Pool-Type LMFBR Plant
1000 MWe Phase A-Extension-1 Design

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ELECTRIC POWER RESEARCH INSTITUTE

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1000 MWe Phase A-Extension-1 Design

PART I: EXECUTIVE SUMMARY

NP-882-SY, Volume 1
Research Project 620-20, 21

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EPRI PERSPECTIVE

PROJECT DESCRIPTION

In 1975 EPRI and the Energy Research and Development Administration (ERDA) initiated the prototype large breeder reactor (PLBR) plant design program with the objective of providing industry with a basis for submitting firm proposals in 1978 for the design and construction of a breeder plant by the late 1980s. This program supported independent design studies by three contractor teams, including the General Electric/Bechtel team.

After the Phase I study of major design options was completed, a 1000-MWe loop-type plant was specified by the sponsors, and Phase II concept studies continued into the second quarter of 1977. ERDA then withdrew as a cosponsor and, after sponsoring a brief Phase II extension study of selected technical areas by the individual contractors, ended its involvement in the PLBR program. This action was taken because of uncertainties in the new administration's nuclear power policies and because of the impending integration of ERDA into the new Department of Energy (DOE).

At this time EPRI decided it was appropriate to reassess the pool concept in the light of altered considerations and growing worldwide experience with the concept. EPRI proceeded with design studies of a 1000-MWe pool-type saturated-steam-cycle LMFBR, with primary emphasis on the areas of the plant unique to the pool concept. The work completed in these studies is presented in the following reports:

> EPRI NP-645, Engineering Aspects of the Pool-Type LMFBR, 1000 MWe, Atomics International, Combustion Engineering, Bechtel, April 1978
>
> EPRI NP-646, Pool-Type LMFBR Plant--1000 MWe Phase A Design, General Electric, Bechtel, April 1978
>
> Westinghouse Large Pool Reactor, Westinghouse, January 1978

These studies indicated that the pool concept is feasible in the context of U.S. design, construction, and licensing practice.

PROJECT OBJECTIVES

The EPRI-funded effort was continued for six months in order to verify further the feasibility of the major technical features unique to the pool concept. This report describes the results obtained by the General Electric/Bechtel team during this additional phase of work. Similar efforts by the Atomics International/ Combustion Engineering/Bechtel team and the Westinghouse/Stone & Webster team are reported in EPRI NP-881 and NP-883.

CONCLUSIONS AND RECOMMENDATIONS

While the results of the work reported here continue to support the feasibility of the pool concept, it is evident that further work is needed to complete reasonable design solutions for the major features unique to the pool concept that might significantly impact design, construction, maintainability, safety, and cost. These additional EPRI-funded studies, now under way, should complete the feasibility analyses of the pool-type LMFBR and serve as the planning foundation for future LMFBR plant design efforts.

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ABSTRACT

A 900 MWe pool-type plant has been developed, to approximately the same level as current U.S. loop-type LMFBR plant designs, in a conceptual design study sponsored by EPRI. Concentrating on areas of the plant unique or especially important to the pool concept, this report covers the reactor deck; the design of internal structures, seismic response and construction of the reactor assembly; the thermal behavior of the plant during part-power operation and during transients; the auxiliary heat transport systems; the instrumentation and control systems; and the Balance of Plant. The main conclusion arising from this work is that the large pool-type LMFBR is a viable concept suitable for use in the United States.

PREFACE

This report describes Phase A Extension 1 work performed between February 8 and July 31, 1978 on the design of a large pool-type LMFBR power plant. The work is the result of a team effort by Bechtel Corporation and General Electric Company which was sponsored and guided by the Electric Power Research Institute (EPRI). The objective of the work was to refine certain areas of design and bring them into better focus than had been provided by the Phase A work performed between April 4 and December 30, 1977.

The Phase A effort produced an initial description of the overall plant, structures and systems. During Phase A, General Electric developed a conceptual design of the overall nuclear steam supply system (NSSS). It defined specific design approaches for selected NSSS components and subsystems after analyzing various design alternatives. Bechtel assumed responsibility for the intermediate sodium piping arrangement , the access area above the reactor deck and the Balance of Plant (BOP). The resulting integrated plant design provided the necessary seismic data for both the NSSS and the BOP.

The special expertise of several subcontractors was used during Phase A; Byron-Jackson provided a preliminary design of the primary sodium pump, Foster-Wheeler provided a preliminary design of the intermediate heat exchanger (IHX), and CBI Nuclear reviewed the reactor deck design and developed a construction sequence for the overall reactor assembly.

The Phase A effort by General Electric, Bechtel and the subcontractors was funded at a level of nearly 1.7 million dollars. Additionaly, General Electric contributed a company-funded effort and both General Electric and Bechtel utilized their backgrounds of prior work on pool-type LMFBRs and extensive interaction with foreign LMFBR organizations. The results of the Phase A work was published by EPRI in April 1978 in report number NP-646, "Pool-Type LMFBR Plant, 1000 MWe Phase A Design".

During Phase A Extension 1, funded at a level of approximately 1.4 million dollars, specific areas established during Phase A received further development and evaluation. These specific areas included the reactor deck, the reactor assembly, the heat transfer system components, the reactor auxiliary systems, and the instrumentation and control systems. Several subcontractors were also used during Phase A Extension 1; Foster-Wheeler designed an alternate IHX, CBI Nuclear evaluated an alternate deck support scheme and further developed the reactor assembly construction sequence, and United Nuclear Industries provided conceptual designs for removable radiation shielding in the deck.

The clear distinction by the reader between "Phase A", and "Phase A Extension 1" is necessary to his comprehension of this report. To enhance this distinction "Phase A Extension 1" is frequently shortened to "Extension 1".

This report of the Phase A Extension 1 work is logically divided into eight parts, which have the general title "Pool-Type LMFBR, 1000 MWe Phase A — Extension 1 Design":

Part I	Executive Summary
Part II	Reactor Assembly — Structures
Part III	Reactor Assembly — Deck
Part IV	Reactor Assembly — Fabrication
Part V	Heat Transport System Components
Part VI	Reactor Auxiliary Systems
Part VII	Instrumentation and Control
Part VIII	Balance Of Plant

The report is physically divided into six volumes as follows:

Volume 1	Part I
Volume 2	Part II
Volume 3	Part III and Part IV
Volume 4	Part V
Volume 5	Part VI and Part VII
Volume 6	Part VIII

A Table of Contents for all volumes is included at the end of every volume.

VOLUME 1

CONTENTS

PART I: EXECUTIVE SUMMARY

A Table of Contents for the entire report appears at the end of each volume.

FIGURES

TABLES

Section 1

INTRODUCTION

This report records the second stage results of a large pool-type LMFBR plant conceptual design study. The first stage was Phase A. This second stage has been designated Phase A Extension 1. To assist in distinguishing between "Phase A" and "Phase A Extension 1", this report will use "Extension 1" to mean "Phase A Extension 1". The primary objective of Extension 1 was to perform studies and evaluations of specific areas in the reference Pool Concept established in Phase A. These five areas are the Reactor Deck, the Reactor Assembly, Heat Transfer System Components, Reactor Auxiliary Systems, and Control and Instrumentation.

The principal objective of Phase A was to develop confidence that the features unique to a pool-type LMFBR could be accommodated in an overall plant design. Features of special interest included the large-diameter deck, vessel and core support (especially their seismic response and deflection under static loads), the thermal barriers which separate the hot and the cold sodium pools, the long-shaft primary sodium pumps, and the low-pressure-drop IHXs. The work performed during Phase A showed the large pool-type LMFBR to be a feasible and promising option for use in the United States. Major features of the design are illustrated in Figures 1-1, 1-2, and 1-3. The key operating parameters are shown in Figure 1-4.

The effort in Extension 1 has involved work areas in four categories: (1) where the work area was not addressed in Phase A; (2) where potential problems were identified during Phase A; (3) where further detail was needed to bring the plant to a more uniform design level; and (4) other areas where general plant improvements were desirable and possible. This last category includes an attempt to reduce the size and complexity of specific structures with the intent of reducing costs, to reduce construction and maintenance schedules, and to develope alternate promising designs to ensure that the best possible concepts are pursued in future work.

1. PRIMARY SODIUM PUMP
2. PRIMARY SODIUM IHX
3. INTERMEDIATE SODIUM PUMP
4. STEAM GENERATOR
5. REACTOR DECK
6. REACTOR POOL W/GUARD VESSEL
7. REACTOR CENTER ISLAND
8. EX-VESSEL STORAGE TANK
9. FUEL TRANSFER CELL
10. COVER GAS CONSTANT PRESSURE TANK CELL
11. MEZZANINE DECK
12. *REACTOR CONTAINMENT*
13. REACTOR OPERATING DECK
14. 450 TON POLAR CRANE
15. REACTOR CONFINEMENT BUILDING
16. IRACS & PRACS STACK
17. FUEL HANDLING CELL
18. REACTOR SERVICE BUILDING
19. SODIUM & GAS PIPEWAY
20. ACCESS CORRIDOR
21. SODIUM WATER REACTION PRODUCTS TANK
22. 25 TON BRIDGE CRANE
23. STEAM GENERATOR BUILDING
24. RAILROAD
25. RAILROAD
26. AUXILIARY BUILDING
27. TURBINE BUILDING
28. FUEL TRANSFER TUBE
29. INTERMEDIATE SODIUM PIPING

POOL - TYPE LMFBR PLANT

NET POWER OUTPUT	923 MWe
NET PLANT EFFICIENCY	31.7%
NUMBER OF PRIMARY PUMPS	4
NUMBER OF IHTS LOOPS	6
IHX'S PER LOOP	1
STEAM GENERATOR PER LOOP	1
REACTOR OUTLET TEMP.	875°
REACTOR INLET TEMP.	595°
TURBINE STEAM CONDITIONS	1000 psia saturated
TURBINE GENERATOR	1800 rpm tandem compound 4 flow ; LWR turbine
DECAY HEAT REMOVAL	2 independent diverse systems PRACS & IRACS both to atmosphere

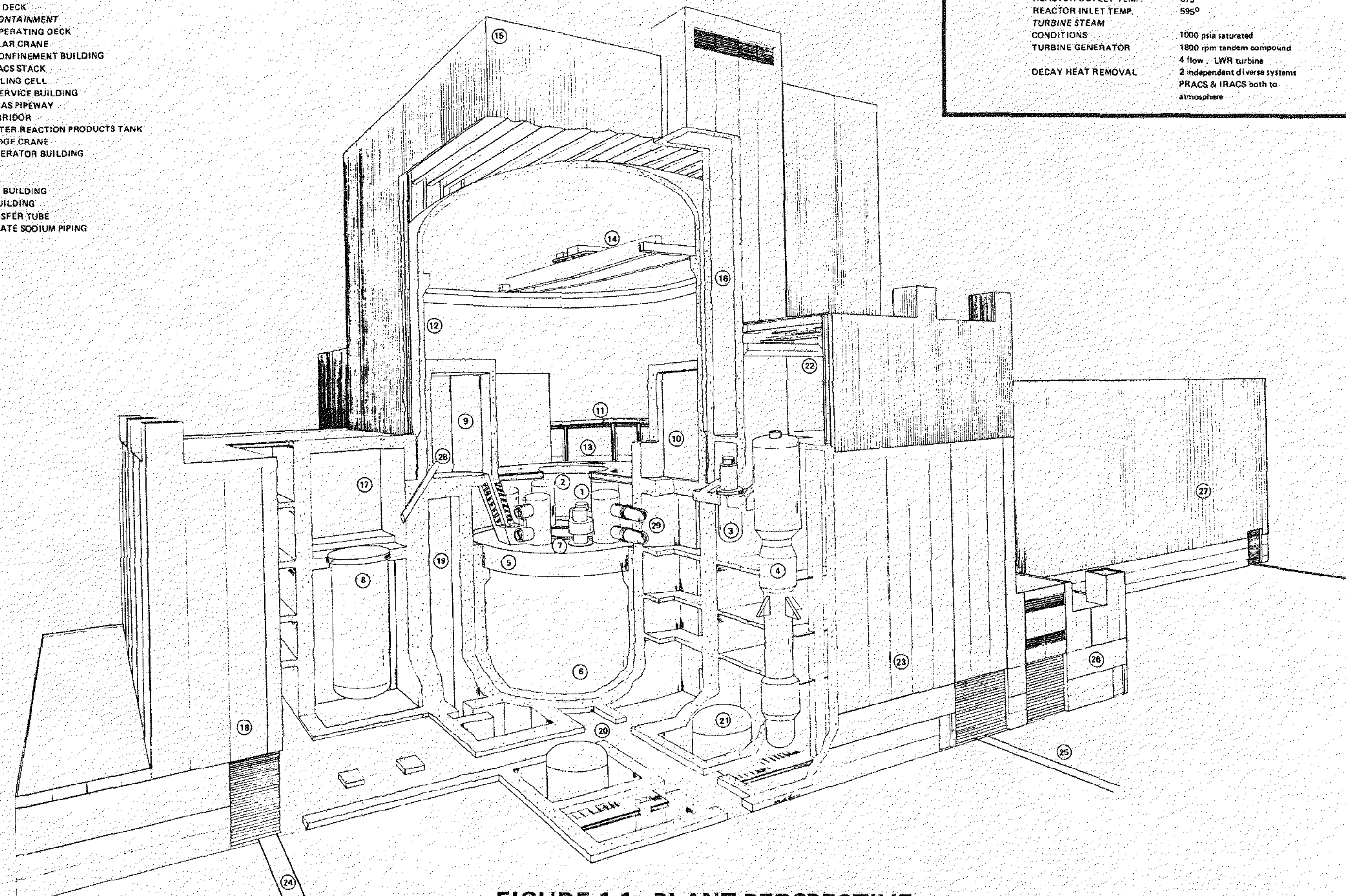
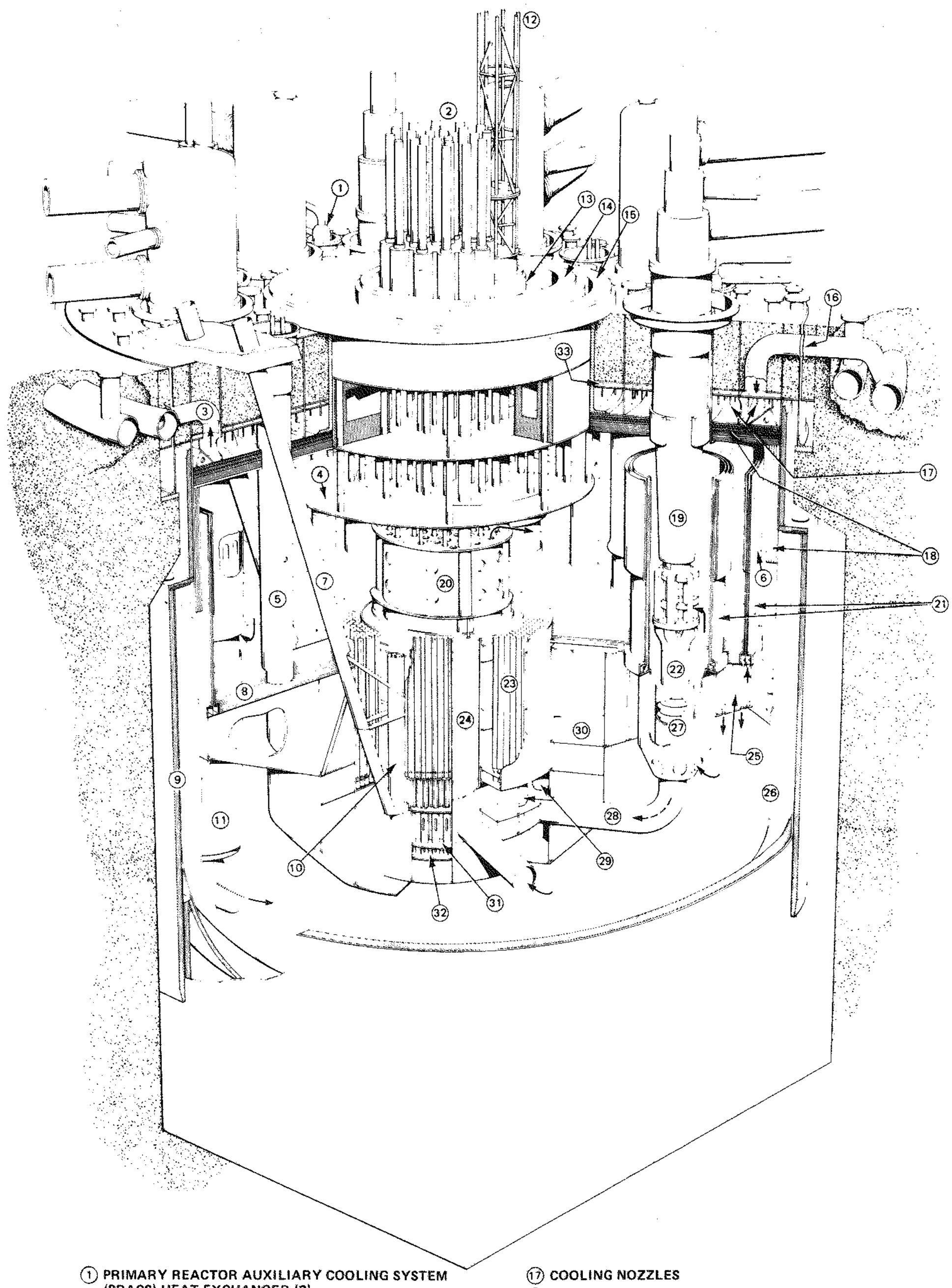


FIGURE 1-1. PLANT PERSPECTIVE



1. PRIMARY REACTOR AUXILIARY COOLING SYSTEM (PRACS) HEAT EXCHANGER (3)
2. CONTROL ROD DRIVES
3. DECK COOLING & ISI ACCESS DUCT
4. HOT POOL SURFACE
5. COLD TRAP (2)
6. COLD POOL SURFACE
7. FUEL TRANSFER TUBE (2)
8. HORIZONTAL INSULATION
9. INSULATED GUARD VESSEL
10. CORE COMPONENT POT (2)
11. INTERMEDIATE HEAT EXCHANGER (6)
12. IN-VESSEL TRANSFER MACHINE (IVTM)
13. SMALL ROTATING PLUG (SRP)
14. INTERMEDIATE ROTATING PLUG (IRP)
15. LARGE ROTATING PLUG (LRP)
16. EXPANSION JOINT
17. COOLING NOZZLES
18. INSULATION
19. PRIMARY PUMP (4)
20. UPPER INTERNALS STRUCTURE (UIS)
21. HOT POOL/COLD POOL THERMAL BARRIERS
22. SHUTOFF VALVE
23. LATERAL NEUTRON SHIELDING
24. CORE BARREL ASSEMBLY AND CORE RESTRAINT
25. THERMAL BARRIER SUPPLY PLENUM
26. PRIMARY VESSEL
27. EXPANSION JOINT
28. INLET PIPING (4)
29. AUXILIARY FLOW MODULES (8)
30. CORE SUPPORT PLATFORM
31. HIGH PRESSURE PLENUM
32. INTERMEDIATE PRESSURE PLENUM
33. COOLING PASSAGE

FIGURE 1-2. REACTOR ASSEMBLY PERSPECTIVE

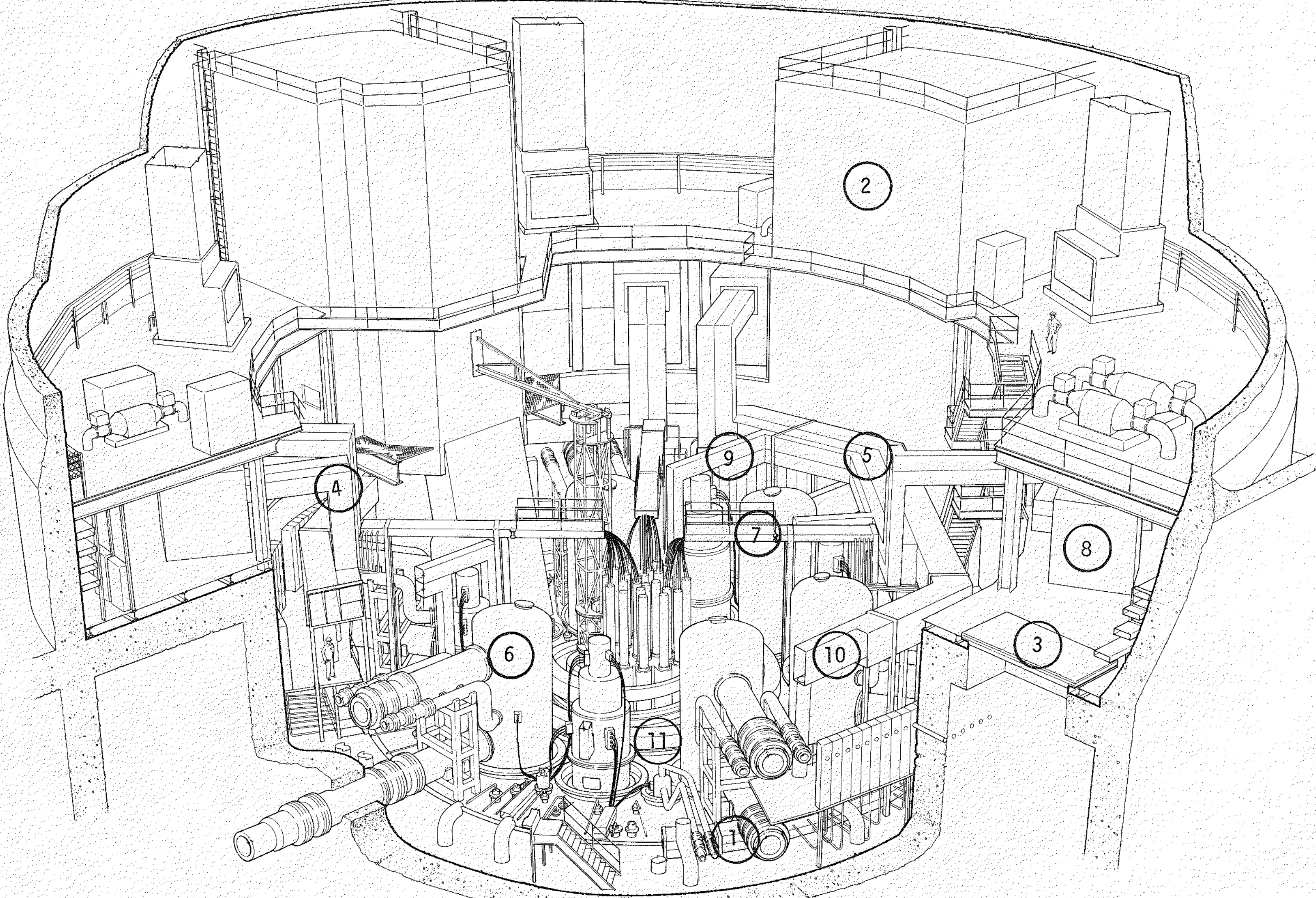


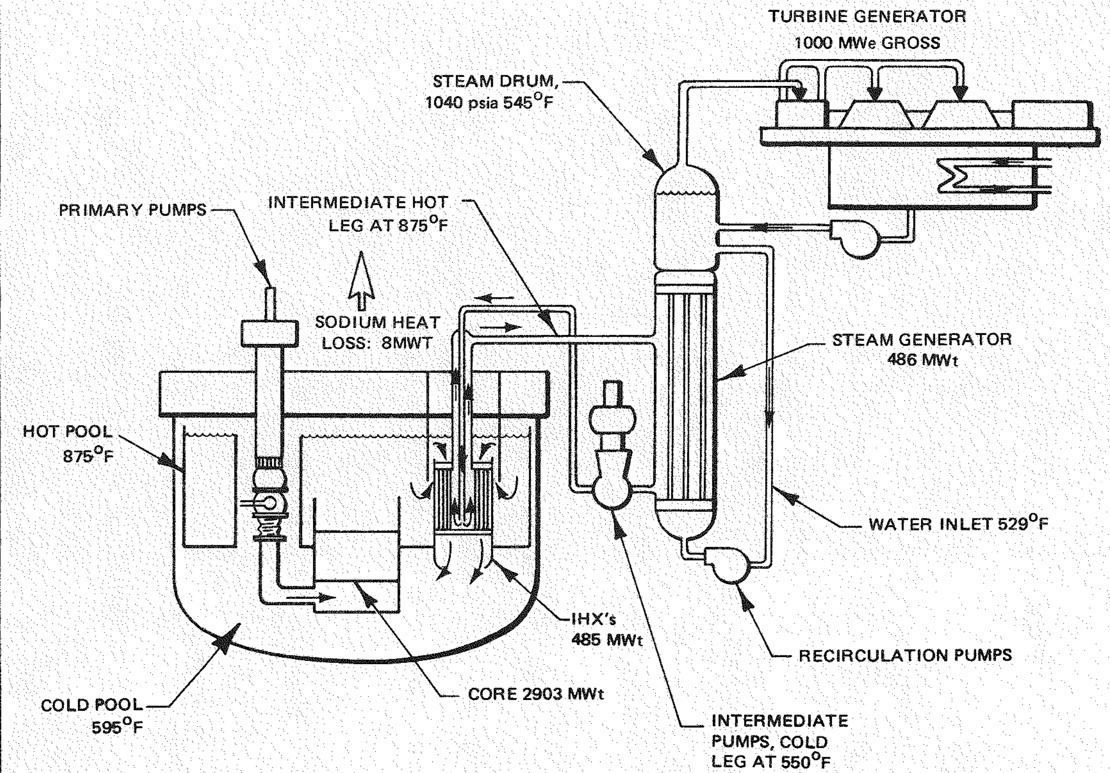
FIGURE 1-3. HEAD COMPARTMENT PERSPECTIVE

Legend

Figure 1-3

During Extension 1 a number of changes in the reactor head compartment layout were accomplished. These changes are identified on the layout, shown as it existed at the end of Phase A, as follows:

1. Many cable trays were re-routed.
2. The two large cells were replaced by one cell and the passive ambient-pressure vessels for cover gas control were replaced by an active pressurized vessel.
3. The equipment hatch was changed from a square shape, 17 feet on a side, to a circular shape 21 feet in diameter.
4. The head compartment cooling ducts were relocated.
5. The head compartment cooling manifold was relocated.
6. The IHX protective enclosure was enlarged.
7. The supports and bridges for the overhead cable trays were re-designed.
8. The primary pump motor cooler was eliminated.
9. The primary pump motor cooler manifold was eliminated.
10. The reactor head gas cooler manifold was relocated.
11. The rotating plug cable trays were relocated.



1000 MWe POOL-TYPE LMFBR

NET PLANT POWER	923 MWe	
NET PLANT EFFICIENCY	31.8%	
PRIMARY PUMPS	4;	COLD POOL SUCTION
IHX's	6;	TUBE-SIDE GRAVITY FLOW FROM HOT POOL
STEAM GENERATORS	6;	STRAIGHT, DOUBLE-WALL TUBES; INTEGRAL STEAM DRUM
TURBINE GENERATOR	1;	STANDARD LWR, 1800 rpm, FOUR-FLOW, TANDEM COMPOUND

FIGURE 1-4. PLANT OPERATING PARAMETERS

Relative to category (1), work areas not addressed in Phase A, the work in Extension 1 has involved the following areas:

- The upper internals structure (UIS).
- The internal neutron shielding.
- In-vessel instrumentation, including the failed fuel element detection and location (FEDAL) system, and the associated cabling and cabinets in the head access area.
- Plant control systems.
- Plant thermal transients and their impact on structures.
- Hot and cold sodium mixing during plant thermal transients.
- Core design.

During Extension 1, some of these areas have been brought to a design level consistent with the majority of the plant. Other areas, such as the understanding of hot and cold sodium mixing in the pool, and the upper internal structure, are long-range in nature and the emphasis has been to assess basic designs. Argonne National Laboratory is performing preliminary core design while General Electric maintains a liaison in order to assess the influence of the core design on the reactor assembly.

Category (2) relates to areas where questions had been raised, or problems were identified, during Phase A. These areas are:

- Capabilities of thermal barriers between the hot and cold sodium pools at off-normal operating conditons.
- Design of reactor deck penetrations for components such as primary pumps and IHXs.
- Reactor cover gas pressure control requirements during plant thermal transients.

The reactor deck penetrations represent an especially complex problem because they are at the interface of many different design areas involving thermal and stress analyses, sodium vapor deposition, radiation shielding, seismic restraint and component removal and installation. Also, the penetration seals and liners form part of the primary system pressure boundary.

Study of further detail, category (3), to bring the integrated plant design to a more nearly uniform level of detail during Extension 1 involved the following areas:

- Reactor deck cooling and insulation.
- Primary pump shutoff valve.
- Seismic analyses.
- Reactor auxiliary cooling system.
- Design of an in-vessel primary cold trap.

Relative to category (4), other areas, the plant improvement area judged most important to investigate was the reactor deck diameter. Reducing the deck diameter would permit a reduction in deck depth which in turn has a host of benefits: better seismic response, shorter IHXs, shorter pump shafts, and a smaller containment building. Toward these goals several different areas were given attention:

- Alternate design for the vessel support.
- Optimization of the triple rotating plugs.
- Smaller IHX diameter.

Other plant improvement areas designated for investigation during Extension 1 were:

- Removable reactor deck shielding to improve in-service inspection.
- Simplified plant control systems.
- A shortened construction sequence for the reactor assembly.
- Backup IHX design.
- Backup pump-to-plenum piping and seals design.
- Backup rotating plug seal designs.

During Phase A it became apparent that several specific items in the BOP area required study in greater depth in order to identify practical means of meeting perceived United States requirements. Those areas selected for further investigation in Extension 1 were:

- Reactor head compartment layout.
- Methods of removal of primary heat transport system components.
- Intermediate heat transport system piping enclosure.
- Reactor assembly support interface.
- Decay heat removal systems — layout and performance.
- Cover gas system.
- Reactor assembly — erection and schedule.

The work performed through Extension 1 is summarized in this volume and covered more fully in the other volumes of this report.

Section 2

RESULTS AND CONCLUSIONS

During Extension 1, General Electric and Bechtel have investigated potential problems identified previously in Phase A, have developed major areas of the integrated plant to consistent levels, and have initiated designs for improvement of many components.

- The large pool-type LMFBR is a viable concept well suited for development into a more detailed design.

- A review of reactor assembly and BOP construction sequences has led to a revision in the recommended construction method. In Phase A, the reactor assembly was to be completed in the reactor containment building by bringing in subassemblies of the reactor structure, which were then welded in place. It is now recommended that the reactor guard vessel, the reactor vessel, and the two halves of the reactor deck be assembled outside, and lifted into the reactor containment building. The necessary lifting equipment is commercially available.

- In Phase A, the design of a straight-tube IHX with a shell bellows was pursued to a point where its feasibility was assured. In Extension 1, the design of a bent-tube IHX with a rigid shell was investigated as an alternate. It has now been concluded that the advantages of a bent tube IHX, although not large, are sufficient for the bent-tube design to be adopted as the reference.

- Preliminary designs of the Upper Internals Structure have been prepared, with particular attention to thermal striping and sodium mixing in the hot pool. Designs using Inconel afford a better means of accommodating thermal striping. Alternate locations of baffles to control the entrance levels of core exit sodium into the hot pool are being evaluated.

- Additional shielding of the core, the IHXs, and the pumps has been added as dictated by more detailed calculations. Otherwise the general reactor assembly and arrangement is essentially that developed in Phase A.

- The deck is supported by a compression skirt projecting downward onto the reactor cavity ledge. The diameter of this skirt has been increased from 78 ft to 80 ft, to allow better access to the region at the top of the reactor vessel, and to facilitate construction by allowing the guard vessel to be installed already assembled, after completion of the vessel support ledge.

- Several alternative methods of supporting the reactor vessel have been investigated with the objective of reducing the deck diameter. The preferred alternate reduces the deck diameter to about 73 ft, but at the expense of increased complexity in the support area.

- The pool reactor has inherent flexibility to operate with major components such as an IHX or a pump out of service. It has been confirmed that this flexibility exists in the reference design, and slow-acting shutoff valves have been provided in both the pump and IHX to allow such operation at the operator's discretion.

- It has been confirmed that the vertical thermal barriers identified in Phase A will permit operation with a pump or IHX out of service.

- To meet the objective that no primary sodium is circulated outside the reactor vessel, it is necessary to have an in-vessel primary cold trap. It has been shown that the design of such a component with an acceptable life (10 years) is feasible.

- Investigation of the constant-pressure, reactor cover gas control system recommended during Phase A showed that its capacity to mitigate pressure transients following a scram was inadequate. The Extension 1 effort proposes that reactor cover gas pressure be controlled by an active feed and bleed system from a pressurized tank. The primary coolant boundary (i.e., reactor tank and deck) can be constructed to withstand the worst pressure transient resulting from post-scram failure of the active system.

- To meet the EPRI guidelines* for Extension 1, the six loops of the Intermediate Reactor Auxiliary Cooling System and the three loops of the Primary Reactor Auxiliary Cooling System might have to have increased heat removal capabilities. The impact of increased ratings on the NSSS and BOP have not been investigated; further study is recommended before a final decision on the capacities of the auxiliary cooling systems is made.

- If complete access to all structural welds in the deck should become a design requirement, it has been shown that removable shielding is possible. The removable shielding would be a combination of pre-formed, boron-magnetite-concrete blocks with an overlayer of boron-lead-polyethylene balls.

- Sufficient space can be allocated within the head access area for the necessary equipment and services with an adequate allowance for maintenance activities. Major components can be removed from the reactor containment building in a single vertical lift, and transferred to the maintenance facility on a special trolley.

- Adequate seismic restraints for the IHX and primary pump can be attached to the deck without the necessity of additional restraint at the core support structure.

- Thermal transients have been studied to provide input to component and reactor structural analysis and no major problems have been identified.

*EPRI guidelines are listed in Appendix IIIB at the end of Part III.

- The saturated cycle pool-type LMFBR has simple control requirements in that precise control of sodium flow and temperatures are not required. This suggests the option of a yet less complex base load plant with fixed two-speed pumps. This option looks attractive, but further work is needed to confirm that the slight loss of operating flexibility is acceptable.

- Instrumentation appears to be generally state of the art, except for failed fuel element detection and location. Several alternatives are under investigation; a reference design has not been selected.

Section 3

NUCLEAR STEAM SUPPLY SYSTEM

REACTOR ASSEMBLY

The reactor and deck assembly proposed by General Electric for the 1000 MWe pool-type LMFBR has been integrated with the reactor vessel and other design areas. The modification to equipment and structures made during Extension 1 are compatible with the arrangement and primary vessel size of Phase A. Figure 3-1 illustrates the general arrangement, and Figure 3-2 shows a 90-degree segment of the deck.

Several uncertainties identified in earlier work have been reduced or eliminated. Among these are:

- Adequate internal neutron shielding can be provided without the addition of further massive structures.
- The major components, i.e. IHXs and pumps, will not require seismic restraint at the core support structure.
- The reactor rotating plug closure now has an acceptable interface with the deck; an independent cooling system for the rotating plug is not required.
- Erection of the reactor and deck assemblies is feasible as whole structures.
- Thermal barrier design alternatives exist which will coordinate with the overall reactor assembly configuration.

Uncertainties still exist, particularly with respect to the design of the upper internals structure and the deck penetrations for major components. None of these uncertainties is expected to have a major impact on the general assembly. Work is required to confirm the adequacy of neutron and gamma radiation shielding, but any required modifications can be accommodated within the present space envelopes.

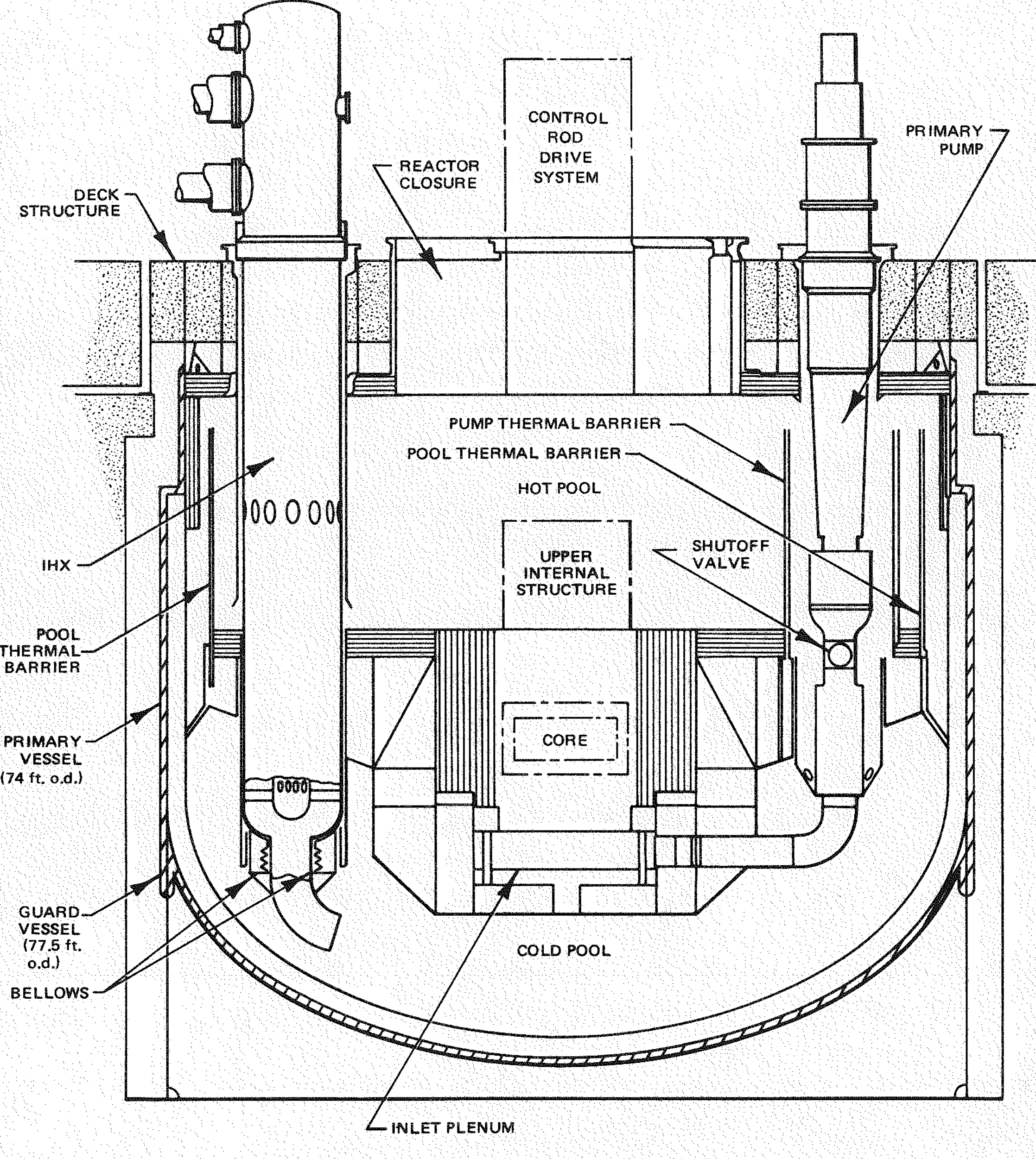


FIGURE 3-1. REACTOR ASSEMBLY

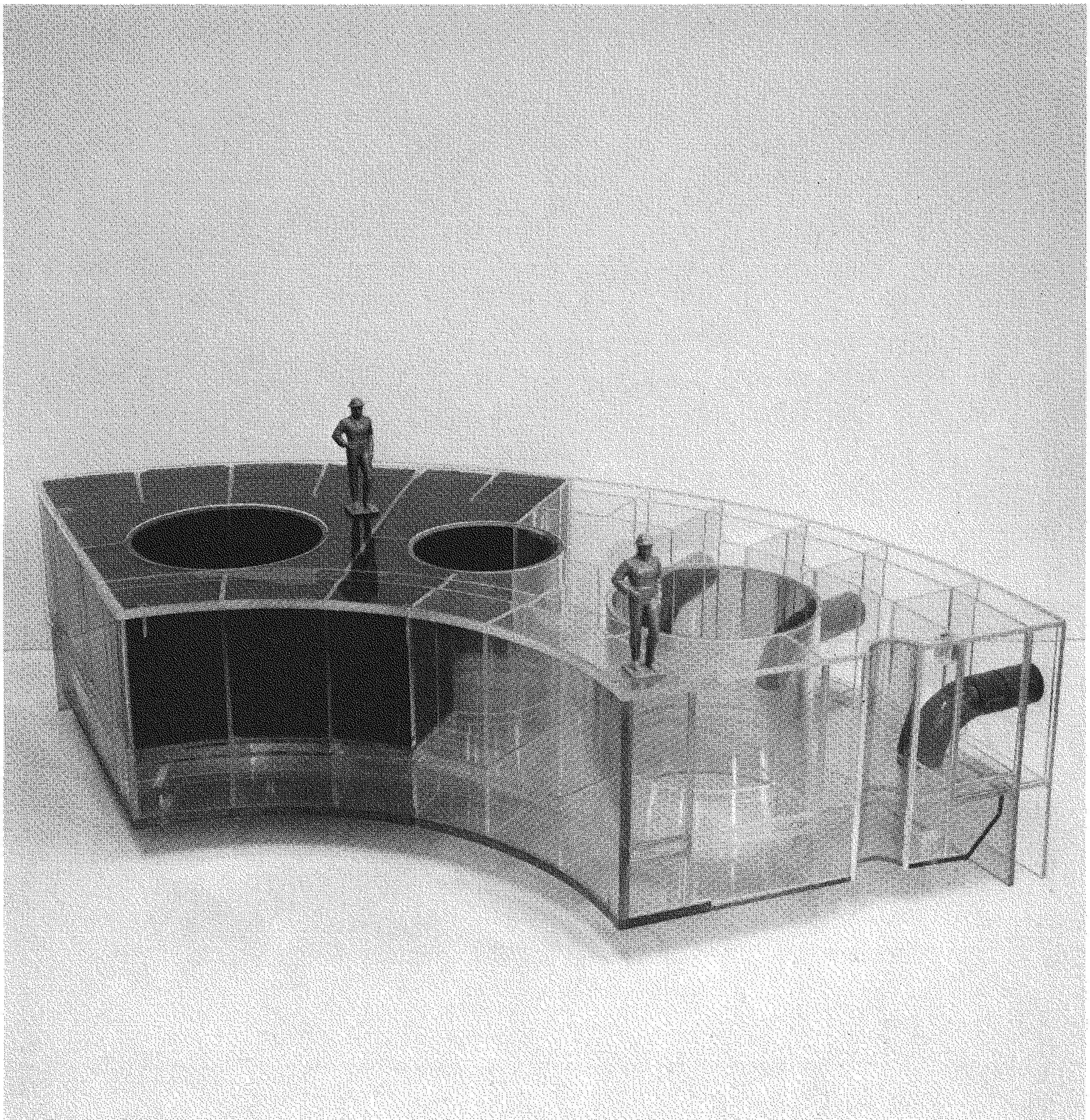


FIGURE 3-2. REACTOR DECK QUADRANT

The only remaining major structural problem which has not been addressed is the design of a redundant core support structure. This task will be accomplished in the next phase. The major findings and conclusions concerning the reactor vessel and deck design are discussed below in more detail.

Upper Internals Structure

Design work related to the upper internals structure (UIS) has been directed toward understanding the associated problems for a commercial-sized, pool-type LMFBR. The starting point has been a design adopted by General Electric for a loop-type plant. It became obvious early in the study that there were two areas where information was limited and these areas received the bulk of the effort. The first area involved the impact of operating conditions on materials requirements for the UIS and the selection of a material for key portions of the structure. The second area focused on the definition of the thermal hydraulic requirements and an attempt to factor these into the UIS design.

Materials selection was complicated by the demanding UIS service conditions. The UIS will be exposed to large, cyclic temperature changes at elevated temperatures, in a liquid sodium environment, and under irradiation conditions. These conditions result in thermal striping (high-cycle, steady-state fatigue) and low-cycle creep and fatigue damage, and preclude the use of conventional SS-316 construction. Inconel 718 has been as providing the best chance of satisfying the prescribed operating conditions.

The UIS will be required to promote mixing within the hot pool during transient events and during steady state operations. Two approaches to the design of a UIS are being pursued which incorporate features to promote mixing locally to destroy the thermal striping potential, and generally to protect downstream components during high rate temperature transients.

One approach may be characterized as a "mixing chamber" design. Upward flowing sodium from the subassemblies is conducted to a mixing chamber in the lower part of the UIS. Here the flow is made turbulent, and mixing is promoted to destroy the striping flow patterns. The mixed sodium is discharged radially by 31 exit ports three to five feet below the sodium surface.

The second approach may be characterized as a "low baffle" scheme. In this design a horizontal plate is suspended about five feet above the core, forcing the sodium to flow radially outwards soon after leaving the subassembly.

The effects of these and other alternate hydraulic designs on mixing in the hot pool region will be evaluated in future work.

Internal Radiation Shielding

Preliminary shielding calculations performed during Extension 1 have confirmed that the internal shielding provisions made during Phase A are generally adequate, but that several modifications are indicated. The following modifications are tentative pending more extensive calculations:

- In order to limit the gamma dose rate to 2 mR/hr at three feet from a main secondary sodium pipe during reactor operation (or 20 mR/hr from activated steel in an IHX or pump 10 days after shutdown) another outer row of B_4C shield assemblies was added, replacing a row of steel shield assemblies. Helium and tritium production in these B_4C assemblies during the 40-year plant life have been shown to be orders of magnitude less than that in the CRBR control rod during one year of full power. Thus, these B_4C assemblies need not be replaced during the life of the plant.
- To account for uncertainties in the calculation of neutron flux at the IHX and pump surfaces, it is desirable to provide for approximately two inches of steel shielding (or possibly less B_4C) at these components. Such additional shielding can be accommodated by a thicker IHX flow shroud and thicker thermal barriers surrounding the pump.
- A total length of 34 inches of upper axial shielding was assumed above both core and blanket assemblies, in order to reduce neutron streaming at the corners of the core.
- Additional steel plate shielding was added to the lower core support structure in the region where the inlet coolant pipes are located to prevent neutron streaming.

A preliminary evaluation of radiation damage to steel structures near the core has been made. While neutron fluence limits have not yet been established, neutron damage criteria have been considered in this evaluation for comparative purposes. This evaluation indicated that more detailed consideration needs to be given to radiation damage in the inner radial steel shield assemblies, the core former, and the upper and lower axial core steel shielding.

Thermal Barriers

Pool-type LMFBRs require effective thermal barriers in order to reduce the primary vessel operating temperature, separate the hot and cold sodium pools, and limit the heat leaked from the hot pool to the cold pool. The vertical barriers consist of a large cylindrical assembly just inside the primary vessel, and smaller cyclindrical assemblies around each primary pump. With this arrangement the cold sodium pool covers the inside vessel wall; the upper part forms an annulus between the large cylindrical barrier and the vessel. The upper cold pool annulus is essentially stagnant, a fact which requires the large cylindrical barrier to provide active cooling (rather than passive insulation) of the hot pool.

Both double-pass and single-pass active vertical thermal barriers were considered during Extension 1. In the single-pass concepts, sodium from the cold pool flows upward through the cylindrical thermal barrier and returns to the cold pool through a tubing arrangement which is not part of the barrier. In the double-pass concept the cold-pool sodium flows upward on the cold side of the cylindrical barrier and returns to the cold pool by flowing downward on the hot side of the cylindrical barrier. Thus the return flow of the double-pass concept forms part of the thermal barrier.

During Phase A the double-pass type appeared most attractive. In Extension 1, the double-pass active vertical thermal barrier concept was further developed and is deemed preferable because of its simplicity, reliability, and thermal performance. The heat transfer through the vertical barrier from the hot pool to the cold pool is approximately 20 MW for the double pass concept.

Three types of horizontal thermal barriers were considered: (1) multiplate insulation of the type proposed by the British for CDFR; (2) gas-filled panels; and (3) parallel plate insulation developed in Phase A. Analysis showed that both of the plate-type insulations required sodium circulation through the core support structure to keep the core support temperature below 750^{o}F. The parallel plate barrier developed during Phase A is judged to be the preferable design because of simplicity, reliability, and acceptable heat transfer (approximately 1.6 MW) from hot to cold pool.

Alternate Vessel Support

In the reference design, the reactor vessel is hung from the deck by a skirt, an extension of the reactor vessel, which penetrates up into the deck as a major circumferential shear web. The resulting bimetallic weld between the stainless steel vessel and the ferritic deck material is in a low-temperature, low-radiation, low-stress area and is in all respects a conventional structural weld.

This method of vessel support does however, contribute to a large deck diameter since the deck must overhang the radius of the reactor support ledge (Figure 3-3). Moreover, the deck depth is a function of deck diameter. Therefore a study of alternate reactor vessel support arrangements was undertaken to see if a basis exists for reducing the deck diameter and depth.

A number of alternates were investigated and one was selected as having important potential benefits. This design is shown conceptually in Figure 3-3. A rigid conical skirt supports a massive flanged and webbed stainless-steel circumferential girder; the reactor vessel is hung on a downward projecting skirt of this girder. A ferritic-steel flange is bolted to the top of this girder, and the deck is hung from a tensile skirt projecting downward from the flange.

This design would meet the primary objective of reducing the deck diameter from 80 ft to 73 ft, and the deck thickness by about 1 ft. It would also facilitate installation of the complete reactor vessel and deck, and eliminate the bimetallic weld. But this would be accomplished by a subtantial increase in complexity and material thicknesses at a critical region of the primary coolant boundary. Also, construction and manufacturing tolerances would be reduced. Seismic design would require a large number of keys, and accurate shimming and wedging at the points where the conical skirt supports the deck.

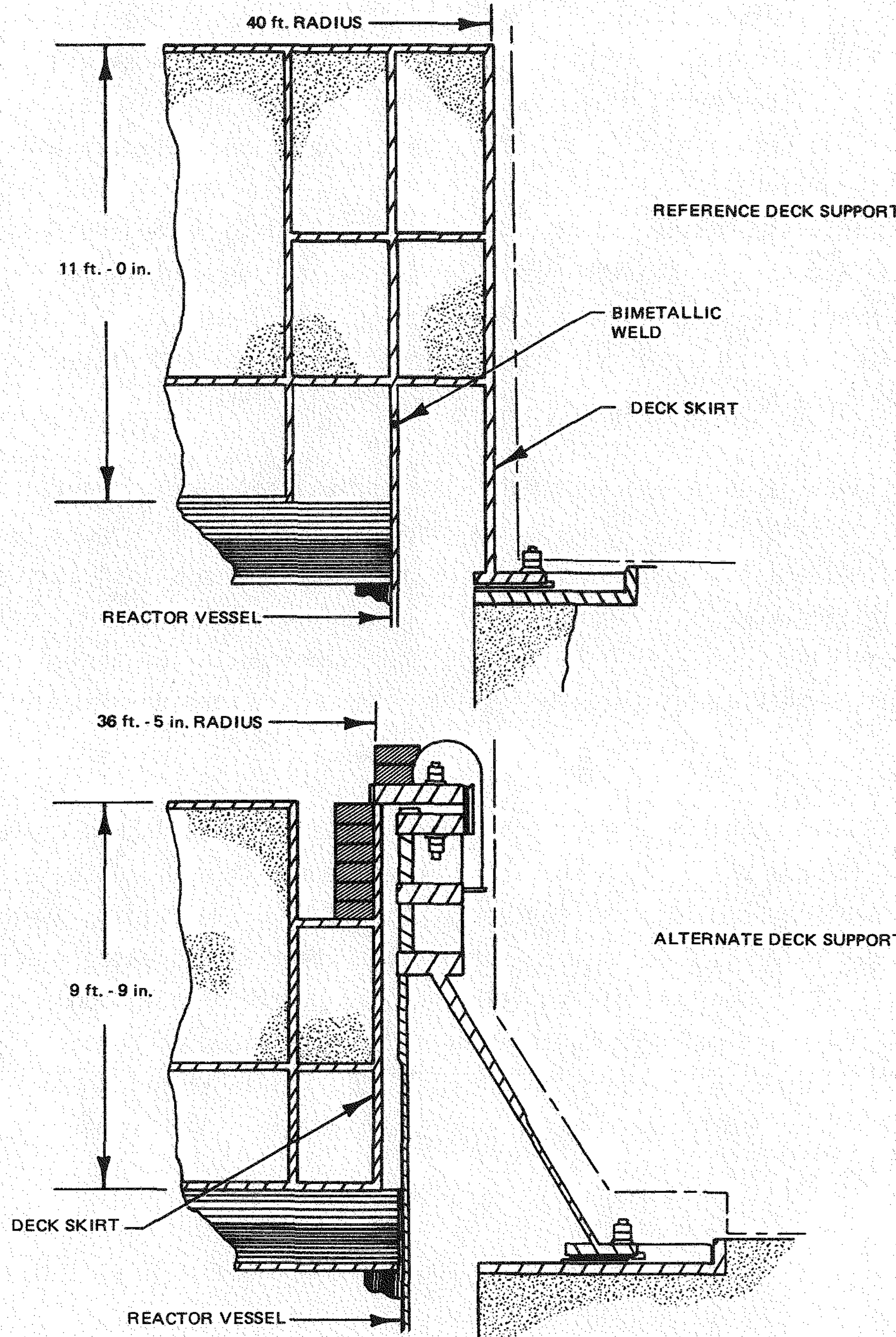


FIGURE 3-3. REACTOR ASSEMBLY SUPPORT CONCEPTS

Deck Insulation and Sodium Vapor Barriers

In Phase A the heat transfer properties of the thermal insulation in the cover gas beneath the deck were estimated from fundamental considerations. They were then compared with available information on similar French and British designs. No work was done on designs of barriers to prevent sodium vapor transport into the large deck penetration annuli, beyond surveying the literature on sodium deposition.

The Extension 1 activities provided a more detailed review which confirmed that heat transport by convection in the thermal insulation plates would be insignificant. In addition, the Extension 1 work produced design concepts for vapor transport barriers for both the pump and IHX penetration, so these designs are based upon current sodium vapor transport and deposition information. The purpose of these barriers is to reduce the heat load on the penetration liner cooling systems and to prevent sodium deposition which might interfere with operation and maintenance of the components. Commercially available stainless-steel insulations have been identified which would be suitable for application in the annular regions around major components.

Deck Penetrations

In the Phase A work, no detailed design of component penetrations was performed, although radiation shielding requirements and possible effects on the deck cooling system were identified as potential problems. In the Extension 1 activities, two IHX penetration designs were developed using the dimensional constraints and shielding requirements established in Phase A and the current IHX design. One design provides nitrogen cooling over essentially the entire length of the penetration liner, while the other design eliminates cooling of the liner in the biological shield region. No detailed design of the IHX shield plug was carried out, and the heat flow from the plug to the liner was estimated making no allowance for a possible plug cooling system. Preliminary thermal and stress analyses of the two penetration designs indicated that both are feasible. However, neither design is sufficiently developed at this time to be regarded as a reference. The study shows that the penetration design is strongly influenced by interface requirements, and that the potential for a superior penetration design exists if a number of these interfacing requirements can be altered.

Such interface areas are:

- Biological shielding requirements and the effect of a stepped plug and penetration.
- Sodium vapor barrier concepts and requirements.
- Periodic inspection requirements for the primary system boundary and deck structural welds.
- Significant reduction in heat flow to the penetration liner by modifying the design of the IHX shroud.

Removable Deck Shielding and In-Service Inspection

During Extension 1 the difficulties associated with ultrasonic examinations of the deck welds have been evaluated.

Provision for periodic examination can be included in the Extension 1 design. It is assumed that a one-twelfth sector could be examined during each yearly refueling period. Thus complete examination of the deck could be accomplished in 12-year cycles on a continuing basis.

The major conclusions of this study are:

- Removable shielding is necessary to provide personnel access to each box beam cell in the deck.
- The only removable shielding arrangement available at reasonable cost and allowing full inspection of all welds is a combination of boron-lead-polyethylene balls.
- Ultrasonic inspection requires an 8-inch by 8-inch clear space along all welds.
- The time for inspecting a one-twelfth deck sector, after waiting ten days for radiation to decay to acceptable levels, is six days, achieved by simultaneously using multiple ultrasonic inspection assemblies.

The removable shielding in each deck cell consists of one large boron-magnetite-concrete island, 18 inches high resting on the middle skin of the deck with an eight-inch clearance to all vertical surfaces. The eight-inch clearance area is filled with smaller brick-sized removable blocks and covered with a layer twelve inches deep of boron-lead-polyethylene balls. When all the shielding is in place, the 8-foot high cell is filled to a depth of 2.5 feet by the shielding. During inspection the boron-lead-polyethylene balls are first removed by vacuum equipment. Then personnel enter the top part of the cell, remove the small blocks in the eight-inch clearance area, and install the ultrasonic inspection assemblies. In this manner the 14 cells above the middle skin in the one-twelfth deck sector are inspected. The 14 cells below the middle skin are inspected without the complication of removable shielding through the access provided in gas cooling spaces. The deck structure inspection is expected to increase the total downtime of the yearly refueling process slightly.

HEAT TRANSPORT SYSTEMS

The heat transport systems carry heat from the reactor to the IHXs and from the IHXs to the steam generators. The major components in the primary heat transport system are the primary sodium pumps and the IHXs; the major components in the intermediate heat transport system are the intermediate sodium pumps and steam generators. The operating temperatures and power handling capabilities of these components are shown in Figure 1-4.

The cold leg primary pump was designed during Phase A. During Extension 1 a shutoff valve has been added. This valve is slow acting and is used only to allow plant operation with a primary pump out of service.

The major design emphasis for the heat transport systems during Extension 1 has been in the following areas:

- Elimination of difficult-to-inspect bellows in the IHX shell: the Phase A IHX has straight tubes and a bellows in the shell; the Extension 1 alternate IHX has bent tubes and no shell bellows.
- Examination of the plant characteristics and limitations with only four or five of the six IHXs in service, and with only three of the four primary pumps in service.
- Identification of the major plant duty cycle events and their number in the 40-year plant life.
- Investigation of the plant thermal transients for the above major events.
- Investigation of sodium flow patterns in the hot pool, especially hot and cold sodium mixing after a reactor scram.

It is estimated that the most severe thermal transients can be accommodated by components such as IHXs, even with conservative assumptions about the mixing of hot and cold sodium.

Alternate IHX Design

An IHX design with straight tubes and a shell side expansion bellows was pursued during Phase A to the point that its feasibility could be assured. To reduce the diameter of the deck penetration, the expansion bellows must be placed in a location difficult to inspect. In Extension 1 an alternate IHX design with bent tubes was explored to improve this situation.

This alternate design has now been completed and is shown in Figure 3-4. Table 3-1 compares some of the more significant thermal, hydraulic and dimensional data. The bent tube IHX concept is acceptable for a 1000 MWe pool-type LMFBR and has been selected as the new reference design.

Table 3-1

IHX CONCEPTS COMPARISON

	Straight Tube Design	Bent Tube Design
Diameter/No. of Tubes	1 Inch O.D./4420	7/8 Inch O.D./4860
Primary (Tube) Pressure Drop	1.6 psi	2.2 psi
Secondary (Shell) Pressure Drop	27.6 psi	13.1 psi
Tube Supports	Perforated Discs	Grid Clamps
Lower Tubesheet	Floating	Fixed

In evaluating the two designs, Table 3-2 was constructed to identify which design feature is considered to be an advantage for each of the requirements indicated. While there are advantages and disadvantages for each concept, a major advantage for the bent tube design is its ability to accommodate temperature differentials. The bent tube arrangement provides enough flexibility to accommodate shell to average tube differential thermal expansion and eliminates the need for a bellows for this function. It also allows tube-to-tube differential thermal expansion and the concept should therefore be more forgiving with respect to flow and temperature maldistributions.

The bent tube IHX also has greater inherent margins to accommodate temperature transients. While transients are judged not to be a major concern for the saturated-cycle pool-type LMFBR, sufficient uncertainty exists with respect to mixing in the pool that extra margin is desirable.

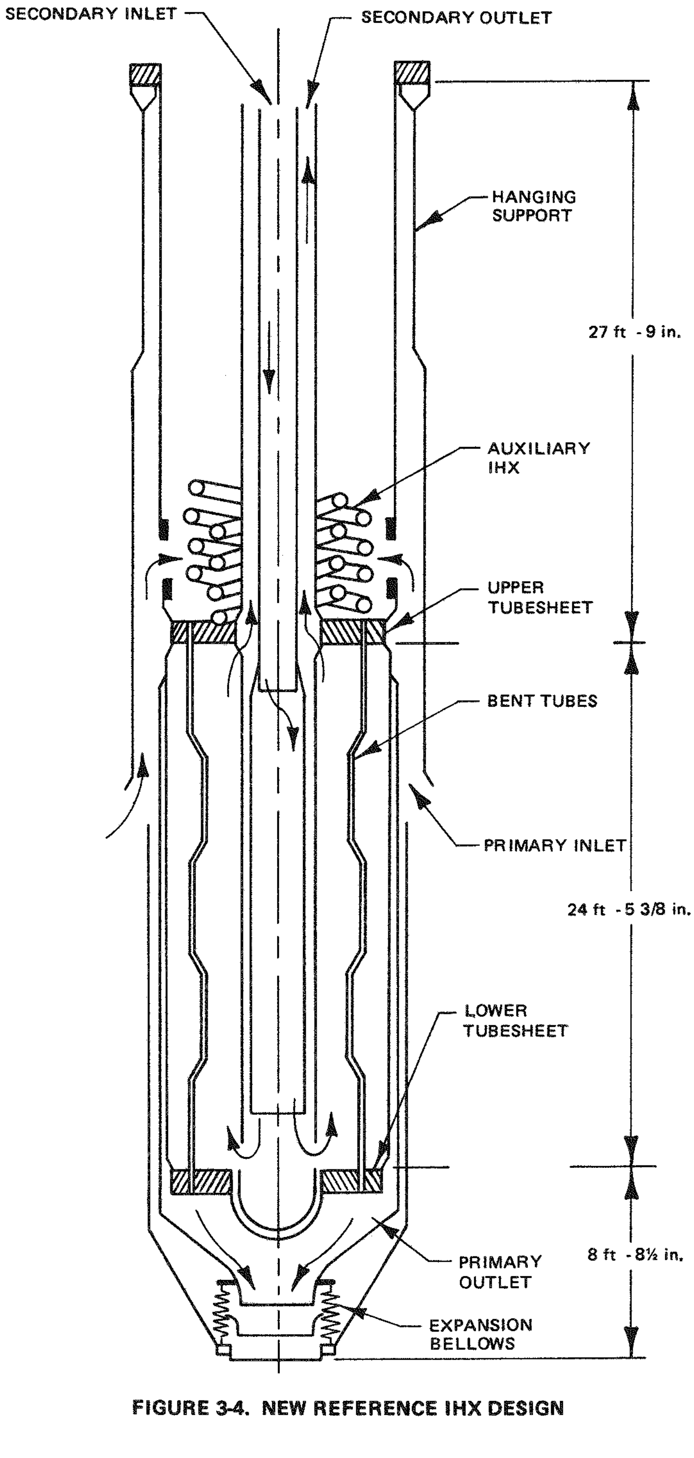


FIGURE 3-4. NEW REFERENCE IHX DESIGN

Table 3-2

IHX CONCEPTS EVALUATION

	Bent Tube Concept	Straight Tube Concept
Costs	—	—
Margin for Transients Plus Flow Mal-Distribution	++	
Shell Bellows	++	
Low Tube ΔP		+
Low Shell ΔP	+	
Seismic Capability		(+ ?)
Fewer Tubesheet Crevices	+	
Ease of Weld Inspection	+	
Simplicity in Design and Fabrication of Tube Supports		+
Ease of Tube Inspection*		+
Resistance to Tube Plugging*		+
Experience	++	

— = Standoff
+ = Nominal Advantage for Indicated Concept
++ = Significant Advantage for Concept
*If required and if possible

The bent tube IHX is inherently less earthquake resistant than its straight tube counterpart. Fortunately, it has proved possible to extend the grid clamp concept developed by the British to the bent tube IHX. This has made it possible to meet guideline seismic criteria.

Operation with Components Out-of-Service

During the operating life of a large LMFBR major components will occasionally be out of service. Plant availability will be maximized by operating the plant at the highest power possible during these periods. In Extension 1 plant operating characteristics and limitations were evaluated for one and two intermediate heat transfer loops out of service, and for one primary pump out of service.

A pool-type plant has the advantage that the primary pumps can operate independently of the intermediate heat transfer loops. Thus in a pool-type plant, when an IHX, intermediate sodium pump, or steam generator is out of service it does not necessarily affect the primary pump operation. In the present plant design the intermediate loop can be isolated by using gas pressure to depress the primary sodium level below the tube-side inlet in the related IHX. All primary sodium pumps can be kept running at a rate commensurate with heat balance and flow limitations. Similarly if one primary pump is out of service, all intermediate loops can be in operation. Flow and temperature asymmetries under such conditions have not yet been investigated but are not expected to present any serious problems.

The plant power available with one intermediate loop out of service depends upon how greatly the plant operating conditions are allowed to be changed. If all plant operating temperatures are held at normal values, then the plant is limited to 83% of full power with the primary pumps operating at 83% of full flow, and the remaining five intermediate pumps at 100% of full flow. On the other hand, using the full pumping capability, 110% of full flow for both primary and intermediate pumps, allows plant operation at nearly 100% power. In this situation the IHX primary outlet sodium and the cold pool would be about 30^{o}F hotter than normal.

The plant power with two intermediate loops out of service is approximately 67% of full power when normal temperatures are maintained. The primary pumps operate at 67% of full flow, and the remaining four intermediate pumps at 100% at full flow. Attempting to use higher primary flow and off-normal temperatures does not improve the 67% power level.

With one primary pump out of service, plant temperatures and flow conditions both depart only slightly from normal. The plant power is about 85% of full power and all six intermediate pumps operate at approximately 85% of their full-flow values. This high power level is possible because there is reduced flow resistance at reduced flow. The three remaining primary pumps each operate about 15% above their normal flow rates. Plant temperatures are affected only slightly. The hot pool operates 5^{o}F below normal and the cold pool 15^{o}F below normal.

Overall it appears that a pool plant will have exceptionally good availability with components out of service. This is true because the primary and intermediate systems in a pool plant are not linked to each other in a one-to-one dependent fashion as they are in a loop plant, and because there is a greater number (six) of intermediate loops.

AUXILIARY SYSTEMS

Work was performed during Extension 1 on the following auxiliary systems associated with safe and reliable operation of the NSSS:

- Primary sodium cold traps
- Cover gas pressure
- Reactor auxiliary cooling

The primary sodium cold traps are located in the primary vessel so that no radioactive primary sodium need be transported outside the vessel. The cold traps are necessary to remove impurities, their capabilities being determined by the amount of impurity in-leakage to the reactor. The in-leakage to the reactor is determined principally by refueling and maintenance. Satisfactory cold traps, able to handle projected in-leakage, were designed in Extension 1.

Following reactor scram the cover gas volume must increase to fill the void left by the sodium as it cools. If the amount of cover gas were not increased, a negative pressure would develop in the primary vessel and, although the pressure would not endanger the vessel's integrity, it would produce substantial in-leakage through the component-to-deck seals. During Phase A, a completely passive cover gas system with two reservoirs in the head access area was defined. During Extension 1 the system has been replaced by an active, pressurized system requiring only one reservoir, thus reducing congestion in the head access area.

The reactor auxiliary cooling system, a multiple, redundant and diverse system designed in Phase A, has been analyzed in more detail and re-designed to ensure that the system adequately meets guidelines for maximum allowable sodium and fuel temperatures. The system is designed to provide adequate cooling even under the pessimistic assumptions that the primary pumps are unavailable and that no heat is removed through the main heat transport system.

Primary Sodium Purification Systems

The Extension 1 effort has shown that purification of the primary sodium in a pool-type LMFBR is best performed by two systems; the Primary Sodium Purification System, and the Primary Sodium Receiving and Purification System. The former purifies primary sodium in the reactor and pool during normal operation. The latter removes impurities from larger quantities of sodium that has no, or only low levels of, radioactivity. This system has one primary function. It will cleanup the entire primary system prior to plant startup. In addition it will clean sodium that has been stored, prior to its reuse.

The Primary Sodium Purification System, which operates during normal plant operation, will have two loops, each loop comprising one cold trap, one economizer, and one electromagnetic pump. During normal plant operation, one loop of the purification system will be in service, and the other will be on standby. However, the two loops can be used simultaneously, if necessary. The physical arrangement requires that these cold traps draw their sodium from the 875°F hot pool. The cold traps will hang from the reactor deck into the hot pool. This fulfills the requirement that primary sodium does not leave the primary tank. The recommended cold trap is an improved design with an increased capacity and lifetime for a given size. An economizer pre-cools the sodium to 340°F and a NaK cooling system further cools it to the 230°F cold trap temperature. The economizer reheats the sodium prior to its return to the system. The electromagnetic pump is located after the economizer to place it at the lower temperature. The flow per purification loop is 50 gallons per minute, yielding sodium impurity levels of less than 2.0 ppm of oxygen and 0.2 ppm of hydrogen. The primary system can be cleaned up following refueling and major maintenance, in about three days.

Because the Primary Sodium Receiving and Purification System will process non-radioactive sodium, it will be located outside the Reactor Containment Building.

Besides cleaning sodium before entry into the primary system, it will also be used to clean up the entire pool and reactor system prior to plant startup. For this operation, the primary sodium will be circulated to the receiving system cold trap for about four days. It is estimated that about 100 lbs of oxygen will have to be removed to lower the oxygen concentration to below 2.0 ppm.

Reactor Auxiliary Cooling Systems

The Reactor Auxiliary Cooling Systems (RACS) prevent excessive reactor and primary sodium system temperatures after reactor shutdown, under all possible combinations of system malfunctions. These two diverse and redundant systems remain essentially the same as defined during Phase A and are shown in Figure 3-5. Both systems transfer reactor decay heat from the primary sodium to the atmosphere via intermediate NaK loops.

One of the systems, the Intermediate Reactor Auxiliary Cooling System (IRACS), is totally passive and will remove heat directly from the primary tank via auxiliary coils located in the primary side of each of the six main IHXs. The primary sodium flow through the core and IHXs, the NaK flow through the six auxiliary system loops, and the airflow through the NaK heat exchangers will all be by natural circulation. With one loop of either RACS out of service, and no other means of heat removal, the core outlet temperature will be less than $1350^{\circ}F$ and the pool temperature will be below $1150^{\circ}F$ following a scram from full power.

The second system is the Primary Reactor Auxiliary Cooling System (PRACS). It has three identical loops with auxiliary IHXs hanging from the deck into the pool. Both the NaK circulation and the air cooling will be forced, and this system is operable from the plant emergency power supply.

The Extension 1 effort was directed toward achieving a better understanding of reactor system temperature and flow transients during decay heat removal by the RACS. In addition, the EPRI requirements for these systems were upgraded.

The Phase A guideline required that, for the unlikely situation that only two-thirds of either auxiliary cooling system would be in service following a scram, and all main heat removal systems were lost, the reactor vessel would be maintained below the faulted temperature limit, about $1100^{\circ}F$. For temperatures below $1100^{\circ}F$, the vessel would retain its integrity. At temperatures above $900^{\circ}F$, it is possible the vessel would require re-qualification before reuse.

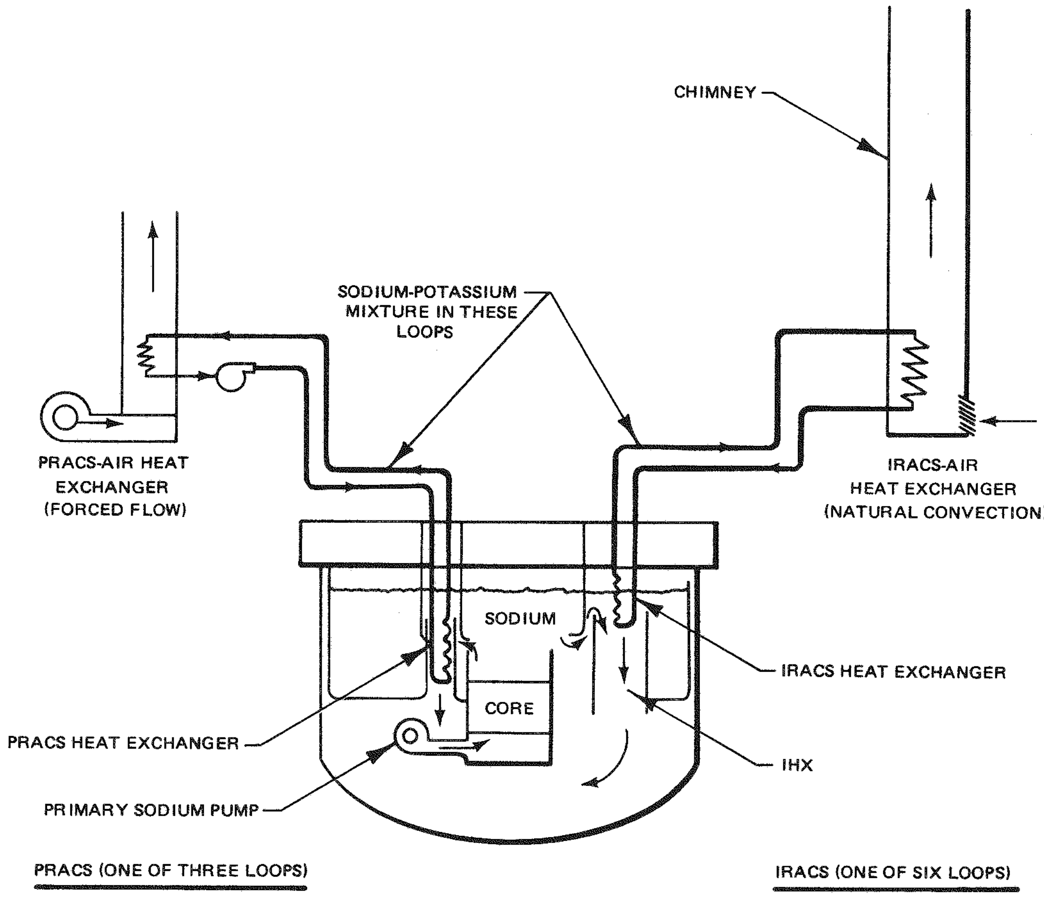


FIGURE 3-5. AUXILIARY COOLING SYSTEMS

For Extension 1 the guidelines were revised to preclude possible damage to the reactor vessel that would require re-qualification, under the above operational assumptions. To achieve this, it was specified that the vessel wall temperature must stay below about 900^{O}F. This is about 200^{O}F above the maximum vessel wall temperature during normal operation.

The Extension 1 work indicates that the RACS as defined for Phase A probably have inadequate capacity to meet the revised design guidelines. Heat input to the vessel wall, and the resulting higher temperatures, stem principally from temperature stratification in the reactor outlet plenum and reduced performance of the thermal barrier separating the hot and cold pools when the pump flow is decreased or stopped.

Several approaches have been identified for keeping the reactor vessel wall suitably cool during emergency events. These include:

- Increasing the capabilities of the RACS.
- Improving the hot-to-cold-pool thermal barrier to reduce the heat-up rate of the upper region of the cold pool following trip of the main pumps.
- Maintaining primary pump operation on pony motors following reactor scram.

All the above approaches require further study in order to select the preferred aproach, or preferred combination of approaches, that integrates best with the entire reactor system.

Cover Gas Pressure Control

The reactor cover gas for a pool-type LMFBR must operate within a narrow pressure range to minimize stresses on the reactor deck and the reactor vessel, and to restrict seal leakage. A tentative cover gas pressure of -6 ±3 inches of water has been selected to assure that reactor seal leakage will result in air and nitrogen leaking into the reactor rather than radioactive cover gas escaping from the reactor. The low-pressure differential between the inside and outside of the reactor enclosure will minimize seal leakage.

During shutdown transient events, cooling will cause sodium and cover gas shrinkage. The cover gas pressure swing during transients should be maintained at less than ±1/2 psi. For emergency events, a pressure differential of up to ±3.0 psi is acceptable. During Extension 1 a computer code was developed to predict cover gas transients for two shutdown scenarios: 1) reactor scram with both the primary and the secondary sodium continuing at full flow, and 2) reactor scram with the primary sodium decreasing to pony motor flow (about 7%). These two procedures envelop all possible shutdowns.

The most severe shutdown procedure is to keep the sodium pumps running after reactor scram. The decreasing temperature causes the hot pool sodium to shrink rapidly in the first few minutes. With no argon make-up, cover gas pressure would decrease about 4 psi after the hot pool temperature reaches about 600^{o}F. To limit the cover gas pressure drop to 1/2 psi, requires at least 360 cubic feet per minute of make-up argon flow for about 3 minutes.

The least severe shutdown procedure is to trip the sodium pumps to pony motor speed. The hot pool temperature will drop very slowly and the cover gas temperature will follow closely. A prolonged 25 cubic feet per minute argon flow is adequate for limiting the gas pressure drop to 1/2 psi. For either shutdown procedure, it will require 4200 cubic feet of make-up argon (at standard conditions) to control the cover gas pressure when the hot pool temperature drops to 600^{o}F and 12,000 cubic feet if the hot pool drops to 400^{o}F.

These analyses have shown that the cover gas constant pressure control system (a "passive" system) proposed for Phase A is inadequate. Its 3500 cubic feet tank capacity will provide only about one-third of the argon required following a scram. The tank capacity of this approach must be increased substantially in order to limit the cover gas pressure drop to within 1/2 psi.

During Extension 1 an "active" cover gas control system was investigated. This system would use actively controlled valves for cover gas pressure regulation. Its principal advantage would be high pressure, low volume, argon storage. A redundant parallel and series valve arrangement would give a system reliability of 1.2×10^{-7} failures per year for losing cover gas pressure control and 2.8×10^{-5} failures per year for overpressurizing the reactor vessel. This active cover gas control system has been selected as the preferred concept.

INSTRUMENTATION AND CONTROL SYSTEMS

Instrumentation consists of sensors, signal processing, and multiplexing that provides the input to the plant monitor displays, the plant control system, and the plant protection system. Redundant and diverse sensors assure high reliability. The largest part of the plant instrumentation is for surveillance.

The first large LMFBR will require more than normal instrumentation to monitor the plant performance, to assure safe operation, to improve plant availability and to verify plant design. Redundant instrumentation will be provided where the instrumentation is complex and does not have a sufficient reliability data base. A large part of the instrumentation will be for making measurements inside the primary vessel. Sensors for these measurements must be supported by the internal structure of the deck and the rotating plugs. A large number of deck and plug penetrations will be needed to connect the sensors to the signal processing and multiplexing instrumentation in the head access area.

Both the loop-type LMFBR and the pool-type LMFBR have a crowded head access area. This area must accommodate the primary pumps, the intermediate heat exchangers, and the auxiliary heat exchangers. In addition a great number of electrical wires pass through deck penetrations and then go to junction boxes. If sufficient space is available, it may be possible to locate multiplexers on the rotating plugs and deck to reduce the number of electrical cables.

A simplified sodium pump control system has been proposed during Extension 1. The main primary and intermediate pump motors run at full speed during part-power and full-power operation, and at reduced speed after a reactor scram. The reduced speed, tentatively chosen as 50% of full speed, is possible because the resulting post-scram temperature transient is not expected to be severe in a pool-type LMFBR.

Failed Fuel Detection

During the 40-year life of a 1000 MWe LMFBR approximately two million fuel pins are expected to spend an average of eighteen months residence time in the core. Even with good quality control some fuel pin leaks are expected. At the present time very little is known about the effects of cladding failure but European experience with operating LMFBR plants show few failures and no serious consequences. Nevertheless, until there is greater experience it will be necessary to detect failed fuel early and take appropriate action.

The type of failed fuel detection system that is selected will depend on the philosophy and methodology relative to failed fuel replacement. The entire core could be tested assembly-by-assembly if fission products exceed acceptable levels in the primary sodium. Such testing could be performed during refueling, or after refueling if the entire core were first reloaded with other fuel assemblies. A preferable method is to have some method of location wherein failed fuel is identified within a small group of assemblies to be tested. Most of the failed fuel detection systems investigated during Extension 1 were directed toward this last alternative.

No single system appears entirely satisfactory and a combination of systems will probably be necessary. The relative emphasis among the elements of the combination should be studied to find the most economical solution depending on fuel failure frequency, detection system costs, reactor availability and fuel costs.

During Extension 1 investigations of several failed fuel element detection and location (FEDAL) systems were begun. The presently recommended approach requires a number of sodium samples to be taken from the above core structure and analyzed for evidence of fuel leakage. Delayed neutrons in the sodium sample streams can be detected if the sample transport time is less than 40 seconds. If enough sample streams are provided the faulty fuel can be isolated within a small number of assemblies. Further isolation of the defective assembly can be achieved off line during refueling. Other systems deserving further investigation include those that detect the presence of local blockage and local boiling of the sodium within a faulted fuel assembly.

Plant Control

In a saturated-steam-cycle LMFBR plant, the steam temperature is a function of the steam pressure only and the major control systems are those that control the steam pressure, drum level, and reactor power. If variable speed sodium pumps are used, the reactor power control system includes sodium flow control. During Phase A, variable speed pump drives powered by motor-generator sets were proposed. While a large building area was required for the motor-generator sets, the Phase A approach minimized congestion in the head access area because relatively small, brushless, three conductor, squirrel cage motors could be used.

During Extension 1 a simpler, two-speed plant control system has been proposed where the three-lead squirrel cage pump motors are powered by either of two different bus lines, one 60 Hz line for full speed and the other a lower frequency line for reduced speed. There is also a lower voltage, 60 Hz line for pony motors. The plant control system is illustrated in Figure 3-6. The prime input to the power control system comes from flux monitors that are periodically calibrated by heat balance measurements. The steam side of the system follows the reactor power by independent pressure control (P, Figure 3-6) and feedwater flow control (F, Figure 3-6) systems. With this type of control, the motor control equipment is smaller than in Phase A because the motor generator sets are eliminated. The motor control system is essentially a switching apparatus which chooses the full-power line, the reduced-power line, or zero power.

With a two-speed pump system there are limitations on the plant power change rate and some design uncertainties. In order to keep thermal stresses acceptable the power change rate should be limited to 0.25% per minute. This rate is based on loop plant calculations, but the expectation is that a pool plant would be allowed a higher rate because of its larger sodium volume. By contrast the Phase A variable sodium flow system would permit a 3% per minute power change rate. In light of the fact that this is a base-load plant, and that some turbine manufacturers recommend less than 1% per minute load changes to increase turbine life, the limit on the power change rate for a constant speed system is probably acceptable. Of greater concern perhaps is the inflexibility of the system. Incorrectly predicted sodium flows might reduce plant performance if they could not be adjusted or compensated for. Such problems will be more important for the first plant than for subsequent plants. It is recommended that such concerns be quantitatively investigated in Extension 2.

PLANT OPERATION

The study of plant behavior in Extension 1 yielded encouraging results. One of the most important plant responses, to the scram transient, can be accommodated by a large pool plant even with conservative assumptions about sodium mixing.

The plant control system is the starting point for analyzing plant operation. In addition, failure rates of instruments, valves, components, etc., and an idea of the maintenance schedule are necessary. Although much of the informa-

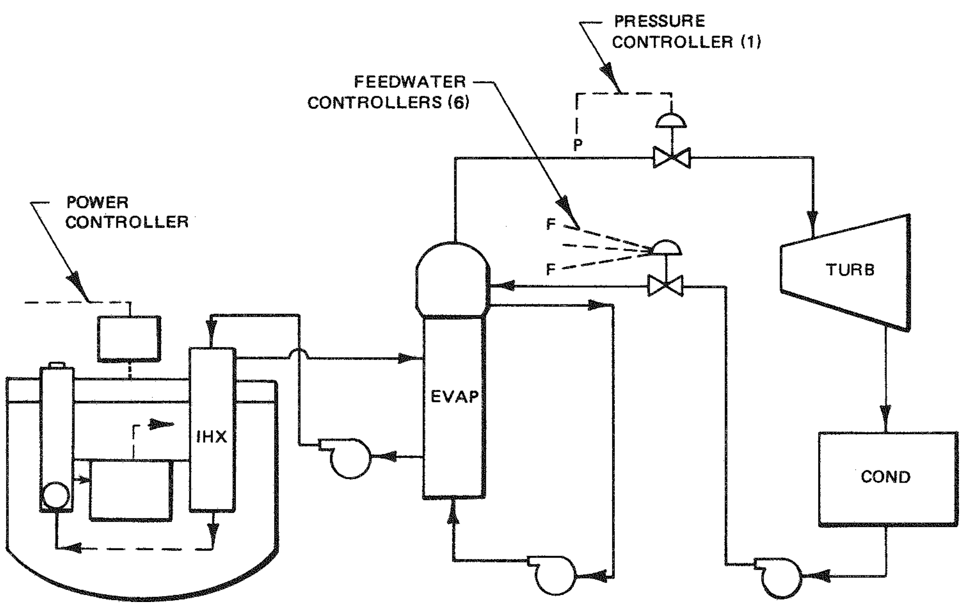


FIGURE 3-6. PLANT CONTROL DIAGRAM

tion for a pool plant is preliminary at this time, a duty cycle has been estimated for the 1000 MWe Pool-Type LMFBR during Extension 1. The major plant transients identified in the duty cycle have been analyzed to determine the overall plant thermal behavior.

Plant Duty Cycle

The plant is designed for base-load operation but is also capable of part-load operation. The 40-year design life consists of about 29 years at full power and about 5 years at part power with the remaining 6 years necessary for activities such as refueling and maintenance. The refueling and hot standby sodium temperatures are 550^{o}F. The major events in the duty cycle are the startups and shutdowns between 550^{o}F and the full power hot pool sodium temperature of 875^{o}F. There are approximately 1000 normal startups, each proceeding over a period of four to six hours. Startups introduce negligible thermal stresses in the heat transport system components. The shutdowns also number approximately 1000 but about 500 of these proceed so slowly as to produce only negligible thermal stesses. The other 500 shutdowns are plant scrams from full power and produce hot pool sodium temperature changes of several hundred degrees within a few minutes. Scrams are upset events which all components must withstand for the plant lifetime. An analysis of post-scram stresses in the most critical region of the IHX has confirmed the adequacy of the IHX design.

Plant Scrams

Saturated-steam-cycle pool plants have the freedom to maintain high primary and intermediate sodium flow rates after scram because the large volume of the hot pool moderates the decreasing primary sodium temperatures and because the steam generator temperatures are stabilized by the saturated steam. Such a system enhances core cooling reliability because pony motor engagement is not necessary on every scram but is always available as a backup.

If the post-scram sodium flow is kept high there are both attractive features and areas of concern. The most attractive feature of high flow is that the active thermal barriers separating the hot and cold sodium pools, dependent on primary sodium flow, remain effective after scram.

The area of most concern with high post-scram sodium flow rates occurs in case scram is caused by the loss of an intermediate sodium pump. In this case the hot primary sodium flowing down through the affected IHX is no longer cooled and causes a thermal shock to the lower part of the IHX and possibly to structures in the vicinity of the IHX primary outlet.

Similarly, if the post-scram sodium is tripped to pony motor flow, there also are attractive features and concerns. The most attractive feature of low flow occurs for loss of an intermediate sodium pump. This event is the same as a normal scram and there is no thermal shock to the IHX primary outlet and its environment.

Areas of concern for low flow occur because IHX performance at a low flow rate is not well understood. Even constant temperature primary sodium entering the tube bundle may be cooled unevenly and produce unacceptable thermal stresses. Also, the vertical thermal barriers, separating the hot and cold pools and using part of the primary sodium flow, would be ineffective at low flow.

The Extension 1 plant design appears adequate to accommodate a post-scram sodium flow which is 50 percent of full flow. At this flow rate, the expectation is that sodium mixing is adequate to produce acceptable thermal transients. If the flow were to be decreased to pony motor flow, the adequacy of the plant design would be very dependent on hot-cold sodium mixing at very low flow rates and on IHX flow distributions. For these reasons the present plant design will be carried forward into Phase A Extension 2 assuming a 50 percent post-scram sodium flow.

If the scram is caused by a pipe break between the primary pump and the high pressure inlet plenum below the core, the design must ensure sufficient core flow that the fuel pin cladding strength remains acceptable. This event is the major factor in determining how many pipes are necessary between each primary pump and the core inlet plenum. When there is only one pipe per pump, a single pipe break represents a flow reduction of approximately one fourth. If two pipes per pump are used then a single pipe break represents only a one-eighth flow reduction. Analyis of the reference plant with one pipe per pump indicates that the core flow is adequate to maintain acceptable fuel pin cladding strength after a pipe break. The criteria for acceptability are not yet well defined and further work is necessary in this area, but it is not expected that two pipes per pump will be necessary.

SEISMIC ANALYSES

Nuclear power plant structures, systems and components important to safety must be designed to withstand the effects of earthquakes without losing the ability to perform their safety functions. Preliminary seismic analyses have been made of the Extension 1 pool-type LMFBR reactor assembly, primary pump, and IHX, to check their design adequacy. More sophisticated seismic models have been used during Extension 1 than were used for Phase A.

During Extension 1 the seismic response of the core and vessel has been evaluated for the alternate vessel support previously shown in Figure 3-3. Table 3-3 compares these results with similar results for the reference vessel support (also shown in Figure 3-3) from Phase A. The reference vessel support is slightly stiffer and allows smaller seismic displacements than the alternate support.

Table 3-3

MAXIMUM SEISMIC DISPLACEMENTS

	Safe Shutdown Earthquake		Operating Basis Earthquake	
	Horizontal	Vertical	Horizontal	Vertical
Phase A				
Core Support	.442"	.420"	.304"	.286"
Vessel Bottom	.574"	1.100"	.390"	.177"
Extension 1				
Core Support	.568"	.521"	.336"	.367"
Vessel Bottom	.757"	1.370"	.449"	.984"

The functional requirements allow ~1.0 inch relative motion between the core and control rod drive systems for the vertical Operating Basis Earthquake. Approximately 2.0 inches is allowed for the vertical Safe Shutdown Earthquake.

In any horizontal excitation, 0.6 inch relative motion is allowed. The maximum horizontal displacement is slightly over this limit in the case of a Safe Shutdown Earthquake. It is believed that a more rigorous analysis will shown this limit satisfied. The maximum vertical displacements are well below the allowed limits.

For a Safe Shutdown Earthquake, the maximum stress must not exceed 70% of the ultimate strength of the material at operating temperature. Allowable stress limits for an Operating Basis Earthquake are considerably lower than for a Safe Shutdown Earthquake and it becomes important to include stresses related to weight, pressure, and temperature. Analyses show that for the main sodium pumps under a Safe Shutdown Earthquake, the maximum combined peak-to-peak horizontal displacement might reach about 3.8 inches at the bottom of the pump and the stress could reach about 20,000 psi in the pump support structure; for an Operating Basis Earthquake, the comparable values are 2.5 inches and 13,000 psi. These stresses are well below the maximum allowable stresses.

The maximum combined peak-to-peak displacement for the IHX could reach about 3 inches for a Safe Shutdown Earthquake. This occurs near the center of the central downcomer. The displacement of the bottom of the IHX is about 1.6 inches. Peak stresses are about 17,000 psi. Without a seismic stop at the upper tubesheet, the displacement of the bottom of the IHX would be 4.6 inches and the maximum stress would be 22.3 ksi.

Overall, the preliminary seismic analysis indicates that the Extension 1 designs of the reactor, IHX, and sodium pump are feasible and will have good safety margins.

Section 4

BALANCE OF PLANT

HEAD COMPARTMENT LAYOUT

Early in the Phase A study it was recognized that the location of major NSSS components on the deck covering the reactor vessel could result in congestion, which could impede normal maintenance. This was of special concern for the pool reactor where the primary pumps, intermediate heat exchangers, control rod drives and refueling provisions are all located on the deck along with other miscellaneous equipment and instrumentation. The conceptual design of this area was therefore developed in depth. It was firmly established that adequate space can be provided within the head compartment for all the equipment and services proposed, and for necessary maintenance, inspection and equipment removal. A typical layout of this area has been illustrated previously in Figure 1-3.

COMPONENT REMOVAL

This study was made to establish concepts and space requirements for removal of major NSSS components from the deck, through the containment, to an on-site maintenance facility. Results indicate removal and transfer of the primary heat transport system components in a vertical position is feasible in the event such requirements become mandatory; for example, to preclude distortion of the primary pump shaft. The scheme for removal of a pump or IHX is illustrated in Figure 4-1, and involves rotating the IHX to the horizontal position once it has been removed (vertically) from the reactor containment building.

PIPING ENCLOSURES

The Phase A conceptual design provided for a means of enclosing the head of the IHX and the connected intermediate heat transport system piping to minimize the fire hazard from a sodium spill within the head compartment area. The objective of the Extension 1 study was to determine a preferred enclosure concept capable of meeting subsequent design requirements with respect to detecting, confining, and accepting pressures associated with a sodium spill. The preferred concept utilizes telescoping guard pipes, which permit inspection and mainte-

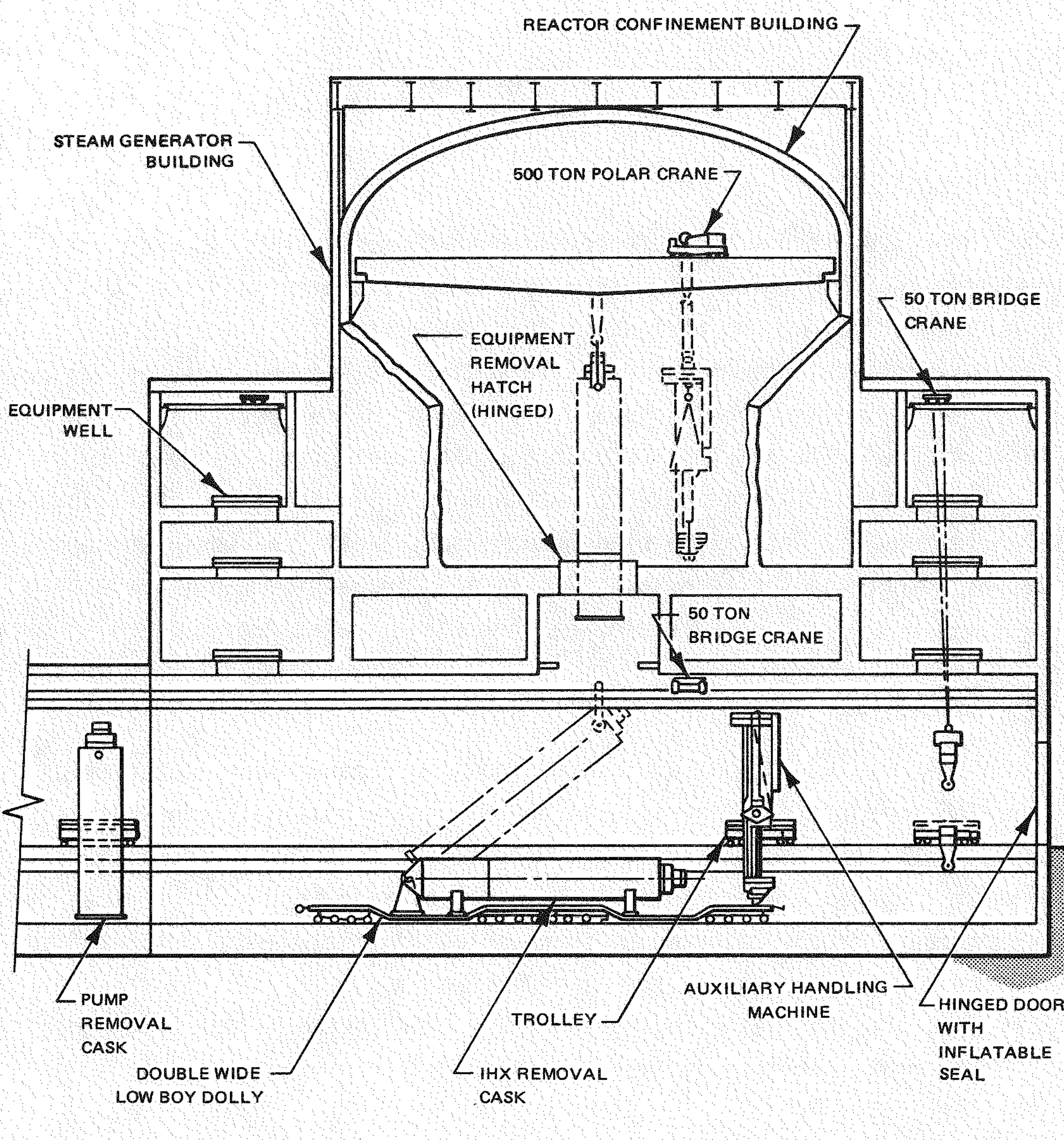


FIGURE 4-1. COMPONENT REMOVAL

nance of the entire closed piping. This concept is illustrated schematically in Figure 4-2. No technical difficulties in developing the detail design of this concept are anticipated.

REACTOR ASSEMBLY SUPPORT

An important aspect of the pool reactor cavity design is its interface with the reactor-vessel/deck-support structures. This study included an investigation of the BOP structural requirements for three alternate deck/vessel support skirt designs, and the structural design of the concrete reactor cavity wall and vessel/deck support ledge. As a result of the work done, it is concluded that it is feasible to support the primary vessel and the deck on a concrete ledge at the top of an 8-foot thick reactor cavity wall. The projection of the ledge could be such that the guard vessel, primary vessel and associated internals could all be installed as complete subassemblies after all support concrete work is done, thus minimizing overall plant construction time. The general configuration of the support scheme chosen is shown in Figure 4-3. An eccentric loading on the cavity wall results from supporting the vessel/deck on the ledge: it has been established that the seismic response is acceptable.

AUXILIARY HEAT REMOVAL SYSTEM

During Phase A, General Electric established the general NSSS features and single point design operating requirements for the conceptual design of auxiliary systems to remove decay heat as shown previously in Figure 3-5. Specific design features of the BOP portion of these systems were not identified at that time. Therefore, in Extension 1 a study was made to determine and develop the conceptual design features and performance capabilities of the BOP portion of the IRACS and PRACS, such as the stacks, piping, fans, and inlet plenums. A further objective was to identify any potential problem areas requiring further investigation. Performance of the IRACS stack over a range of operating and start-up conditions was investigated, and the feasibility of using dampers to control slack performance was evaluated. Further investigation to integrate heat exchangers, piping, and stack performance of the entire system over the expected range of operating and transient conditions will be made in a later phase of the work.

The study shows that a PRACS stack height of 160 ft with a cross-sectional area of approximately 60 sq ft provides adequate margin for design point natural draft operation. This stack would extend approximately 35 ft above the confine-

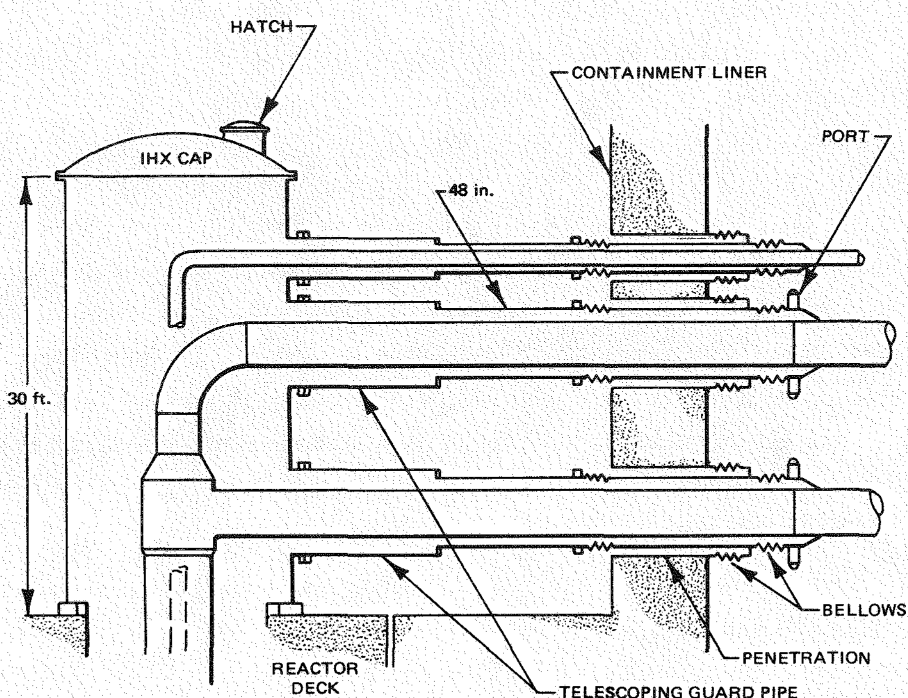


FIGURE 4-2. INTERMEDIATE PIPING ENCLOSURE

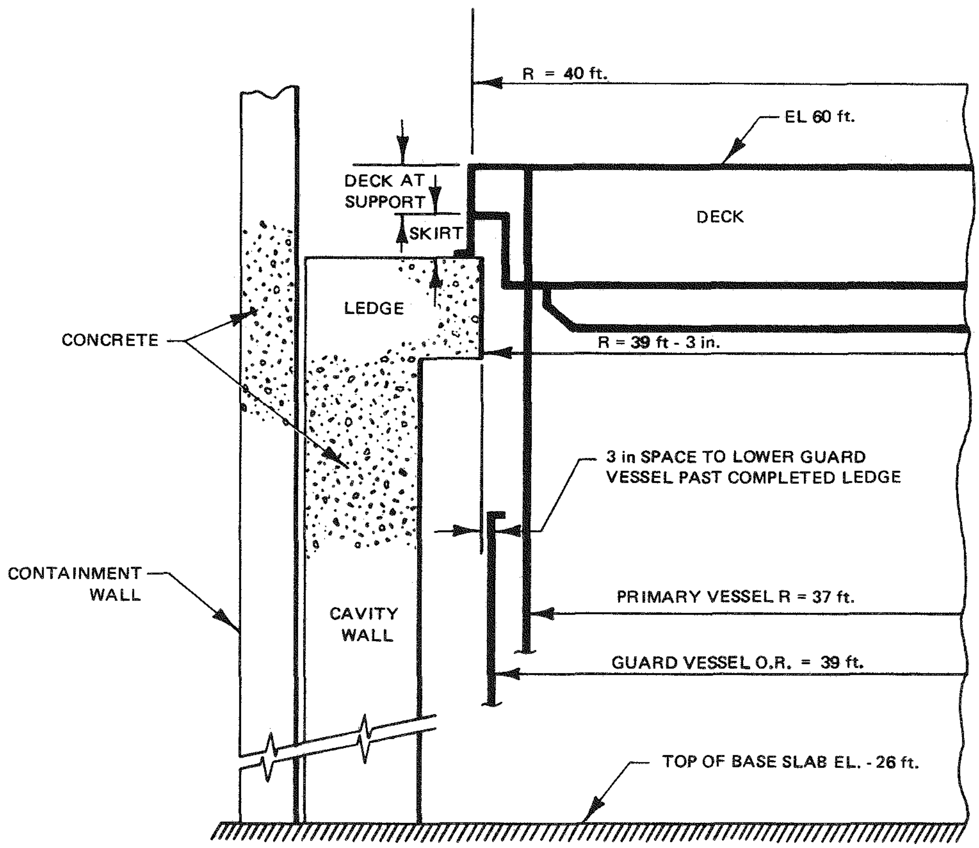


FIGURE 4-3. REACTOR ASSEMBLY SUPPORT

ment building. Potential start-up problems, caused by system inertia, indicate the need for control dampers and investigation of thermal transients. Performance characteristics of the PRACS natural draft stack are shown in Figure 4-4.

COVER GAS SYSTEM

A constant-pressure cover-gas system utilizing a variable volume tank was assumed for the Phase A reference design. Recognizing that safety and licensing issues relating to this constant pressure concept had not been resolved, the reactor cover gas system was further investigated.

Three cover gas pressure control schemes were examined in Extension 1:

- Constant pressure by variable volume.
- Active make-up by mechanical pressure control.
- Constant volume reservoir tank.
- A combined active/passive system proposed by EPRI at the end of Extension 1 remains to be investigated in future phases of the work.

As a result of this study, it was concluded that the "active" system is preferred since it has the potential for highest reliability through redundancy, and meets design and safety codes, is a flexible design with respect to varying set pressures, and minimizes congestion within the containment building. Safety of this system is further enhanced by the inherent structural capability of the reactor vessel to accommodate predicted pressure extremes that could result from a scram without cover gas pressure control. This preferred system is shown schematically in Figure 4-5.

REACTOR ASSEMBLY CONSTRUCTION

In developing the conceptual design for the pool-type LMFBR during Phase A, it was recognized that the large size and weight of the reactor components would require special erection and schedule consideration, and that the amount of in-situ fabrication could adversely impact the overall plant schedule by interference with other construction activities. A means of minimizing such constraints is to essentially complete all fabrication of reactor components (guard vessel, reactor vessel, deck, etc.) before installation in the containment building.

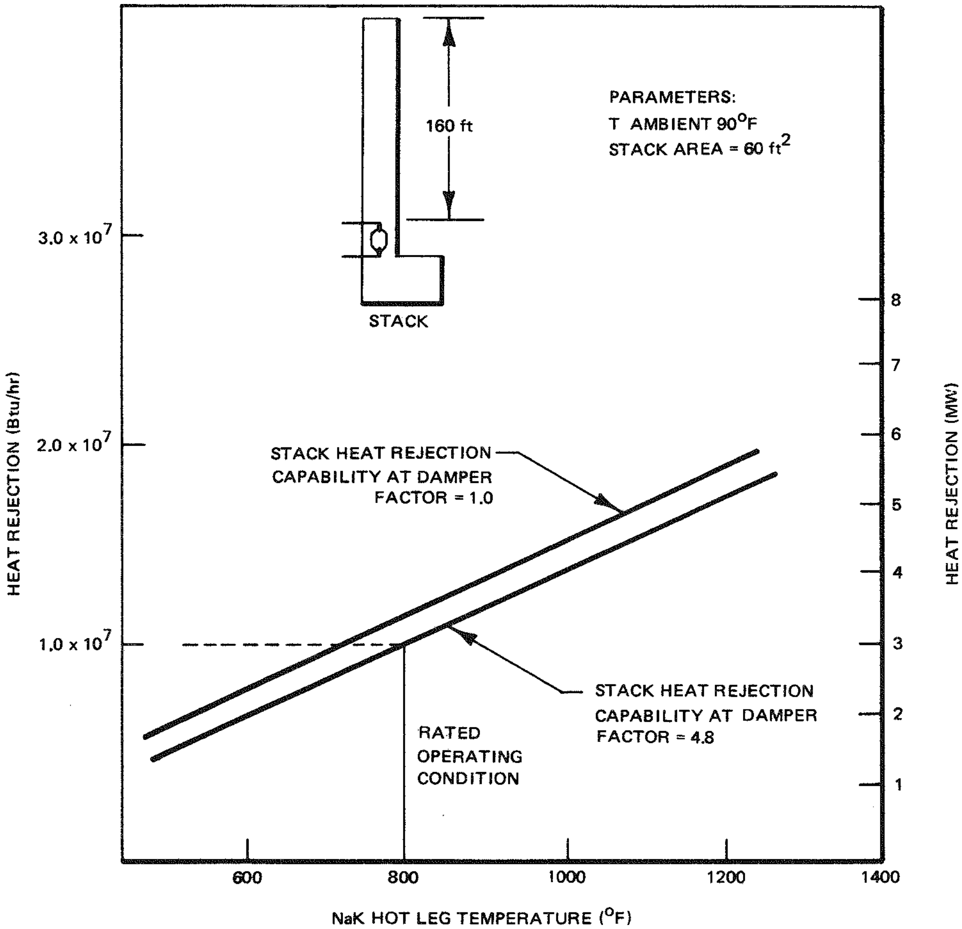


FIGURE 4-4. AUXILIARY COOLING SYSTEM PERFORMANCE

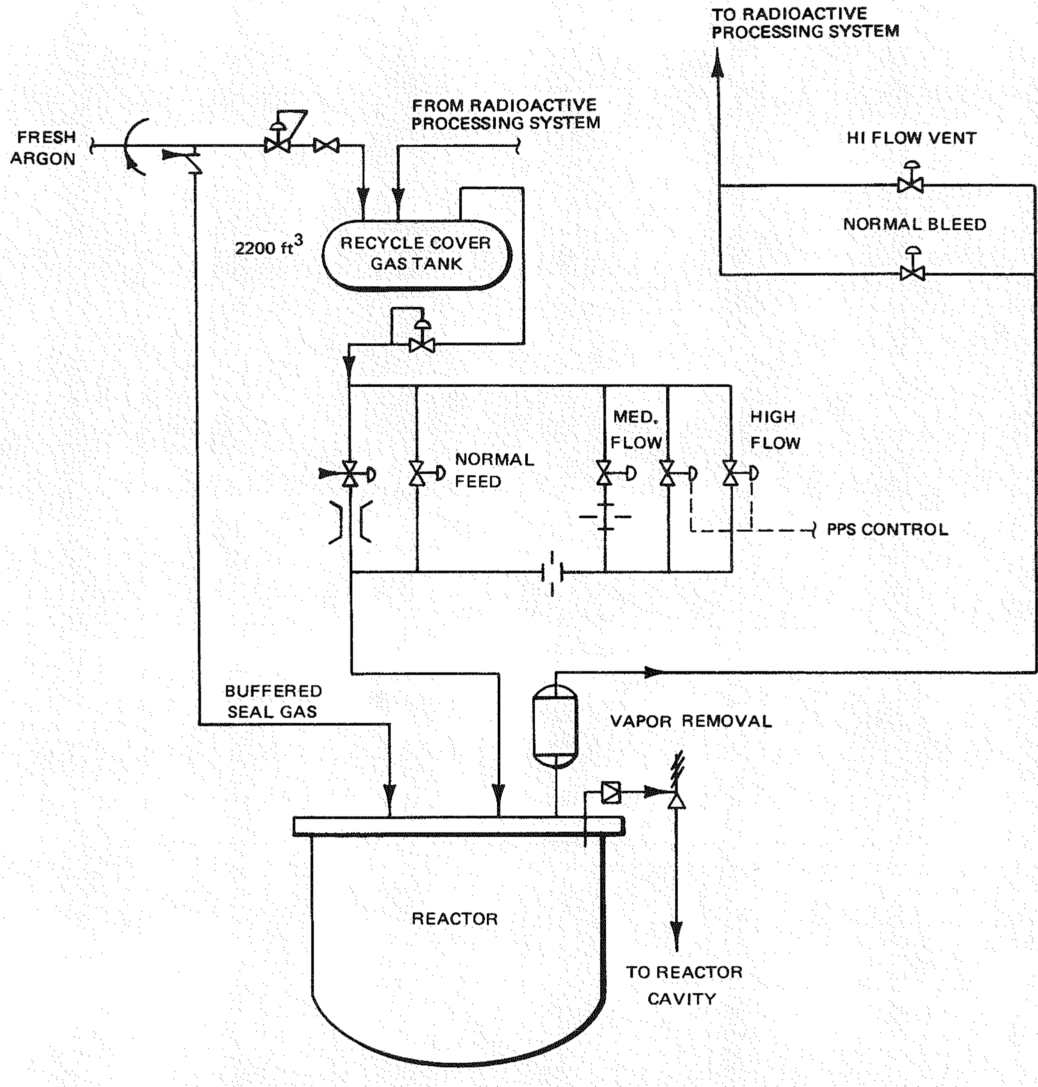


FIGURE 4-5. COVER GAS SYSTEM

During Extension 1 General Electric and CBI Nuclear studied two schemes for final fabrication and erection of the reactor components at the site. One involved partial fabrication in a field shop with final fabrication of these subassemblies within the reactor cavity. In this case, the maximum lift would be under 250 tons. The second scheme is based essentially on complete fabrication of the vessels (less internals) in the field shop, and handling and installing these components as complete units.

The study showed that the type of lifting and handling equipment adopted may also affect the construction sequence and schedule. For example, any lifting scheme which requires that construction of the reactor building structures be stopped at some elevation or point in time could cause schedule delays in the event the reactor components were late. For such reasons, several alternate handling concepts were studied. These included: 1) a large-lift mobile crane (250 ton capability) and a twin tower platform ringer crane as proposed by CBI Nuclear, 2) a gantry crane, and 3) a twin luffing derrick. The latter three are capable of handling the heaviest completed component, i.e., 700 to 750 tons. Of the above schemes, the twin luffing derrick appears to pose the least restraint on plant construction and has the potential to accept a reactor component delay of up to approximately 13 months. This concept is shown schematically in Figure 4-6. The layout of a twin luffing derrick should be further studied in a subsequent phase to determine the effect on construction operations and ensure adequate clearance for other structures.

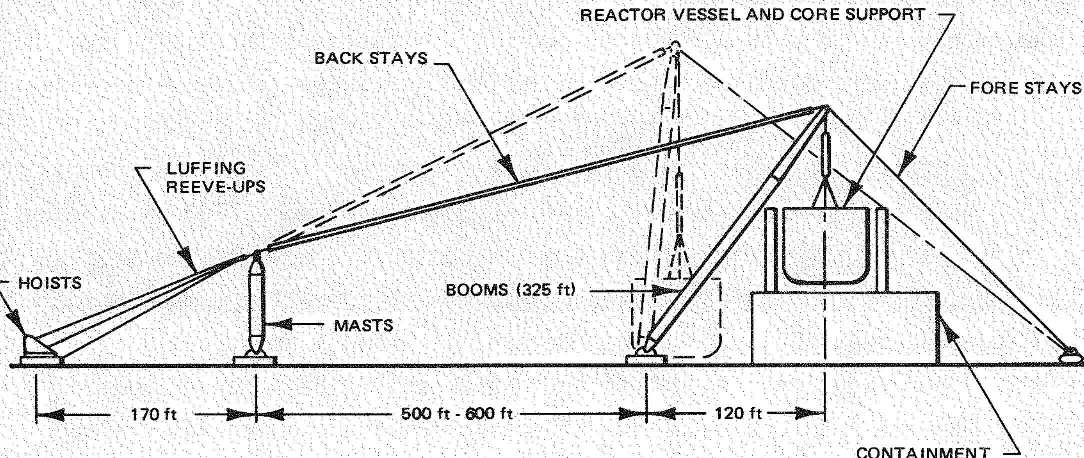


FIGURE 4-6. REACTOR VESSEL RIGGING SYSTEM

POOL-TYPE LMFBR PLANT

1000 MWe PHASE A — EXTENSION 1 DESIGN

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