pumps and drivers used in pressurized water plants. No attempt was made to differentiate the total costs of pumps and drivers for the primary system as compared with the pumps and drivers for the intermediate system.

The computed pump and driver costs for both the primary and intermediate systems are derived from the total electrical requirements as computed by NAPS.

3.3.5. Steam Generators

The steam generators used in this study are the once-through, helical coil type developed by B&W under a study contract to the AEC.² Three steam generators were assumed throughout this study. The computer code calculated their cost as a function of heat load, steam temperature and pressure, and sodium inlet temperature. In general, the code is capable of calculating the cost of the steam generator based on Croloy 2-1/4 in the economizer and boiling section, and a type-304 stainless steel superheater section for all sodium above 1025 F. That is, if sodium enters the steam generator at 1025 F or less, then the superheater section will not require stainless steel, and the cost is computed accordingly.

3.3.6. Reheaters

The reheaters used in this study assume the helical coil concept, as do the steam generators. They were developed under the same contract as the steam generators (above). All reheat cycles use sodium-heated reheaters, except for the 1000 psi dry and saturated steam cycle, which uses a steam reheater. Here again, as in the steam generators, the code calculates the cost of the steam reheaters as a function of heat load, steam temperature and pressure, and sodium inlet and outlet temperature.

In those cycles that require sodium reheat, three reheaters are used. The tube material throughout the reheaters is Croloy 2-1/4.

Table 2. Steam Cycles and Associated Inlet and Outlet Temperatures

Symbols: T_{RO} Reactor outlet temp, F
 T_{RI} Reactor inlet temp, F
 ΔT_R Reactor temp differential, F
 IHX hot end $\Delta T = 75$ F
 IHX cold end $\Delta T = 130$ F

<u>Cycle, psi/F/F</u>	<u>No.</u>	<u>T_{RO}</u>	<u>T_{RI}</u>	<u>ΔT_R</u>
3500/1000/1000	1	1150	900	250
	2	1150	850	300
	3	1150	800	350
	4	1150	725	425
	5	1125	875	250
	6	1125	825	300
	7	1125	775	350
	8	1100	850	250
	9	1100	800	300
	10	1100	750	350
2400/1000/1000	1	1150	900	250
	2	1150	850	300
	3	1150	800	350
	4	1125	850	275
	5	1125	825	300
	6	1125	800	325
	7	1100	850	250
	8	1100	825	275
	9	1100	800	300
2400/950/950	1	1150	900	250
	2	1150	850	300
	3	1150	800	350
	4	1150	775	375
	5	1100	825	275
	6	1100	800	300
	7	1100	775	325
	8	1050	825	225
	9	1050	800	250
	10	1050	775	275

Table 2. (Cont'd)

<u>Cycle, psi/F/F</u>	<u>No.</u>	<u>T_{RO}</u>	<u>T_{RI}</u>	<u>ΔT_R</u>
1800/900/900	1	1150	900	250
	2	1150	850	300
	3	1150	800	350
	4	1150	770	380
	5	1100	850	250
	6	1100	800	300
	7	1100	770	330
	8	1050	800	250
	9	1050	785	265
	10	1050	770	280
2400/950/NRH	1	1150	900	250
	2	1150	850	300
	3	1150	800	350
	4	1150	750	400
	5	1150	700	450
	6	1150	650	500
	7	1100	850	250
	8	1100	750	350
	9	1100	650	450
	10	1050	800	250
	11	1050	750	300
	12	1050	700	350
	13	1050	650	400
1450/1000/NRH	1	1150	900	250
	2	1150	800	350
	3	1150	700	450
	4	1150	650	500
	5	1100	850	250
	6	1100	800	300
	7	1100	750	350
	8	1050	800	250
	9	1050	700	350
	10	1050	600	450
1000/dry & sat.	1	1150	850	300
	2	1150	750	400
	3	1150	650	500
	5	1050	750	300
	6	1050	700	350
	7	1050	650	400

4. RESULTS

4.1. Detailed Procedure

Table 2 lists the reactor outlet temperatures, inlet temperatures, and the temperature differentials for each of the seven steam cycles to be examined. For each of the conditions shown in this table, approximately eight cases (reheat cycles only) were examined to show how increasing the sodium reheater outlet temperature affects the total power generation costs, which include fixed charges, fuel costs, and operating, maintenance, and insurance charges. The variation occurs because the reheater log mean temperature differential increases with an increase in sodium outlet temperature, thereby decreasing the heat transfer area (and the cost) required for a given reheater's thermal duty. As the reheater sodium outlet temperature is increased, the sodium flow through the reheater is increased, thereby decreasing the sodium flow through the steam generator, which in turn decreases the log mean temperature differentials in all of the heat transfer regions. This condition, therefore, increases the heat transfer area requirements and the cost of the steam generator.

Since the cost of the steam generator is not increased by the same amount that the reheater is decreased, a minimum total capital cost of the combined reheater and steam generator can be found. This minimum is reflected in the total power generation costs; Table 3 shows these results with the minimum costs underlined.

Figures 1 through 4 are plots of total power generation cost in mills/kWh versus reactor ΔT for various reactor outlet temperatures for the four reheat cycles. These curves have been plotted from the optimized reheater-steam generator power costs listed in Table 3. In addition, these plots show the maximum fuel cladding temperatures associated with the reactor bulk outlet temperatures.

Table 5 is a compilation of reheater, steam generator, and IHX costs and a number of other pertinent costs for the 2400 psi/950 F/950 F steam cycle. The information in this table is derived from the same computer codes that produced the information for Figure 3. It should be noted that, for the three reactor outlet temperatures, the capital costs increased and fuel costs decreased as the reactor ΔT increased. The method for calculating capital costs for this study is explained in detail in reference 6.

Figures 5 through 7 are the same type of plots for the no-reheat (NRH) cycles. In general, examination of these plots indicates that the total power generation costs decrease with an increase in reactor outlet temperature and also decrease with an increase in reactor ΔT .

Figures 2, 3, 4, and 6 show very comparable power generation costs. Hence, it becomes necessary to investigate those factors that override the cost considerations determined by the NAPS computer code.

4.1.1. Factors Leading to Steam Cycle Selection

The NAPS code mathematical model of the steam generator and reheater used to determine the costs was derived from the results of a study performed by B&W under contract to the Atomic Energy Commission in 1964. Since that time, new information has been derived from more detailed sodium steam generator and reheater studies;³ these studies indicated that a two-material bundle, i.e., a stainless steel superheater section and a Croloy boiling and economizer section, was more expensive than originally concluded. The later study also concluded that a single-material bundle of either stainless steel or Croloy would be comparable to the 1964 results and, hence, compatible with the pricing model in the NAPS code.^{3,6} It was then necessary to make a choice between accepting an all-stainless steel or an all-Croloy tube bundle for the steam generators.

Logical reasoning maintains that with an all-stainless steel bundle, higher inlet sodium temperatures are acceptable for the steam generator. This, then, makes it possible to use a higher-temperature steam cycle, which in turn decreases the total core power required to produce the nominal 1000 MWe. The attendant advantages

are lower fuel costs, smaller IHXs, a smaller reactor vessel and cover plug, and a lower sodium inventory. The major disadvantages are higher clad temperatures, tighter restrictions on sodium purity, higher reactor bulk outlet temperatures and the susceptibility of stress chloride corrosion on the water side of the steam generators.

Of these advantages and disadvantages, all but the disadvantage of stress chloride corrosion are extremely difficult to evaluate. Even here, there are differences of opinion concerning the ability of the feedwater cleanup system to maintain (at all times) the low chloride concentration required for high operational reliability in the system. Nevertheless, it is felt that tight feedwater chemistry control is not a satisfactory solution to this problem, all the advantages cited previously notwithstanding. Therefore, an all-Croloy material was selected for the tube bundle, together with an upper limit of 1025 F on the sodium entering the steam generator.

The original reference concept was based on a 2400 psi/1000 F/1000 F steam cycle, assuming a bimetallic bundle in the steam generator, a 350-F temperature difference across the core, and an 1150-F mixed mean core outlet temperature. An examination of Figure 2 indicates that this condition (1)* was not acceptable in any event, since the maximum cladding temperature of 1327 F at this point exceeded the allowable 1300 F. However, reducing the reactor temperature differential to 300 F would bring the cladding temperature down to an acceptable level (2). In order to satisfy the selected criteria of a maximum inlet temperature of 1025 F to the steam generator and a reactor outlet temperature of 1100 F, a reactor temperature differential of approximately 275 F is required (3) according to Figure 2. This change would increase the total cost of generating power over the modified reference point by approximately 0.07 mills/kWh.

It was apparent that another steam cycle was required to reduce the increased cost in power due to the requirements stated above. As previously mentioned in section 2, the 2400 psi/950 F/950 F, the 1800 psi/900 F/900 F, and the 1450 psi/1000 F/NRH cycles

- - - - -

*Circled numerals refer to the point on the curve under discussion.

showed no significant differences in total power generation cost as determined by the NAPS computer code. On the basis of this observation, these steam cycles alone were considered as the possible candidates for the optimum cycle.

Net plant efficiency was used in eliminating the most undesirable of these three cycles. The lowest efficiencies were associated with the 1800-psi cycle (41%) and the 1450-psi cycle (39%), while the net plant efficiency associated with the 2400 psi/950 F/950 F cycle was 42%. Increased efficiency tends to reduce the duty of the heat transfer equipment, the size of the reactor vessel and cover plug, the primary sodium inventory, and the thermal pollution. An examination of Figures 3, 4, and 6 for possible cost penalties reveals the following:

Steam cycle	Figure No.	Net eff. %	Maximum cladding temp. F	Reactor ΔT	Reactor outlet temp. F	Cost, mills/kWh
2400 psi/950 F/950 F ④	3	42	1254	300	1100	3.448
1800 psi/900 F/900 F ⑤	4	41	1254	300	1100	3.435
1450 psi/1000 F/NRH ⑥	6	39	1275	350	1100	3.426

The cost differences are not considered to be significant enough to compete with the advantages gained by the higher efficiencies. On the basis of this discussion, the 2400 psi/950 F/950 F steam cycle was selected.

4.2. IHX Temperature Optimization

Although the steam cycle evaluation and selection was conducted using hot- and cold-end temperature differentials of 75 and 130 F, it cannot be concluded that a reactor outlet temperature of 1100 F and a core temperature differential of 300 F are necessarily the optimum conditions. Therefore, to determine just what the most advantageous conditions are, and to conform to the contract obligations of investigating the hot-end temperature differentials between the limits of 20 and 100 F, a course of action was undertaken wherein the following points would be investigated for the selected steam cycle:

<u>Combination</u>	<u>Hot-end temp differential, F</u>	<u>Cold-end temp differential, F</u>	<u>Log mean ΔT</u>
1	20	100	50
2	50	110	76
3	75	130	100
4	100	150	125
5	50	50	50
6	76	76	76
7	100	100	100
8	125	125	125

Combinations 1 through 4 were determined by selecting the hot-end range and coupling these to a log mean temperature differential. The cold-end temperature differentials were then calculated. The selection of this range of temperature differentials provides conditions that indicate the combined effect on total power generation costs for variance of the IHX log mean temperature differential, the approach temperature of the intermediate sodium to the steam, the total temperature differential across the steam generator, and the change of total flow rate in the intermediate system.

Combinations 5 through 8 attempt to show the effect on total power generation costs using the same log mean temperature differentials as in combinations 1 through 4, but with hot and cold-end temperature differentials equal to the log mean temperature differential.

Referring to Figures 8 through 17, it can be seen that for all practical purposes the power generation costs are the same if the log mean temperature differential is the same. The problem then reduces itself to selecting the ΔT_M that affords the most desirable economics when combined with the other important plant variables.

Table 4 is a summary of the most attractive cases used to generate Figures 8 through 17. Here again it can be seen that the difference in total power generation cost between case 1 (maximum) and case 3 (minimum) is only 0.024 mill per kWh, which rules out selection based on costs as calculated by the NAPS computer program. Cases 3 and 4 are ruled out because of fuel pin cladding temperatures (F_C) that exceed the allowable 1300 F. Cases 1 and 2 can be eliminated because there is no incentive to have a mixed mean outlet temperature of 1150 F. Cases 7, 9, and 10 have excessively long tube bundle lengths. Case 8 has a

high shell side pressure drop which must be supplied by a static head of sodium, thus increasing the height of the reactor vessel. Cases 5 and 6 are very similar, with case 6 showing a slight edge over case 5.

Based on the preceding discussion, the reference design will include a reactor inlet temperature of 800 F, a reactor outlet temperature of 1100 F, and equal hot-and cold-end temperature differentials for the IHX—100 F.

4.3. IHX Configuration Optimization

Upon completion of 4.2, the last portion of the investigation to establish the final configuration of the IHXs was performed.

Generally speaking, the heat exchanger diameter is directly related to the diameter of the reactor vessel and hence to the reactor cover plug. Therefore, it is intuitive that the cover plug diameter should be minimized without unduly penalizing the height of the vessel. The arrangement of equipment within the reactor vessel and the optimization of the equipment sizes as a function of the cost of the entire unit (reactor vessel, cover plug, sodium inventory, biological shielding, etc.) will require an intensive investigation outside the scope of this effort.

In this study we have elected to minimize the diameter of the IHXs reasonably using engineering judgment to determine the accompanying penalties in the reactor vessel configuration. The effort consisted of determining the heat exchanger diameter, length, primary system pressure drop, and total power generation cost as functions of tube pitch and intermediate side pressure drop for the selected steam cycle and the optimized heat exchanger inlet and outlet temperatures.

The effect of tube size was not investigated, because earlier work had shown the desirability of using 5/8-inch-OD tubes (see 3.3.3).

Figure 18 is a plot of the IHX shroud diameter showing its relation to the sum of the primary sodium pressure drop and the effective length of the heat exchanger tubes. This summation is labeled "minimum reactor pot height." The design point has a minimum reactor pot height of 29.7 (7) feet and a shroud diameter of 72 inches.

The variation in shroud diameter occurs when the IHX is designed to varying allowable tube side pressure drops. As the pressure drop

on the tube side is allowed to rise (for a constant pitch), the shroud diameter decreases, because the overall heat transfer coefficient increases, and consequently the number of tubes decreases. Therefore, it is concluded that for the arrangement in question, it is desirable to use a 0.625-inch tube on a 0.875-inch pitch.

Table 3. Reheater-Steam Generator Optimization Results

Symbols: T_{RI} Reactor inlet sodium temp, F
 T_{RO} Reactor outlet sodium temp, F
 $\Delta T''_{IHX}$ Intermediate heat exchanger temp differential, intermediate sodium, F
 ΔT_{RH} Reheater sodium temp differential, F
Cost Total power generation cost, mills/kWh

Case	T_{RI}	T_{RO}	$\Delta T''_{IHX}$	ΔT_{RH}	Cost
Steam Cycle: 3500 psi/1000 F/1000 F					
1	900	1150	305	305	--
2				280	3.540
3				255	3.532
4				230	3.528
5				205	3.526
6				180	3.527
7				155	3.535
8				130	3.581
9	850		355	355	3.525
10				330	3.517
11				305	3.512
12				280	3.509
13				255	3.508
14				230	3.511
15				205	3.523
16				180	3.566
17	800		405	405	3.540
18				380	3.529
19				355	3.523
20				330	3.521
21				305	3.524
22				280	3.536
23				255	3.568
24				230	3.758
25	725		480	480	3.709
26				455	3.686
27				430	3.713
28	875	1125	305	305	3.569
29				280	3.561
30				255	3.555
31				230	3.551
32				205	3.550
33				180	3.554
34				155	3.572
35				130	3.748
36	825		355	355	3.567
37				330	3.556

Table 3. (Cont'd)

Case	T_{RI}	T_{RO}	$\Delta T''_{IHx}$	ΔT_{RH}	Cost
38	825	1125	355	305	3.549
39	↓	↓	↓	280	<u>3.546</u>
40	↓	↓	↓	255	3.547
41	↓	↓	↓	230	3.557
42	↓	↓	↓	205	3.590
43	775	↓	405	405	3.616
44	↓	↓	↓	380	3.599
45	↓	↓	↓	355	<u>3.591</u>
46	↓	↓	↓	330	<u>3.594</u>
47	↓	↓	↓	305	3.612
48	↓	↓	↓	280	3.678
49	850	↓	305	305	3.632
50	↓	↓	↓	280	3.619
51	↓	↓	↓	255	3.610
52	↓	↓	↓	230	<u>3.605</u>
53	↓	↓	↓	205	<u>3.605</u>
54	↓	↓	↓	180	3.615
55	↓	↓	↓	155	3.665
56	800	↓	355	355	3.663
57	↓	↓	↓	330	3.643
58	↓	↓	↓	305	3.632
59	↓	↓	↓	280	<u>3.629</u>
60	↓	↓	↓	255	<u>3.639</u>
61	↓	1100	↓	230	3.680
62	750	↓	405	405	3.808
63	↓	↓	↓	380	<u>3.779</u>
64	↓	↓	↓	355	<u>3.799</u>

Steam Cycle: 2400 psi/1000 F/1000 F

1	900	1150	305	305	3.459
2	↓	↓	↓	280	3.450
3	↓	↓	↓	255	3.442
4	↓	↓	↓	230	3.435
5	↓	↓	↓	205	3.429
6	↓	↓	↓	180	3.425
7	↓	↓	↓	155	<u>3.421</u>
8	↓	↓	↓	130	<u>3.442</u>
9	↓	↓	↓	105	3.444
10	850	↓	355	355	3.442
11	↓	↓	↓	330	3.426
12	↓	↓	↓	305	3.415
13	↓	↓	↓	280	3.406
14	↓	↓	↓	255	3.399
15	↓	↓	↓	230	3.395
16	↓	↓	↓	205	<u>3.392</u>
17	↓	↓	↓	180	<u>3.393</u>
18	800	↓	405	405	3.483
19	↓	↓	↓	380	3.442

Table 3. (Cont'd)

Case	T_{RI}	T_{RO}	$\Delta T''_{IHx}$	ΔT_{RH}	Cost
20	800	1150	405	355	3.419
21	↓	↓	↓	330	3.404
22	↓	↓	↓	305	3.395
23	↓	↓	↓	280	3.389
24	↓	↓	↓	255	3.387
25	↓	↓	↓	230	3.389
26	850	1125	330	330	3.481
27	↓	↓	↓	305	3.460
28	↓	↓	↓	280	3.446
29	↓	↓	↓	255	3.436
30	↓	↓	↓	230	3.428
31	↓	↓	↓	205	3.423
32	↓	↓	↓	180	3.420
33	↓	↓	↓	155	3.424
34	825	↓	355	355	3.495
35	↓	↓	↓	330	3.464
36	↓	↓	↓	305	3.445
37	↓	↓	↓	280	3.431
38	↓	↓	↓	255	3.422
39	↓	↓	↓	230	3.416
40	↓	↓	↓	205	3.414
41	↓	↓	↓	180	3.418
42	800	↓	380	380	3.539
43	↓	↓	↓	355	3.484
44	↓	↓	↓	330	3.454
45	↓	↓	↓	305	3.436
46	↓	↓	↓	280	3.424
47	↓	↓	↓	255	3.417
48	↓	↓	↓	230	3.415
49	↓	↓	↓	205	3.420
50	850	1100	305	305	3.556
51	↓	↓	↓	280	3.526
52	↓	↓	↓	255	3.506
53	↓	↓	↓	230	3.491
54	↓	↓	↓	205	3.481
55	↓	↓	↓	180	3.474
56	↓	↓	↓	155	3.472
57	↓	↓	↓	130	3.483
58	825	↓	330	330	3.584
59	↓	↓	↓	305	3.538
60	↓	↓	↓	280	3.509
61	↓	↓	↓	255	3.490
62	↓	↓	↓	230	3.477
63	↓	↓	↓	205	3.463
64	↓	↓	↓	180	3.466
65	↓	↓	↓	155	3.476
66	800	↓	355	355	3.657
67	↓	↓	↓	330	3.571
68	↓	↓	↓	305	3.526

Table 3. (Cont'd)

Case	T_{RI}	T_{RO}	$\Delta T''_{IHX}$	ΔT_{RH}	Cost
69	800	1100	355	280	3.499
70	↓	↓	↓	255	3.482
71	↓	↓	↓	230	3.471
72	↓	↓	↓	205	3.469
73	↓	↓	↓	180	3.480

Steam Cycle: 2400 psi/950 F/950 F

1	900	1150	305	205	3.464
2	↓	↓	↓	180	3.461
3	↓	↓	↓	155	3.459
4	↓	↓	↓	130	3.461
5	↓	↓	↓	205	3.486
6	850	↓	355	255	3.434
7	↓	↓	↓	230	3.431
8	↓	↓	↓	205	3.430
9	↓	↓	↓	180	3.433
10	↓	↓	↓	155	3.445
11	800	↓	405	305	3.437
12	↓	↓	↓	280	3.434
13	↓	↓	↓	255	3.435
14	↓	↓	↓	230	3.440
15	↓	↓	↓	205	3.457
16	775	↓	430	330	3.441
17	↓	↓	↓	305	3.439
18	↓	↓	↓	280	3.441
19	↓	↓	↓	255	3.450
20	↓	↓	↓	230	3.475
21	825	1100	330	230	3.452
22	↓	↓	↓	205	3.450
23	↓	↓	↓	180	3.453
24	↓	↓	↓	155	3.467
25	↓	↓	↓	130	3.550
26	800	↓	355	255	3.450
27	↓	↓	↓	230	3.448
28	↓	↓	↓	205	3.452
29	↓	↓	↓	180	3.469
30	↓	↓	↓	155	3.551
31	775	↓	380	280	3.469
32	↓	↓	↓	255	3.468
33	↓	↓	↓	230	3.475
34	↓	↓	↓	205	3.500
35	↓	↓	↓	180	3.648
36	825	1050	280	180	3.564
37	↓	↓	↓	155	3.561
38	↓	↓	↓	130	3.556
39	↓	↓	↓	205	3.610
40	800	↓	305	205	3.559
41	↓	↓	↓	180	3.552

Table 3. (Cont'd)

Case	T_{RI}	T_{RO}	$\Delta T_{IHX}''$	ΔT_{RH}	Cost
42	800	1050	305	155	3.556
43	↓	↓	↓	130	3.587
44	775	↓	330	230	3.565
45	↓	↓	↓	205	<u>3.558</u>
46	↓	↓	↓	180	<u>3.562</u>
47	↓	↓	↓	155	3.593

Steam Cycle: 1800 psi/900 F/900 F

1	900	1150	305	230	3.474
2	↓	↓	↓	205	3.472
3	↓	↓	↓	180	3.469
4	↓	↓	↓	155	3.468
5	↓	↓	↓	130	<u>3.467</u>
6	↓	↓	↓	105	3.472
7	850	↓	355	280	3.439
8	↓	↓	↓	255	3.436
9	↓	↓	↓	230	3.434
10	↓	↓	↓	205	<u>3.433</u>
11	↓	↓	↓	180	<u>3.433</u>
12	↓	↓	↓	155	3.436
13	800	↓	405	330	3.425
14	↓	↓	↓	305	3.421
15	↓	↓	↓	280	3.419
16	↓	↓	↓	255	<u>3.418</u>
17	↓	↓	↓	230	<u>3.419</u>
18	770	↓	435	360	3.424
19	↓	↓	↓	335	3.420
20	↓	↓	↓	310	<u>3.417</u>
21	↓	↓	↓	285	<u>3.417</u>
22	↓	↓	↓	260	3.419
23	850	1100	305	230	3.463
24	↓	↓	↓	205	3.460
25	↓	↓	↓	180	<u>3.459</u>
26	↓	↓	↓	155	<u>3.459</u>
27	↓	↓	↓	130	3.463
28	800	↓	355	280	3.441
29	↓	↓	↓	255	3.437
30	↓	↓	↓	230	<u>3.435</u>
31	↓	↓	↓	205	<u>3.435</u>
32	↓	↓	↓	180	3.439
33	770	↓	385	310	3.440
34	↓	↓	↓	285	3.435
35	↓	↓	↓	260	<u>3.433</u>
36	↓	↓	↓	235	<u>3.434</u>
37	800	1050	305	230	3.502
38	↓	↓	↓	205	3.497
39	↓	↓	↓	180	<u>3.494</u>
40	↓	↓	↓	155	3.494

Table 3. (Cont'd)

<u>Case</u>	<u>T_{RI}</u>	<u>T_{RO}</u>	<u>ΔT_{IHX}</u>	<u>ΔT_{RH}</u>	<u>Cost</u>
41	800	1050	305	130	3.501
42	↓	↓	↓	105	3.540
43	785	↓	320	245	3.497
44	↓	↓	↓	220	3.490
45	↓	↓	↓	195	<u>3.487</u>
46	↓	↓	↓	170	<u>3.487</u>
47	↓	↓	↓	145	3.493
48	770	↓	335	260	3.496
49	↓	↓	↓	235	3.488
50	↓	↓	↓	210	<u>3.484</u>
51	↓	↓	↓	185	<u>3.484</u>
52	↓	↓	↓	160	3.490

<u>Case</u>	<u>T_{RI}</u>	<u>T_{RO}</u>	<u>ΔT_R</u>	<u>Cost</u>
-------------	-----------------------	-----------------------	-----------------------	-------------

Steam Cycle: 1450 psi/1000 F/NRH

1	900	1150	250	3.493
2	800	↓	350	3.421
3	700	↓	450	3.404
4	650	↓	500	3.431
5	850	1100	250	3.483
6	800	↓	300	3.444
7	750	↓	350	3.426

Steam Cycle: 2400 psi/950 F/NRH

1	900	1150	250	3.602
2	850	↓	300	3.558
3	800	↓	350	3.534
4	750	↓	400	3.531
5	700	↓	450	3.561
6	650	↓	500	3.944
7	850	1100	250	3.560
8	750	1100	350	3.529
9	800	1050	250	3.602
10	750	↓	300	3.588
11	700	↓	350	3.612
12	650	↓	400	3.962

Steam Cycle: 1000 psi/546 F/NRH

1	850	1150	300	3.788
2	750	↓	400	3.741
3	650	↓	500	3.733
4	750	1050	300	3.799
5	700	↓	350	3.777
6	650	↓	400	3.774

Table 4. Summary — IHX Temperature Optimization

Symbols:	T_{RI}	Reactor inlet temp, F
	T_{RO}	Reactor outlet temp, F
	ΔT_R	Reactor temp differential, F
	T_{NA-SG}	Intermediate sodium inlet temp to steam generator, F
	ΔT_H	IHX hot end temp differential, F
	ΔT_C	IHX cold end temp differential, F
	ΔT_M	IHX log mean temp differential, F
	Cap.	Capital cost, mills/kWh
	Fuel	Fuel cost, mills/kWh
	Total	Total power generation cost, mills/kWh
	Diam	IHX tube bundle diam, in.
	L	IHX tube bundle length, ft
	$\Delta P'$	IHX primary sodium pressure drop-shell side, psi
	F_C	Maximum fuel pin cladding temp, F
	O,M,&I	Operating, maintenance, & insurance cost, mills/kWh

Case	T_{RI}	T_{RO}	ΔT_R	T_{NA-SG}	ΔT_H	ΔT_C	ΔT_M
1	900	1150	250	1025	125	125	125
2	850	1150	300	1025	125	125	125
3	800	1150	350	1025	125	125	125
4	775	1150	375	1025	125	125	125
5	825	1100	275	1000	100	100	100
6	800	1100	300	1000	100	100	100
7	775	1100	325	1025	75	75	75
8	825	1050	225	1000	50	110	75
9	800	1050	250	1000	50	50	50
10	775	1050	275	1000	50	50	50

Case	Cap.	O,M,&I	Fuel	Total	Diam	L	$\Delta P'$	F_C
1	2.199	0.265	0.991	3.456	92	7.6	3.02	1280
2	2.186	0.265	0.982	3.433	86	8.6	2.53	1306
3	2.205	0.265	0.962	3.432	81	9.5	2.2	1327
4	2.219	0.264	0.957	3.440	79	10.0	2.1	1338
5	2.195	0.266	0.982	3.442	91	9.3	2.6	1242
6	2.199	0.265	0.973	3.438	87	9.9	2.4	1254
7	2.217	0.265	0.967	3.449	88	13	2.04	1265
8	2.226	0.267	1.000	3.492	92	11	3.71	1168
9	2.224	0.266	0.989	3.479	102	14	2.40	1180
10	2.222	0.266	0.980	3.468	99	15	2.15	1192

Cost, \$ $\times 10^6$							
Case	RH	SG	IHX	Total direct	Turbine island	Total indirect	Total capital
1	2.712	8.400	3.885	54.008	43.198	38.882	136.748
2	3.006	8.654	3.717	53.734	43.413	38.859	136.667
3	3.656	8.996	3.586	54.660	43.488	39.259	138.066
4	3.959	9.568	3.537	55.327	43.529	39.543	139.059
5	3.789	7.497	4.103	53.995	43.305	38.920	136.880
6	4.058	7.883	4.019	54.273	43.382	39.062	137.378
7	3.482	8.591	4.465	55.140	43.441	39.431	138.670
8	3.891	7.637	4.480	55.158	43.163	39.329	138.310
9	3.550	6.908	5.720	55.138	43.211	39.340	138.349
10	3.779	7.182	5.650	55.198	43.310	39.403	138.571

Table 5. Principal Costs Associated With Figure 3

T_{RO}	T_{RI}	ΔT_R	Cost, $\$ \times 10^6$							Cost, mills/kWh			
			RH	SG	IHX	Total direct	Turbine island	Total indirect	Total capital	Fixed charges	Fuel cost	O, M, & I	Total power gen. cost
1150	900	250	2.253	9.293	3.948	54.276	43.261	39.000	137.200	2.203	0.990	0.266	3.459
	850	300	2.490	9.484	3.845	53.992	43.420	38.965	137.038	2.192	0.973	0.265	3.430
	800	350	2.999	9.805	3.767	54.871	43.521	39.357	138.409	2.209	0.961	0.264	3.434
	775	375	3.237	10.236	3.741	55.372	43.558	39.572	139.161	2.219	0.956	0.264	3.439
1100	825	275	3.500	8.670	3.896	54.476	43.359	39.134	137.629	2.205	0.980	0.265	3.450
	800	300	3.767	9.099	3.851	54.864	43.428	39.317	138.269	2.211	0.972	0.265	3.448
	775	325	4.091	9.755	3.806	56.104	43.483	39.834	140.080	2.237	0.966	0.265	3.468
1050	825	225	5.280	9.757	4.021	58.222	43.160	40.553	142.595	2.295	0.999	0.267	3.561
	800	250	5.742	10.185	3.953	58.593	43.279	40.749	143.281	2.299	0.987	0.266	3.552
	775	275	6.311	10.778	3.894	59.363	43.367	41.092	144.482	2.314	0.979	0.265	3.558

Figure 1. Effect of Reactor ΔT on Total Power Generation Cost, 3500 psi/1000 F/1000 F Steam Cycle

Figure 2. Effect of Reactor ΔT on Total Power Generation Cost, 2400 psi/1000 F/1000 F Steam Cycle

Figure 3. Effect of Reactor ΔT on Total Power Generation Cost, 2400 psi/950 F/950 F Steam Cycle

Figure 4. Effect of Reactor ΔT on Total Power Generation Cost, 1800 psi/900 F/900 F Steam Cycle

Figure 5. Effect of Reactor ΔT on Total Power Generation Cost, 2400 psi/950 F/NRH Steam Cycle

Figure 6. Effect of Reactor ΔT on Total Power Generation Cost, 1450 psi/1000 F/NRH Steam Cycle

Figure 7. Effect of Reactor ΔT on Total Power Generation Cost, 1000 psi/D&S Steam Cycle

Figure 8. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 250$ F

Figure 9. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 300$ F

Figure 10. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 350$ F

Figure 11. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1150$ F, $\Delta T_R = 375$ F

Figure 12. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1100$ F, $\Delta T_R = 275$ F

Figure 13. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1100$ F, $\Delta T_R = 300$ F

Figure 14. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1100$ F, $\Delta T_R = 325$ F

Figure 15. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1050$ F, $\Delta T_R = 225$ F

Figure 16. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1050$ F, $\Delta T_R = 250$ F

Figure 17. Effect of IHX ΔT_M on Total Power Generation Costs, 2400 psi/950 F/950 F Steam Cycle, $T_{RO} = 1050$ F, $\Delta T_R = 275$ F

Figure 18. Effect of IHX Tube Pitch on Shroud ID and Minimum Reactor Pot Height

APPENDIX

— Study —

Sodium-Heated Vs Steam-Heated Reheater

1. Purpose

This study was carried out to gain information that would aid in the selection of a steam cycle through a comparison of steam-heated and sodium-heated reheaters.

2. Introduction

Because of the limited time allotted to this study, only one technical approach to the evaluation could be followed through. The following approaches were among those considered in making the comparisons:

- (a) Comparisons for steam cycles other than the reference cycle.
- (b) Comparisons holding a constant net power output from the turbine generator.
- (c) Comparisons holding a constant hot reheat temperature.
- (d) Comparisons holding a constant core thermal power and allowing the electrical output to vary.

Of these, the last approach was selected. This approach required the determination of a hot reheat steam temperature in order to proceed with the evaluation of the capital costs and to determine the thermodynamics of the system.

It immediately became obvious that there existed an optimum hot steam reheat temperature that was associated with a minimum total power generation cost; it was apparent that a number of iterations would be required to determine this value. Rather than proceed in this direction, the decision was made to arbitrarily select a temperature of 850 F on the basis of past experience and engineering judgment.

3. Discussion

The reference plant (2400 psi/950 F/950 F cycle), using three sodium-heated steam generators, three sodium-heated reheaters, and a TC6F-30-inch turbine generator, was selected as a base for the study. The duties for the steam generators and sodium-heated reheaters were taken as 7.46×10^9 Btu/hr and 1.32×10^9 Btu/hr, respectively. The turbine generator gross output was rated at 1,033,926 kWe.

The steam-heated reheater plant (2400 psi/950 F/850 F cycle) employs three steam generators having the same pressure ratings and capacity as those of the reference plant. The three steam reheaters have a combined duty of 0.84×10^9 Btu/hr. The 850 F reheater has pressure and temperature ratings on the shell of 2900 psig and 1010 F, and the internal design pressure for the reheater tubes is 750 psig. As with the sodium-heated reheaters, these units are vertically oriented and operate in parallel.

Each unit is 9 feet 7 inches in diameter, 51 feet 6 inches high, and has a hemispherical head. Each has a capacity of 2,000,000 lb/hr of steam at 540 psia and 850 F. The reheat shell is stainless steel, 14 inches thick. The units have helically coiled 1.5-inch-OD, 16 BWG alloy steel tubes with 74,000 ft² of heat transfer area in each unit. The heating steam enters at 2400 psig and 950 F, and discharges as a water/steam mixture at 660 F. On the basis of these design parameters, the estimated cost of each steam-heated reheater is \$3,000,000; therefore, the total estimated cost for the three units is \$9,000,000.

Figure A-1 illustrates a simplified plant heat balance. When this figure is compared with the reference plant heat balance and configuration (as shown in Volume 1, section 2 of this report) the following major differences are apparent:

	<u>Reference plant cycle</u>	<u>Steam-heated reheater cycle</u>
Number of HP heaters	2	3
Feedwater flow to SG, lb/hr $\times 10^6$	7.35	8.75
Feedwater temperature, F	484	484
Steam flow to HP turbine, lb/hr $\times 10^6$	7.35	5.79
Steam flow to IP turbine, lb/hr $\times 10^6$	6.84	5.79
Steam flow to BFP, lb/hr $\times 10^6$	0.28	0.35
Net efficiency, %	42.1	35.7
Gross TG output, MWe	1034.0	877.5

Table A-1 lists the results of an economic analysis of the sodium-heated reheater versus the steam-heated reheater. The table shows

that both the fuel and capital costs are increased by going to the steam-to-steam reheater. Since this evaluation was made for an arbitrarily selected hot steam reheat temperature, it would be pure coincidence if this point turned out to be the actual optimum temperature. However, if we examine the situation qualitatively, we find that, although the exact optimum is unknown, we can conclude that the minimum power generation cost for the steam-heated reheat cycle will always be higher than that for the reference plant. The reasoning is as follows: To reduce the fuel cost increase back to the reference case, the heat rate must be reduced to that of the reference case. To reduce the heat rate, the hot reheat steam temperature must be increased to approach that of the reference case—950 F. Increasing the hot reheat steam temperature decreases the LMTD and increases the design pressure required by the shell. Both of these characteristics tend to increase the cost logarithmically as the reheat steam temperature approaches 950 F.

Reducing the hot steam reheat temperature would tend to decrease the capital cost below that of the reference plant. However, this direction would involve increased heat rates and rapidly rising fuel costs.

4. Conclusions

(a) Although the effort expended to investigate the difference between a sodium-heated and a steam-heated reheater was somewhat limited, the results were in the direction expected, i. e. , steam reheat is more expensive than the sodium reheat.

(b) Because of the cost and technical uncertainties associated with either sodium- or steam-heated reheaters, it appears that the conclusion above is valid and that further engineering studies are not warranted at this time.

Table A-1. Economic Evaluation

	Sodium-steam reheater	Steam-steam reheater
Reactor output, Btu/hr $\times 10^9$	8.37	8.37
Hot reheat temperature, F	950	850
Gross turbine-generator output, kWe	1,033,926	877,500
Net heat rate, Btu/kWh	8096.5	9550.0
Output difference, kWe	Base	-156,426
Heat rate difference, Btu/kWh	Base	+1453.5
Fuel cost, mills/kWh	Base	+0.135
Direct installed construction cost difference, \$		
Feedwater heaters	Base	-480,000
Piping, valves, etc.	Base	+85,000
Turbine-generator unit	Base	-1,240,000
Pumps, condenser, etc.	Base	+150,000
Steam generators	Base	+2,400,000
Reheaters	Base	+1,800,000
Total	Base	+2,715,000
Total capital cost increase, \$	Base	+3,800,000
Fixed charges, mills/kWh	Base	+0.076
Fuel cost + fixed charges, mills/kWh	Base	+0.211

Figure A-1. Heat Balance for Live Steam Reheater
($T_{HRH} = 850 \text{ F}$)

Legend: $\# = \text{lb/hr} \times 10^6$ $F = ^\circ\text{F}$ $P = \text{psia}$
 $B = \text{Btu/hr} \times 10^9$ $h = \text{Btu/lb}$

6. SHIPPING CASK TRADE-OFF STUDY

CONTENTS

	Page
1. INTRODUCTION	1
1.1. Purpose	1
1.2. System Requirements	2
1.3. Summary	3
2. CONCLUSIONS AND RECOMMENDATIONS	4
3. DISCUSSION	6
3.1. General	6
3.2. Fuel Assembly Design	8
3.3. Coolant Performance Capabilities	17
3.4. Fuel Handling	20
3.5. Economic Study	27
4. RESULTS	42
4.1. Discussion of Results	42
APPENDIXES	
A. Thermal Analysis Model, Failure Criteria, and Assumptions	A-1
B. Hypothetical Accident Conditions	B-1
C. Fuel Cleaning Costs	C-1
D. Inventory Charges for the Shipping Period	D-1

List of Tables

Table

1. Mechanical Comparison of Various Fuel Designs	9
2. Heat Generated in a Fuel Assembly	13
3. Minimum Storage Times for Fuel Assemblies Shipped in Various Coolants	18
4. Shipping and Inventory Costs as a Function of the Number of Casks	37
5. Total Costs — Spent Fuel Shipping and Inventory	38

List of Figures

Figure	Page
1. Decay Heat Following Infinite Reactor Operation	14
2. Fuel Assembly, Cask, and Shield Assembly	15
3. Number of Shipping Casks Required Vs Shipping Period	32
4. Cost of Shipping Casks Vs Number of Casks Required	33
5. Costs Vs Number of Shipping Casks	35
6. Total Fuel Cost Vs Spent Fuel Storage Time	40
7. Doubling Time Vs Spent Fuel Storage Time	41

1. INTRODUCTION

The objective of the National Sodium-Cooled Fast Breeder Reactor (LMFBR) Program is to develop a safe, reliable, and economically competitive power plant for the utility environment. Fuel cycle costs are a major item in the economics of this plant, and a very important part of this item could be the cost of shipping the highly irradiated fuel. In the past, this shipping cost has represented a rather small fraction of the total, but with the LMFBR fuel, this may no longer be true. Private fuel ownership; new fuel designs; and higher burnup, radiation levels, and decay heat rates may change this cost fraction considerably. The whole shipping question must be reappraised to ascertain what these costs are, and if necessary, how they can be reduced to be compatible with the objectives of the National Program. As part of this reappraisal, the type of cooling medium—liquid metal or other—will be studied to determine which coolant has the least potential for safe, economical shipment of LMFBR fuels.

1.1. Purpose

This trade-off study is being undertaken to establish the technology necessary to make the shipment of highly irradiated, spent LMFBR fuel safe, reliable, and economical. Since safety considerations are paramount, the maximum accident conditions of transport will be assumed, and the economic penalties (for safe shipment) to the nuclear fuel cycle will be determined. In the process of determining the requirements for safe and efficient shipment, four major design areas will be integrated. These are containment, criticality prevention, shielding, and heat removal. Although it is impossible to separate any of these from the others, major emphasis will be given to the heat removal aspect of design.

For this study, shipping is defined as all those operations on or with the fuel that may be required from the time it is removed from the

reactor to the point at which it enters the fuel reprocessing cycle. Therefore, the study considers the fuel design, decay storage, handling, inspection, cleaning, disassembly, cooling, packaging, and fuel reprocessing as well as the actual transport of the fuel. The shipping technology will be established by optimizing all the parameters with full appreciation of safety, reliability, operating performance requirements, and capital and operating expenditures.

1.2. System Requirements

Neither cask design, coolant medium, nor handling procedures were established in advance of this study. Consequently, to minimize any prejudice toward the results, the design requirements are limited to the following minimum number of broad-based criteria that must be met in handling and shipping irradiated radioactive fuel.

1. The current 1000-MWe LMFBR fuel assembly and the associated functional parameters will be used as a base reference for this study.

2. The reactor site facilities must provide for safe, reliable handling of fuel assemblies prior to shipment, regardless of the coolant medium employed during shipment.

3. Wherever possible, the shipping cask must be designed to existing codes and standards. The cask should meet the requirements for either truck or rail shipment if possible.

4. Sufficient shielding material must be provided to retain gamma exposure dose levels within the limits prescribed in AEC regulations. The cask and transporter must have a cooling system with sufficient backup power sources to maintain the temperature of the shielding material sufficiently low during normal conditions to prevent reduction of the shielding effectiveness or the loss of such material due to vessel failure. The shielding material itself must be selected so that even transport accident conditions (where forced cooling is lost) will not result in the loss or reduction of shielding effectiveness.

5. The shipping cask must sustain the worst accident conditions of transport, as specified by Federal Shipping Regulations, 10 CFR-71, "Packaging of Radioactive Material for Transport" (Appendix B).

6. The maximum fuel cladding temperature shall not exceed 2350 F following the worst accident conditions.

7. The coolant medium that provides the lowest total shipping costs, including outpile inventory charges, will be selected as the reference method of shipment. Costs associated with reprocessing and in-containment handling will not be investigated in detail in assessing shipping costs.

1.3. Summary

The results of this study have established that dissipation of decay heat from the spent fuel assembly is the major area of concern in handling and shipping LMFBR fuel. This problem is more severe than in current shipments of thermal reactor fuel because of the higher rate of heat generation from the LMFBR assembly and the economic penalty to the nuclear fuel cycle and doubling time due to a long reactor-site decay storage period.

Economical shipment of the LMFBR fuel can be attained, as shown in this report, if the cooling medium in the shipping cask is sodium, and if no loss-of-coolant accident must be postulated. The objective can be attained by giving special attention to the following areas:

1. The design of the shipping container to retain the sodium coolant even in the worst transport accident.
2. The design of the reactor-site procedures and equipment for handling and packaging the fuel assemblies which have decayed for only the minimum period.
3. The design of the reprocessing head-end procedures for unloading, disassembly, and removal of the sodium coolant.

2. CONCLUSIONS AND RECOMMENDATIONS

1. The dissipation of decay heat from the spent LMFBR fuel assembly (100,000 MWD/T burnup) has been confirmed as the major shipping problem. To accomplish safe and reliable shipment within one year of the time of discharge from the reactor, only one fuel assembly can be shipped in each cask.

2. Safe shipment of one assembly appears possible after at least 70 days of decay storage if the assembly is sealed into a sodium-filled inner cask container. No circulation of the sodium is required. A secondary cooling system (i.e., nitrogen-to-air) should be provided to cool the cask shielding during normal transport conditions.

3. The shipping cask must be designed to sustain the worst accident conditions of transport, as specified by federal shipping regulations (see Appendix B). This accident could result in the following package conditions and design requirements on the cask:

- a. A maximum fuel cladding temperature of about 2350 F (vented fuel design, only following the worst accident).
- b. The inner cask container must be designed to retain the primary sodium coolant completely.
- c. The inner cask container must maintain its integrity under internal pressures approaching 300 psia and temperatures of approximately 2000 F.
- d. The cask must contain the lead (or alternate) shielding in such an enclosure that molten lead cannot escape the cask or reduce the shielding effectiveness below allowable levels.

4. The shipping cask should be designed for truck shipment if possible.

5. The reactor-site procedures and equipment for handling and packaging fuel assemblies must be designed and fully developed to perform the fuel transfer functions safely and reliably. Here, too, the critical operation is cooling the assembly. A sufficient supply of coolant and redundant safety backups must be provided.

6. Reprocessing head-end procedures for unloading, disassembly, and removal of the sodium coolant must be developed. The recovery techniques themselves may need further adjustment to allow for handling the high radiation levels and the high decay heat from the short-time cooled fuel. The economic gain resulting from reduced decay storage time at the reactor site would be offset by a requirement for lengthy reprocessing site storage.

7. The shipment of spent fuel in a cask with water or any other coolant that has a high vapor pressure does not appear feasible unless at least one year of decay storage can be justified.

8. Shipment in a cask with gas cooling is also feasible, but this, too, is possible only after a very extensive storage period or if forced circulation could be ensured.

9. The economic comparison completed for this study shows that inventory charges are the overwhelming factor. It is highly recommended, therefore, that all procedures that are not absolutely necessary be eliminated from the reactor site.

3. DISCUSSION

3.1. General

An essential operation in the LMFBR nuclear fuel cycle is shipment of the highly irradiated spent fuel from the reactor to a reprocessing facility. Safety and economy are the two major considerations for this shipment. These factors must be considered jointly in studying the LMFBR fuel design, the handling procedures, and the actual transport, in order to establish the methods and equipment necessary for safe, reliable, and economical shipment.

Safety considerations require that adequate protection from radiation, avoidance of accidental criticality, and reliable containment of radioactive materials must be effective for both normal and accident transport conditions. To ensure that these requirements are met, the characteristics of the fuel shipment must be established and the effects of higher decay heat rates and levels of fission product radiation must be clearly understood.

Economic considerations require optimization of the equipment and procedures established for safe fuel shipment so that the effect on other fuel cycle costs can be kept to a minimum. To accomplish this we must study the questions of fuel inventory, decay storage at the reactor site, and handling procedures. Once again, high decay heat rates and high radiation levels affect the decision.

Normally, when irradiated fuel assemblies are removed from the plant they have been cooled for several months to allow the gamma activity and the decay heat generation rate to diminish by natural decay to levels that are relatively easy to handle. With LMFBR fuel, the storage time must be minimized to reduce the inventory charges. A reduced cooling period puts more demanding requirements on the fuel handling system, however, which must then handle increased heating loads and

higher radiation levels. Therefore, fuel handling must also be factored into the optimization.

Next, fuel shipment can be considered. Here the fuel design must again be taken into account along with the increased gamma activity and the higher heating loads of LMFBR fuel. Can the fuel be shipped? The regulations set up by the Interstate Commerce Commission¹ and the Atomic Energy Commission² for shipment of solid radioactive fuel define the normal and abnormal shipping conditions. Using these criteria as the limits, the cask design and cask cooling system must be studied. How much fuel can be shipped in a cask? Should this fuel be assembled or disassembled? Should the cooling medium be sodium, water, or another material? We must also consider the fuel handling capabilities of the reprocessing facility. Assuming that the fuel has arrived successfully at the facility, what complications are presented, and how do they affect costs? These questions must be answered and factored into the overall evaluation.

To meet the objectives of the National Program, the LMFBR fuel must be handled, shipped, reprocessed, fabricated into new assemblies, and recycled in an LMFBR plant in the safest and most economical manner possible. The procedures and details of these steps are to be established. A comprehensive study requires consideration of many design areas in order to provide guidance and direction to a solution of the whole shipping question—both handling and preparation procedures and the shipping cask design; three assumptions were made:

1. Private fuel ownership.
2. Off-site reprocessing.
3. Shipment of B&W reference design vented fuel.

These assumptions were made to narrow the study but with full realization that the conditions set probably impose the maximum problems on fuel shipment. The first, private ownership, emphasizes a decreased reactor decay storage period; thus, the cask design problems are at a maximum. The second assumption, off-site reprocessing, essentially eliminates the possibility of administrative control over a shipment, eliminating the presence of skilled operators who would otherwise be on hand to actuate backup safety or cooling systems. The third assumption,

vented fuel as proposed by B&W, was made primarily to narrow the study decision. Many of the possible LMFBR fuel designs and their effects on the handling and shipping are discussed.

3.2. Fuel Assembly Design

The design of the LMFBR fuel assembly must be studied in combination with the irradiated fuel characteristics in order to establish shipping requirements. Three particular values are highly dependent on the characteristics of the irradiated fuel and must be determined: (1) the problem of criticality, (2) the decay heat rate, and (3) the radiation levels involved.

At this time, the final design of the fuel has not been established. Therefore, in order to determine the values required for this study, the B&W reference fuel design will be assumed. Before proceeding with this evaluation, however, the possible fuel designs are reviewed to establish the mechanical aspects.

3.2.1. Fuel Assembly Design — Mechanical

The possible fuel pin and assembly designs are reviewed to aid in establishing the fuel shipping requirements. The size and configuration of the fuel element can have major effects on the cost of shipping spent fuel. A qualitative analysis is given here.

The fuel for the LMFBR will most likely be assemblies of small fuel pins (about 1/4 inch in diameter) arranged on a triangular pitch to form hexagonal bundles. The final size of these bundles will probably be in the range of 5 to 10 inches across the hexagonal flats. Although there are many variations, the bundle will have pin supports to maintain the pin spacing and top and bottom grid plates, tied together by a stainless steel wrapper, to form a "canned" fuel assembly. The top of this assembly will terminate in a pickup adapter and the bottom has a supporting flow nozzle arrangement.

The physical size and weight of this assembly are very important to its shipment. Most important is the length of the fuel to be shipped. For this qualitative comparison, fuel shipping costs will be assumed as roughly proportional to the length of the fuel shipment. This assumption may not be very accurate, but it serves to show the

effect of assembly length. The assembly lengths of various designs are compared in Table 1. Assumed lengths based on B&W studies are used here to show relative differences, which can be helpful in considering the cask design and in estimating shipping costs. Actual assembly lengths are highly dependent on the particular reactor design.

Table 1. Mechanical Comparison of Various Fuel Designs

<u>Fuel type</u>	<u>Assembly length, ft</u>
1. Non-vented	11
2. Porous vent plug	8
3. Diving bell vent	11
4. Combination of 2 and 3	9
5. Pressure relief valve	9
6. Vent to cover gas	25

Also important for this review is consideration of the pin design; three basic designs are being studied: (1) non-vented, (2) vented to sodium, and (3) vented to the reactor cover gas. The question of vented-versus-non-vented fuel, and which is the preferred design, is a very large one and will be covered by a separate study, but the problems of shipping should be discussed.

3.2.1.1. Non-Vented Fuel Pin Design

The non-vented pin, common to current thermal reactors, contains a fission-gas-collection chamber, or plenum, as an integral part of the sealed fuel cladding. The additional cladding length required for this gas collection increases the shipping length, and therefore the cost, if it is not removed before shipment. The stored gas also increases the hazard of handling the irradiated fuel at the reactor site, during transport, and at the reprocessing site. Each highly irradiated LMFBR fuel pin will contain potentially lethal doses of fission gases, and some failures must be postulated because of the reduced cladding properties.

From the shipping viewpoint, the non-vented pin design has several disadvantages:

1. Failures during transport or handling must be postulated because of the doubtful integrity of the cladding after long reactor exposure. If the fuel is shipped without "canning" or without releasing the gases, a heavy burden is placed on the shipping cask design and on the reprocessing site, both of which must consider some release of the fission gases into the shipping container.

2. The increased length of the fuel cladding, probably approaching the active fuel length or in the range from 3 to 5 feet, penalizes all of the reactor and reprocessing site handling equipment as well as the shipping cask. Assuming a total shipping cost proportional to the length of the assembly being shipped, this penalty could be nearly a factor of two.

3. The shipment of fission gases could increase the shielding requirements of the cask.

4. The ingress of sodium coolant into failed pins adds potential hazards if cleaning or water shipment is used.

5. As is shown later in this report, the non-vented fuel cannot be shipped after the minimum decay storage time that appears possible with the vented design. The higher temperature, considered allowable in the transport accident case with vented fuel, would result in cladding failure in the non-vented designs.

3. 2. 1. 2. Vented-to-Sodium Fuel Pin Design

Several approaches to vented fuel design have been proposed in the LMFBR studies and in other studies of sodium-cooled reactors. The object here is to relieve the buildup of high fission gas pressure without permitting ingress of sodium into the fuel, by venting the fission gases directly into the sodium coolant. Both individual pin venting and collective assembly venting concepts have been proposed. The B&W reference design follows the individual-pin-venting approach and specifies the assembly of a porous plug device in each pin. The increased fuel pin length causes very little penalty in shipping costs;

the B&W designs allow only a 2-inch gas space between the upper blanket material and the vent plug.

Some of the other approaches to venting are: (1) a diving-bell device, (2) a combination of diving-bell and porous plug, and (3) a pressure relief valve. These designs have various advantages, disadvantages, and effects on shipping length (summarized in Table 1). The exact penalty is difficult to establish without detail design. The pressure relief valve approach, however, is considered to have the greatest problems because of the difficulty of reliable operation in sodium.

Again from the shipping viewpoint, two failure modes are postulated for vent-to-sodium devices:

1. Plugging of the vent and a resultant internal pressure and collection of fission gases. The probability of this occurrence is considered very low, however, except with the relief valve design.
2. Failure of the vent to the extent that ingress of the sodium coolant has occurred. This failure mode has been postulated as possible in every fuel pin design we examined.

3. 2. 1. 3. Vented-to-Reactor Cover Gas Fuel Pin Design

The third basic fuel pin design that has been considered is the venting of fission gases directly to reactor cover gas. Individual capillary tubes could be added to the fuel pins to allow constant egress of the gases. Alternately, the assembly could be provided with a collection manifold and one vent tube. Both approaches would complicate the subassembly design as well as the fuel handling problems. In some reactor concepts the length of the fuel could be penalized by as much as 20 feet in additional length; this would greatly increase fuel shipping costs if no disassembly is planned. However, this design does offer what is probably the best insurance against both pressure buildup and sodium leakage into the fuel pins.

3. 2. 2. Fuel Assembly Characteristics

To determine the fuel characteristics as they pertain to shipping, we will now consider the B&W reference fuel design.³ This

design is an assembly of 331 pins arranged on a 0.340-inch pitch and assembled with pin supports, a stainless steel wrapper, and end fittings. The overall assembly length is approximately 8 feet, and the fuel pins are individually vented using porous plugs. The end of life (1005 full power days) corresponds to a core burnup of 100,000 MWD/T. A detailed description will be included in the Follow-On Study Task 2 report.

3.2.2.1. Criticality

To establish the limitations to shipping this reference fuel, the question of avoiding accidental criticality was investigated first. Calculations to determine criticality during fuel handling and storage situations have been completed and were used in this study to determine the effect of a major transport accident.⁴ The results of the calculations showed that a single reference fuel assembly stored in either water or sodium will be subcritical. They also showed that assemblies stored on fixed 12-inch centers in a double-row slab geometry are also subcritical. The calculations were conservative. Applied to the shipping question, criticality of one B&W reference fuel assembly (and possibly more if held on 12-inch spacing or packed with poison material) appears to present no problems, even in transport accident conditions. The only qualification is that the shipping cask must afford enough protection against impact so that the fuel pins in an assembly cannot be separated into a critical configuration. This reservation holds for water shipment or an accident where water could flood the container, but not for sodium shipment. The conservatism of the calculations was sufficient to determine, also, that one assembly in a shipping container was also subcritical even if completely melted into a compact mass.

The possibility of shipping disassembled fuel pins must also be considered. Because of the decay heat dissipation problem, which will be discussed later in this report, disassembly of the fuel bundle at the reactor site should be considered. The reference fuel assembly here is essentially dry or an undermoderated assembly. Caution must be used in disassembly, especially in water, and precautions are necessary to prevent criticality. Considerable effort on the subject has been reported by Battelle Northwest Laboratories.⁵⁻⁸ To

avoid a criticality hazard in disassembling FFTF fuel assemblies, they are considering three methods: (1) mechanical devices to ensure safe geometry, (2) controlled removal of pins from a bundle, and (3) the addition of a soluble neutron poison (boron) to the cell water. Battelle's efforts exemplify the problems of disassembly, but it should be remembered that the difficulties will exist regardless of where the disassembly is done—reactor site or reprocessing site.

3. 2. 2. 2. Decay Heat Rate

Dissipation of the decay heat being generated within the reference fuel by the decay of fission products will be a critical operation in shipping spent LMFBR fuel. As we have already discussed, it will become even more critical as the decay storage period at the reactor site is reduced to obtain the optimum fuel cycle costs. Several fuel shipment possibilities must be considered: (1) one fuel assembly, (2) more than one assembly, and (3) disassembled pins (partial, one assembly, or more than one assembly).

The first step in our study was to establish the heat generation rate of a single B&W reference fuel assembly. Five reactor site decay storage periods—10, 30, 90, and 120 days and 1 year—were selected, and the fraction of operating power was obtained from curves produced in pressurized water reactor studies⁹ and reproduced here as Figure 1. This curve gives the fraction as a function of reactor shutdown for an infinitely exposed fuel assembly. The LMFBR reference design data, using appropriate peaking and hot channel factors, was then used to calculate the decay heat rate. The values for the decay storage periods selected and the relative rates are tabulated in Table 2; these compare favorably with the values given by other available reference curves, including that given in the LMFBR National Program Plan.¹⁰

Table 2. Heat Generated in a Fuel Assembly

<u>Decay storage time, days</u>	<u>Maximum heat generation rate per ft of length, Btu/hr-ft</u>	<u>Decay heat rating, kw</u>
10	50,200	41.6
30	34,970	29.0
90	21,280	17.7
120	18,240	15.1
365	10,330	8.6

Figure 1. Decay Heat Following Infinite Reactor Operation

Figure 2. Fuel Assembly, Cask, and Shield Assembly

Note: Measurements given in inches

3.2.2.3. Radiation Level

The nuclear radiation level will be important in establishing the shielding requirements for all the fuel handling procedures and for the shipping cask. The limitations of the AEC regulations were used to establish the cask requirements. The main provisions of Part 71 are summarized briefly below:

1. The gamma exposure dose under normal conditions of transport must not exceed any of the following (assuming exclusive use of vehicle):

- a. 1000 mr/hr at 1 meter from the surface of the cask.
- b. 200 mr/hr at the outer perimeter of the vehicle.
- c. 10 mr/hr at 2 meters from the vehicle.
- d. 2 mr/hr in the driver's compartment.

2. The cask must remain subcritical at all times under the worst conditions of configuration and moderation.

3. Primary coolant (that which is in contact with the fuel) may not circulate outside the cask.

4. Under accident conditions, the gamma dose level is permitted to increase by a factor of 100, but no release of radioactive material from the cask is permitted, except for gases and contaminated coolant.

5. The cask and its closure must be rugged enough to prevent release or reduction of shielding in the event of impact or fire.

Based on preliminary calculations, the shielding requirement for one B&W reference fuel assembly is 8 inches of lead to meet these regulations. If only one fuel assembly can be shipped in a cask (see section 3.3), the resulting overall shipping cask weight will be within the maximum allowed for truck shipment, even including circulating equipment. This is an important economic consideration.

3.3. Coolant Performance Capabilities

A thermal analysis¹¹ of a B&W reference fuel assembly has been carried out under varied shipping conditions to (1) establish the most desirable coolant for the spent fuel assembly shipment, and (2) to establish reactor on-site decay storage time requirements for safe shipment in various conditions (circulation failure, loss of coolant, and a half-hour, 1475 F fire). Three cooling media were emphasized (argon gas, sodium, and water), although alternate coolants were also considered.

The model chosen for this study was limited to a single fuel assembly within each shipping cask. The analysis of this situation was considered thorough enough to supply the information needed to make the choice of the cask coolant. The model, failure criteria, and assumptions are given in Appendix A. The design criteria did not evolve from an established cask design, although they certainly impose design requirements on the cask; e. g., the cask must be designed to maintain its integrity under the impact, temperature, and pressure conditions.

The maximum cladding temperatures of a fuel assembly were computed for the following conditions:

1. Loss of all coolant from the cask and the fuel assembly in the horizontal position.
2. Loss of coolant circulation (stagnant coolant) with argon, water, and sodium as coolants.
3. A one-half-hour fire as described in the AEC regulations,² reproduced in Appendix B.

The crux of the shipping problem is to choose a coolant that will permit the safe shipment of spent fuel from the reactor site after only a minimum time in storage for the decay of fission products and generated heat. Prolonged on-site storage would penalize the nuclear fuel cycle costs and hence the power generation cost. The coolant and the resulting storage time must still be compatible with the handling and cooling procedures used in the transfer operations.

The results of the thermal analysis are shown in Table 3, and Table 2 gave the heat generation rates used. The analysis considers that emergency cooling is impossible after the accident; therefore, the

cask equilibrium condition will be the controlling condition. The rate of change of the temperatures of the cask and its contents would necessitate detailed knowledge of the cask design and detailed analysis techniques.

Table 3. Minimum Storage Times for Fuel Assemblies Shipped in Various Coolants

Coolant	Storage time required to prevent failure, days		
	Circulation failure	Loss of coolant	1/2-hr, 1475 °F fire, circulation failure
Argon	240	240	>>240
Water	240 ^(a)	240	>>240
Sodium	22	240	70 ^(b)

(a) Any water vapor formed must be vented from the cask to limit cask pressure.

(b) Primary sodium coolant is maintained.

It can be seen in Table 3 that sodium has the best thermal performance; this can be attributed mainly to its high conductivity and low vapor pressure. From the results we can see that shipment may be accomplished safely, even with loss of circulation and with the postulated fire condition. This benefit can be realized only if the cask is designed to ensure against loss of the coolant sodium. Since the vapor pressure of sodium at the maximum allowable cladding temperature of 2350 F is 217 psia, the pressure buildup does not jeopardize the cask coolant containment.

The high vapor pressure of water is definitely its limiting characteristic. The water temperature corresponding to a saturation pressure of 500 psi is 467 F. Therefore, the internal pressure in a sealed container would quickly jeopardize the integrity of the cask containment. This fact requires that the cask be vented to minimize pressure buildup. The shipment must immediately face a boil-off situation, which forces

the water shipment to be governed by the total-loss-of-coolant accident. This is the main reason that water cooling is so much less desirable than sodium. The water shipment is especially complicated by the venting requirements, since the designer must postulate the release of fission gases, even during normal transport, and must provide vent tanks for their collection. The accident condition, which would result in high temperatures, would add unnecessary volume to the vent tanks and possibly a large additional cost to the cask.

Alternate cooling media that might be used in the shipping cask for LMFBR fuel have been studied less extensively. Many of the more commonly considered compounds suffer breakdown or other problems associated with the temperatures involved. They may also introduce handling problems that are detrimental to the overall system. In general, the vapor pressure for the coolants considered fell somewhere between sodium and water, and this fact, coupled with the problems introduced by the use of a new compound, makes these alternates appear undesirable. We also briefly considered metal powder and other forms of metal that would reduce the possibility of coolant loss. In these cases, most metals would be in their melting range at the temperatures involved, and the possibility of loss would be similar to that with sodium. The conductivity of powders or balls is less than that of sodium, and again sodium appears to be best.

The results of this thermal and hydraulic study definitely favor sodium as the coolant for the spent fuel shipping cask. We must be aware of the requirements imposed on the cask design, however; it must be designed to

1. Contain the sodium under the stipulated impact and fire conditions.
2. Maintain its integrity under internal pressures approaching 300 psia and temperatures of about 2000 F.
3. Contain the lead shielding in such an enclosure that molten lead cannot escape from the cask. (Although not a part of this study, it is recognized that depleted uranium has been proposed for use in cask shielding; consideration of this material could eliminate the melting problem.)
4. Handling and shipment of one spent fuel assembly only.

3.4. Fuel Handling

This discussion will cover LMFBR fuel handling procedures at the reactor site and at the reprocessing site as they relate to the shipment of fuel. The possible facilities, equipment, and procedures must be reviewed so that a better picture can be formed for the economic comparison of methods. We must consider again the design of the fuel pins and fuel assembly, as well as the ever-present problem of cooling the spent assemblies.

3.4.1. Reactor Containment Handling

Handling within the reactor containment building affects the fuel shipping question, or more correctly, the high decay heat rate that results from a reduced decay storage period (to obtain the most economical fuel shipment) places limitations on handling procedures.

In every LMFBR proposal to date, storage of the spent fuel assemblies has been planned in either storage racks located right in the reactor pot sodium or in a separate—but still sodium-filled—vessel located as close to the reactor vessel as possible. The first handling procedure, then, is to accomplish this transfer, commonly included in the description of the reactor refueling procedure.

The high heat generation rate within the fuel puts severe time limits on this transfer procedure, and designers have chosen methods that either keep the fuel constantly submerged in sodium or drastically limit the handling time when only gas cooling is available. Our studies have substantiated the need for concern in this procedure. Based on the results of our heat generation studies for the B&W reference fuel assembly and coolant performances (see section 3.3), and on a review of extensive cooling studies carried out by Battelle Northwest Laboratories for the FFTF element,¹⁶ cooling during this transfer will be critical. The B&W assembly would be generating as much as 75 kw of heat (12 hours after reactor shutdown). The thermal and hydraulic characteristics of the reference assembly were analyzed to determine whether adequate cooling could be achieved with only natural circulation in a sodium coolant. The results showed that cooling would be adequate with a maximum fuel cladding temperature of 975 F, assuming a pool temperature of 800 F.¹⁷

An analysis was also performed to obtain similar data for the reference fuel assembly being handled with gas cooling.¹⁸ The time after shutdown used in this study was roughly the same as that in the sodium calculations. The study showed that nearly 3000 pounds of argon gas flow (at 100 F inlet and 2.1 psi pressure drop) per hour was required to maintain a maximum clad temperature of 1000 F. In the event the gas coolant flow was lost, the maximum rate of change of fuel temperature was computed as 6.3 F/sec. Immediate action would be necessary to prevent fuel and cladding meltdown.

The second handling procedure required in the reactor containment building is to transfer a spent fuel assembly from the decay storage pool to some area outside the containment; the development of a safe and reliable method for this transfer is essential. An adverse effect on economics would result if the heat load capacity of the procedure is less than that required for shipment. The heat loads, as a function of time after reactor shutdown, were given in section 3.2.2.2. These loads can be compared with the cask coolant capabilities given in section 3.3 to help in determining the difficulty of this problem.

If shipment of one fuel assembly in sodium is selected, the required storage time is approximately 70 days. The resultant heat load is about 20 kw for the hottest assembly. At this point, transfer could be made in a container of sodium or with gas cooling. Either method appears to be acceptable if sufficient backup systems are provided to ensure heat removal. Even with the failure of coolant circulation, the temperature of the fuel assembly in the sodium would remain below the cladding melting point. The gas-cooled transfer would be more dependent on backup cooling, and meltdown could occur with failure.

If shipment of one assembly in a water-cooled or gas-cooled cask is considered, a storage time approaching one year is required. In this case, cooling is still required for normal circumstances of transfer, but with a heat load of about 8.6 kw, the critical reliance on backup cooling and the consequences of failure would be reduced.

3. 4. 2. Handling Outside the Containment

All of the procedures outside the containment area are very important to the shipping question and to the design of the cask. It appears that very little concentrated effort has been devoted to establishing the handling procedures and storage and equipment requirements for LMFBR fuel after it has been moved into this area from reactor storage. We know that the transfer into the area must be accomplished through a valving arrangement which maintains complete isolation of the reactor containment, and that the assembly will still be wetted with sodium, and that constant cooling must be maintained to prevent fuel damage. We must assume that we have vented fuel or possibly failed cladding. From this information we must assume that this area is either a hot cell with an inert gas environment or that some sort of sealed intermediate containers are provided. In either case, a holdup storage tank in the area appears desirable. This storage tank would be filled with water, sodium, or some alternate liquid dictated by the cooling medium used in the shipping cask and by the required shipping preparation procedures. In addition to the storage tank, the area will probably contain all the facilities, equipment, and tools needed for the preparation procedures, i. e., cleaning, disassembly, canning, and cask loading.

In looking at this out-of-containment handling and storage, we must consider again the fuel assembly design. The length of the fuel assemblies will affect the cell size, the minimum depth of the storage tank, and the equipment used for all the prescribed operations. Radiation shielding must be provided for all operations, and gas purifying might be required to handle large quantities of fission gases.

The question of cleaning sodium residue from the fuel assembly is directly related to the cooling medium used in the shipping cask and also to the fuel pin and fuel assembly design. If shipment in a sodium-filled cask is selected, no cleaning is necessary unless some preparation procedures or possibly inspection procedures require it. If cleaning becomes a requirement, then special procedures should be developed for the sodium shipment case where cleaning agents would be compatible with the sodium. For this study, we will assume that inspection is not required at the reactor site. We feel this assumption is justified for two reasons:

1. The LMFBR plant is a utility-owned, power-producing station of a proven design. Experiments or other testing are not being carried on, and if periodic inspection of the fuel is required, it will be sub-contracted to a laboratory specializing in this type of work.

2. From the fuel cycle cost viewpoint, the decay heat load on the handling equipment at the time of fuel removal will be too great to justify the equipment required to perform these inspections.

If shipment in some other cooling medium is selected, cleaning will depend on its compatibility with sodium. It should be pointed out that almost all the alternate cooling media that we have observed are opaque. This point is made because inspection or other preparation procedures, if required, become more difficult when opaque solutions are considered.

If shipment in water is selected, the fuel assembly must obviously be cleaned of all sodium residue under controlled conditions. One cleaning procedure before water shipment is well known; this is the steam cleaning procedure, which is fairly highly developed at this time. Of course, we must evaluate the procedure in detail for the highly irradiated and relatively very hot spent LMFBR fuel. Further handling or preparation procedures could be accomplished under water after this cleaning. In many cases, proven thermal reactor operations could be used. The major disadvantage to the use of water is the increased likelihood of boiling—procedures must consider this possibility.

As we have already mentioned, the design of the fuel pins and fuel assembly also affects the question of cleaning. In the case of the vented designs, preshipping preparation requirements could make cleaning mandatory. This is especially true for the long vented designs where cutoff is probably required. It may also become a requirement for shorter designs if disassembly of the bundles is required for shipping. Such operations as disassembly, sealing of vents, or "canning" an assembly have generally only been considered as water pool procedures. Once again it should be pointed out that the heat load from the short-time-cooled assembly will complicate these procedures. Referring to section 3.3 again for the coolant capabilities, the B&W reference fuel assembly could be shipped in a sodium cask after about 70 days'

decay time. Cleaning or disassembly operations may very likely extend this required decay time to one year.

3.4.3. Shipping Cask Loading

The normal procedure for loading a shipping cask has been to lower the cask, or at least an internal container, directly into a water pool and transfer the fuel under water. For shipment of the LMFBR fuel in a water-filled cask, this procedure could be followed. Recall again that water shipment requires a decay storage time of about one year and that cask cleanup complications may arise with vented fuel designs; these problems should be considered.

In the case of shipping in a sodium-filled cask, similar loading techniques should be considered. For this case a sodium pool would be required instead of the water pool; however, generally, the system would probably have superior cooling capability. Of major importance would be the enhanced safety obtained by almost-constant cooling in sodium, where (as shown earlier) natural convection is adequate to maintain safe fuel cladding temperatures. However, provisions for the sodium pool and the associated support equipment would probably involve a considerable penalty in capital costs.

A possible alternate procedure involves the transfer of a fuel assembly from the reactor containment directly into a shipping cask or an inner container already filled with sodium. Specifically, this procedure features an inner cask container which is raised from the cask into an intermediate inert-gas loading cell. The container is filled with sodium at this point, then the fuel assembly is loaded, and the container is welded closed. The outside of the container is cooled by an inert-gas system during loading and while the container is again lowered into the cask body. The cask head is then attached, and the cooling system in the shielding provides the required cooling while the cask is loaded onto the transporter. Loss of coolant flow must be prevented by backup systems, and sodium spillage (either partial or complete) cannot be tolerated; complete loss of sodium would result in meltdown in a short time. This method appears to remove the cost penalty that might be associated with the sodium shipment and reactor-site handling procedures of the first method. It also offers a saving over the

method, provided the cleaning procedures are not made a plant requirement for on-site inspection reasons.

In either sodium procedure discussed above, the internal container closure and the design of the cask (to positively prevent loss of sodium from the container) become the critical areas. Because loss of coolant from this container would result in meltdown of the 70-day-cooled assembly, this container and the shipping cask must withstand the maximum transport accident. Multiple shells and positive closures will probably be required.

3.4.4. Handling at the Reprocessing Site

During our discussions on fuel design and reactor site handling, some consideration has been given to handling at the reprocessing site. Some review of these areas will be included in this section along with a brief review of current reprocessing facilities for Light Water Reactors (LWRs).

Safe, reliable, and economical fuel shipment must be attained. The reprocessing facilities are now located an appreciable distance from the nuclear reactor site, and shipping regulations are designed to insure safe and reliable shipment. An additional burden on the shipping cask is to insure the arrival of the fuel in an undamaged condition, so that handling and reprocessing can proceed with a minimum of delay. The LMFBR fuel will arrive at the site contained in a cask which is most probably filled with some cooling medium. The fuel must now be unloaded at the facility and processing must begin without excessive delay. Important here, too, is the almost continuous cooling requirement for the highly irradiated fuel.

The primary task of the reprocessing plant is to recover unexpended fuel. This recovery is now accomplished by the aqueous solvent-extraction "Purex" processes. The fuel materials are dissolved in a nitric acid solution, from which uranium and plutonium are recovered and separated from the fission products by solvent extraction. The products of this operation are nitrate solutions of uranium and plutonium. More important for this study is the requirement of head-end preparation procedures at the reprocessing site. The number and particulars of these procedures are naturally dictated by the specific fuel design and by the

degree of fuel assembly at the time of receipt, but generally some sort of mechanical disassembly, a chopping operation, and an aqueous dissolving process to remove excess cladding material are employed. Because these head-end procedures are costly and because they are usually somewhat specialized for the particular fuel shipment involved, it is desirable to keep such procedures to a minimum and as general in nature as possible.

At the present time, LWR fuel shipments usually arrive at the site in a water-filled cask. This cask is submerged in a water pool for unloading and the fuel is stored in the pool until the reprocessing cycle is scheduled. As stated above, the fuel must arrive undamaged to the extent that it can be unloaded and so that it conforms to the requirements of the head-end operations.

We will now consider the specific shipment of LMFBR fuel and the unloading and head-end procedures. This shipment could be in water, and the reprocessing site operations could then follow the current procedures closely. Similarly, the particulars of the head-end procedures would follow current technology and would depend primarily on the configuration of the fuel—assembled or disassembled, the physical size of the components, the temperature, and the radiation level.

If we now consider shipments in alternate cooling mediums such as sodium, the unloading and head-end procedures at the site must be adjusted for handling this coolant. Most important in considering sodium is the possibility of coolant leakage into the fuel and the volume of sodium in the shipping container. In aqueous reprocessing, at least one head-end procedure will probably be required to remove this sodium before the fuel enters the cycle. These procedures will need to be developed to ensure that the recycled fuel would not retain traces of sodium, which might have detrimental or downgrading effects. It should be pointed out that similar problems exist for failed, non-vented fuel designs and for vented designs, even if shipped in water. In both cases, sodium could be logged within the pins. The bulk contaminated sodium in the cask could be returned to the reactor site by resealing the container.

Alternate fuel reprocessing cycles should also enter this study. Two processes are now being developed: pyrochemical and fluoride-volatility. Operation of the General Electric Company's proposed

fuel recovery plant—using a combination of solvent-extraction, ion-exchange, and fluoride-volatility—is scheduled to begin in 1970. However, these processes are generally only in the development stages, and use for the first-core LMFBR fuel, or even later cores, must still be proved. In addition to this uncertainty, it is important to note that all processes, existing and under consideration, have aqueous head-end stages.

3.5. Economic Study

This study concentrates on the handling and shipping procedures required to get spent LMFBR fuel from the reactor to a reprocessing facility in the safest and most economical manner. Four important cost components are considered:

1. Handling.
2. Shipping.
3. Inventory - shipping.
4. Inventory - storage.

As will be seen, inventory costs far outweigh the other costs, but all components will be discussed here to show this.

3.5.1. Handling Costs

The handling requirements for spent LMFBR fuel have been discussed in detail in section 3.4 and are referred to in several other areas. It is quite difficult to put actual dollar costs on many of these procedures when considering conceptual ideas. A qualitative analysis can be performed, however, and we can estimate differences.

Based on results reported in other sections of this report, the two handling systems that will be compared here are those required for shipping the fuel assemblies in sodium and those required for water shipment. (This is not meant to exclude shipment in other mediums but, as we have shown, other shipments approach the water conditions.) Both systems will have some equipment and procedures that are quite similar. Especially fitting this category will be the prime refueling operation: to remove a spent assembly from the core, install a new assembly, and move the spent assembly into the storage area. This procedure depends much more on the reactor design and the refueling equipment than on shipping conditions. Therefore, no costs for

in-containment handling for either shipping method are given in this study; they are assumed to be roughly equal, although detailed study of the systems must be accomplished to confirm this.

The next operation in handling the spent fuel is its removal from the reactor containment building. Equipment and procedures for this may be very different and quite dependent on the mode of shipment. In our appraisal, however, we have assumed that identical gas cooling systems with proper backups would be used. After the proper decay storage time necessary for shipping, gas cooling appears to be the preferred method for this transfer. Again, these costs are not included in handling costs and are assumed to be roughly equal.

The major area of handling that will affect cost is the out-of-containment procedures, once the previous transfer has successfully passed an assembly. Especially expensive may be the equipment and operation for cleaning that would be required for water shipment. This added expense would also exist for any alternate cask cooling mediums where the cleaning of sodium residue is required. Cleaning is probably not required for sodium shipment.

Again, actual costs for a conceptual system cannot be given. The possible magnitude has been discussed in several articles, most recently in regard to the Fermi reactor facility, and the penalty may be large. Based on a comparison of B&W's Concept IV Report¹⁹ with a conceptual idea of a sodium out-of-containment area, we have estimated the cleaning equipment costs as roughly \$250,000. We feel that this estimate is conservative because it includes engineering, development, fabrication, and installation of a pressure-type cleanup can, drive systems, the steam cleaning equipment, the waste disposal and all the related holdup tanks and filters, and the additional cell volume required. Maintenance and operation must also be added to this estimate. We have assumed a factor for this and have arrived at a cost for cleaning spent fuel of \$2.73/kg of U + Pu (see Appendix C).

The handling required at the processing plant must also be considered in determining the overall handling costs. The requirements for head-end preparation procedures are discussed in section 3.4.4. It was pointed out that these procedures are very dependent on the specific fuel assembly design, the disassembly required before

processing, the coolant in the shipping cask, the number of assemblies in the cask, and the type of fuel processing. It was further pointed out that, even with shipment in a water-filled cask, there is a possibility of sodium-logged pins. Based on the scope of this study, it is not possible to determine whether a penalty for sodium shipment exists or what that penalty should be. Therefore, handling costs at the processing plant are not included, except for those that may exist in the equation used in the following economic study.

3.5.2. Shipping Costs

Shipping costs are usually divided into the following component costs:

Container — If the containers are capitalized, this item covers the annual fixed charges on the investment. If the containers are rented or leased, or are disposable, it covers the annual expense thus incurred.

Freight — This covers the freight charges paid to the carrier. It includes the cost of returning empty containers, where applicable.

Handling — This covers the cost of loading, unloading, crane operations, decontamination, testing, and other miscellaneous operations in connection with the shipment.

Insurance — This covers insurance against loss of or damage to the container and its contents during shipment. It is assumed that fuel shipments within the United States will not require liability insurance, because under the Price-Anderson Act, the liability and indemnification coverage provided for the reactor is extended to cover fuel shipments to and from the reactor.

The methods used in estimating these costs generally follow those suggested by Mr. R. Salmon of Oak Ridge National Laboratory.²⁰ The container cost estimating method is different, however, mainly because of two variables that must be considered: the period over which the discharged fuel will be shipped and the number of fuel assemblies in a refueling batch. The following data are required:

1. Total core loading (B&W reference), kg (U + Pu)	64,091
2. No. of assemblies - fuel plus radial blankets (B&W reference)	464
3. Burnup at time of refueling, MWD/tonne (B&W reference)	100,000
4. Decay cooling time (see section 3.3)	
Minimum for shipment in sodium, days	70
Approximate time for shipment in water or gas, days	365
5. No. of assemblies per cask (see section 3.3)	1
6. Estimated weight of shipping cask, tons	25
7. Weight of fuel assembly, lb	500
8. Refueling cycle	Full core in 3 yr
9. Refueling batch - fraction of full core	1/3, 1/4, 1/5, or 1/6
10. Charges	
Cask fixed charge rate, % per year	15
Fuel value - for insurance, \$/kg (U + Pu)	135
Inventory fixed charge rate, % per year	10
Inventory fuel value, \$/kg fissile Pu	10,000
11. Total final fissile Pu (^{239}Pu and ^{241}Pu), kg (B&W reference)	3480
12. Time required for shipping round trip, days (per Mr. R. Salmon for 1000-mile, one-way distance)	16

1. Number of Shipping Casks Required

The number of casks depends on the number of fuel assemblies per cask, on the number of assemblies in a refueling batch, and on the total period during which the batch of discharged fuel is to be shipped. Refueling batches of 1/3, 1/4, 1/5, and 1/6 of the full core loading (or 155, 116, 93, and 78 assemblies, respectively) were assumed. The number of casks required was calculated as a function of the shipping period. Figure 3 gives these results, and the method is shown below:

$$\text{Avg fuel/assy} = \frac{\text{total fuel (U + Pu) in core}}{\text{No. of assemblies}} = \frac{64,091 \text{ kg}}{464} = 138.13 \text{ kg/assy}$$

(one assy per cask)

$$\text{Avg fuel shipment} = \frac{138.13 \text{ kg/cask}}{16 \text{ days/rd trip-cask}} = 8.63 \text{ kg/day-cask}$$

$$\text{Avg shipment req'd.} = \frac{\text{total core loading} \times \text{fraction of refueling}}{\text{period of shipment}}$$

$$\text{No. of casks req'd.} = \frac{\text{avg shipment req'd.}}{\text{avg fuel/shipment}}$$

Sample: $\frac{1/3 \text{ batch refueling}}{\text{shipment uniformly over 3 months (1/4 year)}}$

$$\frac{64,091 \text{ kg} \times 1/3}{1/4 \text{ yr} \times 365} = 234.12 \text{ kg (avg shipment req'd.)}$$

$$\frac{234.12 \text{ kg}}{8.63 \text{ kg/cask}} = 27.13 \text{ or 28 casks required}$$

2. Cost of Shipping Casks

$$\text{Cost of one cask at } \$1.25 \text{ per lb} = \$1.25 \times 25 \text{ tons} = \$62,500$$

$$\text{Annual fixed charge rate: } 0.15$$

$$\text{Annual charges per cask, } \$/\text{year} = 0.15 \times \$62,500 = \$9375/\text{yr-cask}$$

$$\text{Cost per kg (U + Pu)} = \frac{\$/\text{year-cask} \times \text{No. of casks}}{\text{shipment req'd. /year}}$$

The amount shipped per year is 1/3 of the full core loading

$$\text{Cost per kg} = \frac{\$9375/\text{year-cask} \times \text{No. of casks}}{64,091 \text{ kg} \times 1/3}$$

Figure 4 gives these results, and a sample calculation is given below:

Sample: 28 casks (1/3 batch, 1/4 period)

$$\text{Cost per kg} = \frac{\$9375 \times 28}{21,363.67} = \$12.29/\text{kg}$$

Figure 3. Number of Casks Required Vs Shipping Period

Figure 4. Cost of Shipping Casks Vs Number of Casks Required

3. Freight Costs

Assume that rates are \$0.0204 per lb loaded and \$0.0187 per lb empty
\$/round trip = $(0.0204)(50,000) + (0.0187)(49,500) = \1946 per round trip

Fuel carried per trip, kg (U + Pu) = 138.13

$$\text{Cost per kg} = \frac{\$1946}{138.13 \text{ kg}} = \underline{\$14.09/\text{kg}}$$

4. Handling Costs

$$C_H = 300 + 4 (N+W)$$

N = one assembly per cask

W = 25 tons

$$C_H = 300 + 4 (1 + 25) = \$404 \text{ per round trip}$$

$$\frac{\$404}{138.13 \text{ kg (U + Pu)}} = \underline{\$2.92/\text{kg}}$$

5. Insurance Costs

Cask insurance per round trip = $\$62,500 \times 0.0005 \times 2 = \62.50

Fuel insurance per round trip = $\$135 \times 138.13 \times 0.0005 = \9.31

Total insurance per round trip = \$71.81

$$/\text{kg (U + Pu)} = \frac{\$71.81}{138.13 \text{ kg/trip}} = \underline{\$0.52}$$

6. Total Shipping Cost

	<u>\$/kg</u>
Cask fixed charges	(see Figure 4)
Freight	14.09
Handling	2.92
Insurance	0.52
Total	<u>\$17.53 + cask charge</u>

Figure 5 gives a plot of these costs as a function of the number of shipping casks to be purchased.

Figure 5. Costs Vs Number of Shipping Casks

3.5.3. Inventory Charges - Shipping

(Not included in Mr. Salmon's estimates.)

Inventory charges considered here are only those values directly related to the shipping period as discussed in item 1 of section 3.5.2. These charges are based on the amount of fissile plutonium (^{239}Pu and ^{241}Pu), at a rate of 10% per year and \$10,000 per kg. The B&W reference core (fuel plus blanket assemblies) will have about 3480 kg of fissile plutonium at the 100,000 MWD/T burnup level, giving an average of about 7.50 kg per assembly.

Assuming that the fuel assemblies from one discharge batch will be shipped at a rate of one per day after the required decay storage period, the inventory charge was calculated. The method and results are given in Appendix D. These showed a basic penalty of \$128.49 per kg of fissile plutonium or \$6.92 per kg of total material (U + Pu). Additionally, it is shown that the penalty rate is independent of the batch size and is directly proportional to the shipping period.

These charges can now be added to the total shipping costs (section 3.5.2, item 6) to optimize the number of shipping casks. These results are tabulated in Table 4 and plotted in Figure 5. The stepped curves for inventory charges and combined costs result from the consideration of the period over which the fuel is shipped. It can be noted that 14 casks is the optimum number, and, from previous discussion, corresponds with a 1/6-batch refueling of the B&W reference core and with shipment during a period of about 86 days.

To select a batch refueling of 1/3 of the core for shipment in roughly the same period would require 28 shipping casks. The relative penalty would be \$6.14 per kg of U + Pu. All of this additional cost is directly from the cost of the additional shipping casks. To retain 14 casks to accomplish the shipping of the 1/3 batch would extend the shipping period to roughly 6 months and reflect an increase in inventory charges from \$6.92 to \$13.84 per kg of U + Pu or \$6.92 per kg. Obviously, this is a greater penalty.

Similarly, it can be shown that 21 casks and 17 casks are the optimum number if 1/4- or 1/5-batch refueling is selected.

Table 4. Shipping and Inventory Costs as a Function of the Number of Casks

Number of casks	Shipping costs, \$/kg (U + Pu)	Inventory charge, \$/kg (U + Pu)	Combined costs, \$/kg (U + Pu)
28	29.82	6.92	36.74
21	26.75	6.92	33.67
17	24.99	6.92	31.91
14	23.68	6.92	30.60
11	22.36	13.84	36.20
10	21.92	13.84	35.76
9	21.48	13.84	35.32
7	20.60	13.84	34.44
6	20.16	20.76	40.92
5	19.73	20.76	40.46
4	19.29	27.68	46.97

3.5.4. Inventory Charges — Storage

Also very important to this study are those charges directly associated with decay storage of the spent LMFBR fuel before shipment. Again, inventory charges are based on the amount of fissile plutonium (^{239}Pu and ^{241}Pu) at a rate of 10% per year and \$10,000 per kg. The storage period for this charge is determined by the type of shipment—70 days for sodium shipment and 365 days for water shipment.

To find the rate per kg of fissile material,

$$0.10/\text{yr} \times \$10,000/\text{kg} = \$1000/\text{kg-year}$$

$$\$1000/\text{kg-year} \times \frac{70}{365} = \$191.78/\text{kg} \text{ (70-day storage)}$$

$$\$1000/\text{kg-yr} \times \frac{365}{365} = \$1000.00/\text{kg} \text{ (1-year storage)}$$

Converted to charges per kg of total U + Pu, these are:

$$\$10.41/\text{kg} \text{ (U + Pu) (70-day storage)}$$

$$\$60.07/\text{kg} \text{ (U + Pu) (1-year storage)}$$

3.5.5. Cost Summary

The total costs associated with shipping spent LMFBF fuel assemblies are summarized in Table 5. The two types of shipment, sodium and water, are compared, and a substantial savings for the sodium shipment (\$49.66 per kg of U + Pu) can be seen. The major difference is obviously due to the increased inventory charge associated with the much longer decay storage period for the water shipment. It should be pointed out again that the high vapor pressure of water is definitely its limiting characteristic, and alternate cooling mediums have breakdown or other problems when associated with the temperatures being considered and, in our appraisal, approach the water shipment case in regard to inventory penalty.

Table 5. Total Costs — Spent Fuel Shipping and Inventory
(\$/kg U + Pu)

	<u>Sodium shipment (70-day storage)</u>	<u>Water shipment (1-year storage)</u>
Fuel cleaning costs	--	2.73
Shipping cask costs (14 req'd.)	6.15	6.15
Freight costs	14.09	14.09
Handling costs	2.92	2.92
Insurance costs	0.52	0.52
Inventory charges - shipping period (3 months)	6.92	6.92
Inventory charges - storage	10.41	60.07
Total	<u>41.01 \$/kg</u> or 0.105 mills/kwhr	<u>93.40 \$/kg</u> or 0.239 mills/kwhr
Difference	51.22 \$/kg or 0.134 mills/kwhr	

In terms of mills per kwhr the costs of the two shipment types are calculated as follows (also given in Table 5):

$$\begin{aligned} \text{Sodium shipment: } & \frac{\$41.01/\text{kg} \times 64,091 \cdot \text{kg} \times 1000 \text{ mills}/\$}{1038 \times 10^3 \text{ kwe} \times 1005 \text{ FPD} \times 24 \text{ hr/day}} \\ & = 0.105 \text{ mills/kwhr} \end{aligned}$$

where

$$\text{Plant power} = 1038 \text{ MWe}$$

$$\text{Reactor full power days} = 1005 \text{ FPD}$$

$$\text{Water shipment: } \frac{93.40 \times 64,091 \times 1000}{1038 \times 10^3 \times 1005 \times 24} = 0.239 \text{ mills/kwhr}$$

$$\text{Difference or penalty of water shipment} = 0.134 \text{ mills/kwhr}$$

Although not directly comparable, the total fuel costs calculated for the B&W reference LMFBR are shown in Figure 6. These total costs are output from the Sodium Parameter Study Code and include reprocessing, reconversion, fabrication, and other charges not considered in this study. The figure gives the total fuel costs as a function of the spent fuel decay storage time, which does allow us to see the effect of long storage.

$$\text{Sodium shipment (70 days storage, 0.19 yr)} = 0.901 \text{ mills/kwhr}$$

$$\text{Water shipment (1 year)} = 1.008 \text{ mills/kwhr}$$

$$\text{Difference} = 0.107 \text{ mills/kwhr}$$

Decay storage of spent fuel also has an effect on the reactor doubling time. Figure 7 shows this effect, and again the results are for the B&W reference core and were produced from the Sodium Parameter Study Code. Comparing these values:

$$\text{Sodium shipment (70 days storage, 0.19 yr)} = 15.4 \text{ years}$$

$$\text{Water shipment (1 year)} = 18.25 \text{ years}$$

Figure 6. Total Fuel Cost Vs Spent Fuel Storage Time
(B&W Reference Core)

Figure 7. Doubling Time Vs Spent Fuel Storage Time
(B&W Reference Core)

4. RESULTS

The shipment of only one fuel assembly in a sodium-filled shipping cask was selected as probably the only economical way that spent LMFBR fuel could be shipped with less than one year of decay storage. Several factors influenced this decision, the most important being (1) decay heat rate, (2) length of the fuel assembly, (3) criticality, and (4) handling.

The important economic results are given in Table 5, where sodium shipment is compared with water shipment. Table 3 gives the minimum decay storage times before fuel assemblies can be shipped safely in various liquids. The mechanical design of the fuel was found to be important from two aspects: length of the fuel assembly and whether or not it should be vented. The approximate lengths of various configurations are compared in Table 1. Criticality was not found to be of major concern during shipping, although disassembly before shipping could cause a criticality problem as the fuel pins are separated.

4.1. Discussion of Results

The first interesting fact in this study was that it could not be just a trade-off study of the possible cask cooling media. Instead, it must be a feasibility study—a complete study to determine whether safe shipment of spent LMFBR fuel is possible at all. Possible, that is, within an economical time after its removal from the reactor.

The results of this study have established that safe shipment of spent LMFBR fuel can be accomplished. This shipment can be made in sodium after only about 70 days of decay storage at the reactor site. As expected, there will be a penalty. Not so much a cost penalty, because the cost connected with the shipping procedures will only be about 0.105 mills per kwhr (Table 5). The penalty will be in developing procedures of handling at the reactor site, in adequate cask design, and in reprocessing site procedures.

As discussed, the procedures at the reactor site will be cooling the fuel properly during the transfer from the decay storage to the shipping cask loading point. Gas cooling is recommended, but sufficient redundant systems must be provided to insure cooling. Each LMFBR reactor plant designer must give special emphasis to this procedure.

Also discussed were the more stringent requirements on shipping cask design:

1. A maximum fuel cladding temperature of about 2350 F (vented fuel design, only following the worst accident).
2. The inner cask container must retain the primary sodium coolant completely.
3. The inner cask container must maintain its integrity under internal pressures approaching 300 psia and temperatures of about 2000 F.
4. The cask must contain the lead (or alternate) shielding in such an enclosure that molten lead cannot escape the cask or reduce the shielding effectiveness below acceptable levels. The containment for lead shielding may be a major problem. A material such as depleted uranium should be seriously considered as an alternate.

From the mechanical design aspect, the vented fuel design that was considered (B&W reference assembly) permits the much higher cladding temperature given above. This is not possible in the non-vented fuel designs, where pressure failures must be postulated at much lower temperatures. In the case of non-vented fuel, shipment would not be safe unless at least one year of storage was allowed or unless a very expensive procedure to relieve the gas was added.

The study of fuel design also considered the length of the assemblies. Strictly from the shipping viewpoint, it appears that the length of the assembly should be kept to about 8 feet or less. Remembering that only one assembly can be shipped per cask, this length will most likely permit truck shipment. Increasing the assembly length would increase the cask weight and cost; it is possible that a 10-foot assembly could increase the total cost of shipping by as much as 20%. More important could be the increased heat load to the handling and cask cooling systems. A 20% increase in length could increase the decay storage period needed, and this, as has been shown, rapidly increases overall fuel cycle costs.

The next major area that still needs work before 70-day-cooled assemblies can be shipped is the reprocessing site; the higher decay heat load will again probably be the critical criterion. Another question could be the sodium, and a third item will be the radiation level and criticality problem when the fuel assemblies are disassembled. All these factors still need study. It is of little value to get the fuel to the reprocessing site only to have a mandatory decay storage period scheduled there.

APPENDIX A
Thermal Analysis Model, Failure Criteria,
and Assumptions

A thermal analysis of a B&W reference fuel assembly in a shipping container was carried out to determine the performance of the various coolants proposed for use in the shipping cask. For the analysis, it was assumed that an infinitely long cask, having a heat generation rate equal to the maximum axial condition, was used. Therefore, the blanket contribution is conservatively considered. Three cooling media were examined: sodium, water, and argon gas. The results of this effort, summarized in Table 3, give the required storage periods to prevent failure of the fuel cladding material and the cask structure. The failure criteria chosen for this evaluation of the coolant performance capability were as follows:

1. A maximum center pin cladding temperature of 2350 F (200 F below the melting point of SS 347, only following the worst accident).
2. A maximum internal cask pressure of 500 psig.

These are not the recommended normal criteria but the upper limits under the transport accident. Since the high temperature occurs only after the major accident, the limit appears acceptable for the reference vented fuel design where no pressure-type failure is possible, and the allowance below melting was suggested by Mr. J. C. Glynn.¹²

Assumptions used in the analysis are as follows:

1. The fuel assembly, cask, and shield assembly are as shown in Figure 2 (lead shield requirement of 8-inch thickness was obtained from internal LMFBR data¹³).
2. Ambient air at 70 F for the circulation failure and loss-of-coolant accident cases (heat transfer coefficients were taken from W. H. McAdams¹⁴).
3. With sodium as coolant, the heat transfer mechanism between the fuel pin and cask wall was considered to be conduction only.
4. For air, argon, or water vapor in the coolant space, results by G. O. Geissler were used to predict the center pin temperatures.¹⁵
5. For the fire accident case, the shield was considered to be radiating to a 1475 F surface located an infinite distance from the shield.

APPENDIX B
Hypothetical Accident Conditions²

The following hypothetical accident conditions are to be applied sequentially, in the order indicated, to determine their cumulative effect on a package or array of packages.²

1. Free Drop — A free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected.

2. Puncture — A free drop through a distance of 40 inches striking, in a position for which maximum damage is expected, the top of a vertical cylindrical mild steel bar mounted on an essentially unyielding horizontal surface. The bar shall be 6 inches in diameter, with the top horizontal and its edge rounded to a radius of not more than one-quarter inch, and of such a length as to cause maximum damage to the package, but not less than 8 inches long. The long axis of the bar shall be normal to the package surface.

3. Thermal — Exposure for 30 minutes within a source of radiant heat having a temperature of 1475 °F and an emissivity coefficient of 0.9, or equivalent. For calculational purposes, it shall be assumed that the package has an absorption coefficient of 0.8. The package shall not be cooled artificially until after the 30 minute test period has expired and the temperature at the center of the package has begun to fall.

4. Water Immersion — Immersion in water for 24 hours to a depth of at least 3 feet.

APPENDIX C
Fuel Cleaning Costs

The following method was used to estimate the costs of fuel cleaning that would be required before spent LMFBR fuel assemblies could be in water. An explanation is given in section 3.5.1.

Total core loading of U + Pu (B&W reference core), kg	64,091
Capital cost of cleaning equipment (section 3.5.1), \$	250,000
Maintenance and operating costs, \$/yr	50,000
Life of the reactor, yr	30
Refueling cycle	Full core in 3 years

To find the cost per refueling cycle of cleaning equipment

$$3 \left(\frac{\$250,000}{30 \text{ yr}} + \$50,000 \right) = \$175,000$$

To find the cost per kg of U + Pu

$$\frac{\$175,000}{64,091 \text{ kg}} = \$2.73/\text{kg}$$

APPENDIX D
Inventory Charges for the Shipping Period

To calculate the shipping charges associated with the period required to complete the shipping of a discharged batch of fuel, the following method was used:

$$\begin{aligned} \text{inventory charge} &= \text{amount of fuel (kg of fissile Pu)} \\ &\quad \times \text{rate/yr} \times \text{cost/kg} \times \text{period} \end{aligned}$$

where

amount of fuel =	1160 kg (1/3 batch) (155 assemblies)
	870 kg (1/4 batch) (116 assemblies)
	696 kg (1/5 batch) (93 assemblies)
	580 kg (1/6 batch) (78 assemblies)
rate/year =	10%
cost/kg =	\$10,000
period =	approximately 1/4 year

The time required for one cask round trip is 16 days, which includes one day loading and one day unloading.²⁰

Inventory charges on a shipment are assumed to stop when that shipment arrives at the reprocessing site.

One fuel assembly is shipped per shipment, at 7.50 kg of fissile plutonium per assembly.

Charges on the full batch inventory will exist for the first eight days or until the first shipment reaches its destination. Using the 1/6-batch refueling (78 assemblies) for illustration, the inventory charges were calculated:

$$\begin{aligned} \text{charges (8 days)} &= 78 \text{ assemblies} \times 7.50 \text{ kg/assembly} \\ &\quad \times 0.10/\text{yr} \times \$10,000/\text{kg} \times 8/365 \\ &= \$12,811.15 \end{aligned}$$

For additional charges:

$$\begin{aligned} \text{charges (per day)} &= [\text{No. of assemblies} - (t - 8)] \\ &\quad \times 7.50 \times 0.10 \times \$10^4 \times 1/365 \end{aligned}$$

t = days from start (Note: one day = one assembly); $t \geq 8$

$$\begin{aligned}\text{charges (9th day)} &= [78 - (9 - 8)](7.5) \times 10^3 \times 1/365 \\ &= \$1582.35\end{aligned}$$

Completing these calculations gave a final inventory charge associated with this total shipment of \$74,523.80 or \$128.49/kg of fissile Pu. Converted to charges per kg of total U + Pu, these charges are \$6.92/kg (U + Pu), where the U + Pu for this batch = 138.13 kg/assembly $\times$ 78 assemblies.

Irrespective of the discharge batch, the inventory charges associated with a period during which the fuel is to be shipped will be

- 1/4-year period - \$ 6.92/kg (U + Pu)
- 1/2-year period - \$13.84/kg (U + Pu)
- 3/4-year period - \$20.76/kg (U + Pu)
- 1-year period - \$27.68/kg (U + Pu)

It should be noted that approximately 14 shipping casks would be required to accomplish the sample shipping of 78 assemblies in a total period of 86 days.

REFERENCES

- ¹ "Interstate Commerce Commission Regulation, Title 49, Code of Federal Regulations, Parts 171-179 — Explosives and Other Dangerous Articles," 49 CFR-171, Atomic Energy Law Reports, 1967, Commerce Clearing House, Inc.
- ² "Part 71 — Packaging of Radioactive Material for Transport," 10 CFR-71, Atomic Energy Law Reports, 1966, Commerce Clearing House, Inc.
- ³ Physics Characteristics of the LMFBR Cylindrical Reference Core, B&W reference core information.
- ⁴ Williams, R. L. to Hostetler, D. R., Criticality of Reference to LMFBR Subassemblies, Internal Memo, October 3, 1967.
- ⁵ Brown, C. L., FFTF Criticality Safety System 72 — Underwater Examination Facility Conceptual Design Evaluation, Battelle Northwest Laboratories, BNWL-586, September 1967 (AEC Research and Development Report).
- ⁶ Brown, C. L., FFTF Criticality Safety System 72 — Underwater Examination Facility Conceptual Design Evaluation, Battelle Northwest Laboratories, BNWL-587, September 1967 (AEC Research and Development Report).
- ⁷ Brown, C. L. and Lawrence, L. A., FFTF Criticality Safety System 72 — Underwater Examination Facility Conceptual Design Evaluation, Battelle Northwest Laboratories, BNWL-594, September 1967 (AEC Research and Development Report).
- ⁸ Brown, C. L. and Lawrence, L. A., FFTF Criticality Safety System 72 — Underwater Examination Facility Conceptual Design Evaluation, Battelle Northwest Laboratories, BNWL-595, October 1967 (AEC Research and Development Report).

- ⁹ Howard, J. O. (B&W), Decay Heat From Fission Product and Heavy Isotope Decay, Internal Memo, June 22, 1967.
- ¹⁰ Liquid Metal Fast Breeder Reactor Program Plan — Section 8, Fuel Recycle, prepared by the LMFBR Program Office, Argonne National Laboratory of the AEC.
- ¹¹ Straub, R. V. and Moore, R. (B&W), Cooling of LMFBR Spent Fuel During Shipment, Internal Memo, December 27, 1967.
- ¹² Glynn, James C., "Heat Removal From Casks," Chemical Engineering Progress Symposium, Series 56, Nuclear Engineering, 61, Part 14, 1965.
- ¹³ Stewart, J. E. (B&W), Verbal Communication, December 1967.
- ¹⁴ McAdams, W. H., Heat Transmissions, 3rd Ed., McGraw-Hill Book Co., Inc., 1954 (p 179).
- ¹⁵ Giessler, G. O., Internal Memo 30-2014-20, January 4, 1965.
- ¹⁶ Dickinson, D. R., Gas Cooling of FFTF Fuel During Examination and Disassembly Preliminary Feasibility Study, Battelle Northwest Laboratories, BNWL-524, July 1967.
- ¹⁷ Crane, R. A. (B&W), Decay Storage Tanks for LMFBR, Internal Memo 859-3102, November 10, 1967.
- ¹⁸ Meyer, A., Thermal and Hydraulic Analysis of LMFBR Gas Cooled Fuel Transfer System Concept, Internal Memo 59-3102, November 13, 1967.
- ¹⁹ 1000-MWe LMFBR Follow-On Study, Task 1, Concept IV — Systems Description Report, BAW-1316, Vol 5, August 1967.
- ²⁰ Salmon, R., UC-4-Chemistry, Estimation of Fuel-Shipping Costs for Nuclear Power Cost-Evaluation Purposes, Oak Ridge National Laboratory, ORNL-3943, March 1966.

7. ON-SITE VS OFF-SITE REPROCESSING AND
FABRICATION TRADE-OFF STUDY

CONTENTS

	Page
1. INTRODUCTION	1
1.1. Purpose	1
1.2. Summary	2
2. CONCLUSIONS AND RECOMMENDATIONS	4
3. DISCUSSION	9
3.1. General	9
3.2. Ground Rules	11
3.3. Determination of Trade-Off Points	18
3.4. Fabrication Cost	30
3.5. Reprocessing Cost	38
3.6. Fuel Inventory Charges	42
3.7. Spent Fuel Storage Period	48
3.8. Spent Fuel Shipping Cost	49
4. RESULTS	50
4.1. General	50
4.2. On-Site Reprocessing and Fabrication	51
4.3. On-Site Reprocessing Only	57

List of Tables

Table

1. LMFBR Reference Design Data	13
2. Trade-Off Points — MWe Capacity at One Site, Based on Ground Rule Shipping Costs and 15,000-MWe Centralized Processing Capacity	24
3. Trade-Off Points — MWe Capacity at One Site, Based on \$54/kg Shipping Costs and 15,000-MWe Centralized Processing Capacity	24
4. Trade-Off Points — MWe Capacity at One Site, Based on Ground Rule Shipping Costs and 30,000-MWe Centralized Processing Capacity	25

Tables (Cont'd)

Table	Page
5. Trade-Off Points — MWe Capacity at One Site, Based on \$54/kg Shipping Costs and 30,000-MWe Centralized Processing Capacity	25
6a. Fuel Cycle Costs Based on Off-Site Reprocessing and Fabrication (15,000-MWe Plant Capacity)	26
6b. Fuel Cycle Costs Based on Off-Site Reprocessing and Fabrication (30,000-MWe Plant Capacity)	27
7. Fuel Cycle Costs Based on On-Site Reprocessing and Fabrication, mills/kwhr	28
8. Fabrication Cost Data	36
9. Reprocessing Costs	39
10. On-Site Reprocessing and Fabrication Fuel Holdup Times	44
11. Out-of-Pile Fraction — On-Site Processing	44
12. Trade-Off of Single 1000-MWe Reactor With On-Site Reprocessing and Fabrication.	58

List of Figures

Figure	
1. Schematic Diagram, Fast Breeder Reactor Fuel Cycle . .	16
2. Fuel Cycle Costs for On- and Off- Site Reprocessing and Fabrication, Based on \$10/gm Plutonium and 15,000 MWe Off-Site Processing Capacity.	20
3. Fuel Cycle Costs for On- and Off-Site Reprocessing and Fabrication, Based on \$20/gm Plutonium and 15,000 MWe Off-Site Processing Capacity.	21
4. Fuel Cycle Costs for On- and Off-Site Reprocessing and Fabrication, Based on \$10/gm Plutonium and 30,000 MWe Off-Site Processing Capacity.	22
5. Fuel Cycle Costs for On- and Off-Site Reprocessing and Fabrication, Based on \$20/gm Plutonium and 30,000 MWe Off-Site Processing Capacity.	23
6. Fuel Cycle Cost for On-Site Reprocessing With Off-Site Fabrication, Based on \$10/gm Plutonium and Reference Processing Costs	31
7. Fuel Cycle Costs for On-Site Reprocessing With Off-Site Fabrication and for Off-Site Reprocessing and Fabrication, Based on \$10/gm Plutonium, 15,000 MWe Off-Site Processing Capacity, and Reference Fabrication and Reprocessing Costs.	32

Figures (Cont'd)

Figure	Page
8. Fuel Cycle Costs for On-Site Reprocessing With Off-Site Fabrication and for Off-Site Reprocessing and Fabrication, Based on \$20/gm Plutonium, 15,000 MWe Off-Site Processing Capacity, and Reference Fabrication and Reprocessing Costs.	33
9. Reference Core and Blanket Fabrication Cost Vs Plant Throughput	40
10. Aqueous Fast Reactor Reprocessing Costs	41
11. Reference Reprocessing Cost Vs Plant Throughput	45
12. Simplified Model for Out-of-Pile Plutonium Inventory - One Reactor With On-Site Reprocessing and Fabrication	46
13. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing.	52
14. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing.	53
15. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing.	54
16. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing	55
17. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing.	59
18. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing.	60
19. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing.	61

Figures (Cont'd)

Figure	Page
20. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing.	62
21. Trade-Off Relationship Between On-Site Reprocessing With Off-Site Fabrication - Months of On-Site Storage Plus Reprocessing Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing and for Ground Rules and \$54/kg Shipping Costs	63
22. Trade-Off Relationship Between On-Site Reprocessing With Off-Site Fabrication and Off-Site Reprocessing and Fabrication - Months of On-Site Storage Periods Before Shipment of Fuel to Off-Site Reprocessing and for Ground Rules and \$54/kg Shipping Costs	64

1. INTRODUCTION

1. 1. Purpose

This study is an investigation of the economics of two alternate modes of processing liquid metal fast breeder reactor (LMFBR) fuel—on-site and off-site reprocessing and fabrication. On-site reprocessing and fabrication refers to the location of these facilities at the same site as a reactor or a complex of reactors. Off-site is the more conventional arrangement whereby separate reprocessing and fabrication plants are located at sites most advantageous to their particular requirements; these plants serve a large number of widely scattered reactors. The off-site facilities, called "centralized" facilities, are designed to handle fuel of diverse types and designs from a variety of reactors. An on-site facility, on the other hand, would be designed to handle one particular kind of fuel, and the reprocessing and fabricating operations would be integrated into one continuous process in one facility.

To date, the nuclear power industry has used off-site commercial reprocessing, although this convention is based on only the one operating commercial plant—the Nuclear Fuel Services plant at West Valley, New York. Another announced commercial reprocessing plant (to be constructed by the General Electric Company near Morris, Illinois) will be adjacent to a complex of power reactors and thus minimize the shipping cost; this arrangement is known as "across-the-fence" reprocessing. The GE plant, however, is intended to reprocess fuel from power stations throughout the Middle West, and therefore, it would be principally an off-site reprocessing facility. On the other hand, utilities are taking an increased interest in and are increasing their participation in the fuel cycle; they are procuring ore and arranging for its conversion to UF_6 and for enrichment services, arranging for fabrication and reprocessing, and negotiating for the sale of the residual uranium and plutonium. This increased interest could result in integrated reprocessing and fabrication

arrangements, especially for the huge reactor complexes for water desalination and combination process heat and electric power now being proposed and discussed.

In the course of the LMFBR development program, questions have arisen concerning the feasibility and economics of shipping the fuel—particularly spent fuel—and on-site reprocessing and fabrication have been suggested as a more economical alternative. Fast reactor fuel is much more difficult to ship than is the thermal reactor fuel with which the nuclear industry has had its experience. The higher levels of specific power, power density, and burnup, together with the closer packing of the fuel assembly lattice, present difficult cooling problems. The higher plutonium concentrations and fission product inventories present criticality, shielding, and containment problems. The fuel costs are affected particularly by the cooling difficulties, which have resulted in the specification of longer storage periods to allow decay heat to subside before the fuel can be shipped to reprocessing plants.

This study compares on- and off-site reprocessing and fabrication by utilizing economic trade-off studies to explore the fuel cycle costs using both processing schemes; the trade-off points are identified as those sets of economic parameters and conditions at which the fuel cycle costs under both processing modes are equal.

1.2. Summary

On- and off-site reprocessing and fabrication of LMFBR fuel were compared on the basis of an analysis of fuel cycle costs for each of these recycle methods. A number of trade-off points were determined; at these points various combinations of economic parameters result in equal fuel cycle costs for both methods. The major effort in the study was devoted to comparing on-site reprocessing and fabrication in an integrated plant with the performance of these operations in separate, off-site, centralized (large-capacity) facilities. We also compared the case of on-site reprocessing only—with fabrication in an off-site centralized plant—with the case of totally off-site processing.

Fuel costs were based on the reference design of the LMFBR Follow-On Study, and trade-off points were determined as a function of the following economic parameters: plutonium value, spent fuel cooling

period, shipping cost, off-site processing plant capacity, unit reprocessing and fabrication costs, and the electrical capacity of the on-site reactor complex.

A single 1000-MWe reactor with on-site reprocessing and fabrication is not likely to be economical; on-site reprocessing only, however, is much more attractive. As the reactor complex having on-site processing increases in size, the fuel cycle costs improve because of the lower unit processing costs. The longer the required storage period before shipping, and the higher the shipping cost, the more attractive the on-site processing mode becomes.

The study defines the ranges of economic conditions under which on-site fuel processing would be desirable or feasible and clarifies the importance or sensitivity of the various parameters to the trade-off points. Definite quantitative conclusions, however, are difficult to reach because of the considerable uncertainty that now exists in the costs associated with the various stages of the fuel cycle. Nevertheless, it is hoped that this study can provide a basis for guiding and directing the research and development programs to be pursued in achieving a competitive, self-sustaining LMFBR industrial capability.

2. CONCLUSIONS AND RECOMMENDATIONS

1. The most important conclusion that can be drawn as a result of this study is that the state of LMFBR development and technology is not sufficiently advanced to make accurate cost estimates for the various segments of the fuel cycle in either the on-site or off-site mode. The conclusions listed below, which incorporate absolute values of trade-off points, should, therefore, be viewed in the light of these cost uncertainties.

2. Considerable uncertainty exists with the very question of whether on-site reprocessing and fabrication is desirable or necessary. The incentive for on-site fuel processing derives from the postulated difficulties in shipping the spent fuel to a centralized reprocessing plant. There is a need for more accurate determination of the problems and costs associated with this operation, including the handling problems at the reactor and at the reprocessing plant.

3. Certain aspects of on-site processing were not investigated and may bear heavily on the evaluation of its feasibility and economy. For example, we did not evaluate the inter-effects between the on-site facilities and the reactor plant or reactor complex itself. The presence of the processing plant may impose site penalties and restrictions that require the location of the reactors at more remote sites; this would increase electrical transmission costs. Conversely, we did not evaluate the possible cost savings at the reactor plants that result from eliminating the facilities and personnel that are required to prepare fuel for shipment off-site.

4. On-site reprocessing and fabrication does not appear economically attractive for small reactor capacities. The unit costs for a single 1000-MWe LMFBR with on-site reprocessing and fabrication would have to be reduced to below 30% of the costs predicted for plants

of that capacity using the "reference" estimating methods adopted in this study. It is unlikely that such cost reductions can be realized by integrating the processes.

5. If LMFBR spent fuel can be shipped after the same storage times now used for thermal reactor fuels, and at shipping costs postulated by the ANL ground rules for the LMFBR Follow-On Study (which are three to five times the cost for thermal fuels on a \$ per kilogram basis), then an on-site reactor capacity of about 8600 MWe with integrated reprocessing and fabrication facilities can be competitive with off-site reprocessing and fabrication. This figure is based on the "reference" estimating methods of this study and on 15,000-MWe capacity centralized processing plants. This on-site capacity may not be unreasonable for a reactor complex in the 1980's and 1990's. However, if centralized facilities are twice as large, i.e., designed for 30,000 MWe, then an on-site capacity of over 20,000 MWe is required to be economical. These conclusions are based on a plutonium value of \$10 per gram of fissile isotopes.

6. Unit fabrication and reprocessing costs are the most influential parameters affecting the trade-off points. For example, if on-site fabrication and reprocessing costs can be reduced only 20% from the costs estimated using the "reference" methods, either by integrating the processes into one facility or by technological improvements in the processes, then the on-site reactor capacity required to be economical can be reduced by about 30 to 50%. Thus, referring to item 5 above, the 8600 MWe can be reduced to 5500 MWe, and the 20,000 MWe can be reduced to 11,400 MWe.

7. The requirement that spent breeder reactor fuel may have to remain in storage for as long as a year also has a marked effect on on-site processing competitiveness. An increase from 4 to 12 months in storage time before shipment to off-site reprocessing increases off-site fuel costs to the extent that the competitive on-site reactor complex size can be reduced about 25 to 40%, based on 15,000-MWe centralized processing plants. When 30,000-MWe centralized plants are considered, the economic effect of the longer storage period is greater, and somewhat greater reductions in on-site capacity can be competitive.

8. The effect of plutonium value on the trade-off points is relatively minor. That is, the size of the economic on-site reactor complex is relatively unaffected changing the plutonium value from \$10 to \$20 per gram. This is because the plutonium credit tends to balance inventory cost, independent of unit plutonium value. This effect can be seen in Tables 2 through 5, where, at the same cooling time, the capacity of the on-site reactor (in MWe) is affected very little by a change from \$10 to \$20 per gram in the value of plutonium.

9. Shipping costs also have a relatively small effect on the trade-off points. We compared the effects of the ANL ground rules shipping cost (which amounts to an average of \$17 per kilogram of core and blanket) with a shipping cost of more than three times as high, \$54 per kilogram. The reductions in required capacity with on-site reprocessing and fabrication were about 20 to 30% at the higher shipping cost.

10. Illustrating the strong effect of fabrication and reprocessing cost, a decrease in the size of the centralized plants from 30,000 to 15,000 MWe capacity (with the attendant increases in unit cost) reduces the reactor capacity with on-site reprocessing and fabrication required for competitiveness by about 30 to 40%.

11. The study of on-site reprocessing only, with fabrication in a large-capacity, off-site plant, reveals that relatively small on-site reactor capacities would be economical if the total time the fuel is held at the site for cooling plus reprocessing is relatively short. For example, a 4600-MWe reactor complex with on-site reprocessing would be competitive with totally off-site reprocessing and fabrication (in 15,000-MWe-capacity plants), based on \$10 per gram of plutonium and the ground rules shipping cost, if the post-irradiation fuel holdup time can be limited to about 6 months. This case is also based on a 4-month cooling period before shipment off-site. This conclusion, however, is based on the assumption that the "reference" reprocessing cost for this capacity (i.e., 4600 MWe) applies to a plant that can process the fuel in a 2-month period after the fuel has been stored for 4 months. (Refer to section 3.3.2 for additional discussion.)

In another example, if 12 months of cooling are required before shipping spent fuel off-site, and if the shipping cost is \$54 per

kilogram, then a single 1000-MWe reactor with on-site reprocessing only can be economical, again based on a total post-irradiation inventory period of 6 months, and based on the other economic parameters cited in the paragraph above. Reducing the shipping cost to the ground rules average value of \$17 per kilogram only increased the required size of the on-site reactor capacity slightly—to 1600 MWe.

We should note here that for integrated on-site reprocessing plus fabrication, fuel is held on-site throughout the fuel cycle, thus incurring high inventory charges. There is no freedom to optimize the combination of processing plant throughput (i. e. , processing time) and inventory to achieve the lowest total cost. Thus, in this mode the reprocessing and fabrication throughput is sized to have a production rate that matches the time interval between refuelings.

For on-site reprocessing combined with off-site fabrication, a rapid processing rate could result in overall reduced costs, since the fuel is assumed to be sold to another utility or another reactor of the same utility immediately after reprocessing. This is also true of off-site reprocessing and fabrication.

The problems associated with storage and holdup time were carefully considered. For on-site reprocessing and fabrication, there are important and interesting questions concerning the scheduling of reprocessing campaigns to match reactor refueling schedules, and establishing staggered refueling schedules for the reactors at one site. The calculational model (presented in ANL-7137) for on-site reprocessing inventory charges was studied, and our method was patterned after it. We believe we improved on that method, however, by accounting for the difference between the end-of-life (EOL) and beginning-of-life (BOL) fuel quantities during the outpile portion of the fuel cycle.

In connection with the case of on-site reprocessing combined with off-site fabrication, we would have preferred to choose, or at least to investigate, an optimum combination of plant throughput and fuel inventory for each size of reactor complex, but we did not have a method available for estimating the reprocessing costs for plants operating at less than the reference continuous basis of 260 processing days per year.

Note also that for the case of on-site reprocessing and fabrication, the on-site inventory cost of the BOL loading is not decreased

by accelerated reprocessing (since it must remain on-site in any event to be loaded into the reactor); this cost can be reduced, however, for the "excess" bred plutonium that is sold to another reactor. For our reference design, this quantity is about 37% of the BOL inventory, neglecting process losses.

12. We have not studied certain other problems that arise in comparing on- and off-site reprocessing and fabrication. One is the "chicken and egg" problem: that is, to be economical, FBR's need low fuel cycle costs, which can only be achieved by the use of large-capacity fabrication and reprocessing facilities if off-site reprocessing is to be used. However, these plants can only be operated economically when a large number of reactors are in operation. It may be that, in the interim period when FBR capacity is relatively small, on-site processing could be economical in special instances.

On-site reprocessing and fabrication would be a departure from current or even projected practices in the utility industry in that utilities have not shown any inclination to operate this kind of facility. Joint ventures or other commercial arrangements between utilities and chemical and fuel manufacturing companies are highly conjectural.

13. Additional work is certainly indicated to assess the trade-off between on- and off-site reprocessing and fabrication. The areas of uncertainty have been identified above: (1) the problems and costs of fabricating and reprocessing breeder reactor fuel of various designs in plants of various capacities, (2) the cost savings that can be achieved by integrating these processes into one continuous facility at the reactor site, (3) the siting problems that may exist when reprocessing plants are on-site with reactor power plants, and (4) the problems (associated with the early years of LMFBR industry growth) of providing economical processing facilities for the small loads available.

3. DISCUSSION

3.1. General

The economic trade-off between on- and off-site reprocessing and fabrication regimes hinges on the question of whether reprocessing and fabrication costs in small-capacity, on-site plants can be sufficiently economical to compete with the lower off-site costs, which, however, have the additional costs of shipping and fuel inventory associated with transit and storage time. It is acknowledged that off-site plants have low unit costs because of their large size; in this study we considered plants designed to accommodate up to 30,000 MWe. It is not likely that such large capacities would be at one site. Therefore, the objective of the study was to

1. Determine, for various sets of economic parameters, the minimum reactor complex electrical capacity that would be required to make "on-site" fuel cycle costs equivalent to those using centralized off-site facilities.

2. Determine, for specified sizes of on-site reactor complexes, the on-site reprocessing and fabrication costs that must be achieved in order to make on-site reprocessing and fabrication competitive with off-site.

3. In the case of on-site reprocessing and off-site fabrication, to determine the relationship between the allowable on-site fuel holdup period that would result in economical fuel cycle costs, compared with the totally off-site reprocessing and fabrication mode.

The time limit imposed on completion of this study restricted the number of economic variables that could be investigated, and only those that were thought to have the greatest impact on the trade-off points were selected. The variables used are as follows:

1. Plutonium values of \$10 and \$20 per gram of fissile isotope. The \$10 figure is specified by the ANL ground rules and is used in common practice today for fast reactor evaluation work. The \$20 figure represents the higher value that plutonium is expected to have when breeder reactors come into wide commercial use.

2. Spent fuel cooling periods of 4 months for on-site reprocessing and both 4- and 12-month periods before shipment off-site. The 4-month figure is practiced today for thermal reactor fuels; the 12-month figure is postulated to be necessary because of the difficulty of cooling spent fast breeder fuel, although there are indications that this long period may not be necessary. The 4-month period was used in the cases of integrated on-site reprocessing and fabrication; for the cases of on-site reprocessing only, the on-site storage period was a variable, and periods of up to 24 months were considered. Spent fuel storage periods of about 1 month have been suggested for fast breeder fuels, but only where associated with a nonaqueous mode of reprocessing. For aqueous processing, on which this study is based, the minimum cooling time used in the industry is 4 months (for example, see reference 3).

3. Shipping costs based on the ANL ground rules, which average \$17 per kilogram of core and blanket and \$54 per kilogram of core and blanket. The latter derives from another trade-off study completed in the LMFBR Follow-On Program. These unit shipping costs are intended to cover the range of uncertainty associated with this operation. Shipping costs may be interdependent on the allowable cooling times, and an optimized combination of unit shipping cost and inventory cost could result in shipping costs that are outside the range covered.

4. Centralized fabrication and reprocessing plant capacities of 15,000 and 30,000 MWe. The former has been a popular typical size used in reactor evaluation work; the latter is representative of the larger-sized plants that may be built as the nuclear industry expands.

5. On-site electrical capacity from the equivalent of a single reactor (1000 MWe) to over 12,000 MWe.

6. Reprocessing and fabrication costs are based on the methods discussed in subsequent sections of this report. These methods

are applicable to a range of plant sizes and are applied to both the on- and off-site plants; in this report they are called the "reference" methods. However, for the combined on-site reprocessing-fabrication plants, we evaluated fuel cycle costs based on possible reductions in these processing costs due to the integration of the facilities, and thus we used costs of 80 and 50% of the "reference" method cost.

Integrated plants will also result in reduced costs because of the elimination of facilities and personnel required to prepare the fuel for shipment, and because combined operating and maintenance staffs and facilities can be used. However, there was not enough time to determine the extent of the reductions. The 80% and 50% parameters are possible representative effects of these savings combined with the possible savings due to integration of the reprocessing and fabrication processes.

3.2. Ground Rules

3.2.1. Reference Core Design

The reference core design developed in the B&W LMFBR Follow-On Study was the basis for the fuel cycle costs; Table 1 lists the core design data applicable to this study. The reactor has a nominal rating of 1000 MWe (the actual figure is 1012 MWe) and uses mixed oxide fuel in the core and depleted oxide in the blanket; the core achieves an average burnup of 100,000 MWD/tonne. The core and the axial and radial blankets are vibratorily compacted into stainless-steel tubing.

The reactor is refueled every 1034 full power days; this amounts to a calendar lifetime of 1198 days at the average annual capacity factor of 0.863. Refueling is accomplished by removing all of the core and blanket assemblies and replacing them with fresh assemblies. The spent fuel is then stored and sent to reprocessing, either on or off the site, after the cooling period.

The capacity factor of 0.863 is calculated from the ANL ground rules, which state that the capacity factor is "90% of the plant availability," which is in turn "100% less downtime for refueling." The "refueling" time was subsequently clarified to mean the total scheduled downtime for both refueling and maintenance. A figure of 50 days per

cycle has been assumed in the Follow-On Study as this scheduled downtime. The capacity factor, therefore, is calculated as follows:

$$CF = \frac{FPD}{1.111(FPD) + DT} = \frac{1034}{1.111(1034) + 50} = 0.863$$

where

FPD = full power days per cycle

DT = scheduled downtime per cycle

It would have been desirable to include in this study the effects of reactor refueling schedules other than the reference design, but core designs employing "shuffling" or other partial fuel replacement modes that utilize shorter refueling periods were not available.

Table 1. LMFBR Reference Design Data

General Data

Total thermal power, MWt	2450
Net electrical power, MWe	1012
Fuel type, core	PuO ₂ /UO ₂
Fuel type, blanket	UO ₂ , depleted
Refueling mode	Complete core replacement
Core burnup, avg MWD/tonne	100,000
Fuel and blanket rod loading method	Vibratory compacted
Oxide density, core and axial blanket, % TD	85
Oxide density, radial blanket, % TD	90
Fuel lifetime, full power days	1034
Average reactor capacity factor	0.863
Fuel lifetime, calendar days	1198
Fuel lifetime, calendar years	3.28
Fraction of reactor power produced in core, avg over lifetime	0.940

Mass Balance Data

Beginning of life

Core loading, kg U	19,970
Core loading, kg Pu fissile	2770
Core loading, kg Pu total	3848
Axial blanket loading, kg U	19,064
Radial blanket loading, kg U	15,300
Total core loading, kg U + Pu	23,818
Total blanket loading, kg U	34,365
Total core plus blanket loading, kg U + Pu	58,183

End of life

Core loading, kg U	17,443
Core loading, kg Pu fissile	2999
Core loading, kg Pu total	4224
Axial blanket loading, kg U	18,236

Table 1. (Cont'd)

Axial blanket loading, kg Pu fissile	518
Axial blanket loading, kg Pu total	548
Radial blanket loading, kg U	14,837
Radial blanket loading, kg Pu fissile	279
Radial blanket loading, kg Pu total	289
Total core loading, kg U + Pu	21,667
Total axial blanket loading, kg U + Pu	19,302
Total radial blanket loading, kg U + Pu	15,405
Total blanket loading, kg U + Pu	34,707
Total core plus blanket loading, kg U + Pu	56,374
<u>Core and Blanket Construction Data</u>	
Fuel pin OD, in.	0.280
Fuel pin cladding thickness, in.	0.010
Fuel pin ID, in.	0.260
Core height, in.	34.7
Axial blanket length (each end), in.	14
Radial blanket pin OD, in.	0.610
Radial blanket pin cladding thickness, in.	0.017
Radial blanket pin ID, in.	0.576
Radial blanket height, in.	35
Number of fuel assemblies	288
Fuel and blanket pin cladding material	SS 304
Number of fuel pins per assembly	331
Number of radial blanket pins per assembly	91
Number of fuel pins per core	95,328
Number of radial blanket pins per core	12,558
<u>Processing Throughput Rates (per reactor, as metal)</u>	
Core kg per process day	22.5
Blanket kg per process day	38.8

3.2.2. Fuel Cycle

Figure 1, a schematic diagram of the fuel cycle, shows the operations included in the integrated on-site facility; these include reprocessing, waste disposal, conversion, and fabrication. The fuel cycle for off-site processing is identical except that it includes shipping operations, and the reprocessing and fabrication operations would be in separate facilities. For the case of on-site reprocessing only, the plutonium and uranium nitrate products of the reprocessing plant would be shipped to the fabrication plant, which includes the conversion operations. We have not studied the problems and costs associated with shipping the nitrates as opposed to including the conversion process in the on-site facilities and shipping the oxide powder.

3.2.3. ANL Ground Rules

For this study we attempted to adhere as closely as possible to the ground rules specified by ANL for the LMFBR Follow-On Study. The nature of this particular trade-off study, however, required that certain costs in the ground rules be modified. For example, in the case of the method specified for calculating the reprocessing cost, it was determined that this corresponded to a centralized plant with a capacity equivalent to about five 1000-MWe reactors of the reference design. As explained in section 3.5, it was necessary to develop a method for estimating reprocessing costs over a range of reactor throughputs, for both on- and off-site modes, of from 1000- to 30,000-MWe capacity. However, we developed the required method so that it agrees with the ground rules cost on a \$ per kilogram basis at the 5000-MWe-capacity point.

We also expanded on the ground rules by calculating the fuel cycle costs as a function of additional values of the parameters specified in the ground rules. For example, we evaluated a plutonium cost of \$20 per gram, in addition to the \$10 value specified, and a shipping cost of \$54 per kilogram, in addition to the one specified.

Figure 1. Schematic Diagram, Fast Breeder Reactor Fuel Cycle

The ANL Follow-On-Study ground rules that are applicable to this trade-off study are listed below:

- "1. The LMFBR power plant will be an economic central station, investor-owned, safe, and reliable, based on technology that can reasonably be assumed to be the state of the art in 1980. This date is the assumed target date for sale of the 1000-MWe LMFBR plant.
- "2. The reactor plant will produce 1000 MWe net and will, as a minimum, meet a utility company's base load system demands.
- "3. The reactor fuel cost will be evaluated for the equilibrium fuel cycle with respect to the isotopic composition of fuel and fertile material and will need only fertile material makeup from outside the fuel and blanket cycle. The fuel cycle cost data are to be applied as follows:

Fissile Pu value, \$/gm	10
Interest on core Pu, %/yr	10*
Reprocessing plant Pu criticality limit, % of ²³⁵ U limit	67**
Reprocessing plant charge (including turnaround), \$/day	30,000**
Shipping cost, core and axial blanket, \$/kg	20
Shipping cost, radial blanket, \$/kg	10
Fabrication cost	Adjusted for design
Fuel fabrication charge, % book value	10*
Nitrate conversion to feed material	Adjusted for design
Fabrication losses (included in Pu credit), %	1
Reprocessing losses (included in Pu credit), %	1
Reactor plant availability factor	100% less downtime for refueling
Reactor capacity factor	90% of plant availability"

* Although a 12 to 13% rate on the plutonium interest and fuel fabrication charges would be more realistic at this time, we adhered to the 10% value because the ground rules were intended to provide a common reference point to permit the comparison of reactors and their fuel cycles.

** Not used in this trade-off study.

3.3. Determination of Trade-Off Points

3.3.1. On-Site Reprocessing and Fabrication

The trade-off points between on- and off-site reprocessing and fabrication were determined by plotting fuel cycle costs for the two modes as a function of the various parameters considered in this study; the points of intersection, i.e., where the fuel costs were equal, constituted the trade-off points. For the cases in which we compared integrated on-site reprocessing and fabrication with separate, off-site reprocessing and fabrication facilities, this procedure resulted in Figures 2, 3, 4, and 5, in which the on-site fuel cycle costs are plotted against the electrical capacity of LMFBRs at one site, and the off-site fuel costs are shown as horizontal lines. The on-site costs in each figure are plotted in three curves—one based on the reference method of calculating fabrication and reprocessing costs, one representing 80% of the reference costs, and one for 50% of the reference costs.

We have not attempted to study separate reductions in either fabrication or reprocessing costs. That is, the 80% curve represents a reduction of 20% in the combined fabrication-reprocessing cost as determined by the reference methods of calculation for each part of the operation. The total reduction could therefore result from either the reprocessing part or the fabrication part, or in both steps. However, the fabrication portion of the cost is considerably higher than the reprocessing portion (about 3 to 1), and the reduction would necessarily depend heavily on savings in the fabrication part of the operation.

The fuel cycle costs for reactors using centralized or off-site processing are shown by horizontal lines on these figures, since they are obviously not a function of the on-site reactor capacity; however, they are dependent on other variables. Thus, in each figure there are four horizontal lines representing the off-site fuel costs based on four different combinations of economic parameters. One set of two lines is based on the "ground rules" shipping costs (\$20 per kilogram of core and axial blanket and \$10 per kilogram of radial blanket) as specified by ANL for the LMFBR Follow-On Study; one of these two lines is further based on a 4-month storage period for the spent fuel before shipping, and one is based on a 12-month storage period. The other set of two

horizontal lines refers to an average spent fuel shipping cost of \$54 per kilogram for core and blanket; this figure is derived from the results of another trade-off study completed in the Follow-On Study.² Again, at this shipping cost, there is one line for a 4-month storage period and one for a 12-month storage period.

Hence, on each of the figures, twelve trade-off points are determined. For example, from Figure 2 we see that if on-site, integrated reprocessing and fabrication facilities can achieve unit costs that are 80% of those computed by the "reference" methods, then a 4000-MWe reactor complex will achieve total fuel cycle costs that are equivalent to those for reactors using centralized facilities, when the fuel is stored for 4 months and the shipping cost averages \$54 per kilogram. The additional economic parameters upon which these curves are based are also indicated on each figure: the plutonium value and the size of the off-site reprocessing plant and the fabrication plant. For example, Figure 2 is based on a plutonium value of \$10 per gram of fissile isotope and a 15-reactor (or 15,000 MWe) centralized plant design capacity. Figure 3 uses a plutonium value of \$20 per gram and a 15,000-MWe capacity. Figures 4 and 5 are based on a 30,000-MWe centralized design capacity, the former using \$10 per gram for plutonium and the latter \$20 per gram.

The trade-off points obtained from Figures 2, 3, 4, and 5 have been arranged in Tables 2 through 5. Referring to Table 2, for example, we read that when centralized facilities are designed for 15,000 MWe, and based on ground rules shipping costs, a plutonium value of \$10 per gram and a 12-month storage period, 3800 MWe of reactor capacity on-site, with integrated reprocessing and fabrication facilities having processing costs equal to 80% of those computed by the "reference" method, would yield fuel cycle costs equal to those for the reactors using the off-site facilities.

The trade-off points from Tables 2 through 5 were then plotted in Figures 13 through 20, which are discussed in the Results section of this report (section 4).

Tables 6a and 6b list a breakdown of the fuel cycle costs for the reactors using off-site reprocessing and fabrication. Table 7 lists two representative sets of fuel costs for on-site reprocessing and fabrication.

Figure 2. Fuel Cycle Costs for On- and Off-Site Reprocessing and Fabrication, Based on \$10/gm Plutonium and 15,000 MWe Off-Site Processing Capacity

Figure 3. Fuel Cycle Costs for On- and Off-Site Reprocessing and Fabrication, Based on \$20/gm Plutonium and 15,000 MWe Off-Site Processing Capacity

Figure 4. Fuel Cycle Costs for On- and Off-Site Reprocessing and Fabrication, Based on \$10/gm Plutonium and 30,000 MWe Off-Site Processing Capacity

Figure 5. Fuel Cycle Costs for On- and Off-Site Reprocessing and Fabrication, Based on \$20/gm Plutonium and 30,000 MWe Off-Site Processing Capacity

Table 2. Trade-Off Points — MWe Capacity at One Site, Based on Ground Rule Shipping Costs and 15,000-MWe Centralized Processing Capacity

	On-site reactor capacity, MWe			
	\$10/gm Pu		\$20/gm Pu	
	4	12	4	12
Spent fuel post-irradiation cooling period, mos				
Based on reference processing costs	8600	6100	8000	4800
Based on 80% of reference processing costs	5500	3800	5500	3400
Based on 50% of reference processing costs	2500	1800	3100	2100

Table 3. Trade-Off Points — MWe Capacity at One Site, Based on \$54/kg Shipping Costs and 15,000-MWe Centralized Processing Capacity

	On-site reactor capacity, MWe			
	\$10/gm Pu		\$20/gm Pu	
	4	12	4	12
Spent fuel post-irradiation cooling period, mos				
Based on reference processing costs	6300	4700	6200	3900
Based on 80% of reference processing costs	4000	3000	4300	2800
Based on 50% of reference processing costs	1900	1500	2600	1750

Table 4. Trade-Off Points — MWe Capacity at One Site, Based on Ground Rule Shipping Costs and 30,000-MWe Centralized Processing Capacity

Spent fuel cooling period, mos	On-site reactor capacity, MWe			
	\$10/gm Pu		\$20/gm Pu	
	4	12	4	12
Based on reference costs	>20,000	10,800	>15,000	7,100
Based on 80% of reference costs	11,400	6,500	11,000	4,900
Based on 50% of reference costs	4,300	2,900	4,800	2,800

Table 5. Trade-Off Points — MWe Capacity at One Site, Based on \$54/kg Shipping Costs and 30,000-MWe Centralized Processing Capacity

Spent fuel cooling period, mos	On-site reactor capacity, MWe			
	\$10/gm Pu		\$20/gm Pu	
	4	12	4	12
Based on reference costs	11,200	7,300	11,500	5,600
Based on 80% of reference costs	6,800	4,700	7,300	3,900
Based on 50% of reference costs	3,000	2,200	3,700	2,300

Table 6a. Fuel Cycle Costs Based on Off-Site Reprocessing and Fabrication (15,000-MWe Plant Capacity)

Economic Parameters

Pu value, \$/gm fissile	10	10	10	10	20	20	20	20
Storage time, mos	4	4	12	12	4	4	12	12
Shipping cost, \$	GR ^(a)	54 ^(b)	GR	54	GR	54	GR	54
Centralized plant capacity, MWe × 10 ³	15	15	15	15	15	15	15	15

Fuel cost, mill/kwhr

Fuel Cost Components

Fabrication cost	0.341	0.341	0.341	0.341	0.341	0.341	0.341	0.341
Reprocessing cost	0.0963	0.0963	0.0963	0.0963	0.0963	0.0963	0.0963	0.0963
Fabrication working capital cost	0.0639	0.0639	0.0639	0.0639	0.0639	0.0639	0.0639	0.0639
Reprocessing working capital cost	(0.0296)	(0.0296)	(0.0361)	(0.0361)	(0.0296)	(0.0296)	(0.0361)	(0.0361)
Pu outpile inventory	0.1483	0.1483	0.2490	0.2490	0.2966	0.2966	0.4980	0.4980
Pu inpile inventory	0.4295	0.4295	0.4295	0.4295	0.8590	0.8590	0.8590	0.8590
Pu credit	(0.382)	(0.382)	(0.382)	(0.382)	(0.764)	(0.764)	(0.764)	(0.764)
Shipping cost	0.038	0.1202	0.038	0.1202	0.038	0.1202	0.038	0.1202
Total	0.705	0.7876	0.799	0.882	0.901	0.9834	1.096	1.178

(a) GR: ground rules cost.

(b) \$54/kg average.

Table 6b. Fuel Cycle Costs Based on Off-Site Reprocessing and Fabrication (30,000-MWe Plant Capacity)

Economic Parameters								
Pu value, \$/gm fissile	10	10	10	10	20	20	20	20
Storage time, mos	4	4	12	12	4	4	12	12
Shipping cost	GR ^(a)	54 ^(b)	GR	54	GR	54	GR	54
Centralized plant capacity, MWe × 10 ³	15	30	30	30	30	30	30	30
Fuel cost, mill/kwhr								
Fuel Cost Components								
Fabrication cost	0.2603	0.2603	0.2603	0.2603	0.2603	0.2603	0.2603	0.2603
Reprocessing cost	0.0639	0.0639	0.0639	0.0639	0.0639	0.0639	0.0639	0.0639
Fabrication working capital cost	0.0489	0.0489	0.0489	0.0489	0.0489	0.0489	0.0489	0.0489
Reprocessing working capital cost	(0.0221)	(0.0221)	(0.0289)	(0.0289)	(0.0221)	(0.0221)	(0.0289)	(0.0289)
Pu outpile inventory	0.1295	0.1295	0.2302	0.2302	0.2590	0.2590	0.4604	0.4604
Pu inpile inventory	0.4295	0.4295	0.4295	0.4295	0.8590	0.8590	0.8590	0.8590
Pu credit	(0.382)	(0.382)	(0.382)	(0.382)	(0.764)	(0.764)	(0.764)	(0.764)
Shipping cost	<u>0.038</u>	<u>0.1202</u>	<u>0.038</u>	<u>0.1202</u>	<u>0.038</u>	<u>0.1202</u>	<u>0.038</u>	<u>0.1202</u>
Total	0.5660	0.6503	0.6599	0.7442	0.743	0.8272	0.9376	1.022

(a) GR: ground rules cost.

(b) \$54/kg average.

Table 7. Fuel Cycle Costs Based on On-Site Reprocessing and Fabrication, mills/kwhr

Based on \$10/gm plutonium and reference processing costs								
Reactor capacity, MWe	Fabrication	Reprocessing	Fabr. WCCC(a)	Repr. WCCC(b)	Pu inv. OP(c)	Pu inv., IP(d)	Pu credit	Total
1000	0.705	0.388	0.150	(0.1018)	0.478	0.4295	(0.382)	1.667
2000	0.577	0.280	0.123	(0.0735)	0.280	0.4295	(0.382)	1.234
3000	0.510	0.222	0.1085	(0.0601)	0.2056	0.4295	(0.382)	1.035
4000	0.470	0.199	0.100	(0.0524)	0.120	0.4295	(0.382)	0.935
5000	0.442	0.177	0.094	(0.0466)	0.153	0.4295	(0.382)	0.867
8000	0.345	0.145	0.091	(0.040)	0.133	0.4295	(0.382)	0.772
Based on \$10/gm plutonium and 50% of reference processing costs								
1000	0.352	0.194	0.075	(0.0509)	0.478	0.4295	(0.382)	1.096
2000	0.289	0.140	0.062	(0.0368)	0.280	0.4295	(0.382)	0.781
3000	0.255	0.111	0.054	(0.0300)	0.206	0.4295	(0.382)	0.643
4000	0.235	0.099	0.050	(0.0262)	0.170	0.4295	(0.382)	0.576
5000	0.221	0.089	0.047	(0.0233)	0.153	0.4295	(0.382)	0.534
8000	0.173	0.073	0.046	(0.020)	0.133	0.4295	(0.382)	0.451

(a) Fabrication working capital carrying charge.

(b) Reprocessing working capital carrying charge.

(c) Plutonium inventory, outpile.

(d) Plutonium inventory, inpile.

3.3.2. On-Site Reprocessing With Off-Site Fabrication

In the case of on-site reprocessing combined with off-site fabrication in centralized facilities, we did not study as many combinations of economic variables as we did for the case of on-site reprocessing and fabrication. The on-site reprocessing-only case eliminates the cost of shipping irradiated fuel assemblies and takes advantage of the low fabrication costs that can be achieved in high-throughput centralized plants. It does, however, eliminate the potential of reducing both fabrication and reprocessing costs by integrating both processes in a single continuous process. It is possible that greater savings can be effected by using the high-capacity fabrication plants than by integrating the facilities on-site.

In this comparison, therefore, the fabrication costs of the two alternatives are equal, and the trade-off hinges on offsetting—in the case of off-site reprocessing plus fabrication—the combination of spent fuel shipping cost, inventory charges during storage, and lower processing costs, with the absence of shipping, possibly shorter cooling periods, and the higher reprocessing costs in the on-site reprocessing-only mode.

In performing these calculations, we were faced with the question of on-site inventory costs when only reprocessing is accomplished on-site. In the case of combined on-site fabrication and reprocessing, the fuel never leaves the site, except for the excess bred plutonium, which is sold as it is recovered. In the reprocessing-only case, we did not want to assign a single reprocessing plant holdup time, since the inventory charge has a greater bearing on the trade-off points than it has in the studies referred to in section 3.3.1; a single value or an arbitrary one would mask the problem of determining the sensitivity or importance of this parameter. We would have preferred to choose, or at least investigate, an optimum combination of throughput and inventory cost at each reactor complex size. Obviously, a higher throughput plant would reduce the in-process holdup time and therefore the inventory cost, since the fuel is assumed to be sold by the utility upon completion of reprocessing and only purchased again when needed for fabrication. However, our reference reprocessing costs, although a function of throughput, are based on operating the plant on a continuous basis (260 processing

days per year), and we did not have available a method of adjusting these figures when a particular plant operated at less than this design capacity, which would be the case for a throughput larger than the one that satisfied the refueling cycle.

Consequently, we decided to use the on-site inventory time as a parameter. We first determined the fuel cycle costs for a range of on-site reactor capacities as a function of the on-site reprocessing holdup time. Figure 6 illustrates the case based on \$10 per gram plutonium and a 4-month cooling period before on-site reprocessing. Next, the trade-off points were determined by plotting the fuel cycle costs for three typical reprocessing holdup times taken from Figure 6 (2, 6, and 12 months); these are the three curved lines shown in Figure 7. In this figure, the off-site fuel costs are also plotted (as horizontal lines); four typical off-site cases are shown, as was done in the previous analysis: for the ground rules shipping cost with 4- and 12-month cooling periods; and for a \$54 per kilogram shipping cost at the same two cooling periods.

Trade-off points based on a \$20 per gram plutonium value, with all the other parameters the same as previously mentioned, were obtained as shown in Figure 8. The next step was to plot the trade-off points obtained from these two figures to determine the relationship of on-site holdup time to the other variables. Thus Figures 21 and 22 were obtained; they are described in section 4.

3.4. Fabrication Cost

Since fabrication costs were not established by ANL in the Follow-On Study ground rules, we decided to utilize the fabrication-cost estimating capability of the Oak Ridge National Laboratory, where for several years a computer code has provided fuel fabrication cost estimates for a variety of reactor types and fuels. This code has also provided the basic fabrication cost data to the Fuel Recycle Task Force of the AEC.

Figure 6. Fuel Cycle Cost for On-Site Reprocessing With Off-Site Fabrication, Based on \$10/gm Plutonium and Reference Processing Costs

Figure 7. Fuel Cycle Costs for On-Site Reprocessing With Off-Site Fabrication and for Off-Site Reprocessing and Fabrication, Based on \$10/gm Plutonium, 15,000 MWe Off-Site Processing Capacity, and Reference Fabrication and Reprocessing Costs

Figure 8. Fuel Cycle Costs for On-Site Reprocessing With Off-Site Fabrication and for Off-Site Reprocessing and Fabrication, Based on \$20/gm Plutonium, 15,000 MWe Off-Site Processing Capacity, and Reference Fabrication and Reprocessing Costs

The ORNL code takes specific fuel pin, fuel assembly, and overall core design data and characteristics and computes unit fabrication costs for this particular design over a range of production throughputs. In other words, a unit fabrication cost is computed for a fabrication plant that is designed specifically to produce that type of fuel assembly and that has a specified production capacity or throughput. The computed fabrication cost covers all operations, starting with the fuel feed material (in this case uranium and plutonium oxide powders) and ending with completed fuel elements or assemblies. The cost for preparing the feed material is not included in the ORNL code and was calculated separately, as described below. The reference fuel is vibratory compacted mixed oxide, and the feed material received at the fabrication plant is assumed to be high-density sol-gel oxide particles.

The fabrication cost incorporates the following assumptions: A capital charge rate of 30% per year is used on depreciating capital. The plant operates 260 days per year with an average reject rate of 5%. All materials other than the nuclear fuel (such as the tubing, end caps, fuel rod spacers, fuel cans, end fittings, springs, etc.) are included in the cost. For this type of fuel, it was assumed that all phases of the fabrication process for the fuel assemblies was accomplished with remote handling because of the shielding requirements of the fuel. The radial blanket assemblies are handled as for PWR plants today. The fabrication plant produces both the core assemblies and the radial blanket assemblies; the respective production lines are sized in accordance with the production capacity specified.

We approached ORNL in March 1967, when the LMFBR Follow-On Study was in the early stages of its work and parametric studies and alternate designs were being investigated in order to arrive at a reference design. We needed a tool with which we could produce in-house fabrication cost estimates for a variety of fuel and core designs. All the designs under consideration varied only in certain geometric details but were similar with respect to the use of uranium-plutonium as the core material, depleted uranium as the axial and radial blanket material, and vibratory compaction as the means of loading the tubes. All the designs used SS 316 as the cladding, and all used the same type of fuel and blanket assembly construction. The basic variables were the fuel and blanket

rod diameters and the lengths of the fuel and the axial and radial blankets.

Our approach, therefore, was to submit a series of cases to ORNL, with each case representing a different design with respect to the diameter and length variables cited above, but sufficient cases were selected so that the resultant costs could be correlated with the variables selected. We also submitted preliminary design drawings of the fuel pin and fuel assembly, so that ORNL could include the appropriate materials, hardware, and construction features in the computer program input as required.

Nineteen core cases and 12 blanket cases were submitted, and ORNL generated a set of unit costs for each of the 31 cases. Each set of costs was a tabulation of the unit cost as a function of production throughput, i. e., there was a distinct unit cost for plants designed for discrete capacities. The cases submitted and the corresponding cost data generated by ORNL are listed in Table 8. In this table there is a series of unit costs in \$/kg for the core and a series for the blanket; these unit costs, for four typical production throughputs, were obtained from the ORNL computer printouts.

We then set about to correlate these data into a computer code of our own so that we could generate our own fabrication costs for any values of the selected variables of our choosing. We correlated the ORNL cost data in a manner patterned after ORNL's, i. e., the core cost was made a function of the core rod ID, the core length, and the axial blanket length; and the unit blanket cost was made a function of the radial blanket rod ID and length. Both core and blanket costs were also made a function of production throughput.

Our code was written to generate the same two types of unit costs as the ORNL code and to combine them into a total fabrication cost in the same manner as that of ORNL. The total core-blanket fabrication cost is thus obtained by multiplying the unit core cost generated by the code by the kilograms of core (as metal); then the unit blanket cost generated by the code is multiplied by the kilograms of material (as metal) in the axial and radial blankets. The sum of these two costs is the total fabrication cost for the combined core and blanket.

Table 8. Fabrication Cost Data

Case No.	Core pin ID, in.	Core fuel length, in.	Axial blanket length, in.	Core rods per assembly	\$/kg core - production rate, kg/day				Radial blanket pin ID, in.	Radial blanket length, in.	Blanket rods per assembly	\$/kg axial/radial blanket - production rate, kg per day			
					200	500	1000	2500				350	800	2000	600
21	0.168	20	12	631	658	538	478	410	0.260	28	300	98.3	79.2	50.8	39.8
24	.198	20	12	469	522	409	365	318	.360	28	169	57.8	37.2	24.9	16.1
27	.243	20	12	331	395	305	265	228	.460	28	108	47.3	30.2	19.2	12.6
30	.268	20	12	271	349	266	228	195	.560	28	75	35.8	22.1	13.2	8.1
37	.318	20	12	217	280	208	178	150	.560	28	75	35.8	22.1	13.2	8.1
22	.168	30	12	631	498	403	355	314	.260	38	300	11.9	84.0	64.5	50.8
25	.198	30	12	469	396	314	272	235	.360	38	169	76.0	50.8	36.1	27.0
28	.243	30	12	331	317	232	198	165	.460	38	108	62.9	41.5	28.1	21.1
31	.268	30	12	271	278	204	171	144	.560	38	75	52.1	32.8	21.9	15.1
38	.318	30	12	217	225	161	135	111	.560	38	75	52.1	32.8	21.9	15.1
23	.168	42	12	631	402	318	278	247	.260	50	300	12.2	89.3	69.8	54.8
26	.198	42	12	469	322	249	212	183	.360	50	169	80.1	52.2	39.8	29.4
29	.243	42	12	331	255	188	158	132	.460	50	108	68.0	45.8	31.2	23.0
32	.268	42	12	271	232	169	142	118	.560	50	75	55.8	36.9	24.9	17.4
39	.318	42	12	217	188	132	109	88	.560	50	75	55.8	36.9	24.9	17.4
33	.168	20	24	631	709	591	531	483	.260	28	300	98.3	79.2	50.8	39.8
34	.198	20	24	469	562	458	403	358	.360	28	169	57.8	37.2	24.9	16.1
35	.243	20	24	331	421	335	293	257	.460	28	108	47.3	30.2	19.2	12.6
36	0.268	20	24	271	372	291	252	218	0.560	28	75	35.8	22.1	13.2	8.1

Basie

Vibratory compacted fuel.
Core cladding thickness of 0.016 in.
Radial blanket cladding thickness of 0.020 in.
Core gas plenum of 36 in.
Radial blanket gas plenum of 6 in.
SS 316 tubing material.

To prepare our fabrication cost code, we developed a surface fit of the ORNL fabrication cost data as a function of the fuel rod ID and the production capacity—both at a given active core length and axial blanket length. This is done using a standard Company surface-fit code called SURF, which performs a least-squares fit of the data as a function of two independent variables. This procedure was repeated first for the active core length and then for the axial blanket length, one at a time, until all the data had been surface-fitted. The result is a total of six surface-fit equations, one for each of the three different active fuel lengths at each of the two axial blanket lengths. These equations are incorporated into our fabrication cost code.

The fabrication cost code solves each of the six surface-fit equations for each of the axial blanket lengths contained in the initial data and retains the solution. Next, the code performs a parabolic fit of the fabrication costs obtained from the solutions of the surface-fit equations; in this operation fabrication costs are found as a function of the active fuel length. Using the desired fuel length in each of the parabolic fit equations, the program calculates the fabrication cost that would occur if the fuel pin had either of the two axial blanket lengths in the initial data. Should the fuel pin have some other axial blanket length, the code fits the fabrication costs from the two known points with a straight line. The desired fabrication cost is then determined by using the desired axial blanket length in the straight-line relationship.

The blanket fabrication cost is determined in a manner analogous to that for the core cost described above, except that only three surface-fit equations are generated. The three solutions are then correlated into a single parabolic-fit equation, which is solved for the fabrication cost. The computer program makes a small modification in the ORNL-based costs because the ORNL code used a 22% capital charge rate and we wanted to use 30% as it is more representative of current industrial practice for plants of this kind. We were able to make this adjustment since the ORNL printouts listed the unit fabrication cost broken down into three components: operating, hardware, and capital costs. We made a number of random calculations by increasing the capital cost component by the ratio of 30 to 22 and then computing new total unit fabrication costs for both the core and blanket. We found that the core unit

costs increased by about 11% and the blanket costs by about 10%; therefore we included these factors in our code.

The costs for converting the uranyl and plutonyl nitrate solution products of the reprocessing plant to UO_2 and PuO_2 sol-gel fragments suitable for vibratory compaction were obtained by personal communication with ORNL personnel as follows:

For preparation of the core mixed oxides

$$\$/\text{kg} = 395 (P)^{-0.52}$$

For preparation of the blanket oxide

$$\$/\text{kg} = 383 (P)^{-0.558}$$

where P is the production rate (kg/day). In both cases, kilograms of fuel refers to the metal. These correlations were also included in the computer program.

The total core and blanket fabrication cost, as listed in the fuel cycle cost breakdowns, is then obtained as the sum of the fabrication costs derived from the ORNL data plus the conversion costs. The total core fabrication cost for the reference design (in millions of dollars) is shown on Figure 9 as a function of the core (without blanket) throughput.

We have extrapolated the ORNL computer code fabrication costs and the conversion costs to quite low and high throughputs, i.e., for one reactor and for thirty reactors. It is not known whether these correlations were meant to be used at such ranges, and some inaccuracy may exist. However, we required cost correlations that were continuous functions from the low capacities corresponding to one reactor for the high capacities of 15 and 30 reactors; the ORNL cost correlations accomplish this.

3.5. Reprocessing Cost

The reprocessing cost data specified by ANL in the Follow-On Study ground rules were modified in performing this particular trade-off study. The ground rules specified a reprocessing plant charge (including turnaround) of \$30,000 per day; in addition, it specified that the reprocessing plant plutonium criticality limit was 67% of the ^{235}U limit.

This information indicates that an aqueous reprocessing plant is to be the basis for costing, and we had therefore adopted the AEC Hypothetical Reprocessing Plant⁷ and its throughput rates (using 67% of the data for the plutonium core) in our routine cost estimating work in the Follow-On Study.

However, for this trade-off study, it was necessary to have reprocessing costs representative of plants designed for a range of capacities rather than for the single hypothetical plant. A number of sources in the literature yielded reprocessing cost data for aqueous systems over a range of plant capacities and designed to reprocess fast breeder fuel, but these data showed a rather wide scatter of costs at the same throughput, as can be seen from Figure 10. However, one source of data was chosen for this study, since the costs are based on the Nuclear Fuel Service (NFS) reprocessing plant which is in operation today. In this study Nicholson investigated the feasibility and cost of reprocessing fast reactor fuel in that facility and developed unit reprocessing costs at different throughputs by estimating the cost of modifying the NFS to handle the higher capacities.¹

Nicholson's data on unit reprocessing cost (in mills per kilowatt-hour) at specified reactor capacities (in MWe) are tabulated in the first two columns of Table 9. We converted the mills per kilowatt-hour figures to \$ per kilogram by using Nicholson's average burnup figure of 38,500 MWD per tonne for core and blankets and the 40% electrical efficiency factor. The capacity figures in the table in tonnes per day were calculated from Nicholson's data by using his figure of 18.9 tonnes per year as the throughput per reactor and the NFS operating basis of 227 days per year.

Table 9. Reprocessing Costs

Fast reactor capacity, MWe(a)	Unit reprocessing cost, mill/kwhr(a)	Plant reprocessing costs, \$/kg(b)	Capacity, tonne/day(b)
1,000	0.36	133	0.0833
3,000	0.23	85	0.250
10,000	0.11	40.7	0.837
16,700	0.09	33.3	1.39

(a) From reference 1.

(b) Obtained by appropriate conversion factors.

Figure 9. Reference Core and Blanket Fabrication Cost
Vs Plant Throughput

Figure 10. Aqueous Fast Reactor Reprocessing Costs

The last two columns of the table are plotted in Figure 11 and are identified as ORNL-TM-1784. However, in order to bring these costs closer to a consistent basis with the ANL ground rule cost data, this curve was adjusted so that the ANL ground rule cost would fall on it. This was done by first calculating the unit reprocessing cost for the reference core in \$ per kilogram in accordance with the ground rules, and then determining the throughput of the hypothetical plant for the reference design fuel. This point is also shown in Figure 11. A new curve was then drawn through this point and parallel to the Nicholson report curve and was used as the "reference" reprocessing cost relationship in this study for cost versus plant capacity. (This adjusted curve is also shown in Figure 11.)

3.6. Fuel Inventory Charges

3.6.1. Fuel Value

The electrical utility incurs working capital carrying charges on the value of the fuel throughout the time it owns the material, i. e., both incore and out-of-pile. In this study the fuel value is considered to be the value of the contained fissile plutonium; the value of the depleted uranium is low enough to be neglected in these calculations. Two values of plutonium were used as parameters in the study: \$10 and \$20 per gram of fissile isotope.

3.6.2. Out-of-Pile Inventory — On-Site Processing

Inventory costs for the on-site reprocessing and fabrication alternative were determined in a somewhat different and more complex manner than those for the off-site case; in both instances simplified models were used. In the case of on-site reprocessing and fabrication, the method of calculation can best be explained by illustrating a typical example: the case of a single reactor with its on-site processing facility is explained below.

For the reference design reactor there is a refueling shutdown each 1198 calendar days, based on an average annual capacity factor of 86.3%. At that time, the entire core and the blanket are removed and replaced with fresh fuel and blanket assemblies. The core and blanket assemblies are stored for 120 days and are then sent to reprocessing

for recovery of the plutonium and uranium. The excess bred plutonium is recovered and sold to other reactors; the amount required for the fresh fuel is refabricated, together with the recovered depleted uranium (plus a small amount of makeup uranium) for loading into the reactor. It is assumed that, in addition to the 120-day cooling period, there are additional holdup and storage periods totaling 45 days. Thus the time available in a cycle (between reactor refuelings) for reprocessing and fabrication is $1198 - (120 + 45)$ or 1033 days; the fraction of the discharged fuel that is processed in a reactor cycle is therefore $1033/1198$, or 0.862. The remaining fraction, 0.138, is held over to the next cycle for completion of the reprocessing and fabrication operations. These holdup times, based on data in ANL-7137, are summarized in Table 10.

Figure 12 is a simplified flow chart for the on-site fuel, showing the duration of the storage and processing periods and the quantities of fuel in each step, expressed as a fraction of the end-of-life (discharged) or fresh (beginning-of-life) fuel. Again, the figure pertains to a single reactor. During the reprocessing and fabrication operations, it is assumed that the excess bred fuel, which is the difference between the EOL and BOL quantities, is continuously recovered and sold. Thus, to simplify the calculations, an average quantity is assumed to be in-process, as indicated in the diagram. The effects of fuel losses are neglected. As shown by the calculations below the diagram, the average out-of-pile fraction is determined to be 1.34 times the beginning-of-life quantity.

In the case for two or more reactors with on-site reprocessing and fabrication, it is assumed that the reactor refueling periods are staggered to provide equal intervals, and that as fuel becomes available from the reprocessing-fabrication facility, it is loaded into either one of the reactors at the earliest possible opportunity, rather than recycled to the reactor from which it was discharged. Table 11 lists the out-of-pile fraction for on-site reprocessing and fabrication of from one to eight reactors, calculated in accordance with the simplified method discussed above.

Table 10. On-Site Reprocessing and Fabrication Fuel Holdup Times

	<u>Days</u>
Holdup time for cooling discharged fuel	120
Holdup time in reprocessing	10
Analysis and batch preparation	5
Holdup in fabrication	15
Holdup storage of fabricated fuel at reactor	<u>15</u>
Sum of holdup times above	165

Table 11. Out-of-Pile Fraction — On-Site Processing

<u>No. of reactors on-site</u>	<u>Out-of-pile fraction per reactor^(a)</u>
1	1.34
2	0.784
3	0.569
4	0.476
5	0.426
8	0.364

(a) The outpile inventory is the fraction times the fresh fuel core loading.

Figure 11. Reference Reprocessing Cost Vs Plant Throughput

Figure 12. Simplified Model for Out-of-Pile Plutonium Inventory - One Reactor With On-Site Reprocessing and Fabrication

$$\begin{aligned}
 \text{Out-of-pile fraction} &= \left[150 D(EOL) + 1033 D \frac{BOL + EOL}{2} + 15 D(0.862 + 0.138)BOL \right. \\
 &\quad \left. + 15 D(0.138) \frac{BOL + EOL}{2} + 1183 D(0.138) \frac{BOL + EOL}{2} \right] / 1198 D \\
 &= 1.34(BOL)
 \end{aligned}$$

where
 EOL = discharge loading, kg
 BOL = fresh fuel loading, kg
 D = days

Note: Figures in boxes denote duration and average inventory in that segment of cycle.

3.6.3. Out-of-Pile Inventory — Off-Site Processing

For the off-site reprocessing and fabrication cases, the out-of-pile inventory cost was calculated on the following basis:

1. Two alternate values of spent fuel cooling were used as parameters: 120 days and 1 year.

2. Spent fuel shipping plus holdup at the reprocessing plant requires two months.

3. Reprocessing time was determined by dividing the cycle time of 1198 calendar days by the reprocessing plant design capacity, expressed as the number of 1000-MWe reactors. Two sizes of reprocessing plants were considered in the study: those for 15 and 30 reactors. For the first, the reprocessing time is 2.65 months, and for the second case it is 1.33 months.

4. Upon completion of reprocessing, the utility sells the entire lot of plutonium recovered; it purchases the amount it requires for a fresh fuel loading only when fabrication begins.

5. The spent fuel inventory period is therefore 8.65 months after reactor discharge in the case of a 15-reactor-capacity reprocessing plant, and 7.33 months for a 30-reactor-capacity plant.

6. Fresh fuel fabrication time is calculated in the same manner as reprocessing time; a fabrication plant handling 15 reactors requires 2.65 months, and one handling 30 reactors requires 1.33 months.

7. Shipping the fresh fuel to the fabrication plant and from the plant to the reactor site is assumed to require an additional 20 days.

8. The entire fresh fuel reactor loading is assumed to be in storage at the reactor site for 30 days before refueling begins.

3.6.4. In-Pile Inventory

The in-pile inventory cost, which is the same for both on- and off-site cases, is based on the arithmetic average of the initial and final plutonium loadings in the core and blanket; this quantity times the cycle time and a charge rate of 10% yields the in-pile inventory charge per cycle.

3.7. Spent Fuel Storage Period

Storage and cooling periods of 4 and 12 months (before shipping spent fuel assemblies) were used as parameters in computing off-site fuel costs in order to determine the sensitivity of the trade-off points to this parameter. Commercial reprocessing facilities now require that spent thermal reactor fuel be stored for a minimum of 150 days after reactor discharge before reprocessing can begin. This period of time is dictated by the reprocessing plant to allow for (1) iodine decay (since the storage capacity and emission rate of iodine by the plant is limited), and (2) reduction of the total heat and radioactivity of the fuel to prevent the destruction of the solvents and to permit adequate cooling of the reprocessing solutions. The 4-month cooling time (the additional 30 days is taken up in shipment and storage time at the reprocessing plant) has therefore become a standard for thermal reactor fuel cycle cost evaluation work and is commonly used for breeder reactors. However, there has been concern about the feasibility of shipping fast breeder fuel after only 4 months of cooling, due to the high power density of the fuel, the close-packed lattice of the fuel rods in an assembly, and the shipping regulations which required the assumption that any coolant could be lost in an accident. These three conditions would make adequate cooling of the fuel in an accident difficult or impossible until about a 12-month period of storage has elapsed, when the decay heat would be low enough to make cooling feasible. The long storage periods are of particular concern since the high monetary value of the fuel would incur excessive working capital carrying charges. The high iodine content of fast breeder fuel may also require long cooling or storage periods.

The requirement for assuming that all coolant is lost from the shipping cask during an accident has recently been deleted from the shipping regulations. The cooling problem has thus been alleviated, and a one-year storage period may no longer be necessary. However, it is not known if the 4-month period that is suitable for water reactor fuel will be satisfactory for fast reactor fuel, since other constraints may become operative. For example, the high fission product activity (at the short cooling period) may degrade the solvents used in reprocessing operations. While there are now known methods for adequately treating degraded solvents, they may be inadequate if the solvent is sufficiently degraded by

prolonged contact with highly active fast reactor fuel solutions. In summary, there appeared to be sufficient question regarding the storage period that may be required to justify investigating this parameter.

3.8. Spent Fuel Shipping Cost

Spent fuel shipping costs are of concern for the reasons discussed in section 3.7. Unit shipping costs for breeder reactor fuel are expected to be considerably higher than those for thermal reactor fuel (on a per kilogram basis) because of the limitations on the amount of fuel that can be carried per cask; this can be attributed to the difficult heat removal problems and the high concentration of fissile material. An additional difficulty in shipping may be encountered if breeder fuel consists of vented pins; the complications of cask design and of handling both at the reactor and at the reprocessing plant have not been fully explored.

The ANL ground rules for the Follow-On Study specified shipping costs of \$20 per kilogram for the core and axial blanket and \$10 per kilogram for the radial blanket. A recent study² performed under the Follow-On Study determined that costs could be as high as \$54 per kilogram, averaged over the core and blanket (the ground rule costs amount to an average of \$17 per kilogram for core and blanket). Faced with this wide difference in cost, we decided to perform the trade-off study using both sets of values, in the hope that they would bracket the range of interest.

The shipment of fresh fuel is assumed to be of comparatively small cost and is not included in our fuel cycle costs.

4. RESULTS

4.1. General

The results of this trade-off study are summarized in a number of figures which indicate graphically:

1. The minimum reactor complex size required to make fuel cycle costs with on-site reprocessing and fabrication equivalent to those using off-site processing facilities.

2. The reprocessing and fabrication costs that must be achieved in integrated on-site reprocessing and fabrication facilities to produce fuel cycle costs that are competitive with costs using off-site facilities.

3. The maximum on-site fuel inventory period permissible with on-site reprocessing but off-site fabrication to produce fuel costs that are competitive with reactors using totally off-site processing facilities.

The graphical results also indicate the relationship and sensitivity of the various economic parameters to the trade-off points.

Figures 2 through 5 and Tables 2 through 5 present the minimum reactor complex size trade-off points referred to in item 1 above. These figures and tables are included and discussed in section 3 of this report.

The relationships between the required on-site processing costs and the size of the reactor complex that are trade-offs with centralized processing fuel costs, referred to in item 2 above, are shown in Figures 13 through 20 and are discussed in this section of the report.

The on-site inventory period trade-off in the case of on-site reprocessing only, referred to in item 3 above, is shown in Figures 21 and 22. The on-site reactor capacity trade-off points in the comparison

of this alternative with totally off-site processing fuel costs are shown in Figures 7 and 8 and discussed in section 3.3.2.

The trends produced in all of the various trade-off relationships are predictable and can be summarized briefly as follows: On-site reprocessing and fabrication tends to become more economically competitive with the use of off-site processing facilities when

- there are a large number of reactors at one site and the on-site facilities have large capacities and therefore low unit costs;
- longer cooling periods are required before spent fuel can be shipped to off-site reprocessing plants;
- spent fuel shipping costs are high;
- on-site reprocessing and fabrication costs are considerably lower than the "reference" costs used in this study;
- centralized, off-site facilities have small capacities and therefore high unit costs.

A more detailed discussion of the trade-off results is presented in sections 4.2 and 4.3; a summary of the conclusions that can be drawn from the study is given in section 2, Conclusions and Recommendations.

4.2. On-Site Reprocessing and Fabrication

Figures 13 through 20 summarize the economic trade-off between on- and off-site reprocessing and fabrication. Each figure is a plot of the LMFBR capacity at a single site (in MWe), with on-site reprocessing and fabrication, versus the percentage of the "reference" reprocessing and fabrication cost that must be achieved in these on-site processing plants to realize fuel cycle costs equal to (i.e., a trade-off with) the costs in reactors at separate localities using off-site or centralized reprocessing and fabrication facilities. In each figure there are two curves—one for a 4-month fuel storage period before shipping to off-site reprocessing, and one for a 12-month storage period.

Other variables are also represented. Figures 13 and 14 are for the "ground rules" shipping cost and for centralized processing facilities designed to handle fifteen 1000-MWe reactors; two values of plutonium are used—\$10 and \$20 per gram, as indicated in the figures. Figures 15 and 16 are for a \$54 per kilogram average shipping cost for core and blanket, with the other variables the same as in Figures 13 and 14.

Figure 13. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figure 14. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figure 15. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figure 16. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figures 17 and 18 are for centralized processing plants designed for thirty 1000-MWe reactors, for the "ground rules" shipping cost, and for the two values of plutonium mentioned above. Figures 19 and 20 are for the \$54 per kilogram shipping cost, with the other parameters the same as in Figures 14 and 15.

These figures can be used to determine the degree to which the fabrication and reprocessing costs adopted as the "reference" for this study must be reduced by integrating both processes into a single facility in order to make on-site processing economical. The "reference" costs are not a single unit fabrication cost or a unit reprocessing cost, but rather, they refer to the costs computed by the "reference" methods described in this report—methods which we believe are the most consistent with the ANL ground rules for the LMFBR Follow-On Study (of which this trade-off study is a part) and which are being used by the AEC Fuel Recycle Task Force in the evaluation of the national nuclear economy.

As an example, when we examine Figure 14, we see that if only a 4-month cooling period is required before shipping fuel to the reprocessing plant, an 8000-MWe reactor complex with on-site reprocessing and fabrication is required (based on using the "reference" fabrication and reprocessing costs for the on-site facilities) to achieve fuel costs that can compete with those for reactors shipping fuel to off-site reprocessing and fabrication plants that are designed to handle 15,000 MWe of reactor capacity. If a reactor capacity of only 3000 MWe is at one site, the on-site fabrication and reprocessing costs must be 50% of those calculated by the "reference" method for reprocessing and fabrication facilities designed for that capacity.

The case of a single 1000-MWe reactor with its own on-site reprocessing and fabrication facilities may be of particular interest. The fraction of the "reference" fabrication and reprocessing costs that must be achieved to make fuel costs competitive with centralized facility costs is obtained by extrapolating the curves of Figures 13 through 20. Table 12 lists the results of this extrapolation. As might be expected, in all cases studied, a drastic reduction in cost must be effected to achieve a competitive fuel cost. From the table, we see that the costs are from 2% to 35% of the "reference" costs. It is doubtful that such

a reduction could be achieved merely by combining the two operations into an integrated facility.

We may take a different viewpoint, however, in assessing the percentage of "reference" fabrication and reprocessing cost as a parameter. In sections 3.4 and 3.5 of this report are given the actual values of the "reference" fabrication and reprocessing costs as well as an explanation of their derivation or source. These costs are expressed, in the case of fabrication, as millions of dollars to fabricate the entire core and blanket of the reference design as a function of the fabrication plant design throughput. For the reprocessing cost, it is expressed as \$ per kilogram versus the design throughput. Thus, in our trade-off studies, a 50% reduction in cost at a particular throughput, for example, can be viewed as a target to be achieved by any means, i.e., by a change in the process itself (as for example, using fluoride volatility or pyrochemical reprocessing instead of aqueous), or by taking the position that the "reference" costs for the processes they represent are either too optimistic or too pessimistic, or, by integrating the two processes into a continuous process in a single facility.

4.3. On-Site Reprocessing Only

The results of the trade-off between on-site reprocessing only with off-site fabrication in centralized, large scale facilities, and totally off-site reprocessing and fabrication are shown graphically in Figures 21 and 22. In these figures the on-site reactor complex size (with on-site reprocessing) is plotted against the post-irradiation period of storage plus the reprocessing time for the on-site reprocessing mode. Figure 21 is based on a \$10 per gram plutonium value, and Figure 22 is based on \$20 per gram. Both figures plot the relationship between reactor capacity and on-site time as a function of two shipping costs for irradiated fuel assemblies and two typical cooling periods before such shipment is permitted. Both figures are also based on centralized processing plants of 15,000 MWe capacity.

These two figures indicate the sensitivity of the on-site storage and reprocessing time to the size of the economic reactor complex. For example, in Figure 21 we see that a relatively small increase in on-site time greatly affects the size of the reactor complex required

to be competitive with the case where fuel assemblies can be shipped after only 4 months. The effect on required complex size is much less when a 12-month storage period is required.

Viewing the curves in Figure 21 differently, we see that if fuel can be reprocessed on-site after 6 months (including both storage and in-process time), and if a 12-month period is required for off-site shipment, plus a \$54 per kilogram shipping cost, then a single 1000-MWe reactor with on-site reprocessing would be economical. Even at the more optimistic conditions of a 4-month preshipping time plus the lower ground rule shipping costs, only 4600 MWe in a reactor complex is required to be competitive, again based on a 6-month total, post-irradiation, on-site inventory period.

The use of these figures as well as the figures on which they are based (Figures 7 and 8) depends on the assumption that the "reference" reprocessing costs used in the calculations are valid over a range of on-site total inventory periods. This matter is discussed in section 3.3.2.

Table 12. Trade-Off of Single 1000-MWe Reactor
With On-Site Reprocessing and Fabrication

		\$10/gm fissile Pu value							
		4				12			
		Ground rules		\$54/kg		Ground rules		\$54/kg	
Cooling period, mos									
Shipping cost									
Off-site capacity -									
No. of reactors		15	30	15	30	15	30	15	30
% of reference fabrication									
& reprocessing cost ^(a)		22	18	27	28	30	26	35	32
		\$20/gm fissile Pu value							
		4				12			
		Ground rules		\$54/kg		Ground rules		\$54/kg	
Cooling period, mos									
Shipping cost									
Off-site capacity -									
No. of reactors		15	30	15	30	15	30	15	30
% of reference fabrication									
& reprocessing cost ^(a)		18	2	11	20	18	10	21	20

^(a) Percent of reference reprocessing and fabrication cost required to achieve competitive fuel costs with centralized processing. Values are obtained by extrapolation of Figures 13 through 20 and are, therefore, approximate.

Figure 17. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figure 18. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figure 19. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figure 20. Trade-Off Relationship Between On- and Off-Site Reprocessing and Fabrication - % of Reference Reprocessing and Fabrication Costs Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing

Figure 21. Trade-Off Relationship Between On-Site Reprocessing With Off-Site Fabrication - Months of On-Site Storage Plus Reprocessing Vs On-Site Reactor Capacity for 4- and 12-Month Storage Periods Before Shipment of Fuel to Off-Site Reprocessing and for Ground Rules and \$54/kg Shipping Costs

Figure 22. Trade-Off Relationship Between On-Site Reprocessing With Off-Site Fabrication and Off-Site Reprocessing and Fabrication - Months of On-Site Storage Periods Before Shipment of Fuel to Off-Site Reprocessing and for Ground Rules and \$54/kg Shipping Costs

REFERENCES

- ¹ Nicholson, E. L., Preliminary Investigation of Processing Fast Reactor Fuel in an Existing Plant, ORNL-TM-1784, May 1967.
- ² Masseth, F. X., Shipping Cask Trade-Off Study, TRG-68-4, February 1968.
- ³ Levenson, M., Trice, V. G., Jr., and Mechan, W. J., Comparative Cost Study of the Processing of Oxide, Carbide, and Metal Fast Breeder Reactor Fuels by Aqueous, Volatility, and Pyrochemical Methods, ANL-7137, May 1966.
- ⁴ Safety and Economic Characteristics of a 1000-MWe Fast Sodium-Cooled Reactor Design, GEAP-4939.
- ⁵ Carlsmith, R. S., Delene, J. G., and Jung, L., Startup of Fast Reactors on ²³⁵U Fuel, ORNL-TM-1800, May 1967.
- ⁶ Roberts and Stockdale, ORNL Memo August 11, 1967.
- ⁷ Guide to Nuclear Power Cost Evaluation, Vol 4, TID-7025, March 15, 1962.

CONTENTS

	Page
1. INTRODUCTION	1
1.1. Purpose	1
1.2. Summary	2
1.3. System Requirements	3
2. CONCLUSIONS	6
3. DISCUSSION	7
3.1. Introduction	7
3.2. Ground Rules	8
3.3. System Description — Pot Arrangement With Under-the-Plug Refueling	10
3.4. System Description — Pot Arrangement With Hot Cell Refueling	23
3.5. System Description — Spread-Out Arrangement With Through-the-Plug Refueling	29
3.6. System Description — Spread-Out Arrangement With Hot Cell Refueling	37
4. RESULTS	70
4.1. General	70
4.2. Evaluation of Pot Arrangements	70
4.3. Evaluation of Spread-Out Arrangements	74
4.4. Selection of Most Desirable Reactor System	75
APPENDIXES	
A. Equipment and Facilities List	A-1
B. Refueling Time Estimates	B-1
C. Differential Capital Costs	C-1

List of Tables

Table

1. Pot Arrangement Comparative Advantages	78
2. Spread-Out Arrangement Comparative Advantages	78
3. Comparative Advantages	79

Tables (Cont'd)

Table	Page
A-1. Pot Arrangement — Equipment and Facilities List	A-2
A-2. Spread-Out Arrangement — Equipment and Facilities List	A-4
B-1. Summary — Refueling Time Estimates	B-4
B-2. Pot Arrangement Under-the-Plug Refueling	B-5
B-3. Pot Arrangement Under-the-Plug Refueling	B-6
B-4. Pot Arrangement Hot Cell Refueling	B-7
B-5. Pot Arrangement Hot Cell Refueling	B-8
B-6. Spread-Out Arrangement Through-the-Plug Refueling . . .	B-9
B-7. Spread-Out Arrangement Through-the-Plug Refueling . . .	B-10
B-8. Spread-Out Arrangement Hot Cell Refueling	B-12
B-9. Spread-Out Arrangement Hot Cell Refueling	B-13
C-1. Pot Arrangement Differential Capital Costs — Summary	C-3
C-2. Pot Arrangement Differential Capital Costs — Details . .	C-4
C-3. Spread-Out Arrangement Differential Capital Costs — Summary	C-6
C-4. Spread-Out Arrangement Differential Capital Costs — Details	C-7

List of Figures

Figure	
1. Reactor System Selection Diagram	4
2. Pot Arrangement With Under-the-Plug Refueling — Plan View	49
3. Pot Arrangement With Under-the-Plug Refueling — Section Z-Z	50
4. LMFBR Fuel Element Diagram for Under-the-Plug Refueling	51
5. LMFBR Under-the-Plug Refueling Mechanism	52
6. LMFBR Under-the-Plug Fuel Assembly Grapple Tool . . .	53
7. LMFBR Fuel Transfer Machine	54
8. LMFBR Fuel Transfer Machine — Cross Section Near Base	55
9. Pot Arrangement With Hot Cell Refueling — Plan View . .	56
10. Pot Arrangement With Hot Cell Refueling — Section Z-Z	57
11. LMFBR Fuel Element in Hot Cell — Flow Diagram . . .	58
12. Pot-Type Hot Cell Fuel Handling Machine	59
13. Pot-Type Hot Cell Fuel Handling Machine — Cross Sections	60
14. Spread-Out Arrangement With Through-the-Plug Refueling — Plan View	61

Figures (Cont'd)

Figure	Page
15. Spread-Out Arrangement With Through-the-Plug Refueling — Section Y-Y	62
16. Spread-Out Arrangement With Through-the-Plug Refueling — Section Z-Z	63
17. Fuel Element Flow Diagram — LMFBR Spread-Out Concept With Through-the-Plug Refueling.	64
18. Spread-Out Arrangement With Hot Cell Refueling — Plan View.	65
19. Spread-Out Arrangement With Hot Cell Refueling — Section Y-Y.	66
20. Spread-Out Arrangement With Hot Cell Refueling — Section Z-Z.	67
21. LMFBR Fuel Element Flow Diagram — Spread-Out Concept With Hot Cell.	68
22. LMFBR Spread-Out-Type Hot Cell Fuel Handling and Fuel Transfer Machine	69

1. INTRODUCTION

The solution of many technical problems is required for the development of a successful LMFBR. The LMFBR Program Plan outlines an approach to these problems;¹ included are detailed studies of pot- and loop-type reactor vessels and various refueling methods. The conclusions may not be clear cut, since many facets of plant design are involved in both studies. Safety factors will probably be the primary consideration in the final analysis, but economics, performance, ease of maintenance, and customer preference will also influence the design choice.

1.1. Purpose

This trade-off study was conducted to assess, on a conceptual basis, the relative merits of different plant arrangements and refueling schemes and to make a preliminary choice of the most desirable combination of plant arrangement and refueling method. To accomplish this aim, the two trade-offs described below, which are specified in B&W's Follow-On Study contract, were combined into one study.

1. A single primary tank versus a system having primary sodium pumps and IHXs placed external to the core.
2. An open cell versus an under-the-plug refueling system.

A through-the-plug refueling system was also studied, so that the overall objective—to study all of the possible combinations of arrangement and refueling method—would be met. The investigation was thorough enough to establish the safety, performance, maintenance, economic, and development requirements of the system selected. Many possible combinations of features were studied to narrow the field of candidate systems, and all the features examined cannot be presented in detail.

The final selection of features for detailed study, however, included both plant arrangements and three refueling systems.

Efforts in the study of refueling systems have been concentrated largely on refueling, handling, and storage of new and spent fuel assemblies and their movement into and out of the reactor containment structure. Fuel reprocessing and waste disposal have received limited attention.

Efforts in the study of plant arrangement have been concentrated on systems and components that differ from one arrangement to another. These changes and their effects on plant cost and maintenance were studied, and any design features that could be used without change in either arrangement with little or no penalty were retained. For example, tandem pump-IHX units are shown in each arrangement, even though the B&W conceptual design was changed to separated units. Because this study had already begun before that decision was final, and because our appraisal showed that possible penalties for the two arrangements were roughly equal, the tandem pump-IHX concept was retained for this work.

Each concept is described in considerable detail to minimize the need for reference to other documents. Descriptions and drawings of each plant layout are included, as are descriptions and sketches of the major components. Each concept is represented by time studies and cost estimates, which are used, along with an examination of the safety, performance, and maintenance factors, in a final evaluation to select the best combination of plant arrangement and refueling method for the LMFBR.

1.2. Summary

All of the arrangement-refueling combinations indicated by the objectives in section 1.1 of this report have been studied. The results of our Task I efforts and the four major combinations studied formed the basis for the work reported herein. The plant designs were perturbed by varying the refueling schemes. The major objective, to reduce the number of candidate combinations, was accomplished by discarding the less-favorable schemes on the basis of comparison; that is, the spread-out arrangement with through-the-plug refueling (SOTP) was selected over the same arrangement with under-the-plug refueling (SOUP) on the

basis of design complexity, estimated reactor vessel cost, and estimated R&D requirements.

Based on this preliminary work, the following four arrangement-refueling combinations were selected for detailed trade-off study:

1. Pot arrangement with under-the-plug refueling (PUP).
2. Pot arrangement with hot cell refueling (PHC).
3. Spread-out arrangement with through-the-plug refueling (SOTP).
4. Spread-out arrangement with hot cell refueling (SOHC).

Although this selection was somewhat arbitrary, it utilized the best information available and covered the intended scope—both plant arrangements and all of the refueling methods.

As shown in Figure 1, the next stages of the study involved (1) comparing the two-pot arrangements to determine the best refueling method for this layout and (2) comparing the two-loop (spread-out) arrangements to determine the best refueling method for this type of plant. These two "optimum" combinations were then evaluated and compared to select the most desirable reactor system.

Since refueling downtime contributes to overall plant economics, detailed time estimates for refueling were prepared. The plant arrangement and the type of refueling will cause variations in plant equipment; hence, effects will be felt in both plant capital cost and the performance and maintenance of the components. Therefore, differential costs were estimated, and performance and maintenance were assessed. These factors were given major emphasis in the study.

The results of the evaluations indicated that the pot arrangement with under-the-plug refueling is the prime candidate for use in the LMFBR system.

1.3. System Requirements

The overall function required of the reactor plant is to provide economical electric power, and therefore the systems within the complex must perform economically. Each system and each component therein must also have stringent requirements for safety, reliability, and maintenance, so that the overall plant requirements can be met.

Figure 1. Reactor System Selection Diagram

The study of the plant arrangement-refueling combinations requires consideration of almost all the plant systems and, hence, almost all of the detailed requirements. The major systems in the LMFBR are listed below.

1. Reactor vessel system.
2. Reactor core system.
3. Reactivity control system.
4. Reactor coolant system.
5. Reactor support systems (refueling).
6. Safety system.

The specific requirements of these individual systems are many, especially where interfaces between systems are considered. Further complications in defining the requirements of this study are proposed component designs that are either completely new or not fully developed. Therefore, for the purposes of this study, the following requirements were defined, and each arrangement-refueling combination candidate will be evaluated for its compliance to the requirements.

1. Vented fuel assemblies are assumed.
2. The rate of decay heat generation from a spent fuel assembly is based on an average exposure of 100,000 MWd/T.
3. A minimum decay storage time at the reactor site is planned.
4. The transfer of radioactive sodium will be minimized.
5. The removal of components to permit refueling will be minimized, e. g. , reactivity control drives.
6. Although they are not necessarily detailed, the evaluation of refueling components will give due consideration to push-pull capability, force-limiting ability, instrumentation, and the capacity for handling more than one fuel assembly.
7. All components, although (again) not in detail, should be evaluated on the basis of such factors as reliability and the requirements for cooling, maintenance, and inspection.
8. All components will be designed and fabricated in accordance with the requirements of Section III of the ASME Code (Nuclear Vessels) or other applicable specifications.
9. All components that become an integral part of the reactor cover structure will be designed to meet the requirements set forth for that structure. (Note that the internal surfaces of the hot cell structure and the fuel transfer machine become a part of the reactor vessel when the plugs are removed.)

2. CONCLUSIONS

The trade-off study shows that the pot arrangement with under-the-plug refueling is the most desirable design for the LMFBR system. The next best design is the compact spread-out arrangement with through-the-plug refueling. From this evaluation, no strong economic or performance incentive was observed that would justify the use of hot cell refueling with either a pot or spread-out arrangement.

Even though the pot with under-the-plug refueling has the largest reactor vessel, it has a greater concentration of primary equipment closer to the reactor center line and, hence, a considerably smaller reactor building. As a result, the capital costs for this system are lower than those for the compact spread-out arrangement with through-the-plug refueling. The pot arrangement with under-the-plug refueling requires much less refueling downtime than do the other plants studied in this trade-off. A savings of 36% is realized over the spread-out arrangement with through-the-plug refueling.

During refueling of the pot-under-the-plug arrangement, the fuel assemblies being transferred remain completely immersed in the circulating sodium pool; this provides greater assurance against the loss of coolant during spent fuel transfer. In the pool concept, all of the primary sodium is contained in the reactor vessel instead of being dispersed in many separate vessels and associated piping. Hence, the possibility of a leak in the primary system is greatly reduced, and plant safety and reliability are enhanced. This also reduces the leak-tightness requirements for piping, IHXs, and other equipment, since a leak only means slightly lower efficiency and not a breach of primary system boundaries. The large sodium pool is also more capable of reducing some thermal shocks that might arise, and this leads to longer service life for the primary system components.

3. DISCUSSION

3.1. Introduction

In performing a trade-off study of this type, one of the first questions to arise is how broad the study can be and still produce meaningful results. A number of basic plant arrangements and refueling methods are possible. In addition, many variations to, and combinations of, the basic plant arrangements and refueling methods expand the number of plant concepts that are possible.

During the Task I effort of this contract, four arrangement-refueling combinations were investigated:²

- Concept I — Pot arrangement with through-the-plug refueling.
- Concept II — Compact spread-out arrangement with hot cell refueling.
- Concept III — Spread-out arrangement with hot cell refueling.
- Concept IV — Pot arrangement with under-the-plug refueling.

These combinations were selected as stated in section 3.1, reference 2.

The concepts covered all of the major types of refueling but not all of the arrangement-refueling combinations that required evaluation for a satisfactory trade-off. The study did indicate a strong incentive to use under-the-plug instead of through-the-plug refueling, and in-pot instead of out-of-pot storage in conjunction with the pot arrangement. It also indicated an incentive to utilize the compact spread-out arrangement instead of the conventional spread-out arrangement and an out-of-pot storage for the spread-out plant.

In outlining this trade-off, it was felt that the Task I effort was useful, but that a three-step approach was necessary to obtain a more comprehensive evaluation:

1. Determine the best refueling method for the pot arrangement.
2. Determine the best refueling method for the compact spread-out arrangement.
3. Compare and evaluate these two "optimum" combinations to select the most desirable reactor system.

Using the information from Task I and the contract requirements for the trade-offs (see section 1.1), the four reactor systems that formed the basis for this trade-off evolved:

1. Pot arrangement, in-pot fuel storage, under-the-plug refueling.
2. Pot arrangement, in-pot fuel storage, hot cell refueling.
3. Compact spread-out arrangement, out-of-pot fuel storage, through-the-plug refueling.
4. Compact spread-out arrangement, out-of-pot fuel storage, hot cell refueling.

Each of these combinations is described briefly in subsequent sections.

3.2. Ground Rules

In an effort to reduce the variables between the four plants studied, facilities and equipment were selected on the basis of the following ground rules:

1. Direct visual observation of equipment or operations in sodium is impossible. Periscopes, TV cameras, shielded windows, or mirrors will be used wherever they can provide a checking function on machine or mechanism movements and operations outside the pool.
2. Because the fuel assemblies and the primary sodium are always radioactive, direct contact with the fuel assemblies is impossible. Thus, remote handling techniques are mandatory, and all the handling equipment and methods selected are for semiautomatic operation. The operator must supply the intelligence by initiating all instructions, but he must be constantly supplied with information through interlock circuits that indicate progress and next-step data. He must be constantly protected by preventive control, which will preclude his initiating the wrong operation.

3. Mechanisms to operate in sodium, especially in sodium vapor, must be held to a minimum. Because the accuracy and reliability of moving parts can be affected by this environment, it was decided to eliminate all such mechanisms, if possible. Where it is impossible or impractical to eliminate mechanisms, it was further decided that such assemblies should be designed for removal from the environment when they are not in use, and that the sodium-wetted subassemblies should be designed for easy replacement or cleaning and maintenance. This approach also produced a desirable advantage by limiting the radiation exposure to these mechanisms.

4. All core assemblies (fuel, control rod, blanket, and neutron shield assemblies and special elements) that must be transferred into, out of, or within the reactor vessel have shapes, sizes, and handling weights that are compatible with the handling equipment used for fuel assemblies. Identical pickup adapters are assumed for this study. The safety problem is recognized, and the final design will probably provide for different pickup adapters.

5. The core size remains constant for all the pot and spread-out arrangements considered in this study.

6. All core assemblies are hexagonal in cross section. For this reason, it was considered desirable to maintain their orientation by positive means during all phases of handling.

7. The core outlet coolant temperature must be lowered to about 800 F for under-the-plug and through-the-plug refueling to allow sufficient ΔT to convectively cool a fuel assembly that is being transferred without exceeding its upper design temperature. The temperature must be down to about 400 F for hot cell refueling in order to reduce the amount of sodium vapor available for deposition on the walls and equipment of the hot cell during a refueling cycle (see sections 4.2 and 4.3 for justification).

8. Cooling the fuel assemblies to remove the decay heat generated internally after a long core exposure was considered to be a major problem for the handling equipment. If possible, the assembly should always be submerged in sodium coolant. This can be done by

in-reactor storage or by designing the handling machine to carry along a volume of sodium. These machines must have sufficient and reliable cooling systems to permit the completion of a handling procedure.

9. Decay heat storage facilities for approximately one-third of the fuel assemblies of a full core are considered a minimum requirement. These storage facilities are in the reactor primary sodium pool for the pot arrangements and outside the pool for the spread-out arrangements. The fuel assembly support within the storage cell and the access ports to the cell should duplicate those for the core, if possible. Because of the relatively high rate of decay-heat generation from an irradiated fuel assembly, storage in cooled circulating sodium was considered to be mandatory. The storage time for highly irradiated fuel was to be about 3 months. Methods of handling ruptured fuel elements or leakers must be considered.

10. Nuclear and process instrumentation will not be considered a major influence in this study. The degree of development required and all related costs of such equipment will be assumed equal for every plant arrangement-refueling method combination.

11. The most compact arrangement for each plant is to be strived for.

3.3. System Description — Pot Arrangement With Under-the-Plug Refueling

3.3.1. Plant Layout

The primary system for this plant is an integral-pot arrangement with an under-the-plug refueling scheme. In this system the entire primary system is submerged in a pool of sodium contained in the reactor vessel. Refueling is accomplished remotely and completely under sodium without removing the reactor vessel cover structure. This results in a compact arrangement that allows all of the equipment to be serviced by the main overhead rotary crane. Figures 2 and 3 show this arrangement, and Appendix A lists the major equipment and facilities.

The reactor vessel is located at the geometric center of the reactor building and is surrounded by biological shielding. The center section of the reactor vessel houses the core vessel, which comprises

the fuel and blanket assemblies, core internals, holddown units, neutron shield assemblies, tension support members, grid plate, and meltdown pan. In the annulus between the core vessel and the reactor vessel are the six tandem pump-IHX assemblies, cold traps, and two rotating fuel storage racks. The tandem pump-IHXs are radially arranged from the center of the core in two groups of three assemblies. The primary pumps are in the cold leg of the primary pipework, and the pump motors are above the top surface of the reactor vessel.

The top of the reactor vessel is sealed by an 8-foot-thick cover structure, and rotating plugs (of the same thickness) over the top of the core vessel form an integral part of the cover structure. The plugs are supported on bearings to facilitate indexing.

The reactor building has several floors which house the various reactor components. The upper floor is called the operating floor. Equipment in the vicinity of the reactor is assembled below this level wherever possible. The pump-IHX units are spaced so that a maintenance valve box can be placed over them to permit the removal of a single unit for repairs.

Secondary sodium enters and leaves the IHX through its upper portion, and this pipework runs southward to the steam generating building. Once clear of the reactor pressure vessel, the sodium pipes are contained in concentric ducts above the operating floor.

Primary equipment storage pits in the south section of the reactor building are provided for the partial decay of elements and for minor repair work. These storage areas are within reach of the main overhead rotary crane, which has a capacity of about 200 tons and is used as the transfer vehicle.

An equipment entrance door is provided at the east end of the reactor building. A penetration door and a blast door ensure the integrity of the reactor building during normal operation. The reactor is always in a fully shut down condition before the equipment door is opened.

The personnel airlock at the west end of the reactor building has a double set of hydraulically or pneumatically operated steel doors that are gas-sealed to ensure the integrity of the reactor building.

Emergency air cooling equipment in both the east and west areas of the operating floor level supplies cooling air to the dome through ducts.

3.3.2. Reactor Vessel

The reactor vessel contemplated at this time is cylindrical and has an ellipsoidal bottom head and a flat top head. The vessel contains the primary sodium in which the core, the pump-IHX assemblies, the decay heat spent fuel storage drums, and all other internals are immersed.

The top reactor cover structure contains shielding material and three rotating refueling plugs. The structure supports the reactor vessel, core vessel, pump-IHX assemblies, control rod drives, and fuel handling and purifying equipment.

3.3.3. Reactivity Control

There are approximately 25 mechanically driven, top-mounted control rods, all of which are mounted on the rotating plugs. The overall height of each unit is approximately 15 feet above the top of the rotating plug. Although the drive actuator shafts penetrate the reactor cover structure, a disconnect is provided to permit the actuator shaft to remain in its sealed linear bushing. A lifting device, such as an eye bolt, is provided on each unit. The approximate weight of each unit is 3500 pounds. This system is not described in detail, since all the components in the under-the-plug refueling system are also used in the hot cell refueling system.

3.3.4. Reactor Shielding

Thermal, neutron, and biological shielding is provided to protect plant personnel and equipment during normal reactor operation. The shielding system also protects plant personnel and the general public from the results of all credible accidents. Since most of the shielding components are used without change in both the under-the-plug and hot cell refueling systems, they are not described in detail.

3.3.5. Reactor Auxiliary Systems

The functions of the auxiliary systems associated with the pot arrangement with under-the-plug refueling are described briefly below.

1. NaK Service System — This system purifies primary and intermediate sodium and removes waste heat from the cold traps, the fuel storage pits, and the components that require direct interchange of heat from a sodium stream to a NaK stream.

2. Decay Heat Removal System — This system removes decay heat from the core during a planned shutdown or a scram condition.

3. Preheat System — This system heats to 300 F all of the tanks, pipes, heat exchangers, and the like that contain sodium.

4. Inert Gas System — The inert gas system supplies cover gas to the primary and secondary sodium systems and supplies the gas for various purge operations. The cleanup portion of the system purifies part of the gas for recirculation. Gas that cannot be cleaned is routed to the waste gas disposal system.

5. Sampling System — The sampling system extracts specimens of all heat transfer media, cover gas, and plant effluents in order to quantitatively determine the constituents foreign to the fluid in question.

6. Reactor Building Heating and Cooling System — This system maintains the temperature in the reactor building at not more than 120 F during normal operation and not less than 60 F during shutdown.

7. Waste Disposal Facility — The waste disposal system processes liquid, solid, and gaseous wastes for removal from the plant site.

3.3.6. Fuel Handling System

The fuel handling system for this concept requires operations in three major areas: the reactor, the reactor building, and the

fuel transfer tube. To perform the necessary functions, six major pieces of equipment are required: the reactor building crane, the rotating shield plugs in the reactor cover structure, the refueling mechanism, the rotating fuel storage racks in the reactor vessel, the large, shielded cask-type fuel transfer machine, and the fuel transfer tube. Some of these components are described in more detail in subsequent paragraphs.

Refueling is accomplished under the plug by a refueling mechanism mounted on the inner rotating plug. Fuel assemblies are transferred (under sodium) from the core to either of two rotating storage racks outside the core vessel in the sodium pool of the reactor vessel. A shielded cask-type fuel transfer machine moves the decayed assemblies from the storage rack through the shielded cover structure to an area outside the reactor building. Figures 2 and 3 illustrate the reactor system as it applies to fuel handling.

Sodium coolant is used in the reactor and in the fuel decay storage racks. Because the sodium is opaque and thus precludes visual observation, and because both the sodium and the fuel assemblies are highly radioactive, complete remote handling has been selected. Automated procedures are used wherever practical, economical, and reliable. The spent fuel assemblies are submerged in sodium during the prime handling operation and until the internal decay heating rate has been reduced to a point where alternate cooling can be safely accomplished.

The design of the overall fuel handling system utilizes conventional hardware and proven methods wherever possible. Lead-in chamfers are provided for all make-and-break connections, and all access port adapters are standardized to limit the handling procedures.

Semiautomatic methods are used for all fuel handling procedures. The operator must supply the intelligence by initiating instructions. He is constantly supplied with information through interlock circuits which indicate progress and next-step data. He must initiate the operation, however, and preventive control precludes his initiating the wrong operation. Visual observations of rotating plug motions are possible, since the reactor building is accessible during normal running and refueling periods.

The prime function of the fuel handling equipment is to remove a spent fuel assembly from the reactor core and install a new assembly in its place. The equipment, however, must be capable of handling all assemblies that are to be loaded into the reactor grid support. Figure 4 is a diagram of the fuel handling operations, which are as follows:

1. Replace spent fuel assemblies in the reactor core with new fuel assemblies.
2. Shuffle fuel assemblies in the core from one position to another.
3. Move spent assemblies into the storage rack, and pick up new assemblies from the rack.
4. Move spent assemblies out of the reactor building to a cleanup, disassembly, packaging, or shipment station.
5. Replace spent blanket assemblies with new blanket assemblies.
6. Move new fuel and blanket assemblies into the reactor building and into the storage racks in preparation for core loading.
7. Remove and replace spent regulating control or special assemblies in the reactor core and move the assemblies out of the reactor building.
8. Handle and install the refueling mechanisms and regulating control rod actuating mechanisms in the reactor closure plugs.

All of these functions must be performed with maximum safety and reliability. The prime function of fuel replacement is accomplished with the reactor in the off-load condition and must be completed as quickly as possible to minimize reactor downtime. Each step of each function has sufficient cooling, instrumentation, and backup systems to prevent damage or propagation of damage.

Item 7 above covers the handling, removal, and replacement of control or miscellaneous assemblies that may be inserted in the core. These assemblies are designed with lifting adapters identical to those of the fuel assemblies, and the size, length, and configuration of these assemblies are compatible with the equipment described for fuel handling. The same handling machines, tools, storage, and exit and entrance are used.

Since the reactor is an integral or pot-type system, spent fuel assembly storage space is provided directly adjacent to the core vessel but within the large, sodium-filled reactor pot. The key indexing or "finding" operation of the system is accomplished by rotating shield plugs which are mounted on bearings and are completely sealed to their respective containments. Rotation is by a gear drive, and indexing is programmed to automate the procedure. An additional visual check of the exact scribed markings on the plug faces is proposed to ensure complete safety during subsequent handling. No adjustment of the sodium coolant level is required in this concept. A refueling mechanism in the smaller rotating shield plug accomplishes the prime refueling function. In combination with rotation of the shield plugs, this mechanism transfers elements back and forth between the reactor core and the storage racks, keeping the active fuel section fully immersed in the sodium coolant at all times. Secondary handling functions are accomplished by a large, shielded, cask-type transfer machine. This machine is cooled with inert gas and provides access for the installation and removal of all reactor components that must pass through the reactor closure (shield) plugs. The transfer machine also transfers assemblies into and out of the reactor building through the fuel transfer tube.

The fuel handling scheme does not require extensive preparation for refueling. After a shutdown period of about 3 hours, the regulating rod drive extensions are disconnected and raised to the "up" position, the refueling mechanisms are installed into the shield plug access ports, and the rotating plugs are unbolted and seated on their bearings. The system is especially adaptable to partial refueling or fuel shuffling programs in which only a relatively few assemblies are to be handled.

3.3.6.1. Refueling Mechanism

The prime handling function is the movement of fuel assemblies and other core components between the reactor core and the two storage racks. This procedure is accomplished completely under the large reactor cover structure by using the rotating shield plugs and a refueling mechanism. The mechanism is assembled through an access port in the smaller rotating plug after the reactor has been shut down and cooled to refueling conditions.

Two identical refueling mechanisms are proposed in this concept. The spare mechanism permits maximum flexibility and the greatest reliability. The mechanism itself is mounted on bearings within the port and allows for orientation of the hexagonal fuel assemblies with the guide tubes and grapple tool.

The refueling mechanism (see Figure 5) uses standard commercial components, where possible, assembled to provide maximum safety and reliability of operation. The assembly consists of a stepped housing which fits and is sealed to the access opening; a grapple tool drive system, which is essentially located within the upper section of the housing; the grapple tool, which is suspended inside the lower housing section on a cable; the grapple tool drive motor mounted above the housing at floor level; a housing drive system mounted on the reactor cover structure; and shield plug sections mounted within the housing.

The mechanism's housing is an all-welded, stainless-steel structure which is stepped from 14 to 11 inches in diameter just below the operating floor level to seat on bearings, and is then stepped again about 6 feet below this level into a hexagonal shape 7.5 inches across the flats. The hexagonal section guides the grapple tool. The overall length of the housing is 20 feet, and in the installed position it extends 12 feet below the reactor shield plug. A machined bearing surface under the top housing shoulder seats on commercial bearings mounted in the access port. Double O-ring seals are fitted on the outside of the 11-inch-diameter section below this shoulder to protect the bearings and to maintain the reactor environment. The bearings can be replaced or maintained with the operating shield plug assembled in the port opening. Before each scheduled use of the mechanism, the O-rings will be replaced on the housing while the machine is in the nonoperating storage position.

The top of the mechanism's housing is fitted with a 24-inch-thick shield plug section which also has double O-ring seals. This plug seats on the top step within the housing, and from it is suspended the grapple tool drive system. This system consists of a 6-inch-diameter grooved drum (on which a hoist cable and a gas hose are wound) and a pulley arrangement (which accomplishes level winding

and direction of the cable and hose down the center of the lower hexagonal housing tube). The drum is mounted just under the top plug, and the shaft extends upward through seals to engage with the drive motor assembled on top of the plug. This motor is a variable-speed electrical unit having a gear reduction unit and an electrically actuated clutch and brake which are designed to be failsafe. The pulley arrangement adjacent to the drum is mounted on a ball-screw shaft and is geared to the drum shaft. Thus, the system is synchronized to level-wind both the hose and cable simultaneously at opposite ends of the drum. A lower shield plug section, 24 inches thick, is also suspended from the top plug. This plug is shaped to fit the lower transition section of the housing; in this position it protects the drum and pulley area from the bottom. The shield has no gas seals, but provides a close mechanical fit to prevent sodium vapor from entering the drum area.

The grapple tool is suspended from the hoist cable described above. This tool is hexagonal to guide within the hexagonal housing tube. The main section of the tool, about 84 inches long, provides continuous guidance as the tool traverses the space between the fixed tube and the core structure. The top of the tool is terminated in a cable attachment fixture. The bottom of the hexagonal tube is welded to an adapter section which reduces the size to about 3.5 inches across the flats. This section is split to form six individual grapple fingers which are machined with a hook shape (Figure 6) on the inside for positive latching with the fuel assembly. Two distinct operations are required to unlatch the fingers, and accidental unlatching is impossible as long as the weight of the fuel assembly is on the fingers.

Machined on the inside of the fingers is a tapered surface which acts as the fulcrum against which an actuating piston works to open the fingers. The actuating piston is positioned just above the tapered surface of the fingers on the end of an argon-actuated cylinder. The cylinder is mounted on a fixed platform inside the tube and is completely sealed from the environment. A rigid gas line is routed up the tube to a position adjacent to the cable attachment, where the flexible gas hose is attached.

The drive motor for the refueling mechanism housing is on the reactor cover structure. This constant-speed geared

motor provides rotation of the complete mechanism within the access port at one revolution per minute. The motor's beveled pinion gear drives against a mating ring gear mounted on the top surface of the mechanism. The motor is manually assembled after the refueling mechanism has been installed into the access port. The top surface of the mechanism is scribed with position markings that orient with similar scribings around the port. Accurate positioning is accomplished by visually aligning these markings; therefore, the drive becomes independent, and no precision gearing or instrumentation is provided.

Service lines to the refueling mechanism enter at the operating floor level. These consist of electrical leads for the two drive motors and for instrumentation and two argon gas lines. One gas line is connected through a fixed, stepped tube in the plug to the flexible gas line on the drum. Similarly, a second line feeds directly into the mechanism's housing. This gas volume pressurizes the housing to force sodium down out of the tube as the grapple tool is lowered into the reactor. This keeps the cable and gas hose from collecting sodium, which could cause complications when the cable and hose are wound on the drum.

The complete refueling mechanism is designed for easy maintenance. The internal mechanism can be replaced as a unit while located in its nonoperating storage position, or, after a cleaning operation, it can be repaired for reuse. The drive motors are top-mounted and can be repaired or replaced even in their operating position. Failure of the hoist cable, by snarling or breaking, is considered highly unlikely if proper servicing and checkout are used. Likewise, gas hose failure is considered unlikely in this design, although such a failure could result in a difficult maintenance problem if a spent fuel assembly were locked within the grapple fingers. In this situation, the fingers could not release the assembly and, although this is a "safe" condition, reactor operation would be held up until the condition could be cleared. Current plans are to use the large shielded fuel transfer machine to remove the entire mechanism and element.

3.3.6.2. Fuel Storage Racks

Two fuel storage racks are located inside the reactor vessel (or pot) as shown in the sectional view of the reactor

building (Figure 3). The racks provide intermediate storage for both new and spent fuel assemblies. New assemblies are loaded into the racks during reactor operation and are transferred into the core during the refueling procedure. After reactor shutdown, spent assemblies are placed in the racks immediately after their removal from the core and are held there for about three months. This storage period allows the radiation and internal heating rates to decay to levels that are tolerable for the next handling procedure. The spent assemblies can be removed from the racks during reactor operation. The operation and location of the storage racks within the large reactor coolant pool permits refueling without removing fuel assemblies from the sodium.

The complete rack assembly is supported from the reactor cover structure by a 10-inch-diameter center shaft welded directly to the rack base structure. The storage tubes are fastened to the shaft near their top by a web structure. The tops of the rack tubes are located 17 feet below the underside of the reactor cover structure. A structure, built upward from the core vessel's support structure, does not support the racks but provides a center guide bearing at the bottom center of the rack. This bearing is a noncritical, loose-tolerance fluid type which simply prevents extreme motions.

The storage rack drive system is flush with the operating floor level and is fastened directly to the reactor cover structure. This consists of a housing which seals to the plug and provides the secondary gas seal. The drive unit is a commercial, constant-speed, a-c electric motor that drives the rotor shaft through a gear reduction system at a speed of 1/4 revolution per minute. The motor drive shaft acts through a shaft coupling, seals, and a slip clutch. Instrumentation for rotor positioning is taken directly from the rotor shaft, which eliminates in-reactor vessel instrumentation.

3.3.6.3. Fuel Transfer Machine

The fuel transfer machine (Figure 7) is a large, shielded, cask-type mechanism which installs and removes all reactor components that must pass through the reactor cover structure. (This does not include the primary sodium pumps or heat exchangers.) The machine is handled by the large reactor containment crane, which lifts

or lowers the machine vertically and moves it laterally between various stations.

The main component of the machine is a stainless-steel pressure vessel approximately 39.5 feet high. This vessel has an elliptical cross section approximately 4 feet by 2 feet. It is closed on the bottom by a gas valve and is completely surrounded by thermal insulation and cast iron shielding material. The shield is 23 inches thick on the lower 19 feet and is tapered to about 6 inches thick at the top. The shielding provides man-access to the immediate area during all handling operations. All penetrations and externally mounted equipment are equally shielded to protect against the maximum radiation doses.

The bottom gas valve provides a full 18-inch access opening and has a tapered adapter, a locking device, and seals. This permits positive sealing and locking when the machine is on the port valve box. Electrical interlocks are provided between the locking device, the valves, and the crane drives to prevent accidental movement of the crane while the valves are open and to prevent the valves from opening if they are not locked properly.

The machine access port valve box is a manually installed adapter assembly which is sealed and bolted to any of the access openings where a gas seal with the transfer machine is required. This assembly is equipped with an 18-inch valve similar to the machine bottom valve; it has an adapter surface for mating with the machine. The assembly is a heavy-walled structure designed to support the full weight of the transfer machine, and its height of 2 feet provides protection to mechanisms, such as the refueling mechanism, that protrude above the loading face level. Three of these boxes are required—one for location at each end of a planned handling procedure. They are equipped for connection to gas purging lines and to the electrical service line for operating the valve.

In the top of the shielded pressure vessel is a 9-inch-diameter gas valve through which a grapple hook is connected to a cable drive unit; the drive and hoist unit is mounted on and sealed to this upper valve. This gas valve is only used to seal the pressure vessel in the event that the grapple hook drive has to be removed for servicing. The hoist unit has a horizontally mounted winding drum which

is automatically traversed to ensure that the cable is always directly above the center guide tube. The hoist unit has a dual-speed control, giving 5 ft/min for slow handling operations and 60 ft/min for intermediate handling. An overload clutch is provided to prevent damage to elements where binding may occur. A failsafe electrical brake is provided on the drum shaft to prevent accidental rotation. A hand-wind crank can be fitted to the reduction gearbox to complete hoisting if a motor or power failure occurs.

The elliptical internal shape of the pressure vessel (Figure 8) allows for two holding positions—one to either side of the bottom valve opening. Each of these positions is fitted with a bottom holddown and support fixture capable of accepting all the components to be handled. One of these base positions is fitted with the argon gas cooling-heating outlet nozzle. This position is for fuel assembly holding and for handling other core components. The alternate position is for support of the access port shield plug when components are handled through that port.

The grapple tool (and the component being handled) is moved between the center lift position and the two holding positions by a lateral displacement mechanism in the upper portion of the machine. Two independent hydraulic cylinders operate a shaped displacement tool to move the grapple from the exact center to an exact displaced position. The operation is programmed for the procedure involved, and interlocks prevent the operation of more than one unit. The complete driving mechanisms for this system are mounted outside the machine vessel on sealed access plates to eliminate vapor and temperature complications and to allow for maintenance and replacement.

An argon-gas heat exchanger system is fitted directly to the machine. The system heats new fuel assemblies before they are installed in the rotating fuel storage rack, and cools spent fuel assemblies that are removed. The outlet nozzle is routed into the holding position for these assemblies in order to direct the flow up through the assembly.

The complete internals of the transfer machine are stainless steel, which protects against sodium vapor and high temperature as required. To prevent malfunctions, the number of

mechanisms are minimized and tolerances are generous. The overall machine measures about 45 feet from the seating face to the crane hook, and the overall handling weight is about 200 tons. Lifting points are provided above the center of gravity of the machine, and a strongback and lateral supports to a single lifting grab facilitate lifting operations.

3.3.6.4. Fuel Transfer Tube

This tube permits transfer of fuel assemblies and other core components from the reactor building to the adjacent fuel cleanup cell. One tube is used for all operations; it has a gas valve at the reactor operating floor level and another valve at the roof level in the discharge area directly below.

The transfer tube is formed by a 10-inch-diameter, stainless-steel tube permanently installed in the concrete structure. The tube terminates at both ends with a gas valve to form a gas-tight seal. In combination with the valves, this tube forms an integral cell arrangement and maintains complete separation of the reactor containment and the discharge area as long as either valve is closed.

The tube is within the range of the reactor building handling crane. A valve box, identical to the one used on the reactor access ports, is sealed to the floor level face for connection with the fuel transfer machine. The lower end of the tube is designed with an adapter to connect with the discharge area handling equipment.

3.4. System Description — Pot Arrangement With Hot Cell Refueling

3.4.1. Plant Layout

The primary system is an "integral pot" arrangement with a hot cell refueling scheme. The general arrangement of the plant equipment within the reactor building is essentially the same as that for the under-the-plug refueling scheme (see Figures 9 and 10). This description will cover only the differences between the features associated with the hot cell and the under-the-plug refueling schemes. Appendix A lists the major equipment and facilities.

A hot cell that eliminates the rotating plugs of the under-the-plug refueling scheme is built directly over the core to effect a hot cell refueling system for a pot. The cylindrical cell has fuel transfer

isles protruding north and south at the operating floor level. It is 83 feet high and 38 feet in diameter and has 4-foot-thick concrete shielding walls. The total length of the fuel transfer isles, including the circular section, is 89 feet. The inside of the cell is steel lined and insulated to ensure that it will not leak and to protect the concrete shielding from sodium vapor and heat.

The core vessel is directly beneath the hot cell and is covered by a removable plug approximately 8 feet thick. The removable plug is mounted in the reactor vessel cover structure, which is also 8 feet thick.

The hot cell is refueled by a fuel handling machine mounted on a jib crane in the cell. After the core vessel cover plug is removed, fuel assemblies are transferred (under sodium) from the core to either of two rotating storage racks outside the core vessel in the sodium pool of the reactor vessel. A secondary gas-cooled fuel handling machine moves the fuel assemblies from the storage racks through the reactor vessel cover structure to the discharge area via the hot cell.

3.4.2. Reactor Vessel

The vessel for the hot cell refueling scheme is approximately 10 feet larger in diameter than the vessel for the under-the-plug refueling scheme. To accommodate the hot cell walls, the primary pumps and IHXs must be moved outward, thereby increasing the diameter.

3.4.3. Reactivity Control

The reactivity controls for the hot cell are similar to those used for the nonhot cell. The major differences are (1) all disconnects and lifting lugs are designed for remote manipulation, and (2) all external surfaces of the actuator will operate in an inert atmosphere that may contain sodium vapor.

3.4.4. Reactor Shielding

The shielding for the under-the-plug refueling system is essentially the same as that for the hot cell refueling system. An additional 4 feet of concrete shielding is required around the hot cell.

3.4.5. Reactor Auxiliary Systems

The choice of a refueling scheme does not materially affect the NaK service system, the decay heat removal system, or the preheat system. Other auxiliary systems are affected as follows:

1. Inert Gas System — For hot cell refueling, the initial charge of inert gas is increased by an amount equal to the volume of the hot cell. To minimize the contamination of the inert gas contained in the hot cell during the refueling operations, the capacity of the cleanup portion of the inert gas system will have to be increased to clean up the cover gas before the reactor vessel is opened.

2. Sampling System — The hot cell refueling scheme requires additional monitoring equipment for sampling the atmosphere of the hot cell.

3. Reactor Building Heating and Cooling System — This system is rated to maintain the reactor building temperature between 60 and 120 F during normal shutdown and operation. Emergency power is available to keep the system operational when the plant is shut down. The reactor building heating and cooling system appears to be essentially independent of the method of refueling. However, additional heating is required in the hot cell to facilitate the fuel handling operation.

4. Waste Disposal System — The waste disposal system processes liquid, solid, and gaseous wastes for removal from the plant site. The hot cell refueling scheme has a disadvantage in the gaseous waste area. The capacity of the decay storage tanks will have to be increased sufficiently to accept the waste gas associated with the cleanup of the hot cell atmosphere.

3.4.6. Fuel Handling System

The fuel handling system for this concept requires operations in three major areas: the reactor, the hot cell, and the fuel transfer tube. To perform the necessary functions, five major pieces of equipment are required: the hot cell polar crane, the fuel handling machine, the rotating fuel storage racks in the reactor vessel, the fuel transfer machine, and the fuel transfer tube. Some of these components are described in more detail in subsequent paragraphs.

This fuel handling procedure requires the building of a hot cell directly over the reactor core. Figures 9 and 10 give the general shape and size of the hot cell and illustrate the reactor system as it applies to fuel handling. The primary pumps, intermediate heat exchangers, emergency core cooler, cold traps, and hot traps penetrate the reactor cover structure outside the hot cell.

The general hot cell fuel handling scheme centers on raising the reactor shield plug completely clear of the reactor vessel. This requires unlatching all control assemblies that penetrate the plug and extend into the reactor core. Thus, all plug-mounted drives and extensions are removed as a unit, and the shield plug leaves the reactor vessel completely open for fuel handling access. The fuel handling machine is then placed over the reactor, and the lower portion is telescoped into the sodium to grapple the spent fuel assembly and raise it into the machine. The next steps are retraction of the telescoped tube clear of the core vessel and movement, under sodium, to the adjacent fuel storage rack where the spent assembly is deposited and a fresh assembly is picked up. After reactor startup and a suitable decay time, the fuel transfer machine moves the decayed assemblies from the storage rack through the shielded cover structure to an area outside the hot cell and the reactor building.

Sodium coolant is used in the reactor and in the fuel decay storage racks. Because the sodium is opaque and therefore precludes visual observation, and because both the sodium and the fuel assemblies are highly radioactive, complete remote handling has been selected for this design. Automated procedures are used where practical, economical, and reliable. The spent fuel assemblies are submerged in sodium during the prime handling operation until the internal decay heating rate has been reduced to a point where alternate means of cooling can be used safely.

The design of the overall fuel handling system utilizes conventional hardware and proven methods wherever possible. Lead-in chamfers are provided for all make-and-break connections, and all access port adapters are standardized to limit the handling procedures.

The system selected follows "hot cell" techniques for handling radioactive materials; the methods are semi-automatic. The

operator must supply the intelligence by initiating instructions. He is constantly supplied with information through interlock circuits that indicate progress and next-step data. He must initiate the operation, however, and preventive control precludes his initiating the wrong operation. Machine movements are observed through shielded windows and by shielded television cameras, but all operations are traced primarily by position indicators. Bench marks are provided at several points within cells where visual sightings can be made; these serve as alignment points to supplement the automatic motions of the handling machines.

The prime function of the fuel handling equipment is to remove a spent fuel assembly from the reactor core and install a new assembly in its place. The equipment, however, must be capable of handling all assemblies that are to be loaded into the reactor grid support. Figure 11 is a flow diagram of the fuel handling operations, which are as follows:

1. Replace spent fuel assemblies in the reactor core with new fuel assemblies.
2. Shuffle fuel assemblies in the core from one position to another.
3. Move spent assemblies into the storage rack and pick up new assemblies from the rack.
4. Move spent assemblies out of the hot cell to a cleanup, disassembly, packaging, or shipment station.
5. Replace spent blanket assemblies with new blanket assemblies.
6. Move new fuel and blanket assemblies into the hot cell and into the storage racks in preparation for core loading.
7. Remove and replace spent regulating control, or special assemblies in the reactor core and move the assemblies out of the hot cell.

All of these functions must be performed with maximum safety and reliability. The prime function of replacing components in the core is accomplished with the reactor in the off-load condition and must be completed as quickly as possible to minimize reactor downtime. New fuel assemblies are loaded into the storage racks with the reactor

on-load. Each step of each function has sufficient cooling, instrumentation, and backup systems to prevent damage or propagation of damage.

Item 7 above covers the handling, removal, and replacement of control or miscellaneous assemblies that may be inserted in the core. These assemblies are designed with lifting adapters identical to those of the fuel assemblies, and the size, length, and configuration of these assemblies are compatible with the equipment described for fuel handling.

3.4.6.1. Fuel Transfer Tube

This fuel transfer tube is the same as that for under-the-plug refueling except that it is located inside the hot cell (see 3.3.6).

3.4.6.2. Fuel Handling and Fuel Transfer Machines

Two machines effect fuel handling within the reactor hot cell. One of these, the fuel handling machine, is used exclusively for operations between the reactor and the fuel storage racks; it performs all loading and unloading of fuel assemblies or other core assemblies. The second, the fuel transfer machine, is used only for operations between the fuel storage racks and the fuel transfer tube. This machine moves fuel assemblies in and out of the hot cell.

The fuel handling machine (Figures 12 and 13) is designed to transfer new and spent fuel assemblies between the reactor core and the fuel storage racks. The principal operation is to secure a fuel assembly and move it from one location to another. To accomplish this, the machine must reach into the reactor or the fuel storage racks, grapple the desired fuel assembly, and raise it for movement to another position. The major components of the machine are a jib crane, a housing, an indexing mechanism, a screw extension mechanism with pickup tool, and a guide.

Each component in the core is located by rotating the jib crane about its base and actuating the housing radially along the jib crane rails. The fuel assemblies are latched by inserting the pickup tool into an assembly and rotating. Since the rotating operations

can be performed only when the pickup tool is in the down position, unlatching of an assembly is impossible during the transfer operation.

The pickup tool is lowered to its latching position by the screw mechanism. The guide tube surrounding this central screw mechanism holds down the assemblies adjacent to that being removed and prevents rotation of the assembly relative to the pickup tool.

The fuel transfer machine is designed to transfer new and spent fuel assemblies between the fuel storage racks and the transfer tube. It is assembled on a trolley which is mounted on rails that span the distance between the fuel transfer tube and the farthest storage rack.

The major difference between the handling and transfer machines is that a cooling gas is supplied to the latter to cool the decayed spent fuel assemblies. The cooling is accomplished by forced circulation of the hot cell atmosphere along the assembly.

3.4.6.3. Fuel Storage Racks

The fuel storage racks are identical to those in the pot arrangement with under-the-plug refueling (see 3.3.6).

3.5. System Description — Spread-Out Arrangement With Through-the-Plug Refueling

3.5.1. Plant Layout

The primary system is a spread-out arrangement with a through-the-plug refueling scheme. The primary sodium pump-IHX combinations are in individual pits and are connected to the reactor vessel by straight-line coaxial pipes. Figures 14, 15, and 16 show this arrangement, and Appendix A lists the major equipment and facilities.

The containment building, which has several floors and houses all primary and auxiliary equipment required for reactor operation, is a 150-foot-diameter cylinder with ellipsoidal ends. The operating floor is approximately at ground level. The reactor vessel, which contains the fuel and blanket assemblies, holddown units, thermal shields, grid plate, and meltdown pan, is at the geometric plan center of the building.

The top of the reactor vessel is covered by rotating plugs that are approximately 8 feet thick and are supported on bearings

to facilitate indexing. The rotatable plugs are part of the reactor vessel's cover structure, which is also 8 feet thick.

Refueling is accomplished through the plug by a shielded cask-type transfer machine, which is maneuvered by the main overhead crane. Fuel assemblies are transferred from the core, through the shield plug, to a decay storage pit outside the reactor vessel. The same machine is used to move the decayed assemblies from the decay storage pit to the cleanup area.

Six combined pump-IHX assemblies are positioned radially around the reactor vessel, in two groups of three, just outside the reactor biological shield. Two assemblies are coupled in parallel to serve one steam generator. The primary pumps are in the hot leg of the primary circuit. Secondary sodium enters and leaves the IHXs through their upper portions and is piped southward to the steam generators. The pump-IHX assemblies are arranged so that a valve box can be used for repair and replacement. These combined pump-IHX units are described in more detail in reference 2.

Plant personnel are protected from radiation by biological shielding surrounding the reactor vessel and the primary sodium. This also shields the IHX to minimize activation of the secondary sodium.

Storage and access pits extending downward from the operating floor are provided as necessary. The pump storage pit, which is used for decay storage and minor repair work, is in the south section of the containment building. A storage and decay area for regulating control rod actuator assemblies is provided adjacent to the pump storage pit. The heat exchanger decay storage and repair work pit is also in the south section of the building. The access hatch to the lower floors is in the northeast section of the building near the main equipment entrance lock. The access hatch and all the storage areas are within reach of the main overhead crane, which has a capacity of about 200 tons.

An equipment entrance airlock is provided at the north side of the containment building. A single hydraulically or pneumatically actuated seal door ensures the integrity of the containment building during reactor operations. (The reactor is always fully shut down before the equipment door is opened.) To minimize the size of the penetration, low-slung dollies are used to transport equipment through the door.

A personnel access door (with airlock) at the southwest side of the building has hydraulically or pneumatically operated gas-sealed steel doors to ensure the integrity of the containment building.

Air-conditioning equipment on both the east and west sides of the building at the charge face level supplies air to the dome through ducts.

3.5.2. Reactor Vessel

The cylindrical reactor vessel has an ellipsoidal bottom head and a flat top head. It retains and supplies pressurized liquid sodium coolant to the core and supplies heated sodium to the IHXs. The vessel is suspended from the top of the surrounding biological shield wall. This stainless steel vessel is 30 feet in diameter and 38.5 feet long (excluding the top head thickness); it is insulated externally.

The flat top head (reactor vessel cover structure) consists of two rotating refueling plugs and a ring support flange attached to the vessel body. The larger of the two rotating plugs is mounted concentric to the vessel center line. The smaller plug is mounted in the larger plug and is eccentric to the vessel center line. The control rod actuating mechanisms and fuel element holddown structure are attached to the smaller rotating plug. The reactor vessel cover structure is constructed as a hollow structure, then filled with shielding material to complete the reactor biological shield. All penetrations of the reactor vessel are above the core level to prevent the loss of sodium in the core.

A cylindrical flow divider and thermal shield is attached and sealed to the reactor vessel just below the lower face of the shield plug, but above the maximum level of the hot sodium. The combination flow divider and thermal shield consists of four cylindrical layers of stainless steel plates that extend downward from the attachment point to below the bottom of the core.

A combined grid plate and support structure is permanently attached to the lower end of the flow divider to support the fuel and blanket assemblies. The grid plate takes the form of a rhombus to match the geometry of the hexagonal fuel and blanket assemblies.

A fuel meltdown pan is placed below the lower grid plate near the bottom of the reactor vessel. Its geometrical configuration prevents the formation of a critical mass.

A holddown and guide structure is provided above all in-core assemblies. It is permanently attached to the underside of the eccentric rotatable plug. The holddown structure locates the fuel and blanket assemblies on their correct pitch and provides a holddown force. Lateral support and guidance for the control rod extension rods are also provided by the holddown structure.

3.5.3. Reactivity Control

The control scheme for the spreadout arrangement with through-the-plug refueling is identical to the one described in section 3.3.3 for the pot arrangement with under-the-plug refueling.

3.5.4. Reactor Shielding

This system includes the thermal, neutron, and biological shielding used to protect plant personnel and equipment during normal operation. The shielding system also protects plant personnel and the general public from the results of all credible accidents. Most of the shielding components are held constant in the through-the-plug refueling system and in the hot cell refueling system; therefore, detailed descriptions of these are not given.

3.5.5. Reactor Auxiliary Systems

The reactor auxiliary systems associated with the spreadout arrangement with through-the-plug refueling are identical to those for the pot arrangement with under-the-plug refueling. Brief functional descriptions of these systems are given in section 3.3.5.

3.5.6. Fuel Handling System

The fuel handling system for this concept requires operations in three major areas: the reactor, the fuel storage cell, and the fuel transfer tube. To perform the necessary functions, five major pieces of equipment are required: the reactor building crane, the rotating shield plugs in the reactor cover structure, the large shielded fuel handling machine, the rotating shield plug in the fuel storage cell,

and the fuel transfer tube. Some of these components are described in more detail in subsequent paragraphs.

Fuel handling includes all fuel assembly handling from the time a fresh fuel assembly is received at the reactor site until it is loaded and unloaded in the core, transferred into and out of the decay storage cell, and passed through a cleanup, disassembly, packaging, storage, or shipment station.

The fuel handling procedure centers on raising the core holddown structure clear of the core and rotating the shield plugs to the desired position. This requires unlatching of all control assemblies that penetrate the plug and extend into the core. The reactor building crane then places the fuel handling machine over the reactor access port. After the access port plug has been removed by the machine, a section of the machine is telescoped into the sodium to grapple a spent fuel assembly. The assembly is then raised into the telescoped section, and a valve is closed to retain sodium in the section. The next steps are retraction of the telescoped section clear of the reactor vessel, replacement of the access port plug, and (with inert-gas cooling) movement of the fuel handling machine to the fuel storage cell where the spent assembly is deposited and a fresh assembly is picked up. Figures 14, 15, and 16 illustrate the reactor system as it applies to fuel handling.

Sodium is used as a coolant in the core and the fuel storage cell. It is also used as a heat transfer medium in the fuel handling machine, in which inert gas is the main coolant. Because the sodium is opaque and therefore precludes visual observation, and because both the sodium and the fuel assemblies are highly radioactive, complete remote handling has been selected for this design. Automated procedures are used where practical, economical, and reliable. The spent fuel assemblies are submerged in sodium during the prime handling operation until the internal decay heating rate has been reduced to a point where alternate cooling can be safely accomplished.

The design of the overall fuel handling system utilizes conventional hardware and proven methods wherever possible. Lead-in chamfers are provided for all make-and-break connections, and all access port adapters are standardized to limit the handling procedures.

Semi-automatic methods are used for all fuel handling. The operator must supply the intelligence by initiating instructions. He is constantly supplied with information through interlock circuits which indicate progress and next-step data. He must initiate the operation, however, and preventive control precludes his initiating the wrong operation. Visual observations of rotating plug motions are possible because the reactor building is accessible during normal running and refueling periods.

The prime function of the fuel handling equipment is to remove a spent fuel assembly from the reactor core and install a new assembly in its place. The equipment, however, must be capable of handling all assemblies that are to be loaded into the reactor grid support. Figure 17 is a flow diagram of the fuel handling operations, which are as follows:

1. Replace spent fuel assemblies in the reactor core with new fuel assemblies.
2. Shuffle fuel assemblies from one position to another in the core.
3. Move spent assemblies into the fuel storage cell and pick up new assemblies from the cell.
4. Move spent assemblies out of the reactor building to a cleanup, disassembly, packaging, or shipment station.
5. Replace spent blanket assemblies in the reactor core with new blanket assemblies.
6. Move new fuel and blanket assemblies into the containment structure and into the fuel storage cell in preparation for core loading.
7. Remove and replace spent control, safety, or special assemblies in the reactor core, and move the assemblies out of the containment structure.

All of these functions must be performed with maximum safety and reliability. The prime function of the fuel replacement is accomplished with the reactor in the off-load condition and must be completed as quickly as possible to minimize reactor downtime. Each step of each function has sufficient cooling, instrumentation, and backup systems to prevent damage or its propagation. Item 7 covers the handling, removal, and replacement of control or miscellaneous assemblies that

may be inserted in the core. These assemblies have lifting adapters that are identical to those of the fuel assemblies, and the size, length, and configuration of these assemblies are compatible with the equipment described for fuel handling.

3.5.6.1. Fuel Transfer Tube

The fuel transfer tube is identical to the one for the pot arrangement with under-the-plug refueling (see section 3.3.6).

3.5.6.2. Fuel Storage Cell

The fuel storage cell is inside the reactor building adjacent to the reactor (see the sectional view of the reactor building in Figure 16). This location permits minimum reactor shutdown for refueling by providing minimum travel distance.

The fuel storage cell provides intermediate storage for both new and spent fuel assemblies. New assemblies are loaded into the storage position and heated to refueling temperature during reactor operation and are subsequently transferred into the reactor during the refueling procedure. The key indexing or "finding" operation is accomplished by rotating the shield plugs that form the upper closure of the storage cell. These plugs are mounted on bearings and are sealed to their respective environments. Rotation is achieved through a gear drive, and position is indicated by scribed markings on the face of the plugs. Visual observation of each position is planned, since the operating floor is accessible during normal operation. Spent assemblies are placed in the cell directly after their removal from the reactor and are held there for about 3 months. This storage period allows the radiation and internal heating rates to decay to tolerable levels for the next handling procedures. The fuel storage vessel is contained in a steel-lined pit approximately 17.5 feet in diameter by 46 feet deep. The pit is sealed at the top by a 21.5-foot-diameter by 8-foot-thick shield plug that supports the fuel storage vessel. The circular vessel is 15.5 feet in diameter and has an elliptical bottom end. It is insulated externally and has a flow divider and a combination flow distributor and fuel assembly support structure inside. The storage positions in the cell are formed by concentric rings with radial spacers. The flow divider is circular and forms a 6-inch annulus

between itself and the fuel storage vessel. It is attached and sealed to the storage vessel approximately 2 feet below the shield plug and extends downward to support the fuel assembly support structure.

Access to all fuel storage positions in the vessel is through six openings that penetrate the rotating plug mounted in the shield plug. The rotating plug is 15.5 feet in diameter and is supported on bearings to facilitate indexing. The fuel storage positions are arranged in six concentric rings that have radii to match the radial positions of the six access openings. Each of the access openings has a removable plug.

The sodium coolant, which has its own sodium-to-air heat exchanger and cleanup system, enters the storage vessel just beneath the attachment point of the flow divider. It flows down the annulus to the bottom plenum, then upward through the stored fuel assemblies, and exits the storage vessel between the attachment point of the flow divider and the bottom of the shield plug.

3.5.6.3. Fuel Handling Machine

The fuel handling machine is the exclusive means of handling fuel assemblies within the reactor building and of transferring assemblies into and out of the reactor building. It is maneuvered by the general-purpose containment crane. The major components of the machine are a circular, leak-tight shielded vessel, two fuel assembly handling mechanisms, a refueling plug hoist, and a gas cooling system.

The circular shielded vessel has an internal operating space that is 3 feet in diameter by 40 feet high. Its 2-foot-thick wall and ends protect operating personnel from any internal components that may be radioactive. An eccentric opening at the bottom end is provided for all components that enter and leave the core.

A plug closes the opening in the shielded vessel. This plug acts as an adapter to the reactor refueling plug, and, in conjunction with the refueling plug hoist inside the shielded vessel, is used to lift the refueling plug into the shielded vessel during refueling.

The two fuel handling mechanisms and the refueling plug hoist are supported in a circular pattern by a rotating platform in the upper part of the vessel. Rotation of the platform will

place the hoist or one of the two fuel handling mechanisms over the opening at the bottom of the vessel. The platform is rotated by a central support shaft that extends through the vessel's top cover to an electrically driven motor.

Each fuel handling mechanism has a guide tube that is attached to the rotating platform, a telescoping tube mounted inside the guide tube, and a grapple tool installed inside the telescoping tube. The actuating systems for the telescoping tube and the grapple tool are similar. Both systems comprise an electric motor and a chain and sprocket combination that is capable of push-pull operation.

The grapple tool is a double-jawed, cam-operated gripper and an extension shaft. The gripper fingers utilize a pin joint and are cammed open and closed by the relative motion of the grapple tool and the telescoping tube.

The telescoping tube, which is used to extract spent fuel assemblies from the core, is built with cooling gas flow passages in its wall and has a closure valve at its bottom end. The closure valve is used to trap sodium coolant with a fuel assembly that is being transferred, and the tube wall acts as a sodium-to-gas heat exchanger to remove decay heat from the element. In the tube that is used (1) to transfer spent fuel assemblies from the decay storage cell through the transfer tube to an area outside the reactor building or (2) to handle non-irradiated assemblies, the cooling or heating gas enters at the top end of the tube, flows downward through the assembly, and exits at its lower end.

3.6. System Description — Spread-Out Arrangement With Hot Cell Refueling

3.6.1. Plant Layout

The primary system is a spread-out arrangement with a hot cell refueling scheme; it is identical to Concept II.³ The general arrangement of the plant equipment within the containment building is essentially the same as that for the through-the-plug refueling scheme. Figures 18, 19, and 20 show this arrangement, and Appendix A lists the major equipment and facilities.

A hot cell is built directly over the reactor core to effect a hot cell refueling scheme for the spread-out arrangement. The cell is

a rectangular concrete structure with a tower located around the reactor core center line. The structure is above the operating floor level and protrudes from the reactor building geometric plan center southward toward the steam generating building. The concrete hot cell structure is 85 feet long, 28 feet wide, and 38 feet high; it has a tower that is 36 feet across and rises to a height of 81 feet above the operating floor. The walls of the cell are approximately 4 feet thick and are constructed of medium-density concrete to meet biological shielding requirements. The inside surface of the hot cell is steel lined, and there is insulation between the concrete and the steel liner.

Within the hot cell are the top of the reactor vessel, the fuel transfer machines, the fuel storage cell, the plug hoisting equipment for the reactor cover, the reactivity control rod actuators, miscellaneous tools, and the fuel transfer tube. All of this equipment is operated remotely, but five leaded glass windows are provided in the walls of the cell for direct viewing of fuel handling and associated operations.

3.6.2. Reactor Vessel

The reactor vessel for the spread-out arrangement with hot cell refueling has general configuration features that match those of the through-the-plug refueling scheme (see 3.5.2). The major differences are that the reactor vessel for hot cell refueling is 17.75 feet in diameter instead of 30 feet, and that a removable refueling plug is used in place of rotating refueling plugs. Elimination of the rotating plugs enables the decrease in the size of the vessel.

3.6.3. Reactivity Control

The reactivity controls for the hot cell are generally the same as those used for the nonhot cell (see section 3.3.3). The major differences are that all disconnects and lifting lugs are designed for remote manipulation, and all units will operate in an inert atmosphere that may contain sodium vapor.

3.6.4. Reactor Shielding

The shielding described for the under-the-plug refueling system is essentially the same as that for the hot cell refueling system

(see section 3.5.4). The additional shielding required is the 4 feet of concrete surrounding the hot cell.

3.6.5. Reactor Auxiliary Systems

The reactor auxiliary systems for the spread-out arrangement with hot cell refueling are identical to those for the pot arrangement with hot cell refueling. Brief functional descriptions of these systems are given in section 3.4.5.

3.6.6. Fuel Handling System

This handling procedure requires a hot cell directly over the reactor core. Figures 18, 19, and 20 give the general shape and size of the hot cell and illustrate the reactor system as it applies to fuel handling.

The fuel handling system requires operations in three major areas: the hot cell, the fuel storage cell, and the fuel transfer tube. To perform the necessary functions, four major pieces of equipment are required: the hot cell crane, the fuel handling machine, the fuel transfer machine, and the fuel transfer tube. Some of these components are described in more detail in subsequent paragraphs.

The general fuel handling scheme centers on utilizing the hot cell crane to raise the reactor shield plug completely clear of the reactor pressure vessel. This requires unlatching of all control assemblies that penetrate the plug and extend into the reactor core. Thus, all plug-mounted drives and extensions are removed as a unit with the shield plug, leaving the reactor vessel completely open for fuel handling access. The fuel handling machine is then placed over the reactor and the lower portion is telescoped into the sodium to grapple the spent fuel assembly. The assembly is raised into the machine and a valve is closed to capture sodium with the assembly. The next step is retraction of the telescoped tube clear of the reactor vessel, and, with inert gas cooling, movement to the adjacent fuel storage cell where the spent assembly is deposited and a fresh assembly is picked up. After reactor startup and a suitable decay time, the fuel transfer machine is used to move the decayed assemblies from the storage cell to an area outside the hot cell and the reactor building by way of the fuel transfer tube.

Sodium is used as the cooling medium in the reactor, in the decay storage pool, and in the fuel handling machine. Since the sodium is opaque and therefore precludes visual observation, and because both the sodium and the fuel assemblies are highly radioactive, complete remote handling has been selected for this design.

The system selected follows "hot cell" techniques for handling radioactive materials; the methods are semi-automatic. The operator must supply the intelligence by initiating instructions. He is constantly supplied with information through interlock circuits that indicate progress and next-step data. He must initiate the operation, however, and preventive control precludes him from initiating the wrong operation. Machine movements are visually observed through shielded windows and by the use of shielded television cameras, but all operations will be traced primarily by position indicators. Bench marks are provided at several points within cells where visual sightings can be made; these serve as alignment points to check the automatic motions of the handling machines.

The overall fuel handling system utilizes conventional hardware and proven methods wherever possible. The system has rugged, long-lived components and generous clearances on moving parts. Machine handling motions are vertical, and the mechanisms in sodium are limited to the pickup grapple tool and a hinged valve. (See Fuel Handling Machine in this section.) Horizontal movements are performed by the cell cranes on an X-Y basis to minimize problems of "finding" operations and to simplify automation of the motions. Problems of orienting assemblies are eliminated by this system, and, in fact, rotation of assemblies is always restricted mechanically.

The prime function of the fuel handling system is removal of a spent fuel assembly from the reactor core and installation of a new assembly in its place. The equipment, however, must be capable of handling all assemblies that are to be loaded into the reactor grid. Figure 21 is a flow diagram of the fuel handling operations, which are as follows:

1. Replace spent fuel assemblies in the reactor core with new fuel assemblies.
2. Shuffle fuel assemblies from one position to another in the core.

3. Move spent fuel assemblies to a storage pool, and pick up new assemblies.
4. Move spent fuel assemblies out of the hot cell to a cleanup, disassembly, packaging, or shipment station.
5. Replace spent blanket assemblies in the reactor core with new blanket assemblies.
6. Move new fuel and blanket assemblies into the hot cell and into the storage pool in preparation for core loading.
7. Remove and replace spent control, safety, or special assemblies in the reactor core and move the assemblies out of the hot cell.

All these handling functions must be performed with maximum safety and reliability. The prime function of fuel replacement is effected with the reactor in the off-load condition and must be completed as quickly as possible to minimize reactor downtime. Each step of each function has sufficient cooling, instrumentation, and backup systems to prevent damage or propagation of damage.

Item 7 covers the handling, removal, and replacement of control or miscellaneous assemblies that may be inserted in the core. These assemblies have lifting adapters that are identical to those of the fuel assemblies, and their size, length, and configuration are compatible with the equipment described for fuel handling.

3.6.6.1. Fuel Transfer Tube

The fuel transfer tube is identical to the one used for the pot arrangement with under-the-plug refueling except that it is inside the hot cell (see section 3.3.6).

3.6.6.2. Fuel Storage Cell

The fuel storage cell is inside the hot cell adjacent to the reactor (see Figure 20). The proximity to the reactor minimizes equipment travel during refueling and hence reduces reactor shutdown time. The shutdown time is also kept low by providing access to the storage cell through a large plug opening where no seating or sealing is required.

The fuel storage cell provides intermediate storage for both new and spent fuel assemblies. New assemblies are

loaded into storage positions during reactor operation by the fuel transfer machine and are subsequently transferred into the reactor during the refueling procedure. Spent assemblies are placed in the cell directly after their removal from the reactor and are held there for about 3 months. This storage period allows the radiation and internal heating rates to decay to tolerable levels for the next handling procedures.

The fuel storage cell is a steel-lined pit approximately 20.5 feet square by 41 feet deep. It is closed at the operating floor level by a 17-foot-diameter, 5-foot-thick, stepped shield plug which provides radiation shielding to the hot cell area above. The shield plug is fitted with static O-ring seals to complete the gas-tight closure of the pit. It has 14 smaller access openings, each of which contains a stepped shielding and sealing plug when not in use. The first opening is 60 inches in diameter and gives access to all storage positions within the cell during the refueling procedure with the telescoping fuel handling machine. Each of the remaining 13 openings is 10 inches in diameter and is located to give access to all the storage positions on a particular radius. These openings are used during fuel transfer into and out of the hot cell while the reactor is on-line. In each of the 13 openings is a fixed guide tube extending down to within 2 feet of the storage positions. The storage positions within the cell are formed by a lattice structure of hexagonal stainless-steel tubes which are supported 36 feet below the operating floor on a grid structure.

This assembly is contained within a double-walled, stainless-steel vessel. The inside diameter of the inner vessel is 14 feet; it is 29.5 feet deep and is filled with sodium coolant (approximately 29,000 gallons). This vessel-within-a-vessel design is used to prevent inner vessel leakage from voiding the storage positions. The structure between the vessels gives vertical stability, and the space contains detectors to warn of inner vessel leakage. The entire vessel assembly is mounted on commercial roller bearings from the floor of the cell. A ring gear is mounted on the periphery of the bottom of the vessel, and the assembly is rotated through a worm gear arrangement which eliminates backlash and overtravel. An electric motor outside the cell, with its shaft extending through the seals in the wall, provides the driving force.

The top of the vessel is about 1.5 feet below the large shield plug, and the outer vessel wall is flanged outward to provide a seal face. Controlled-leakage seals are mounted on a skirt assembly which extends into the cell from the 17-foot-diameter plug liner. This forms a gas area between the top of the sodium coolant and the shield plug. Argon gas is used to inert both the cell around the storage vessel and the space between the double storage vessels.

The sodium coolant in the storage vessel is circulated through its own loop system. The inlet to the vessel enters through the skirt wall above the vessel and down a 1-foot-diameter pipe on the center line into a bottom plenum. The flow is then directed up through the assembly positions, out the exit pipe, and back to the pump. Baffles and flow deflectors positioned in the bottom plenum equalize flow. The vessel, supply and return piping, and the heat exchanger are heated electrically to maintain a minimum sodium temperature of 350 F during any periods when spent assemblies are not stored in the cell.

3.6.6.3. Fuel Handling Machine

The fuel handling machine is used exclusively for operations between the reactor and the fuel storage cell. It accomplishes all the loading and unloading of fuel assemblies in the core.

The fuel handling machine (see Figure 22) is assembled on a trolley which operates on rails about 20 feet above the operating floor. The trolley also contains the auxiliary hoist, which has a 10-ton capacity and is used to handle storage cell plugs. This hoist has a latching adapter to permit remote attachment. The trolley performs all lateral movements on an X-Y basis to maintain orientation of the machine at all times; this eliminates the need for all rotary motions and simplifies programming of the machine's movements. Viewing windows and TV cameras in the hot cell allow observation of the machine's motions, and proper positioning of the machine is checked constantly against scribed rail markings.

The operating principle of the fuel handling machine is to capture a volume of sodium coolant along with the fuel assembly and, accompanied by external gas cooling, to move the assembly from one location to the other. To accomplish this, the machine must extend

into the reactor or the fuel storage cell, grapple the desired fuel assembly, raise the assembly into a tube, valve off the required coolant, and then retract so that movement to another position can be completed.

The machine comprises three tube sections of different diameters, which telescope vertically to extend into the reactor and the fuel storage cell. Guide rollers between overlapping sections of the tubes maintain rigidity of the assembly and prevent rotation. The upper (largest) tube is mounted on the trolley structure and provides the operating base for the mechanism. The two inner sections are raised and lowered within this fixed tube to provide a total stroke of approximately 25 feet. The fully retracted position results in a 1-foot minimum clearance above the operating floor level, and the fully extended tube is mechanically stopped about 2.5 feet above the reactor core.

The drive system for the entire machine, contained within the closed housing on the trolley, is cooled by the argon gas flow. Argon gas under high pressure continually flows from the closed drive housing (which encloses the upper fixed tube on the trolley) down the center of the telescoping tubes and out around the lower telescoping tube; it is controlled by limited-leakage seals.

The cable hoist unit has a horizontally mounted winding drum which is automatically traversed to ensure that the cable is always directly above the center of the tubes. This unit is driven by a variable-speed electric motor and is synchronously geared to an argon cylinder hose reel which is likewise traversed automatically to ensure hose position. This hose provides argon to actuate the fuel assembly grapple tool. The cable and hose for these systems are trailed behind the trolley along with the cooling gas line.

The innermost tube in the telescoping assembly contains the fuel grapple tool which "reaches" the remaining 2.5 feet to latch with the fuel assembly and provides the anchor point for the hoisting cable. The grapple tool is basically an 8-inch-diameter tube with a top fitting to which the hoisting cable is attached, and a finger mechanism at the bottom. With the tubes fully extended, raising the grapple tool against a fixed stop in the lowest tube starts the nesting of the tubes. Retraction continues as the grapple tool is raised further, and each successive tube reaches a fixed stop and starts the nesting of that tube. In

reverse, lowering the grapple tool allows the tubes to telescope downward.

The fuel assembly is latched by a finger mechanism on the bottom end of the grapple tool. The 8-inch grapple tool tube is reduced at this end by a hexagonal adapter section which is split into six individual fingers at its corners. These fingers are machined with a hook shape to provide positive latching with the assembly. Two distinct operations are required to unlatch the fingers, and accidental unlatching is impossible as long as the weight of the assembly is on the fingers. A tapered surface machined on the inside of the fingers acts as a fulcrum against which an actuating piston works to open the fingers. The piston that actuates the grapple fingers is suspended just above this tapered surface on the end of an argon-actuated cylinder. This cylinder is mounted on a fixed platform just below the cable connection at the top of the tool; this position keeps the cylinder above the sodium coolant at all times.

Movement of the grapple tool within the lowest telescoping tube also operates the coolant flapper valve at the bottom of the tube. As the tool is raised the last 10 inches (before it begins to raise the lowest telescoping tube), it strikes a spring-loaded shoulder. This shoulder is connected to two connecting rods which extend down the annulus between the grapple tool and the tube and are hinged to the flapper valve. Completion of the stroke captures the sodium coolant.

Continued upward movement of the grapple tool starts the nesting of the lowest tube section into the next section. This accomplishes about 9.5 feet of the total 25-foot retraction stroke before it encounters the next stop and starts movement of the next tube section.

Maintenance of the fuel handling machine is expected to be minimal because of the simplicity of the system; however, the grapple tool can be lowered completely out of the telescoping tube assembly. The hoist cable and cylinder gas hose can then be remotely disconnected from the top of the tool and the tool can be replaced. Maintenance of the drive requires removal of the gas housing. Normal servicing can be accomplished with the housing in place and will be coordinated with servicing on the trolley.

3.6.6.4. Fuel Transfer Machine

The fuel transfer machine is used only for operations between the fuel storage cell and the fuel transfer tube. It is designed for safe and reliable handling of both new and spent fuel assemblies between the fuel storage cell and the cleanup area outside the hot cell. It is assembled on the same trolley as the fuel handling machine, and similar programmed X-Y motions of the trolley position the machine over the desired access positions. Argon gas is supplied to the machine to cool the decayed spent fuel assemblies.

The transfer machine consists primarily of a system of two telescoping tubes and a fuel grapple tool. The grapple tool is identical to that used in the fuel handling machine. It does not actuate the telescoping tube, however, but operates vertically within the lower tube and within the fixed guide tube in the fuel storage cell access port or in the transfer tube. Vertical movements of the grapple tool are controlled by the hoist cable, and grapple finger actuation is again controlled by the argon-operated cylinder mounted on the tool.

The stroke of the telescoping tubes (1 foot) in this machine is governed by argon pneumatic cylinders mounted on the upper tube section, which is fixed in the trolley structure. The motion is required only for mating with the fuel transfer tube. The lower telescoping tube is terminated with an electrically operated gas valve and an adapter that connects with a similar arrangement on the transfer tube. No mating is required above the fuel storage cell. The overlapping tube section has guide rollers to maintain the rigidity of the assembly and prevent rotation.

The hoist cable and the cylinder gas hose are routed up the center of the tubes and are wound on drums that automatically traverse to ensure a correct path down the tubes. The cable drum is driven by a variable-speed electric motor and is geared to the hose drum to synchronize their movements. The supply cable and hose for these systems are trailed behind the trolley.

The complete drive system is enclosed in a housing to which high-pressure argon gas is supplied. The gas cools the drive and is channeled down the tubes to cool the fuel assembly. Normal gas cooling flow is directly down the tubes and out through the

gas valve. The connection with the fuel transfer tube requires closure of the bottom gas valve. Pressure buildup inside the assembly is controlled by limited-leakage seals at the telescoping joint.

Maintenance of the fuel transfer machine is expected to be minimal because of the simplicity of the system; however, the grapple tool can be lowered completely out of the telescoping tube assembly. The hoist cable and cylinder hose can be disconnected remotely from the top of the tool and the tool can be replaced. Maintenance of the drive requires removal of the gas housing; normal servicing can be accomplished with the housing in place and will be coordinated with servicing on the trolley.

Figure 2. Pot Arrangement With
Under-the-Plug Refueling —
Plan View
(B&W Drawing A31061E-1)

LEGEND

- 1 FUEL TRANSFER CART
- 2 CLEAN UP AREA
- 3 FUEL & REGULATING CONTROL ROD TRANSFER TUBE
- 4 PUMP STORAGE PIT
- 5 EQUIPMENT WELL
- 6 BLAST DOOR
- 7 EQUIPMENT PENETRATION DOOR
- 8 PUMP & IHX VALVE BOX
- 9 EMERGENCY COOLING UNIT
- 10 PRIMARY PUMP
- 11 INTERMEDIATE HEAT EXCHANGER
- 12 REACTOR BUILDING AIR CONDITIONING EQUIPMENT
- 13 CLOSED LOOP GAS SHIELD COOLING SYSTEM
- 15 ROTATING DECAY HEAT STORAGE DRUM
- 16 HONEYCOMB CORE STRUCTURE & IHX STORAGE PIT
- 17 SMALL ROTATING PLUG
- 18 REFUELING MACHINE
- 19 PERSONNEL LOCK
- 20 BLAST WALL
- 21 INTERMEDIATE ROTATING PLUG
- 22 OUTER ROTATING PLUG
- 23 PRIMARY SODIUM STORAGE TANK PIT
- 24 COLD TRAP
- 25 SECONDARY SODIUM LINES
- 26 PUMP MOTOR STOWAGE "MOBILE"
- 27 REACTOR PIT
- 28 CRANE MAX. PICK-UP DIAMETER
- 29 REACTOR BUILDING
- 30 HONEYCOMB CORE STRUCTURE VALVE BOX
- 31 REGULATING ROD DROP PIT
- 32 UNOBSTRUCTED AREA, 4'-0" MIN. PASSAGEWAY
- 33 SPENT FUEL STORAGE
- 34 SPENT FUEL TRANSFER TUBE
- 35 FUEL TRANSFER CRANE
- 36 RUPTURED FUEL CANNING STATION
- 37 NEW FUEL TRANSFER TUBE
- 38 NEW FUEL STORAGE

NOTE:

1. FOR SECTIONS SEE DWG. NOS. A31066E & A31067E

Figure 3. Pot Arrangement With Under-the-Plug Refueling — Section Z-Z
(B&W Drawing A31066E-1)

LEGEND

- 1 OVERHEAD ROTARY CRANE BEAM RAIL
- 2 REACTOR BUILDING
- 3 OVERHEAD ROTARY CRANE
- 4 CRANE CIRCULAR BEAM RAIL
- 5 CRANE SUPPORT COLUMN
- 6 AIR CONDITIONING DUCT
- 7 AIR CONDITIONING UNIT
- 8 POT ANNULUS COOLING UNIT OPENING
- 9 POT ANNULUS COOLING DUCTS OPENING
- 10 REACTOR VESSEL
- 11 PRIMARY PUMP & IHX UNIT
- 12 CLEAN UP AREA
- 13 TRANSFER TUBE OPENING
- 14 CHARGE FACE SHIELD PLUG
- 15 OUTER ROTATING PLUG
- 16 INTERMEDIATE ROTATING PLUG
- 17 SMALL ROTATING PLUG
- 18 REACTOR VESSEL PIT (CONCRETE)
- 19 REACTOR VESSEL BEARING SUPPORT PIT (CONCRETE)
- 20 REACTOR BUILDING PENETRATION
- 21 SECONDARY SODIUM LINES
- 22 LOWER EQUIPMENT ROOMS (TYP)
- 23 REACTOR VESSEL INSULATION
- 24 PRIMARY SODIUM INLET PIPE
- 25 MELT DOWN PAN
- 26 CORE VESSEL
- 27 ELEMENT GRID PLATE
- 28 NEUTRON SHIELD (LOWER)
- 29 SUPPORT GRID
- 30 ROTATABLE DECAY HEAT STORAGE RACK
- 31 TENSION SUPPORT MEMBER
- 32 SHIELD PLUG BEARING PLATE
- 33 REFUELING MACHINE
- 34 CORE
- 35 NEUTRON SHIELD (UPPER)

Figure 4. LMFBR Fuel Element Diagram for Under-the-Plug Refueling

Figure 5. LMFBR Under-the-Plug Refueling Mechanism

Figure 6. LMFBR Under-the-Plug Fuel
Assembly Grapple Tool

Figure 7. LMFBR Fuel Transfer Machine

Figure 8. LMFBR Fuel Transfer
Machine — Cross
Section Near Base

Figure 9. Pot Arrangement With Hot Cell Refueling — Plan View
(B&W Drawing A31076E)

Figure 10. Pot Arrangement With Hot Cell Refueling — Section Z-Z
(B&W Drawing A31075E)

LEGEND

- 1 REACTOR BUILDING
- 2 ROTARY CRANE (REACTOR BUILDING)
- 3 HOT CELL CLOSURE PLUG
- 4 ROTARY CRANE (HOT CELL)
- 5 HOT CELL
- 6 REACTIVITY CONTROL DRIVES
- 7 REMOVABLE SHIELD PLUG
- 8 GUIDE MECHANISM
- 9 OVERHEAD FUEL TRANSFER RAIL (REMOVABLE SECTION)
- 10 OVERHEAD FUEL TRANSFER RAIL
- 11 JIB CRANE
- 12 HOT CELL ACCESS PLUG
- 13 FUEL HANDLING MACHINE
- 14 EQUIPMENT HANDLING CART
- 15 EQUIPMENT HANDLING CART RAIL
- 16 SHIELD PLUG
- 17 SHIELD PLUG BEARING PLATE
- 18 REACTOR VESSEL
- 19 REACTOR VESSEL INSULATION
- 20 EQUIPMENT ROOMS
- 21 TENSION SUPPORT MEMBERS
- 22 ROTATABLE DECAY HEAT STORAGE RACK
- 23 SUPPORT GRID
- 24 CORE VESSEL
- 25 MELT DOWN PAN
- 26 FUEL TRANSFER REMOVABLE PLUG
- 27 FUEL TRANSFER MACHINE WITH AUXILIARY HOIST
- 28 FUEL TRANSFER TUBE
- 29 TRANSFER TUNNEL
- 30 FUEL TRANSFER CART
- 31 NEUTRON SHIELD (UPPER)
- 32 ACTIVE CORE
- 33 CORE BLANKET
- 34 ELEMENT GRID PLATE
- 35 NEUTRON SHIELD (LOWER)
- 36 FUEL TRANSFER CART RAIL
- 37 HOT CELL ACCESS DOOR

Figure 11. LMFBR Fuel Element in Hot Cell —
Flow Diagram

Figure 12. Pot-Type Hot Cell Fuel Handling Machine

Figure 13. Pot-Type Hot Cell Fuel Handling Machine —
Cross Sections

SECTION A-A

SCALE: 1" = 1'-0"

SECTION B-B

SCALE: 1" = 1'-0"

Figure 14. Spread-Out Arrangement With
Through-the-Plug Refueling —
Plan View
(B&W Drawing A31063E-2)

LEGEND

- 1 PRIMARY PUMP
- 2 INTERMEDIATE HEAT EXCHANGER
- 3 CORE VESSEL
- 4 NEW FUEL STORAGE AREA
- 5 TOP SHIELD PLUGS
- 6 PUMP MOTOR STORAGE "MOBILE"
- 7 COLD TRAPS
- 8 FUEL TRANSFER TUBE
- 9 CLEAN UP AREA
- 10 REACTOR BUILDING
- 11 EQUIPMENT AIR LOCK
- 12 PERSONNEL AIR LOCK
- 13 BLAST DOOR
- 14 BLAST WALL
- 15 WATER STORAGE POOL "SPENT FUEL"
- 16 NEW FUEL TRANSFER TUBE
- 17 EMERGENCY CORE COOLER
- 18 PUMP OR INTERMEDIATE HEAT EXCHANGER STORAGE
- 19 FUEL STORAGE CELL
- 20 EQUIPMENT WELL
- 21 PRIMARY SODIUM STORAGE TANKS
- 22 FUEL TRANSFER CART
- 23 RUPTURED FUEL CANNING STATION
- 24 SPENT FUEL TRANSFER TUBE
- 25 FUEL TRANSFER CRANE
- 26 CONTROL ROD DRIVE STORAGE
- 27 VALVE BOX "MOBILE"
- 29 AIR CONDITIONING EQUIPMENT

NOTE:
1 - REACTOR BUILDING THICKNESS ~4'-0"

Figure 15. Spread-Out Arrangement With
Through-the-Plug Refueling —
Section Y-Y
(B&W Drawing A31064E-1)

- LEGEND
- 1. CORE VESSEL
 - 2. FUEL ASSEMBLY
 - 3. TOP SHIELD PLUGS
 - 4. THERMAL SHIELD
 - 5. DRAIN PAN
 - 6. TOP BLANKET
 - 7. BOTTOM BLANKET
 - 8. INSULATION
 - 10. OVERHEAD CRANE
 - 13. REACTOR BUILDING
 - 14. PRIMARY PUMP
 - 15. INTERMEDIATE HEAT EXCHANGER
 - 16. PRIMARY SODIUM STORAGE TANKS

NOTE:

- 1. REACTOR BUILDING THICKNESS - 4'-0"
- 2. POSITION OF RE-FUELING MACHINE FOR FUEL REMOVAL.

Figure 16. Spread-Out Arrangement With
Through-the-Plug Refueling —
Section Z-Z
(B&W Drawing A31065E-2)

- LEGEND**
- 1. CORE VESSEL
 - 2. FUEL ASSEMBLY
 - 3. TOP SHIELD PLUGS
 - 4. THERMAL SHIELD
 - 5. DRIP PAN
 - 6. TOP BLANKET
 - 7. BOTTOM BLANKET
 - 8. INSULATION
 - 9. CHARGE FACE
 - 10. FUEL STORAGE CELL SHIELD PLUG
 - 11. FUEL STORAGE CELL
 - 12. FUEL STORAGE CELL
 - 13. TRANSFER TUBE
 - 14. RE-FUELING MACHINE
 - 15. TRANSFER CART
 - 16. CLEAN-UP AREA
 - 17. WATER STORAGE POOL
 - 18. AIR CONDITIONING EQUIPMENT
 - 19. OVERHEAD CRANE
 - 20. BLAST DOOR
 - 21. EQUIPMENT AIR LOCK
 - 22. REACTOR BUILDING
 - 23. SUPPORT RAFT

NOTE:
1 - REACTOR BUILDING THICKNESS - 4'-0"

Figure 17. Fuel Element Flow Diagram — LMFBR
Spread-Out Concept With Through-the-
Plug Refueling

Figure 18. Spread-Out Arrangement With Hot Cell Refueling — Plan View (B&W Drawing A31017E)

LEGEND

- 1 PRIMARY PUMP
- 2 INTERMEDIATE HEAT EXCHANGER
- 3 CORE VESSEL
- 4 NEW FUEL STORAGE AREA
- 5 TOP SHIELD PLUG
- 6 PUMP MOTOR STORAGE "MOBILE"
- 7 RE-FUELING MACHINE
- 8 FUEL TRANSFER TUBE
- 9 CLEAN UP AREA
- 10 REACTOR BUILDING
- 11 EQUIPMENT AIR LOCK
- 12 PERSONNEL AIR LOCK
- 13 BLAST DOOR
- 14 BLAST WALL
- 15 WATER STORAGE POOL "SPENT FUEL"
- 16 NEW FUEL TRANSFER TUBE
- 17 REGULATING CONTROL ROD TRANSFER TUBE
- 18 PUMP OR INTERMEDIATE HEAT EXCHANGER STORAGE
- 19 HOT CELL FUEL STORAGE
- 20 EQUIPMENT WELL
- 21 PRIMARY SODIUM STORAGE TANKS
- 22 FUEL TRANSFER CART
- 23 RUPTURED FUEL CANNING STATION
- 24 SPENT FUEL TRANSFER TUBE
- 25 FUEL TRANSFER CRANE
- 26 VIEWING WINDOW
- 27 VALVE BOX "MOBILE"
- 28 HOT CELL
- 29 AIR CONDITIONING EQUIPMENT

Figure 19. Spread-Out Arrangement With Hot Cell Refueling — Section Y-Y (B&W Drawing A31018E)

- LEGEND
- 1. CORE VESSEL
 - 2. FUEL ASSEMBLY
 - 3. REACTOR COVER PLUG
 - 4. THERMAL SHIELD
 - 5. Drip PAN
 - 6. TOP BLANKET
 - 7. BOTTOM BLANKET
 - 8. INSULATION
 - 9. COVER PLUG WINCH HOUSE
 - 10. OVERHEAD CRANE
 - 11. HOT CELL
 - 12. VIEWING WINDOW
 - 13. CONTAINMENT BUILDING
 - 14. PRIMARY PUMP
 - 15. INTERMEDIATE HEAT EXCHANGER
 - 16. SECONDARY SODIUM STORAGE TANKS

Figure 20. Spread-Out Arrangement With
Hot Cell Refueling — Section
Z-Z (B&W Drawing A31016E)

Figure 21. LMFBR Fuel Element Flow Diagram — Spread-Out Concept With Hot Cell

Figure 22. LMFBR Spread-Out-Type Hot Cell Fuel Handling and Fuel Transfer Machine

4. RESULTS

4.1. General

The four concepts described in section 3 have been evaluated using quantitative estimates of refueling times and capital costs (shown in the appendixes) as well as qualitative conclusions derived from studying the construction, operation, and maintenance aspects of each plant.

The pot arrangements are evaluated in section 4.2, and the spread-out arrangements are evaluated in section 4.3. Section 4.4 discusses the relative merits of the "optimum" pot arrangement versus the "optimum" spread-out arrangement to arrive at the selection of the most desirable reactor system.

The following abbreviations are used throughout this section:

- PUP — Pot Arrangement With Under-the-Plug Refueling
- PHC — Pot Arrangement With Hot Cell Refueling
- SOTP — Spread-Out Arrangement With Through-the-Plug Refueling
- SOHC — Spread-Out Arrangement With Hot Cell Refueling

4.2. Evaluation of Pot Arrangements

This evaluation applies to the PUP and PHC plants described in sections 3.3 and 3.4. The comparative advantages of the two plants are summarized in Table 1.

It can be seen in Appendix C, Table C-1, that using a PHC scheme results in higher capital costs. This is due primarily to the factors listed below.

1. The cost of the hot cell.
2. The cost associated with the larger-diameter reactor vessel needed to accommodate the hot cell.
3. The cost of a taller reactor building to accommodate the hot cell.
4. The cost of remotely operated manipulators to perform handling and maintenance procedures within the hot cell.

Additional operating costs are also incurred with the hot cell since 13% more downtime is required for one-third of a core refueling (see Appendix B, Table B-1). Although no numerical estimates were made, additional operating costs will probably be incurred with the hot cell because of the necessity for performing remote maintenance procedures inside the hot cell.

The PUP system is especially adaptable to partial refueling or fuel shuffling programs, since it requires less extensive reactor shutdown and startup times than does the PHC system. This occurs primarily because, with under-the-plug refueling, the core coolant outlet temperature need not be reduced below approximately 800 F to perform a fuel handling function; but with hot cell refueling, a fuel handling function cannot be accomplished until the core coolant outlet temperature has been reduced to approximately 400 F.

The 800 F refueling temperature for the PUP system was established by considering the cooling requirements of an irradiated, decay-heat-producing fuel assembly. The results of the analysis of the thermal and hydraulic characteristics of the reference fuel assembly show that adequate cooling can be achieved with natural convection, so that the maximum cladding surface temperature does not exceed 975 F, assuming an 800 F fuel assembly inlet temperature.

The refueling temperature for the PHC system was established at 400 F by considering the melting temperature and vapor release characteristics of the sodium coolant. The minimum temperature of 400 F should be sufficiently above the solidification temperature to prevent any portion of the coolant system from cooling to the point where any of the coolant could freeze. When the sodium coolant temperature is reduced to the lowest point that is safely possible without danger of solidification, there is a corresponding reduction in the mass of sodium vapor available for deposition on the walls and equipment of the hot cell. For instance, when the sodium temperature is lowered from 800 to 400 F, the mass of sodium vapor per cubic foot is reduced by a factor of approximately 3000.

In addition to the shorter shutdown and startup times, the PUP method requires less refueling preparation time and less actual refueling time. Further proof that the PUP system is more efficient than the PHC system is presented in Appendix B, Tables B-1 through B-5.

The PUP system allows plant maintenance personnel to use contact methods to remove, service, replace, or repair equipment that is on or above the operating floor level. Since the operating floor level is a man-access area, the reactivity control rods and the fuel handling mechanisms can be actuated manually if need be; and manual bolting and unbolting of the rotating shield plugs and refueling access ports are possible. However, with the PHC system, the equipment within the hot cell will require completely remote removal and operating and maintenance procedures, since radioactive contaminants and sodium vapor will enter the hot cell during a refueling cycle.

To effect these remote procedures requires that special remotely operated tools be installed in the hot cell and that all equipment to be maintained, removed, or replaced be designed with bolts, nuts, quick disconnects and power input connectors that can be manipulated by these tools. Manual actuation of the reactivity control rods or the fuel handling machine is virtually impossible, and even if accomplished it would be by remote means.

The ability to perform all the maintenance required in a PHC system remotely appears to be questionable. Indications are that provisions will have to be made for man access into the hot cell to assist in certain maintenance functions. However, when the refueling cell is opened to the primary sodium pool, it will inevitably become contaminated by some of the primary coolant and some of the fission products contained in the coolant plus some of the fission products from the cover gas. As a result, the radiation background of this type of cell may not be able to be reduced low enough to give sufficient time for a man to do a reasonable amount of work without receiving an unacceptable radiation dose. This dictates that some means must be devised whereby the exposed surfaces of the refueling cell and all exposed equipment can be decontaminated to allow the maintenance to be accomplished without significant mechanical, radiation, or contamination problems.

The problems associated with the cleanup of a hot cell are formidable even without being compounded by having to perform the operations in a cell that is sitting above a huge mass of hot, radioactive liquid sodium. Normal cleaning procedures, such as reactive and nonreactive cleaning methods, cannot be used here, since both methods require

flushing as a final step, which presents the possibility of introducing foreign matter into the primary coolant via inleakage of the flushing agent. It has been demonstrated that sodium can be cleaned from surfaces by vacuum distillation; however, this requires temperatures in the 900 to 1000 F range and pressures below 1 mm Hg absolute. The temperature is objectionable from the standpoint of equipment within the cell, and the vacuum is intolerable because it would tend to pull more fission products and sodium vapor from the reactor sodium pool.

In the area of stuck core component removal, the PHC scheme has an advantage over the PUP scheme, since full access to the core is realized when the removable plug is lifted; however, the rotary plugs of the under-the-plug refueling scheme are adaptable to the removal of such components since they, too, can be lifted by incorporating a valve box into the lifting procedure.

The fact that full access to the core is realized when using the PHC scheme creates three conditions that are reduced or eliminated by the PUP scheme. First, if the heavy plug or any other object were dropped onto the unprotected core, it could wreck the core arrangement. Second, if the hot cell must be designed to withstand a DBA during refueling, additional cost is imposed on the PHC system because the present hot cell is not designed for this condition. The PUP system provides protection against a DBA during refueling as well as during operation. Third, the total surface area exposed to contamination is increased considerably.

Since the hot cell does not cover the reactor shield plug in its entirety, and since the IHXs, cold traps, primary pumps, and emergency cooling units penetrate the shield plug external to the hot cell, the cell was considered to be of minimal value in sodium fire prevention and missile protection. This observation is based on the assumption that it appears feasible and mandatory to incorporate a DBA-resistant design into the cover structure of the primary tank, regardless of whether or not a hot cell is incorporated to accomplish refueling.

The sealing of penetrations into a space that is contaminated by radioactive fission products is a problem experienced by both systems. It appears that the PUP scheme has an advantage if one considers the extra penetrations required to install the viewing ports, equipment access hatches, electrical lines, instrumentation lines, and cooling lines in the hot cell walls.

By weighing all the foregoing factors as objectively as possible, it appears that the PUP scheme is superior to the PHC scheme.

4.3. Evaluation of Spread-Out Arrangements

This evaluation applies to the SOTP and SOHC plants described in sections 3.5 and 3.6. The comparative advantages of the two plants are summarized in Table 2.

As can be seen in Appendix C, Table C-3, an SOHC scheme results in slightly higher capital costs than does the SOTP method. The cost differential is small because of offsetting costs of major equipment in the two plants. The offsetting effects are (1) the SOTP requires rotating plugs, which increase the size of the reactor vessel and cover structure; and (2) the SOHC requires a hot cell and more fuel handling equipment.

Refueling costs tend to be higher for the SOTP, since about 11% more downtime is required for one-third core refueling (see Appendix B, Table B-1). Although no numerical estimates were made, these costs will probably be offset by added costs associated with remote maintenance procedures required for the SOHC plant. More downtime is required for the SOTP system, because the cask refueling machine requires make-and-break seal connections every time it is moved to a different location. The SOHC system does not have this requirement.

The SOTP system is especially adaptable to partial refueling or fuel shuffling of $1/6$ of a core or less, since it requires less extensive reactor shutdown and startup times and less refueling preparation time than does the SOHC system. This occurs primarily because, with through-the-plug refueling, the core coolant outlet temperature need not be reduced below about 800 F to perform a fuel handling function; but with hot cell refueling, a fuel handling function cannot be accomplished until the core coolant outlet temperature has been reduced to about 400 F.

A discussion of these sodium temperature conditions was given in section 4.2, and the requirements given for the PUP system also apply to the SOTP system. Similarly, the requirements given for the PHC apply to the SOHC. Also directly applicable are the discussions that were given for operation and maintenance.

It is difficult to select the best refueling method for the spread-out arrangement. Although the study shows the difference to be very small,

the SOTP system was chosen on the basis of our assessment of hot cell operating expenses, the difficulties that would be encountered with electrical components operating in an argon environment, and the capital expenditure to design and build the hot cell for DBA conditions, if this becomes a requirement.

4. 4. Selection of Most Desirable Reactor System

The evaluations of sections 4.2 and 4.3 indicate that the PUP is the best pot-refueling arrangement and that the SOTP is the best spread-out-refueling arrangement. This section compares these two plants to select the most desirable reactor system.

The PUP system is briefly described in section 3.3, and the SOTP plant is described in section 3.5. The advantages of these two plants are compared in Table 3.

From Appendix C, Tables C-1 and C-3, it appears that the differential capital cost of the SOTP scheme is about 2% more than that of the PUP. Also, as noted in Appendix C, the extra cost associated with additional piping and a pump in the hot leg for the SOTP combined pump-IHX units is not included in these estimates. This increases the cost differential and favors the PUP system even more. Because the PUP has a larger reactor vessel and cover structure and the requirement for two fuel handling mechanisms instead of one, these items cost more than for the SOTP. These costs are more than offset in the SOTP by the larger reactor building and the extra cost attached to the separate fuel storage vessel.

Refueling time is much less for a PUP than for a SOTP. From Table B-1, it can be seen that the SOTP system requires about 36% more refueling time than does the PUP system. Although a small amount of additional time (about 3%) is required to prepare the PUP plant for refueling, this is completely overshadowed by the extra time (about 130%) required by the SOTP plant for actual refueling.

The long SOTP refueling time is directly attributable to the necessity of moving the fuel handling machine from the fuel storage cell to the reactor and back to the fuel storage cell for each fuel assembly replacement cycle. Also, each time the fuel handling machine is relocated, gas-tight seals between the machine and its mating surface (either the

reactor vessel cover structure or the fuel storage cell cover structure) have to be made or broken.

During the PUP refueling, the fuel assemblies being transferred remain completely immersed in the circulating sodium pool. This provides greater assurance against the loss of cooling during spent fuel transfer. Although the fuel assembly is also immersed in sodium in the SOTP refueling machine, the main coolant is inert gas. The consequences of losing gas cooling during transfer, as discussed in reference 4, require that immediate action be taken to provide backup cooling to the fuel assembly.

Having under-the-plug refueling with spent fuel storage provided in the reactor vessel, along with having all the primary system equipment inside the reactor vessel, requires that the reactor vessel for the PUP system be considerably larger than that for the SOTP system. The large reactor vessel dictates at least partial, if not complete, field fabrication, whereas the SOTP reactor vessel is amenable to shop fabrication. Even though the reactor vessel is larger, a smaller reactor building (110 feet versus 150 feet diameter) is possible in the PUP plant because of a greater concentration of primary equipment closer to the reactor center line.

In the pool concept, the entire primary sodium inventory is contained in the reactor vessel instead of being dispersed in many separate vessels and associated piping having a larger combined wetted surface. Hence, the possibility of a leak occurring in the primary system is greatly reduced, and the plant safety and reliability are enhanced. The PUP system also reduces the leak-tightness requirements on piping, IHXs, and other equipment, since a leak means only slightly lower efficiency and not a breach of the primary system boundaries. Without relaxing the ASME Code requirements, this results in less stringent tolerances during fabrication and an associated reduction in the cost of these components. Credit for this cost reduction was not taken in this comparison.

The large sodium pool in the PUP plant is more capable of reducing any thermal shocks that might arise. This is possible because there is more sodium that must absorb or give up heat for any change in temperature to occur. A reduction in thermal shock tends to lead to longer

service life of primary system components by reducing the cyclic stresses for which the unit must be designed.

The requirements for neutron activation protection of the reactor vessel are much less stringent than for the protection of secondary sodium. Therefore, the spread-out plant has less neutron shielding than does the pot plant. Also, because of the biological shielding between the reactor vessel of the SOTP and the IHXs, there is less potential to activate the secondary sodium. This decreases the possibility of having contaminated secondary components and the associated maintenance problems.

The pot concept using the combined pump-IHX is as compact a concept as could be conceived, and thus it is considered that a more conventional piped spread-out system would, by the very nature of its spread-out characteristics, cost more. In addition, the conventional spread-out system has a much larger sodium-wetted surface, and the possibility of accidental sodium drainage below the fuel level is increased.

Utilizing the information available at this time, the results of this trade-off strongly indicate that the pot arrangement with under-the-plug refueling is the prime candidate for the most desirable LMFBR system.

Table 1. Pot Arrangement Comparative Advantages

PUP	PHC
1. Contact maintenance procedures.	1. Full access to core for removal of stuck core components.
2. Greater protection to core from accidental droppage of tools, etc.	2. Better access for removal of reactor internals.
3. Less surface area exposed to contamination.	3. Greater protection against DBA during operation.
4. Greater protection against DBA during refueling.	4. Some core component inspection capability.
5. Shorter refueling time.	
6. Lower capital costs.	
7. Smaller reactor building.	
8. Fewer sealing problems during operation and refueling.	
9. Under normal conditions, most of the electrical components operate in an air environment.	

Table 2. Spread-Out Arrangement Comparative Advantages

SOTP	SOHC
1. Contact maintenance procedures.	1. Full access to core for removal of stuck core components.
2. Greater protection to core from accidental droppage of tools, etc.	2. Shorter refueling time.
3. Less surface area exposed to contamination.	3. Better access for removal of reactor internals.
4. Greater protection against DBA during refueling.	4. Greater protection against DBA during operation.
5. Fewer sealing problems during operation and refueling.	5. Fewer make and break connections during refueling.
6. Lower capital costs.	6. Some core component inspection capability.

Table 3. Comparative Advantages

PUP	SOTP
1. Primary coolant entirely contained in one vessel.	1. Smaller reactor vessel and cover structure.
2. Shorter refueling time.	2. Less neutron shielding required.
3. Reduced leak tightness requirements for primary system components.	3. Shop built reactor vessel.
4. Lower thermal shock potential.	4. Less potential to activate secondary sodium in IHX.
5. Fewer make-and-break connections during refueling.	5. Easier to install and remove primary system components.
6. Smaller reactor building.	
7. Greater assurance against loss of cooling during fuel transfer.	
8. Less wetted surface contaminated by primary sodium.	
9. Lower capital costs.	

APPENDIX A
Equipment and Facilities List

Included in this appendix are Tables A-1 and A-2 itemizing the major equipment and facilities for the pot and spreadout arrangements. Each table shows where major differences exist in an arrangement as a result of a different refueling scheme. Most of these variations are reflected in the capital cost estimates included in Appendix C.

Table A-1. Pot Arrangement—Equipment and Facilities List

Unit	No. of units	
	Under the plug	Hot cell
Pump - IHX	6	6
Cold traps	2	2
Emergency core coolers	1	1
Rotatable fuel storage racks	2	2
Rotatable plugs (refueling)	3	0
Removable plugs (refueling)	0	1
Fuel Transfer Tubes	1	1
Equipment entry door	1	1
Personnel entry door	1	1
Blast door	1	1
Pump storage pit	1	1
IHX storage pit	1	1
Reactivity control rods	25	25
General purpose crane	1	1
Fuel transfer mechanism	1	0
Fuel transfer cask	1	0
Fuel transfer machine	0	1
Fuel handling machine	0	1
Removable plug hoist	0	1
Remote maintenance equipment	None	As required
Remote viewing equipment	None	As required
Equipment well	1	1
Access ports (refueling)	4	2
Primary sodium dump tanks	2	2

Table A-1. (Cont'd)

Unit	Size, ft-in.	
	Under the plug	Hot cell
Reactor vessel (dia)	50-9	62-0
Top head (dia)	60-3	72-0
Reactor vessel pit (dia)	55-9	67-0
Outer rotatable plug (dia)	29-3	--
Intermediate rotatable plug (dia)	21-8	--
Small rotatable plug (dia)	6-5	--
Removable plug (dia)	--	24-0
Cold trap (dia)	8-0	8-0
Emergency core cooler (dia)	4-0	4-0
Pump - IHX flange (dia)	10-8	10-8
Rotatable fuel storage rack (dia)	15-8	15-8
Core vessel (dia)	12-6	12-6
Reactor building (dia)	110-0	110-0

Unit	Location, ft-in.	
	Under the plug	Hot cell
Pump - IHX (radius)	20-0	25-6
Cold traps (radius)	21-3	25-9
Emergency core cooler (radius)	23-2	25-6

Unit	Weight or capacity, tons	
	Under the plug	Hot cell
General purpose crane (capacity)	200	100
Removable plug hoist (capacity)	--	200
Primary sodium inventory	1750	2625

Table A-2. Spread-Out Arrangement — Equipment and Facilities List

Unit	No. of units	
	Through the plug	Hot cell
1. Pump - IHX	6	6
2. Cold traps	2	2
3. Emergency core coolers	1	1
4. Decay fuel storage cell	1	1
5. Rotatable plugs (refueling)		
a. Over reactor	2	0
b. Over fuel storage cell	1	0
6. Removable plugs (refueling)		
a. Over reactor	0	1
b. Over fuel storage cell	0	1
7. Fuel transfer tubes	1	1
8. Equipment entry door	1	1
9. Personnel entry door	1	1
10. Blast door	1	1
11. Pump storage pit	1	1
12. IHX storage pit	1	1
13. Reactivity control rod	25	25
14. General purpose crane	1	1
15. Fuel handling machine	0	1
16. Fuel transfer machine	0	1
17. Fuel handling machine (shielded cask)	1	0
18. Removable plug hoists	0	1
19. Equipment well	1	1
20. Access ports (refueling)		
a. Reactor	1	1
		(this is the same unit as Item 6a)
b. Fuel storage cell	6	13
		(including Item 6b)
21. Primary sodium dump tanks	2	2
22. Refueling equipment and control rod extension maintenance and storage pit	1	0
23. Remote maintenance equipment	None	As required
24. Remote viewing equipment	None	As required

Table A-2. (Cont'd)

Unit	Size, ft-in.	
	Through the plug	Hot cell
Reactor vessel (dia)	30-0	17-9
Top head (dia)	37-4	27-6
Reactor vessel pit (dia)	34-0	21-9
Rotatable plugs (reactor head)		
a. Concentric plug (dia)	29-3	--
b. Eccentric plug (dia)	21-8	--
Rotatable plugs (fuel storage cell)	15-6	--
Removable plugs		
a. Reactor head (dia)	--	18-2
b. Fuel storage cell (dia)	--	5-0
Cold trap (dia)	8-0	8-0
Emergency core cooler (dia)	4-0	4-0
Pump - IHX flange (dia)	10-8	10-8
Fuel storage vessel (dia)	15-6	14-0
Reactor building (dia)	150-0	150-0

APPENDIX B
Refueling Time Estimates

Basis for Refueling Time Estimates

The refueling times for each of the arrangement-refueling method combinations considered in this tradeoff were estimated. Eight basic operations were identified for each combination:

1. Reactor shutdown.
2. Physics testing before refueling.
3. Refueling preparation.
4. Refueling procedure.
5. Physics testing during refueling.
6. Preparation of reactor for startup.
7. Physics testing before startup.
8. Startup procedure (to 50% of full power).

The refueling times for the eight basic operations are summarized in Table B-1. These refueling times represent the best estimates possible consistent with the level of detail that was achievable in this study. Although some absolute quantities may be questionable, the relative values between the concepts appear realistic.

The estimating basis for each of the eight major operations is presented below. Except as indicated, the estimates are shown in Table B-1.

Operation 1 — Reactor Shutdown

Reactor shutdown as used here refers to cooldown of the core outlet sodium from an operating temperature of 1100 F to the appropriate refueling temperature. For under-the-plug or through-the-plug refueling, a primary sodium coolant temperature of 800 F was established. For hot cell refueling, a primary sodium coolant temperature of 400 F was established. The rate at which these temperature changes can take place was established at an average value of 50 degrees per hour for the main body of sodium coolant (800 to 400 F) and at approximately 100 degrees per hour for the core outlet coolant (1100 to 800 F). The slower rate of temperature change for the main body of sodium coolant was deemed necessary to reduce the thermal racking of the primary vessel and top shield plug and thereby enhance their structural integrity and leaktightness. The higher rate of temperature change for the core outlet coolant was considered acceptable since the thermal shields or the core vessel containing this flow are internal to the primary vessel and are subjected to a very small pressure differential.

Operations 2, 5, and 7 — Physics Testing

No sound basis exists for the physics testing times assumed in these operations. However, they appear to be adequate for the expected testing.

Operation 3 — Refueling Preparation

Details of these estimates for all the arrangement-refueling method combinations are listed in Tables B-2, B-4, B-6, and B-8.

Operation 4 — Refueling Procedure

Details of these estimates for all the arrangement-refueling method combinations are listed in Tables B-3, B-5, B-7, and B-9. These refueling estimates assume batch refueling of 1/3 of the core. Since there are 375 fuel and blanket assemblies in the core, 125 assemblies are replaced at each refueling.

Operation 6 — Preparation of Reactor for Startup

This operation is the reverse of operation 3, and therefore the time estimate is identical to that for operation 3.

Operation 8 — Startup Procedure to 50% Full Power

The return to power assumes that the rate for increasing sodium coolant temperature is the same as that for decreasing sodium temperature as described in operation 1. Fifty percent full power, as used here, refers to attaining a core outlet coolant temperature of 950 F (100% = 1100 F). This corresponds to slightly less than 20% of the maximum power generation.

The percentage refueling times in Table B-1 are based on a cycle of reactor operation of 350 full power days and a 90% plant load factor. These percentages were calculated as follows:

$$\% \text{ Refueling Time} = \frac{(\text{Total Refueling Time in Days})}{(1.111)(350 \text{ Full Power Days}) + (\text{Total Refueling Time in Days})}$$

$$\% \text{ Refueling Time} = \frac{RT}{388.85 + RT}$$

Table B-1. Summary — Refueling Time Estimates (hrs)

Operation	Pot arrangement		Spread-out arrangement	
	<u>Under-the-plug</u>	<u>Hot cell</u>	<u>Through-the-plug</u>	<u>Hot cell</u>
1. Reactor shutdown	3	11	3	11
2. Physics testing before refueling	24	24	24	24
3. Refueling preparation	7.9	10.9	7.7	11.3
4. Refueling procedure	60.8	66	141	87.7
5. Physics testing during refueling	24	24	24	24
6. Preparation of reactor for startup (reverse of item 3)	7.9	10.9	7.7	11.3
7. Physics testing before startup	100	100	100	100
8. Startup procedure to 50% full power	1.5	9.5	1.5	9.5
9. Total estimated refueling time				
Hours	229.1	256.3	308.9	278.8
Days	9.5	10.7	12.9	11.6
%	2.4	2.7	3.2	2.9

Table B-2. Pot Arrangement Under-the-Plug Refueling
(Basis for Estimate)

Operation 3 — Refueling Preparation After 3-Hour Shutdown

<u>Step</u>	<u>Time, min</u>
1. Disengage drive extensions from regulating column	25.0
2. Raise drive extension 3 in. at 0.25 in./min (25 total)	300.0
3. Release locks on rotatable shield plugs and raise plugs 6 in. to lift upper grid plate clear of core assemblies	30.0
4. Install refueling mechanism (this is concurrent with steps 1 through 3)	--
5. Rotate plugs to check motion and drive operation, position refueling mechanism over one fuel storage rack access plug	30.0
6. Lower the refueling mechanism grapple 8.5 ft at 0 - 60 fpm	0.28
7. Actuate the grapple cylinder - opens fingers	0.25
8. Lower grapple tool 6 in. at 0 - 5 fpm	0.20
9. Deactivate grapple cylinder - closes fingers (latches storage rack access plug)	0.25
10. Raise plug 18 in. at 0 - 5 fpm	0.6
11. Rotate shield plugs to access plug over storage bracket	1.0
12. Lower plug 12 in. at 0 - 5 fpm	0.4
13. Deactivate grapple fingers - opens fingers	0.25
14. Raise grapple tool 12 in. at 0 - 5 fpm	0.4
15. Rotate shield plugs to position refueling mechanism over second fuel storage rotor access plug	2.0
16. Lower grapple tool 18 in. at 0 - 5 fpm	0.6
17. Repeat steps 9 through 14 to remove and store second plug	2.9
18. Rotate shield plugs to position the refueling mechanism over the first spent assembly to be removed	2.0
19. Rotate refueling mechanism to orient	0.5
20. Lower 9 in. at 0 - 5 fpm	0.3
Total	396.9 (6.6 hr)
Total time for operation 3 at 80% efficiency	7.9 hr

Table B-3. Pot Arrangement Under-the-Plug Refueling
(Basis for Estimate)

<u>Operation 4 — Refueling Procedure</u>	
<u>Step</u>	<u>Time, min</u>
1. Lower grapple tool 9 in. at 0 - 5 fpm	0.6
2. Deactivate grapple cylinder - closes fingers (latches spent fuel assembly)	0.25
3. Raise assembly 8 ft - 9 in. at 0 - 5 fpm	3.5
4. Rotate shield plugs to position assembly over storage rack access	2.0
5. Rotate refueling mechanism to orient	0.25
6. Lower assembly 9 ft - 9 in. at 0 - 5 fpm	3.9
7. Actuate grapple cylinder - opens fingers	0.25
8. Raise grapple tool 9 in. at 0 - 5 fpm	0.30
9. Rotate fuel storage rotor to position new assembly	0.5
10. Lower grapple tool 9 in. at 0 - 5 fpm	0.30
11. Deactivate grapple cylinder - closes fingers	0.25
12. Raise new assembly 9 ft - 9 in. at 0 - 5 fpm	3.90
13. Rotate shield plugs to position new assembly over empty core position	2.0
14. Rotate refueling mechanism to orient	0.25
15. Lower assembly 8 ft - 9 in. at 0 - 5 fpm	3.50
16. Actuate grapple cylinder - opens fingers	0.25
17. Raise grapple tool 9 in. at 0 - 5 fpm	0.3
18. Rotate shield plugs to position refueling mechanism over next spent assembly to be removed	2.0
Total time per assembly	24.3
Total time per 1/3 core loading (125 assemblies)	3037.5 (50.6 hr)
Total time per 1/3 core loading for operation 4 at 80% efficiency	60.8 hr

Table B-4. Pot Arrangement Hot Cell Refueling
(Basis for Estimate)

Operation 3 — Refueling Preparation After 11-Hour Shutdown

<u>Step</u>	<u>Time, min</u>
1. Disengage drive extension from regulating column	25.0
2. Raise drive extension 3 in. at 1/4 in. /min (25 total)	300.0
3. Unbolt shield plug	150.0
4. Break gas seal	15.0
5. Raise shield plug 8 ft at 1 fpm	8.0
6. Raise shield plug 59 ft at 0 - 8 fpm	14.8
7. Raise shield plug 2 ft - 9 in. at 1 fpm	2.8
8. Actuate mechanical locks and check	5.0
9. Lower shield plug to seat 1 ft at 1 fpm	1.0
10. Make gas seal and pressurize	5.0
11. Move fuel handling machine over one storage rack access plug and check position	2.0
12. Lower guide tube 24 ft at 0 - 60 fpm	0.8
13. Lower guide tube 1 ft at 0 - 5 fpm	0.3
14. Lower pickup tool 8 ft at 0 - 60 fpm	0.8
15. Lower pickup tool 9 in. at 0 - 5 fpm	0.3
16. Rotate pickup tool to lock	0.2
17. Raise plug 9 in. at 0 - 5 fpm	0.3
18. Raise guide tube 18 in. at 0 - 5 fpm	0.6
19. Move handling machine over plug storage position and check correct position	2.0
20. Lower guide tube 6 in. at 0 - 5 fpm	0.2
21. Lower plug 9 in. at 0 - 5 fpm	0.3
22. Rotate pickup tool to unlock	0.2
23. Raise pickup tool 9 in. at 0 - 5 fpm	0.3
24. Raise guide tube 6 in. at 0 - 5 fpm	0.6
25. Move handling machine over second storage rack access plug and check correct position	2.0
26. Lower guide tube 18 in. at 0 - 5 fpm	0.6
27. Repeat steps 15 through 24 to store second plug	5.0
Total	543.1 (9.1 hr)
Total time for operation 3 at 80% efficiency	10.9 hr

Table B-5. Pot Arrangement Hot Cell Refueling
(Basis for Estimate)

Operation 4 — Refueling Procedure

<u>Step</u>	<u>Time, min</u>
1. Move handling machine over spent fuel assembly position and check correct positioning	2.0
2. Lower guide tube 18 in. at 0 - 5 fpm	0.6
3. Lower pickup tool 9 in. at 0 - 5 fpm	0.3
4. Rotate pickup tool to lock	0.2
5. Raise assembly 8 ft - 9 in. at 0 - 5 fpm	3.5
6. Raise guide tube 1 ft at 0 - 5 fpm	0.4
7. Move handling machine over storage cell access port and check correct positioning	2.0
8. Lower guide tube 1 ft at 0 - 5 fpm	0.4
9. Lower assembly 9 ft - 9 in. at 0 - 5 fpm	3.9
10. Rotate pickup tool to unlock	0.2
11. Raise pickup tool 9 in. at 0 - 5 fpm	0.3
12. Rotate storage cell to position new assembly beneath storage cell access port	1.0
13. Lower pickup tool 9 in. at 0 - 5 fpm	0.3
14. Rotate pickup tool to lock	0.2
15. Raise assembly 9 ft - 9 in. at 0 - 5 fpm	3.9
16. Raise guide tube 1 ft at 0 - 5 fpm	0.4
17. Move handling machine over position from which spent assembly was removed and check correct positioning	2.0
18. Lower guide tube 1 ft at 0 - 5 fpm	0.4
19. Lower assembly 8 ft - 9 in. at 0 - 5 fpm	3.5
20. Rotate pickup tool to unlock	0.2
21. Raise pickup tool 9 in. at 0 - 5 fpm	0.3
22. Raise guide tube 1 ft at 0 - 5 fpm	0.4
Total time per assembly	26.4
Total time per 1/3 core loading (125 assemblies)	3300.0 (55 hr)
Total time per 1/3 core loading for operation 4 at 80% efficiency	66 hr

Table B-6. Spread-Out Arrangement Through-the-Plug Refueling
(Basis for Estimate)

Operation 3 — Refueling Preparation After 3-Hour Shutdown

<u>Step</u>	<u>Time, min</u>
1. Disengage drive extensions from regulating column	25
2. Raise drive extension 3 in. at 0.25 in./min (25 total)	300
3. Release locks on rotatable reactor shield plug and raise plug 6 in. to lift upper grid plate clear of core assemblies	30
4. Rotate reactor plug to check drive operation and position access port over first spent assembly	30
Total	385 (6.4 hr)
Total time for operation 3 at 80% efficiency	7.7 hr

Table B-7. Spread-Out Arrangement Through-the-Plug Refueling
(Basis for Estimate)

Operation 4 — Refueling Procedure

The fuel handling machine (FHM) and storage cell have been prepared for the refueling operation during the reactor on-line period or during the reactor shutdown period. During this same period, the first new fuel assembly has been loaded into the fuel handling machine.

<u>Step</u>	<u>Time, min</u>
1. Move FHM from its position over fuel storage cell to reactor access port 31 ft - 6 in. at 0 - 150 fpm	0.4
2. Actuate FHM to reactor plug seal and shielding mechanism, purge interspace	5.0
3. Rotate access port to unlock	1.0
4. Raise plug into FHM 14 ft - 0 in. at 0 - 60 fpm	0.5
5. Rotate FHM indexing table to position handling mechanism over access port	1.0
6. Lower telescoping tube 26 ft - 5 in. at 0 - 60 fpm	0.9
7. Lower grapple tool 9 ft - 0 in. at 0 - 60 fpm	0.3
8. Lower telescoping tube 0 ft - 6 in. at 0 - 5 fpm to open closure valve	0.2
9. Lower grapple tool 0 ft - 6 in. at 0 - 5 fpm to open gripper finger	0.2
10. Lower telescoping tube 0 ft - 8 in. at 0 - 5 fpm	0.3
11. Raise grapple tool 0 ft - 4 in. at 0 - 5 fpm to close gripper finger	0.2
12. Lower telescoping tube 0 ft - 5 in. at 0 - 5 fpm to seat it against adjacent assemblies	0.2
13. Raise assembly 8 ft - 2 in. at 0 - 5 fpm	0.3
14. Raise telescoping tube 1 ft - 7 in. at 0 - 5 fpm	0.6
15. Raise grapple tool with assembly 1 ft - 0 in. at 0 - 5 fpm to close valve	0.4
16. Start cooling gas	1.0
17. Raise telescoping tube 26 ft - 5 in. at 0 - 60 fpm	0.9
18. Rotate FHM indexing table to position new assembly for insertion into core	1.0
19. Lower telescoping tube 28 ft - 0 in. at 0 - 60 fpm	0.9

Table B-7. (Cont'd)

<u>Step</u>	<u>Time, min</u>
20. Lower grapple tool with assembly 9 ft - 2 in. at 0 - 5 fpm	3.6
21. Raise telescoping tube 0 ft - 5 in. at 0 - 5 fpm	0.2
22. Lower grapple tool 0 ft - 4 in. at 0 - 5 fpm to open gripper fingers	0.2
23. Raise telescoping tube 0 ft - 8 in. at 0 - 5 fpm	0.3
24. Raise grapple tool 0 ft - 6 in. at 0 - 5 fpm	0.2
25. Raise telescoping tube 26 ft - 11 in. at 0 - 60 fpm	0.9
26. Rotate FHM indexing table to position access port plug over port	1.0
27. Lower plug 14 ft - 0 in. at 0 - 60 fpm	0.5
28. Rotate plug to lock	1.0
29. Purge interspace and check for leakage	4.0
30. Actuate FHM to reactor plug seal and shielding mechanism	1.0
31. Move FHM to storage cell and position over access port	0.4
32. Repeat steps 2 through 5	7.5
33. Lower telescoping tube 26 ft - 11 in. at 0 - 60 fpm	0.9
34. Lower grapple tool 0 ft - 6 in. at 0 - 5 fpm	0.2
35. Lower telescoping tube 0 ft - 8 in. at 0 - 5 fpm	0.3
36. Raise grapple tool 0 ft - 4 in. at 0 - 5 fpm	0.2
37. Lower telescoping tube 0 ft - 5 in. at 0 - 5 fpm	0.2
38. Raise grapple tool with assembly 9 ft - 2 in. at 0 - 5 fpm	3.6
39. Raise telescoping tube 28 ft - 0 in. at 0 - 60 fpm	0.9
40. Rotate FHM indexing table to position spent assembly handling mechanism over access port	1.0
41. Reverse steps 6 through 17	5.5
42. Repeat 26 through 30	7.5
Total time per assembly	56.4
Total time per 1/3 core loading (125 assemblies)	7050.0 (117.5 hr)
Total time per 1/3 core loading for operation 4 at 80% efficiency	141.0 hr

Table B-8. Spread-Out Arrangement Hot Cell Refueling
(Basis for Estimate)

<u>Operation 3 — Refueling Preparation After 11-Hour Shutdown</u>	
<u>Step</u>	<u>Time, min</u>
1. Disengage drive extensions from regulating column	25.0
2. Raise drive extension 3 in. at 1/4 in. /min (25 total)	300.0
3. Lower reactor coolant level 4 feet	30.0
4. Unbolt shield plug	150.0
5. Break gas seal	15.0
6. Raise shield plug 6 ft at 1 fpm	8.0
7. Raise shield plug 55 ft at 0 - 8 fpm	14.0
8. Raise shield plug 2 ft at 1 fpm	2.0
9. Actuate mechanical locks and check	5.0
10. Lower shield plug 1 ft at 1 fpm	1.0
11. Make gas seal and pressurize	5.0
12. Actuate handling trolley (aux hoist over 60-in. plug) avg 30 ft at 0 - 150 fpm	0.4
13. Lower plug adapter 18 ft at 0 - 60 fpm	0.6
14. Make plug-adapter connection	2.0
15. Lift plug 7 ft at 2 fpm	3.5
16. Move to storage hole 25 ft at 0 - 150 fpm	0.4
17. Lower into storage hole 7 ft at 2 fpm	3.5
18. Raise plug adapter 18 ft at 0 - 60 fpm	0.6
19. Move handling trolley (handling machine) over spent assembly avg 52 ft at 0 - 150 fpm	0.7
Total	566.7 (9.4 hr)
Total time for operation 3 at 80% efficiency	11.3 hr

Table B-9. Spread-Out Arrangement Hot Cell Refueling
(Basis for Estimate)

Operation 4 — Refueling Procedure

At the start of this operation, the handling machine is assumed to be in position over a fuel assembly in the reactor and is ready to be activated.

<u>Step</u>	<u>Time, min</u>
1. Lower grapple tool 35 ft at 0 - 60 fpm (telescopes tubes)	1.17
2. Lower grapple tool 2.5 ft at 0 - 5 fpm	1.00
3. Actuate grapple cylinder - opens fingers	0.25
4. Lower grapple tool 6 in. at 0 - 5 fpm (complete extension)	0.20
5. Deactivate grapple cylinder - closes fingers	0.25
6. Raise fuel assembly 12.5 ft at 0 - 5 fpm	5.00
7. Raise fuel assembly 25.5 ft at 0 - 60 fpm (retraction)	0.85
8. Move handling machine to storage cell 32 ft at 0 - 150 fpm	0.43
9. Actuate storage rotor (motion concurrent with steps 1 through 8)	--
10. Lower fuel assembly into cell 25.5 ft at 0 - 60 fpm	0.85
11. Lower fuel assembly 12 ft at 0 - 5 fpm	4.80
12. Lower grapple tool 6 in. at 0 - 5 fpm (complete extension)	0.20
13. Actuate grapple cylinder - opens fingers	0.25
14. Raise grapple tool 6 in. at 0 - 5 fpm	0.20
15. Deactivate grapple cylinder - closes fingers	0.25
16. Raise grapple tool 12 ft at 0 - 60 fpm	0.40
17. Raise grapple tool 25.5 ft at 0 - 60 fpm (complete retraction)	0.85
18. Actuate rotor to position new assembly (avg time 1 min)	1.00
19. Actuate handling trolley 4 ft at 0 - 150 fpm	0.05
20. Repeat steps 1 through 7 (complete retraction)	8.72
21. Move handling machine to reactor avg 32 ft at 0 - 150 gpm	0.43
22. Lower fuel assembly into reactor 25.5 ft at 0 - 60 fpm	0.85
23. Repeat steps 11 through 17 (complete retraction)	6.95
24. Move handling machine to next spent assembly avg 8 ft at 0 - 150 fpm	0.12
Total time per assembly	35.07
Total time per 1/3 core loading (125 assemblies)	4383.8 (73.1 hr)
Total time per 1/3 core loading for operation 4 at 80% efficiency	87.7 hr

APPENDIX C
Differential Capital Costs

Basis for Pot and Spread-Out Arrangement Capital Cost Estimates

Different refueling methods imposed on the same basic reactor arrangement cause variations in plant equipment which affect the capital cost of the plant. Under subcontract to Babcock & Wilcox, Sargent & Lundy Engineers prepared cost estimates for a pot arrangement with both under-the-plug and hot cell refueling, and for a spread-out arrangement with both through-the-plug and hot cell refueling. S&L provided estimates for equipment outside the reactor which is normally supplied by an A-E. B&W estimated other pertinent equipment costs.

The majority of the estimates are based on previous nuclear plant experience with similar components or materials of construction. Where no such precedent existed, attempts were made to arrive at realistic or slightly pessimistic estimates. Accuracy in the absolute costs was strived for to the extent possible. However, because of the nature of the design effort involved in this trade-off and the preliminary nature of the effort, some of the absolute costs may be questioned. Nevertheless, the accuracy in the differential costs between the two methods of refueling is considered quite good.

Labor costs are based on a 40-hour week with no premium paid for overtime or transportation. The construction costs are the estimated total direct construction costs and include all costs related to the purchasing, manufacturing, shipping, and erection of the material. These costs do not contain such items as engineering services, insurance, interest, administrative and general overhead, taxes, personnel training, miscellaneous construction, or contingency on plant material, equipment, or labor.

The construction site used is the Atomic Energy Commission's hypothetical site in western Massachusetts as specified in the AEC Guide to Nuclear Power Cost Evaluation.

Since the same basic pot and spread-out reactor concepts are utilized with both types of refueling, items of equipment or systems that were identical in, or exhibited no apparent differences between, the four plants were not estimated. Equipment and systems falling in this category are as follows:

1. The reactor building heating and ventilation system.
2. The reactor building lighting and electrical system.
3. The reactor building fire protection system.
4. The closed-loop argon system.
5. The emergency shutdown cooling system.
6. The primary coolant and secondary coolant purification and storage system.
7. The nuclear fuel handling and storage system external to the reactor building.
8. Combined IHXs and pumps (some extra cost is involved for concentric piping and for having the pumps in the hot leg for the spread-out concepts, but no estimates were prepared).
9. Control rods.
10. Instrumentation.
11. Reactor internals—lower grid plate, meltdown pan, and top holddown structure.

The pertinent cost estimates are shown in Tables C-1 and C-2 for the pot arrangement with under-the-plug refueling versus the pot arrangement with hot cell refueling, and in Tables C-3 and C-4 for the spread-out arrangement with through-the-plug refueling versus the spread-out arrangement with hot cell refueling. The costs are summarized in six main categories: reactor building, reactor vessel cover structure, reactor vessel, hot cell, fuel handling system, and fuel storage.

Table C-1. Pot Arrangement Differential Capital Costs — Summary

Category	Capital costs, \$ × 10 ³		
	Under-the-plug	Hot cell	Differential
A. Reactor building	5,093.0	5,467.8	374.8
B. Reactor vessel cover structure	1,163.0	1,015.0	148.0 ^(a)
C. Reactor vessel	7,077.6	7,947.6	870.0
D. Hot cell	--	1,687.7	1,687.7
E. Fuel handling system	835.0	725.0	110.0 ^(a)
F. Fuel storage	198.0	198.0	--
Total	14,366.6	17,041.1	2,674.5

(a) Areas where under-the-plug refueling costs more than hot cell refueling.

Table C-2. Pot Arrangement Differential Capital Costs — Details

		Capital costs, \$ × 10 ³		
		Under-the-plug	Hot cell	Differential
<u>A. Reactor Building</u>				
1.	Excavation, backfill and disposal	410.0	410.0	--
2.	Concrete	853.9	916.7	62.8
3.	Reinforcing steel	790.5	877.5	87.0
4.	Form work	1,055.1	1,098.1	43.0
5.	Exterior waterproofing (below grade)	9.0	9.0	--
6.	Structural steel	116.0	151.0	35.0
7.	Steel liner—reactor building	1,420.0	1,700.0	280.0
8.	Steel liner—reactor vessel pit	70.5	94.5	24.0
9.	Miscellaneous steel (stairs, ladders, rails, etc.)	25.6	29.0	3.4
10.	Exterior painting	11.0	14.0	3.0
11.	Polar crane—100 tons	--	168.0	168.0
12.	Polar crane—200 tons	218.0	--	218.0 ^(a)
13.	Refueling equipment and control rod extension storage and maintenance pit	113.4	--	113.4 ^(a)
Total		5,093.0	5,467.8	374.8
<u>B. Reactor Vessel Cover Structure</u>				
1.	Serpentine concrete	142.0	206.0	64.0
2.	Plugs—rotatable	500.0	--	500.0 ^(a)
3.	Plugs—fixed	446.0	646.0	200.0
4.	Plugs—removable	--	100.0	100.0
5.	Plugs—fuel transfer access	60.0	45.0	15.0 ^(a)
6.	Insulation	15.0	18.0	3.0
Total		1,163.0	1,015.0	148.0 ^(a)
<u>C. Reactor Vessel</u>				
1.	Primary tank	1,892.0	2,420.0	528.0
2.	Insulation	56.0	83.0	27.0
3.	Primary sodium inventory	630.0	945.0	315.0
4.	Core vessel and skirts	645.2	645.2	--
5.	Internal thermal shields	3,445.0	3,445.0	--
6.	Core vessel support ring	326.4	326.4	--
7.	Core vessel tension support members	83.0	83.0	--
Total		7,077.6	7,947.6	870.0

(a) Areas where under-the-plug refueling costs more than hot cell refueling.

Table C-2. (Cont'd)

		Capital costs, \$ × 10 ³		
		<u>Under-the-plug</u>	<u>Hot cell</u>	<u>Differential</u>
<u>D. Hot Cell</u>				
1.	Concrete	--	77.0	77.0
2.	Reinforcing steel	--	29.0	29.0
3.	Form work	--	93.0	93.0
4.	Steel liner	--	832.0	832.0
5.	Cover plug hoist—250 tons	--	94.0	94.0
6.	Jib crane—2 tons	--	5.0	5.0
7.	Insulation	--	79.7	79.7
8.	Heating system	--	50.0	50.0
9.	Television monitoring system	--	60.0	60.0
10.	Leaded glas windows	--	48.0	48.0
11.	Remotely operated tools and equipment	--	300.0	300.0
12.	Maintenance plug	--	20.0	20.0
	Total	--	1,687.7	1,687.7
<u>E. Fuel Handling System</u>				
1.	Cask type fuel transfer machine	535.0	--	535.0 ^(a)
2.	Plug mounted refueling mechanism	300.0	--	300.0 ^(a)
3.	Monorail mounted fuel handling machine (including crane)	--	375.0	375.0
4.	Jib crane mounted fuel transfer machine (including crane)	--	350.0	350.0
	Total	835.0	725.0	110.0 ^(a)
<u>F. Fuel Storage</u>				
1.	Rotating storage drums and drives	198.0	198.0	--
	Total	198.0	198.0	--
	Grand Total	14,366.6	17,041.1	2,674.5

(a) Areas where under-the-plug refueling costs more than hot cell refueling.

Table C-3. Spread-Out Arrangement Differential
Capital Costs — Summary

Category	Capital costs, \$ × 10 ³		
	Through-the-plug	Hot cell	Differential
A. Reactor building	6,692.1	6,447.4	244.7 ^(a)
B. Reactor vessel cover structure	755.9	383.8	372.1 ^(a)
C. Reactor vessel	5,332.5	3,227.2	2,105.3 ^(a)
D. Hot cell	--	2,545.1	2,545.1
E. Fuel handling system	535.0	810.0	275.0
F. Fuel storage	1,267.2	1,267.2	--
Total	14,582.7	14,680.7	98.0

^(a) Areas where through-the-plug refueling costs more than hot cell refueling.

Table C-4. Spread-Out Arrangement Differential
Capital Costs — Details

		Capital costs, \$ × 10 ³		
		Through-the-plug	Hot cell	Differential
<u>A. Reactor Building</u>				
1.	Excavation, backfill, and disposal	609.0	609.0	--
2.	Concrete	992.5	1,010.5	18.0
3.	Reinforcing steel	1,101.0	1,115.5	14.5
4.	Form work	912.9	960.9	48.0
5.	Exterior waterproofing (below grade)	11.0	11.0	--
6.	Structural steel	150.0	165.0	15.0
7.	Steel liner—reactor building	2,050.0	2,050.0	--
8.	Steel liner—reactor vessel pit	37.4	22.6	14.8 ^(a)
9.	Steel liner—IHX pits	197.0	197.0	--
10.	Steel liner—fuel storage cell pit	136.0	136.0	--
11.	Miscellaneous steel (stairs, ladders, rails, etc.)	30.9	38.9	8.0
12.	Exterior painting	16.0	16.0	--
13.	Polar crane—250 tons	325.0	--	325.0 ^(a)
14.	Bridge crane—75 tons	--	115.0	115.0
15.	Refueling equipment and control rod extension storage and maintenance pit	123.4	--	123.4 ^(a)
	Total	6,692.1	6,447.4	244.7 ^(a)
<u>B. Reactor Vessel Cover Structure</u>				
1.	Serpentine concrete	56.9	41.8	15.1 ^(a)
2.	Plugs—rotatable	495.0	--	495.0 ^(a)
3.	Plugs—fixed	174.0	104.0	70.0 ^(a)
4.	Plugs—removable	--	223.0	223.0
5.	Plugs—fuel transfer access	15.0	--	15.0 ^(a)
6.	Insulation	15.0	15.0	--
	Total	755.9	383.8	372.1 ^(a)
<u>C. Reactor Vessel</u>				
1.	Primary tank	1,093.2	601.2	492.0 ^(a)
2.	Flow dividers	373.4	198.2	175.2 ^(a)
3.	Thermal shields	2,474.9	1,237.6	1,237.3 ^(a)
4.	Insulation—reactor vessel	42.0	16.4	25.6 ^(a)
5.	Insulation—IHXs	55.0	55.0	--
6.	Primary sodium inventory	364.0	188.8	175.2 ^(a)
7.	IHX outer shell (6)	930.0	930.0	--
	Total	5,332.5	3,227.2	2,105.3 ^(a)

^(a) Areas where through-the-plug refueling costs more than hot cell refueling.

Table C-4. (Cont'd)

		Capital costs, \$ × 10 ³		
		Through-the-plug	Hot cell	Differential
<u>D. Hot Cell</u>				
1.	Concrete	--	74.7	74.7
2.	Reinforcing steel	--	28.0	28.0
3.	Form work	--	77.0	77.0
4.	Steel liner	--	1,608.8	1,608.8
5.	Cover plug hoist—250 tons	--	94.0	94.0
6.	Jib crane—2 tons	--	5.0	5.0
7.	Insulation	--	141.7	141.7
8.	Heating system	--	87.9	87.9
9.	Television monitoring system	--	60.0	60.0
10.	Leaded glass windows	--	48.0	48.0
11.	Remotely operated tools and equipment	--	300.0	300.0
12.	Maintenance plug	--	20.0	20.0
	Total	--	2,545.1	2,545.1
<u>E. Fuel Handling System</u>				
1.	Fuel handling machine (shielded)	535.0	--	535.0 ^(a)
2.	Fuel handling machine (nonshielded) including crane	--	435.0	435.0
3.	Fuel transfer machine including crane	--	375.0	375.0
	Total	535.0	810.0	275.0
<u>F. Fuel Storage</u>				
1.	Vessel	306.0	306.0	--
2.	Flow dividers	130.0	130.0	--
3.	Insulation (vessel and cover)	21.2	21.2	--
4.	Cover plugs	144.0	144.0	--
5.	Internal racks and drives	103.0	103.0	--
6.	Sodium inventory	63.0	63.0	--
7.	Sodium circulating system	500.0	500.0	--
	Total	1,267.2	1,267.2	--
	Grand Total	14,582.7	14,680.7	98.0

(a) Areas where through-the-plug refueling costs more than hot cell refueling.

REFERENCES

- ¹ LMFBR Program Plan, Section 3 - Components, Argonne National Laboratory.
- ² 1000-MWe LMFBR Follow-On Study, Task I Report, Babcock & Wilcox, BAW-1316, Vol 1, Lynchburg, Virginia, June 1967.
- ³ 1000-MWe LMFBR Follow-On Study, Task I, Concept II System Description Report, Babcock & Wilcox, BAW-1316, Vol 3, Lynchburg, Virginia, June 1967.
- ⁴ Masseth, F. X., Shipping Cask Trade-Off Study, Babcock & Wilcox, TRG-68-4, Lynchburg, Virginia, February 1968 (also part of BAW-1328, Volume 4).