Pool-Type LMFBR Plant 1000 MWe Phase A-Extension-2 Design

Volume 1: Executive Summary

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ELECTRIC POWER RESEARCH INSTITUTE

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1000 MWe Phase A-Extension-2 Design

PART I: EXECUTIVE SUMMARY

NP-1014-SY, Volume 1
Research Project 620-20, 21

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NOTICE

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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, and focused on a 1000 MWe (gross) loop-type plant. The design teams had completed preliminary conceptual loop-type plant designs by mid-1977, when ERDA withdrew as co-sponsor, and the PLBR program schedule became indefinite pending results of "non-proliferation" and other energy policy studies which were initiated by the new national administration and are still in progress.

EPRI then decided to take advantage of some of this interim time period to sponsor continuation of large LMFBR plant design studies by the contractor design teams, but with the studies re-directed to the pool-type plant in the light of continuing successful worldwide experience with this concept. The work was performed in three stages, and this is the third and final report which describes results obtained by the General Electric/Bechtel team during the course of this work on a 1000 MWe (gross) pool-concept plant with a saturated steam cycle. EPRI Reports NP-1015 and NP-1016 describe the final results of the parallel design studies of pool-concept plants by the Rockwell International (AI)/Bechtel team and the Westinghouse/Stone & Webster team.

Results of the EPRI-sponsored pool-concept studies are already being utilized in the Concept Design Study (CDS) program which was initiated by the federal Department of Energy (DOE) during the last quarter of 1978.

PROJECT OBJECTIVES

The project was initiated with the end objective of bringing the level of preliminary design of a pool-concept plant up to that of the loop-concept PLBR plant. The pool concept is outwardly characterized by the large diameter of the vessel and shield-deck structure which contain the reactor core and surrounding primary sodium system pool, and the compact arrangement of the intermediate heat exchangers and primary system components which extend from above the shield-deck down into the primary sodium pool, in close proximity to the reactor core. The design work was performed with major emphasis on the pool-unique design considerations of concern, such as:

- Seismic stability and constructability of the large diameter pool vessel, vessel internal, and shield-deck structures.
- Constructability and performance of in-sodium thermal barriers between hot and cold pools within the vessel.
- Thermal/hydraulic performance of large pool heat transport systems, and the influence of pool volume and pool flow mixing characteristics on thermal shock effects of transient events.
- Adequacy of accessibility for inspection, maintenance and replacement of equipment and components in the compact arrangement of a pool-concept plant.

The design work was therefore focused primarily within the containment building, with design of the Balance of Plant generally covered by carry-over adaption of features from the loop-concept PLBR plant preliminary design.

CONCLUSIONS AND RECOMMENDATIONS

The study effectively demonstrates the feasibility of the design and construction of a large pool-concept plant within current U.S. design and construction practices and licensing requirements, and indicates that multiple design options are available in many of the design areas that are unique to this type of plant. The identified additional design verification and development requirements are moderate and do not present any serious obstacles to the pool-concept. Although a conclusive comparative evaluation of pool vs. loop concepts has not yet been performed, it is evident that the pool-concept is a strong contender for selection for development of the LMFBR in the commercial size range.

Both the pool-concept and loop-concept designs are now at a stage where their constructability, operability, inspectability, and maintainability should be reviewed and assessed with a power plant owner/operator viewpoint. The review will provide a more substantial basis for comparative evaluation of the pool and loop concepts, and will also contribute to better focusing of technical direction and priorities in follow-on design efforts.

At the time of this writing EPRI has already taken the initial step of issuing a request for proposals to perform such a review and evaluation.

James G. Duffy, Project Manager
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ABSTRACT

A 900 MWe (net) pool-type LMFBR 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. A major guideline specified the plant to have an 875°F reactor outlet temperature and 550°F, 1000 psi steam with the design effort concentrating on areas of the plant unique or especially important to the pool concept. These areas include the reactor deck; the reactor assembly and its internal structures, the thermal behavior of the plant during steady state and transients; the auxiliary heat transport systems; the primary sodium pumps; the intermediate heat exchangers; the plant seismic response; maintenance and inspection; 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 2 work performed between August 1 and December 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 complete the conceptual plant design established during Phase A and Phase A Extension 1 during the period from April, 1977 through July, 1978.

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. Additionally, 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 results of the Phase A Extension 1 work was published by EPRI in September 1978 in report number NP-882, "Pool-Type LMFBR Plant, 1000 MWe Phase A-Extension-1 Design."

During Phase A Extension 2, funded at a level of approximately 1.4 million dollars, the reactor assembly, the reactor deck, heat transport systems, auxiliary systems, seismic analysis, maintenance studies, construction studies, safety studies and the balance-of-plant design were all brought to levels consistent with a completed concept. Assistance in several specialized areas was supplied by subcontractors; Chicago Bridge and Iron provided a fabrication scheme for the thermal barrier design, Foster-Wheeler developed the IHX Design, Bryon-Jackson continued work on the primary sodium pump design, Engineering Decision Analysis Corporation verified and further developed the seismic studies, and CBI Nuclear studied construction of alternate vessel supports and fabrication of the upper internals structure.

"Phase A Extension 1" and "Phase A Extension 2" are frequently shortened to "Extension 1" and "Extension 2" when referred to in this report.

This report of the Phase A Extension 2 work is logically divided into thirteen parts, which have the general title "Pool-Type LMFBR, 1000 MWe Phase A - Extension-2 Design:"

Part I	Executive Summary
Part II	Plant Summary Description
Part III	Reactor Assembly
Part IV	Reactor Deck
Part V	Heat Transport Systems
Part VI	Auxiliary Systems
Part VII	Plant Control and Instrumentation
Part VIII	Seismic Analysis
Part IX	Constructibility and Fabricability

Part X	Maintainability and Inspectibility
Part XI	Safety
Part XII	Balance of Plant - Plant Description
Part XIII	Balance of Plant - Evaluation of Pool-Related Areas

The report is physically divided into twelve volumes as follows:

Volume I	Part I
Volume 2	Part II
Volume 3	Part III, Sections 1 through 2.5
Volume 4	Part III, Sections 2.6 through Appendix IIIB
Volume 5	Part IV
Volume 6	Part V
Volume 7	Parts VI, VII and VIII
Volume 8	Parts IX, X and XI
Volume 9	Part XII
Volume 10	Part XII Appendices
Volume 11	Part XIII
Volume 12	Part XIII Appendices

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

Pool-Type LMFBR Plant

1000 MWe Phase A-Extension-2 Design

Part I: Executive Summary

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FIGURES

1. INTRODUCTION

The major, long-range goal of the LMFBR program in the United States is to create a practical substitute for coal, gas, oil and U-235 in making electricity. There are a number of characteristics that appear essential in achieving this goal, among them high availability, low power generation costs, practicality and safety. Although the pool-concept LMFBR was first conceived in the United States with the 60 MWe ERR-II plant, which became operational ten years ago, the steady progression of research, development and design efforts towards these goals in the United States has mainly involved loop-type plants such as the Fast Flux Test Facility, the Clinch River Breeder Reactor Plant, and the Prototype Large Breeder Reactor designs. On the other hand, foreign pool-type plants in the several hundred MWe range, such as the British Prototype Fast Reactor and the French Phenix have already been built and operated. Additionally, the French 1200 MWe Super-Phenix plant is presently under construction and the British 1300 MWe Commercial Demonstration Fast Reactor is being designed. It is clear that pool-type plants represent viable concepts but the United States, at the beginning of 1977, had no advanced pool-plant design. Accordingly, EPRI elected to critically examine the pool concept for large plants and bring it to a level of development equivalent to that of the loop concept.

The main thrust of this study was to examine the impact of features unique to the pool concept. Features expected to have a significant impact on design, construction, maintainability and costs included the large diameter reactor/pool vessel, and the large, deep deck which must support the primary sodium pumps, the IHXs, and the control rods without excessive seismic deflection. Also included are the under-deck insulation, the thermal barriers which must insulate the hot sodium pool from the vessel wall and from the primary sodium pumps, and thermal-hydraulic phenomena in the large hot and cold sodium pools, especially when cold sodium enters the hot pool after a reactor scram. Significant findings, conclusions, and remaining questions in these and other areas are highlighted in the remainder of this volume. The work performed to date shows the large pool-type LMFBR to be a feasible and promising option for use in the United States.

For convenience, a summary description of the plant, a detailed plant parameter list, and the major plant and component drawings are collected into Volume 2 of this report.

2. PLANT DESCRIPTION

This design study deals with a pool-concept LMFBR having a gross power of 1000 MWe, a reactor outlet temperature of 875°F, a temperature rise through the core of 280°F, a heterogeneous core, a cylindrical containment building, and 1000 psia saturated steam driving one light-water-reactor-type turbine generator. EPRI guidelines also specified four primary pumps, all taking suction from a common cold sodium pool, and six intermediate sodium loops. The six intermediate heat exchangers (IHXs) are fed by primary sodium coolant by gravity flow from the hot sodium pool where the free surface is 7 feet higher than that of the cold sodium pool when the pumps operate at full flow.

In addition, choices in the fuel handling area included: a triple rotating plug with a straight-pull, in-vessel fuel transfer machine; inclined fuel transfer to a shielded transfer cell; and ex-containment spent fuel storage. These conditions were selected to provide a high assurance of plant reliability and safety with minimal dependence on research and development.

The balance of plant design is based on a circular containment building within a rectangular confinement structure surrounded by steam generator buildings, a service building, an auxiliary and control building, and a peninsular turbine building. The major weights, dimensions and plant operating parameters are listed in Table 1. The general arrangement of the buildings and structures is illustrated in Figure 1.

The reference plant design is illustrated in Figures 2, 3,4 and 5. Both the reactor and the primary sodium heat transport system, including the four primary pumps and six IHXs, are placed within the primary (reactor/ pool) vessel which is hung from the reactor deck. A rotating plug assembly supports the control rods and the upper internals structure, and positions the in-vessel fuel transfer machine. This rotating plug assembly is surrounded by the IHX and pump penetrations. The core is supported from the primary vessel wall by a structure placed high in the vessel near the core's center of gravity in order to minimize seismic deflections. Above the core is the hot sodium pool with ten vertical, cylindrical penetrations for the IHXs and the primary pumps arranged on a circle around the centrally located upper internals structure (UIS).

TABLE 1

Major Design and Performance Characteristics

Nominal Plant Rating	2900 MWt/1000 MWe Gross
Electrical Output	922 MWe Net
Plant Net Efficiency	31.7%
Steam Conditions	Saturated, 1000 psia/554°F
Operational Mode	Base Loaded
Primary System Configuration	Hot Pool/Cold Pool
Reactor Outlet/Inlet Temperature	875°F/595°F
Number of Primary Sodium Pumps	4, Cold Pool Suction
Number of IHXs	6, Gravity Flow from Hot Pool
Number of Steam Generators	6
IHTS Hot/Cold Leg Temperature	815°F/595°F
PHTS Pump Flow Rate	68,000 gpm
IHTS Loop Flow Rate	46,000 gpm
Core Concept	Radially Heterogeneous
Fuel Handling Concept	Triple Rotating Plug with Straight Pull In-Vessel Transfer Machine
Fuel Transfer Concept	A-Frame in Shielded Transfer Cell
Fuel Storage Concept	Ex-Containment (plus 24 storage positions in the reactor vessel)

TABLE 1 (continued)

Primary Sodium Cleanup System Location	Entirely within the Reactor Vessel
IHTS Loop Isolation Method	Loop Isolation Valves
Decay Heat Removal Concept	Two Independent and Diverse Sytems; Both Reject Heat from Primary Hot Pool to Atmosphere
Containment Building	Cylindrical, Stepped Step elevation - 30 ft, Lower Diamter - 104 ft, Upper Diameter - 170 ft
Containment Building Polar Crane Capacity	500 Tons
Turbine	Tandem-Compound, Four-Flow, 1800 rpm
Gross Generator Output	1000 MWe
Net Turbine Heat Rate (5% Blowdown)	9,974 Btu/kWh
Turbine Heat Rejection	Mechanical Draft Cooling Towers
Primary (reactor/pool) Vessel	Diameter - 74 ft, Height - 75 ft, Wall thickness - 1 in.
Reactor Deck	Outer Diameter - 82 ft, Inner Diameter - 33 ft, Depth - 11 ft
Reactor Deck Penetrations	IHX - 11 ft Pump - 8 ft Pitch Circle - 53 ft

The plant operating parameters include a nominal reactor outlet temperature of 875°F but there are margins in the plant design to account for uncertainties and the rated plant power is expected to be achieved with a reactor outlet temperature of 860°F.

A novel feature of the reference design for the pool-type plant is the stepped circular containment configuration illustrated in Figure 6. The step-change in containment diameter occurs at 30 feet above the reactor deck. Above this level, the inside diameter of the containment is 170 feet while the lower part is 104 feet. This allows removal of major components in a vertical position through an equipment hatch at the operating floor level. Below the operating floor level the smaller containment diameter minimizes intermediate sodium piping runs within containment.

The IHXs or primary pumps and their associated casks can be transferred by the containment building's polar crane, but the UIS, if it were to be encased in a cask and removed whole, may require a crane of greater capacity. While the UIS repair options investigated include providing more crane capacity, possible options using the reference crane have also been developed such as removing the UIS piecemeal.

A heterogeneous core has been specified as a design basis because of the low reactivity introduced if sodium boils or voids. This characteristic, which is expected to eliminate energetic core accidents as a licensing issue, is achieved by interspersing fuel assemblies with some blanket assemblies as shown in Figure 7. The hexagonal fuel, blanket and control assemblies, and shielding are contained within an 18 foot core barrel. This dimension affects the size of the rotating plug assembly above the core, affects the placement of the six IHXs and four primary pumps around the rotating plug, affects the diameter of the primary vessel, and thus partially determines the 82 foot diameter of the deck structure.

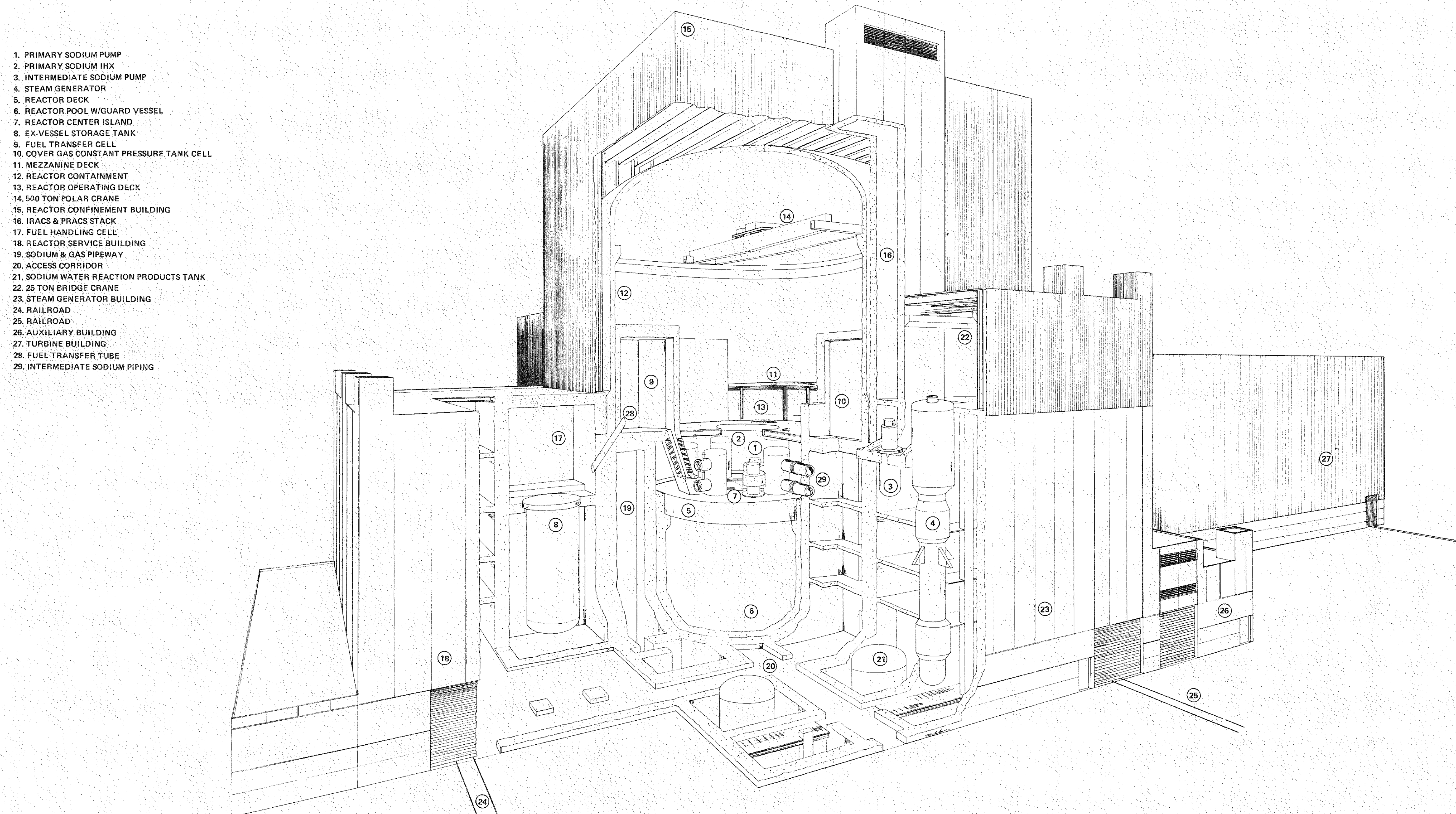


FIGURE 1. GENERAL ELECTRIC-BECHTEL 1000MWe POOL-TYPE LMFBR PLANT

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

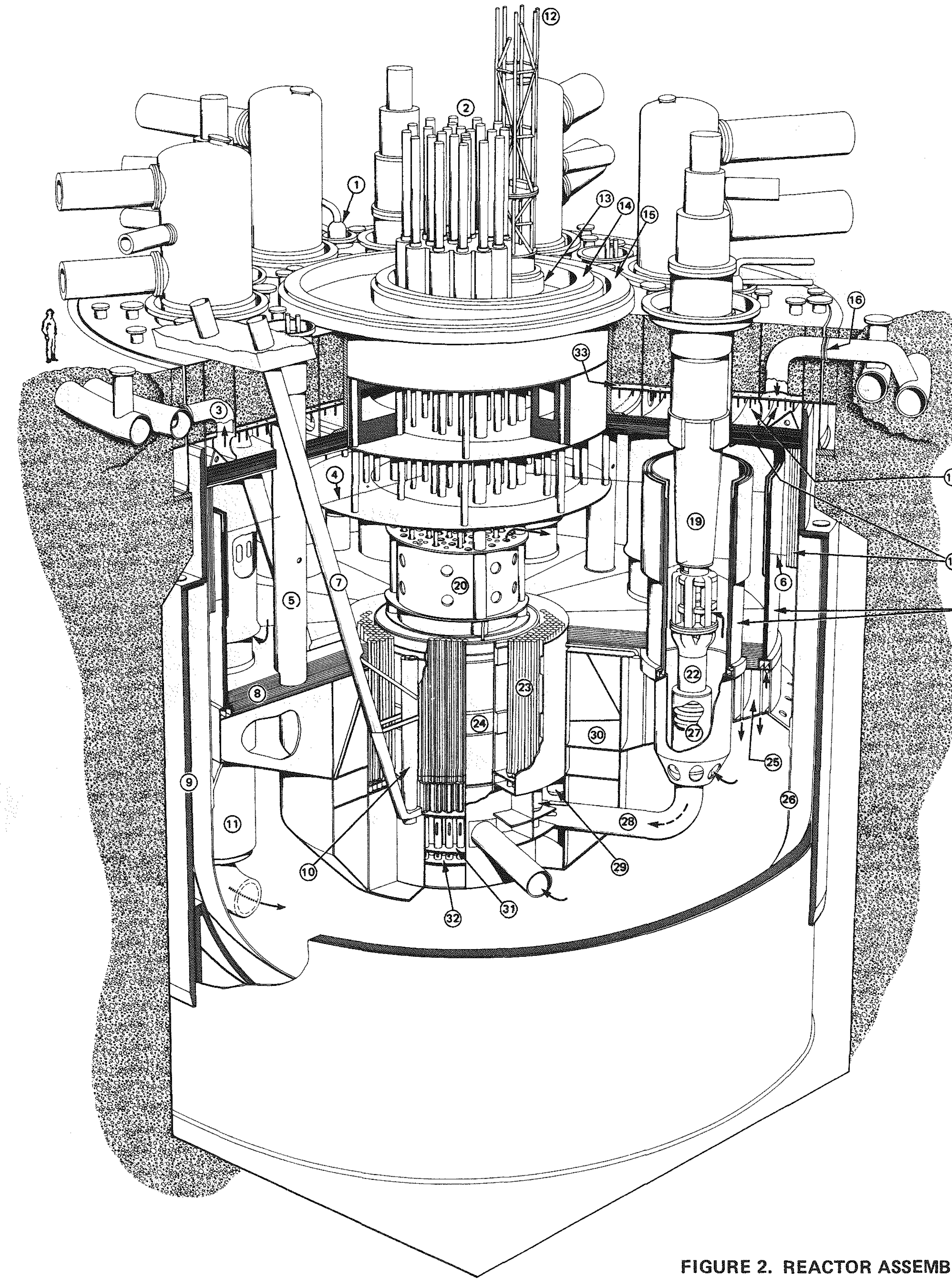


FIGURE 2. REACTOR ASSEMBLY

(1) TOP OF PRIMARY SODIUM PUMP (MOTOR REMOVED)
(2) PRIMARY PUMP MOTOR
(3) INTERMEDIATE HEAT EXCHANGER (IHX)
(4) IHX ENCLOSURE
(5) INTERMEDIATE REACTOR AUXILIARY COOLING SYSTEM (IRACS) PIPING
(6) CONTROL ROD DRIVES
(7) PRIMARY REACTOR AUXILIARY COOLING SYSTEM (PRACS) PIPING
(8) TOP OF IN-VESSEL HANDLING MACHINE (IVHM)
(9) AUXILIARY HANDLING MACHINE (AHM) RESTING IN CRADLE
(10) AHM POSITIONER
(11) FUEL TRANSFER CELL
(12) GAS COOLING LINES
(13) EQUIPMENT HATCH

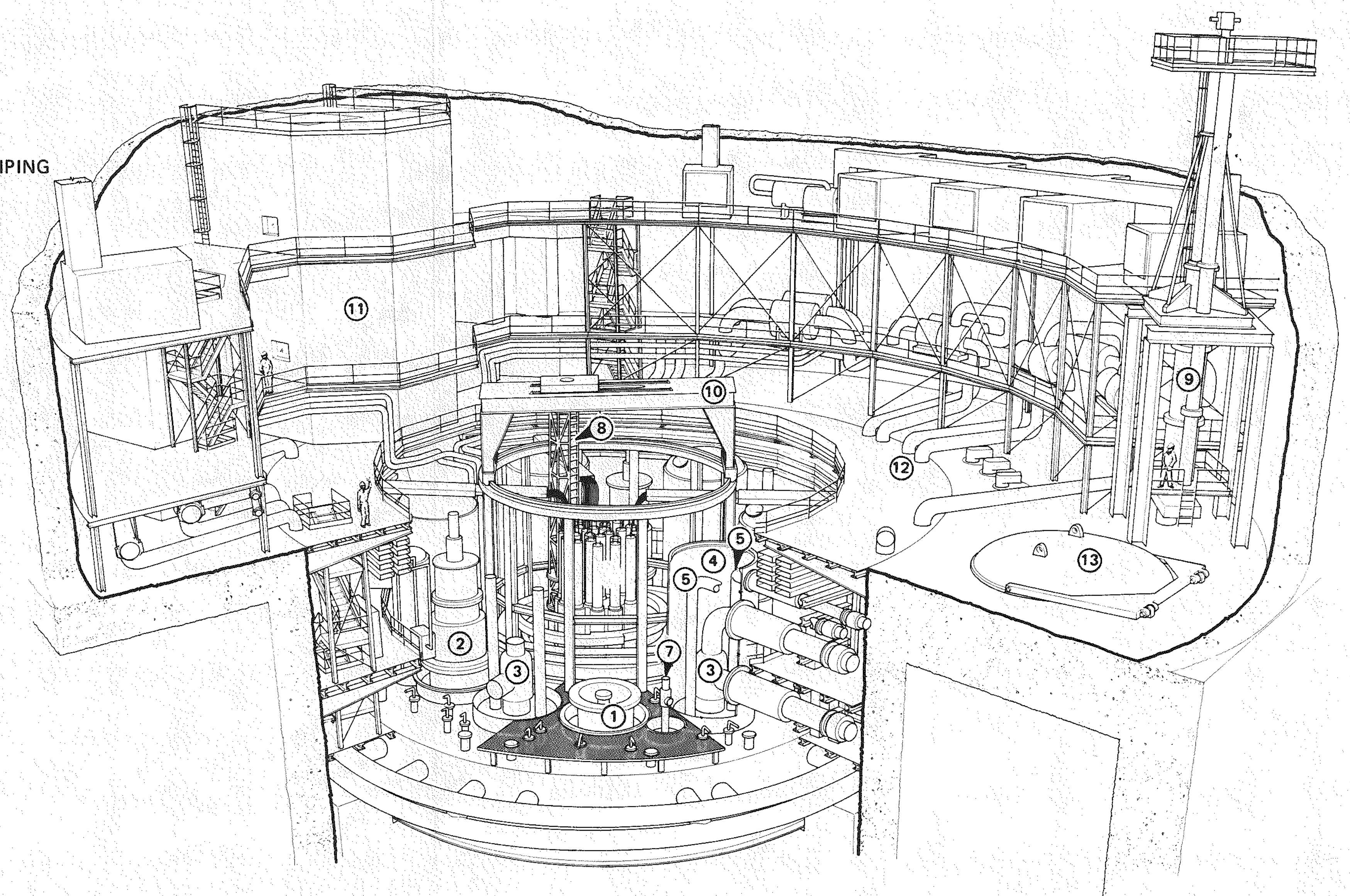


FIGURE 3. HEAD ACCESS AREA

This photograph of a 1/32 scale model of the reactor assembly and head access area realistically illustrates many of the items previously identified in Figure 3. In this model the floor and walls are transparent to reveal interior details. The human figure in the upper left quadrant illustrates the true size of the surrounding structures.

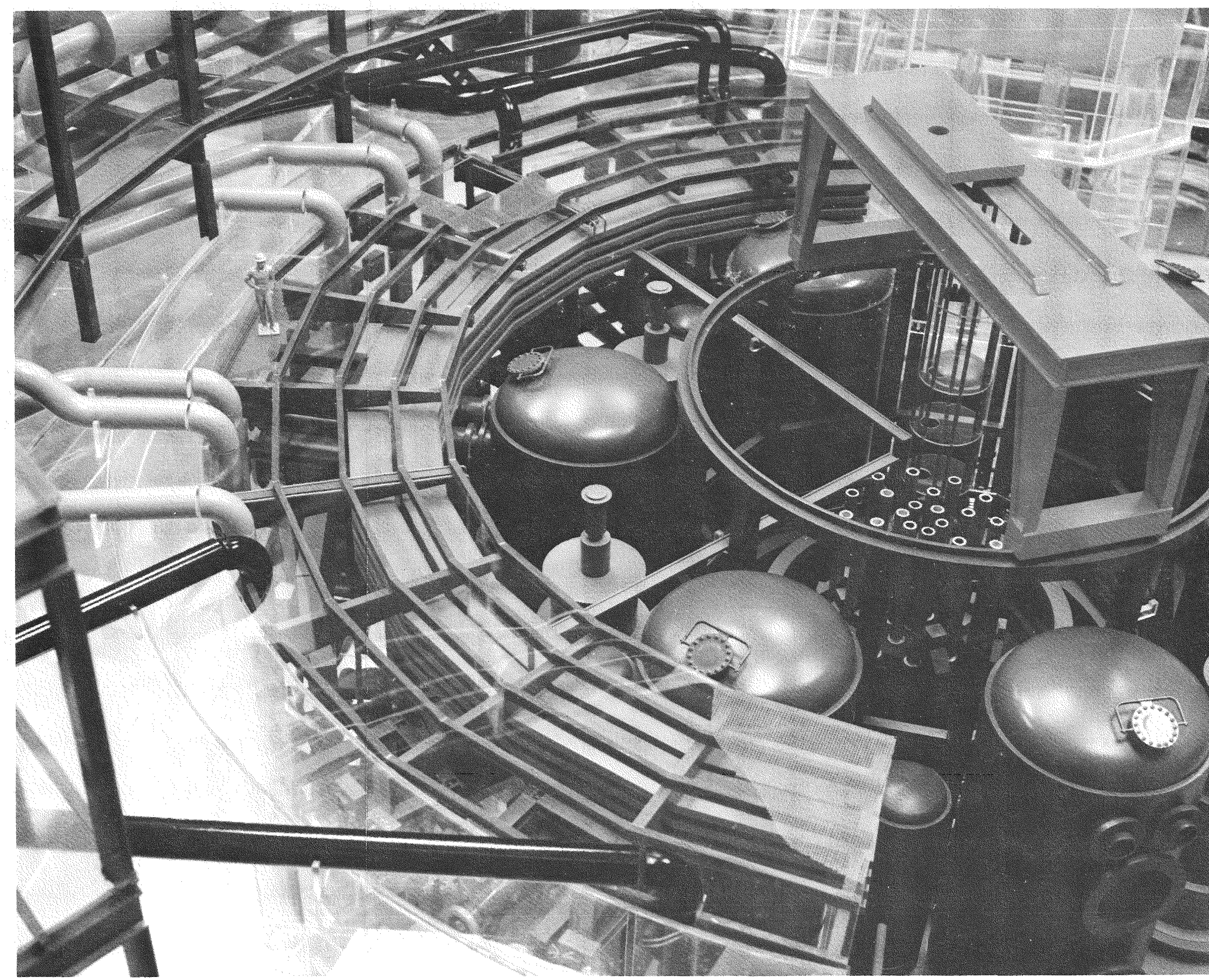
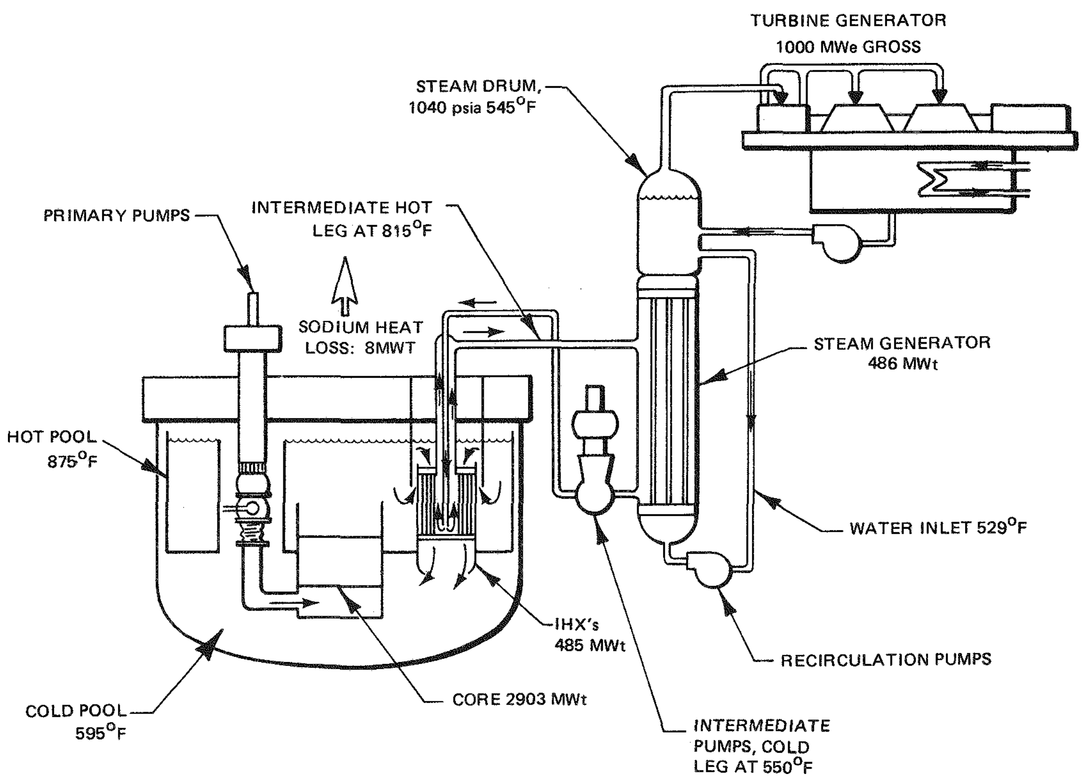


FIGURE 4. REACTOR MODEL



1000 MWe POOL-TYPE LMFBR

NET PLANT POWER	922 MWe	
NET PLANT EFFICIENCY	31.7%	
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 5. PLANT OPERATING PARAMETERS

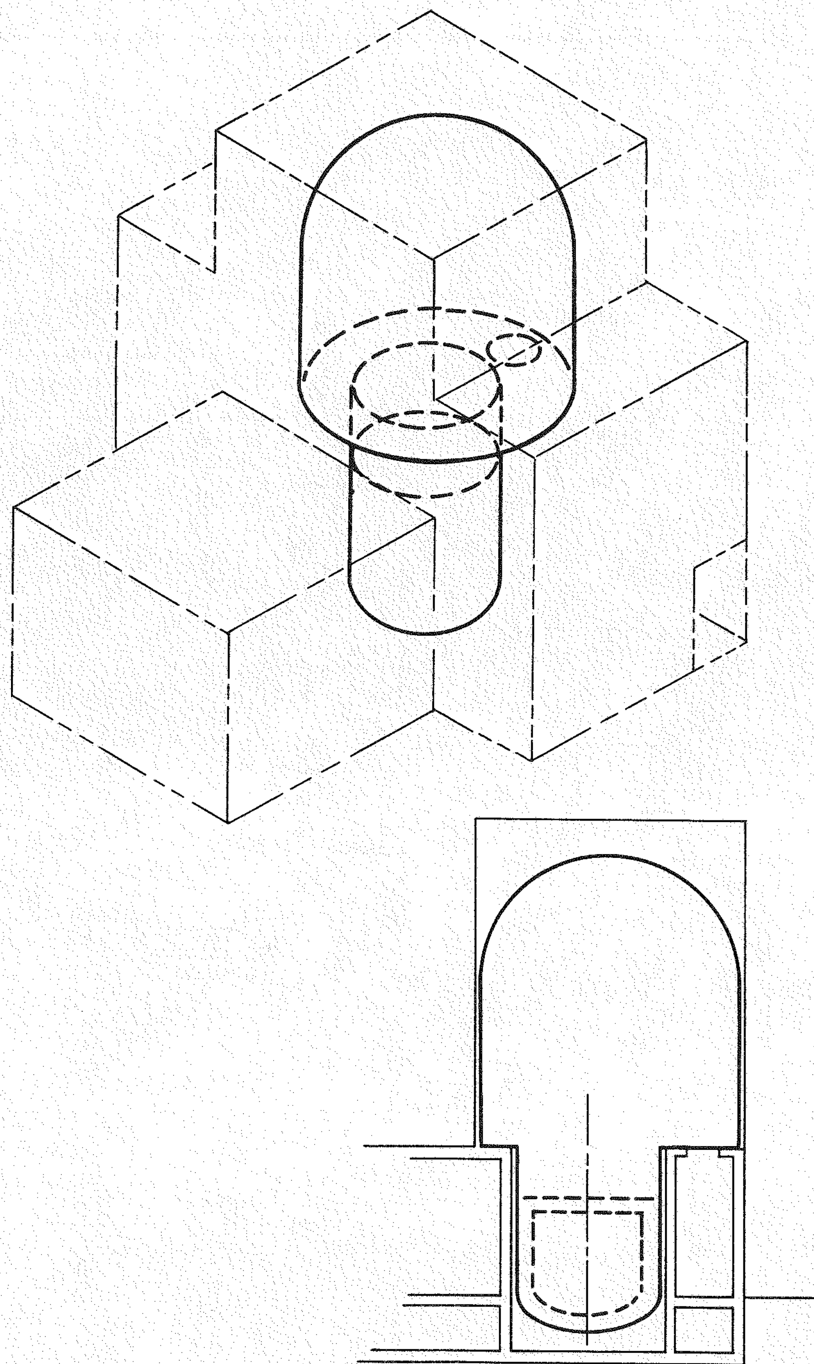
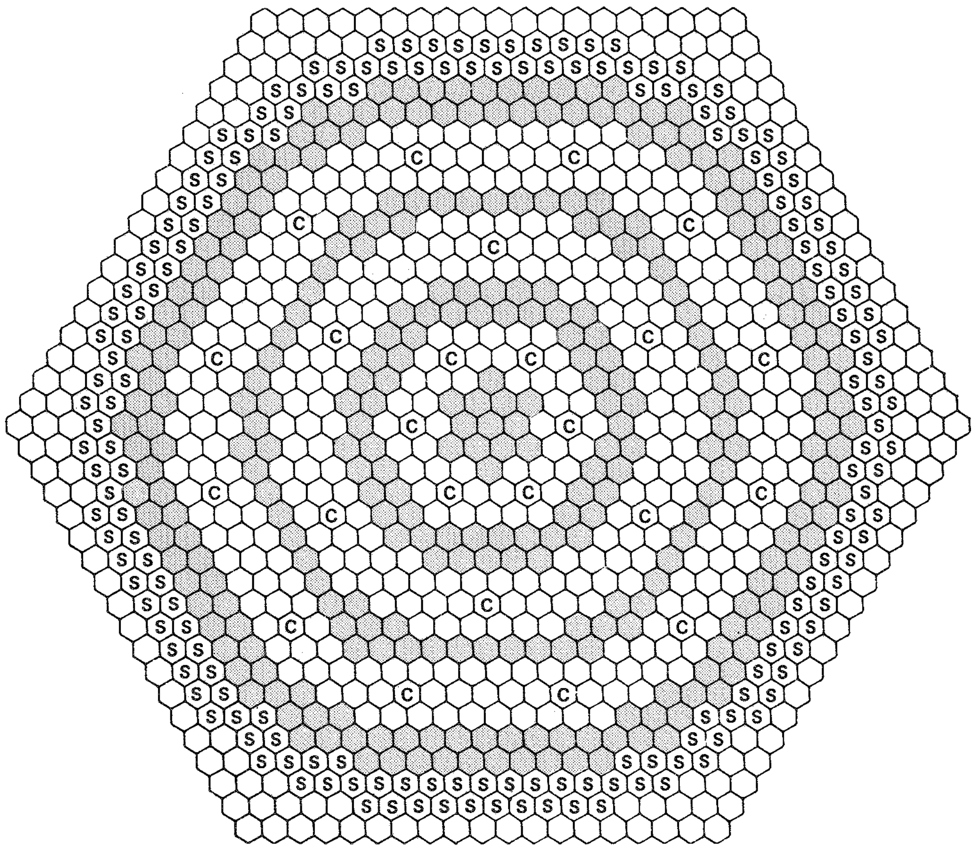


FIGURE 6. STEPPED CONTAINMENT CONCEPT



1. BLANKET ASSEMBLY
   145 INNER BLANKET ASSEMBLIES
   162 OUTER BLANKET ASSEMBLIES
2. FUEL ASSEMBLY NUMBER = 360
3. S = REMOVABLE SHIELD NUMBER = 180
4. C = CONTROL ROD ASSEMBLY NUMBER = 24
5. HEXAGONAL MODULE CENTER-TO-CENTER = 6.045"
6. DIAMETER OF CIRCUMSCRIPT CIRCLE AROUND REMOVABLE SHIELD ASSEMBLIES = 16.2'
7. HEIGHT FROM SEAT TO TOP = 181"

FIGURE 7. HETEROGENEOUS CORE LAYOUT

3. SIGNIFICANT FINDINGS

Many areas of the plant unique to the pool concept have been conceptually resolved and these are discussed below. There remains some work to be done, generally consisting of experimental testing to confirm assumptions made in the conceptual design. Such work is discussed in the next Section.

Deck Structure

The reactor deck, illustrated in Figures 8 and 9, is a dominating structure in a pool reactor. It has a large diameter (82 ft.) in order to support the reactor vessel, core, primary sodium pumps, IHXs, control rods, above-core structure, etc., and in order to provide refueling access. It must be rigid enough to minimize displacement of control rods relative to the core during a seismic event. It must accommodate a transition from the 875°F, high-radiation environment below the deck to radiation levels and surface temperatures above the deck that are compatible with man access.

These requirements are satisfied by a circular box-beam structure similar to that of Super-Phenix. Constructed almost entirely of 1-1/2 inch mild steel plate, the radial and circumferential shear webs surround a central opening for the rotating plug system. Such a design meets the EPRI guideline that the central opening deflect under static load no more than 1/4 inch. This design has low stress levels at its deflection limit and provides adequate margin against excessive deflection during a seismic event.

The stressed-skin, circular box-beam deck structure offers the advantage of the multiple shear webs which create such a high degree of redundancy that no single weld failure, or any credible combination of multiple weld failures, could compromise deck integrity. The depth of the deck is sufficient to allow the nitrogen cooling system in the bottom region and the radiation shielding in the top region to provide comfortable temperatures and low enough radiation levels that the deck surface is accessible for a normal 40-hour week during full power operation.

Temperature control of the deck is accomplished by nitrogen gas directed downward onto the deck's bottom skin, a common technique used, for instance, to control the temperatures of glass being formed into large sheets. In addition,

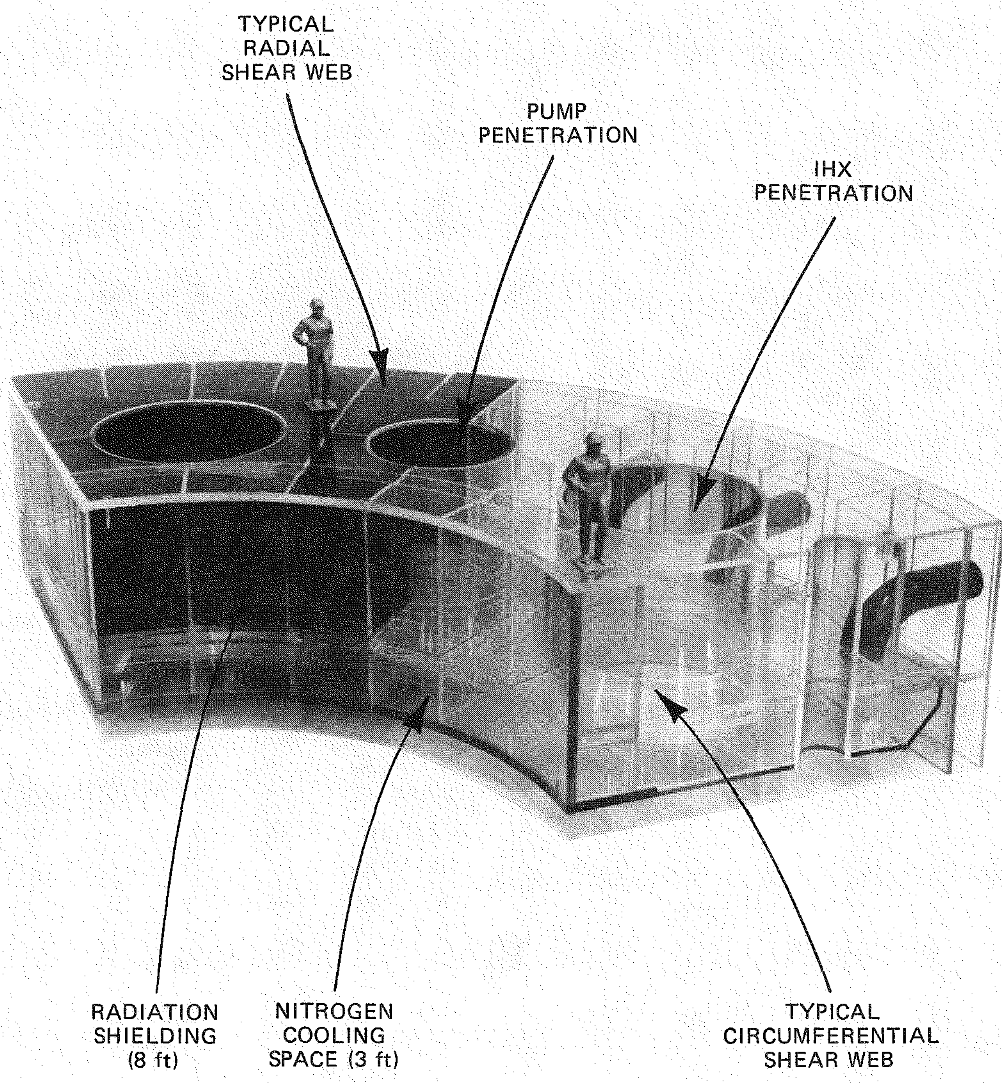


FIGURE 8. REACTOR DECK QUADRANT

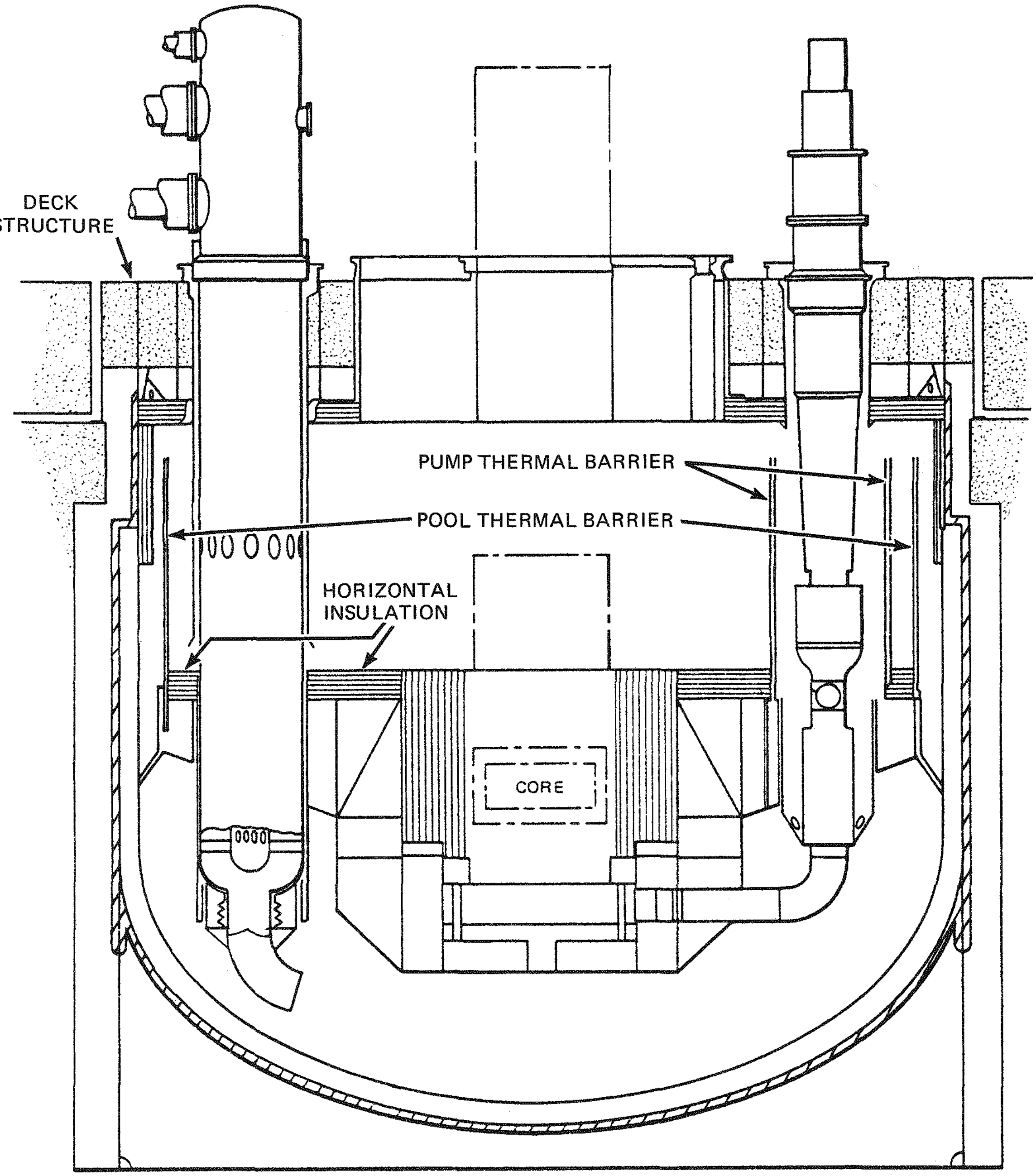


FIGURE 9. REACTOR DECK STRUCTURE AND THERMAL BARRIER LOCATIONS

the bottom skin of the deck is insulated from the hot argon cover gas atmosphere and radiant heat above the hot sodium pool by multiple barriers of perforated plates and woven screen mesh. In this manner, the deck temperatures are controlled to lie between 100°F on the bottom skin and an average of 80°F on the top surface. Even if the nitrogen cooling system were to fail, it would take 100 to 150 hours for the deck bottom skin to reach 200°F and the deck top surface to reach 100°F, a equilibrium condition which could then be maintained by the containment building's heat removal system.

Thermal Barriers

The stainless steel reactor/pool vessel wall must be kept at a temperature below its creep range during plant operation. This is accomplished by allowing only the cold-pool sodium to contact the vessel wall. The hot sodium pool is completely surrounded by a layer of cold-pool sodium and the transfer of heat from the hot pool to this cold-pool layer is minimized by a large, cylindrical, vertical thermal barrier between them. The primary pump penetrations inside the hot pool are each surrounded by smaller, cylindrical, vertical thermal barriers similar to the much larger vertical barrier protecting the vessel wall. Adequate insulation at the bottom of the hot pool is provided by a passive barrier consisting of a stack of horizontal plates to break up convection currents and minimize heat transfer. This arrangement is illustrated in Figure 9.

It is clear that limiting heat transfer to the vessel wall must include an active system somewhere because the layer of cold-pool sodium is stagnant and will slowly heat up in the absence of heat removal or forced circulation of colder sodium into the stagnant region. The reference design includes the active system in the vertical thermal barrier itself, and accomplishs the necessary heat removal from the inside of the cold-pool layer protecting the vessel wall. The vertical barriers are formed of three concentric cylinders enclosing two annular gaps as shown in Figure 10. Cold sodium flows up the inner annulus and down the outer annulus. Another cylindrical wall inside the main inner wall traps stagnant sodium on the hot-pool side. The colder, entering sodium is initially heated by the hot pool and by the warmer, exiting sodium in the lower part of the barrier, and reaches its maximum temperature near the top of the barrier.

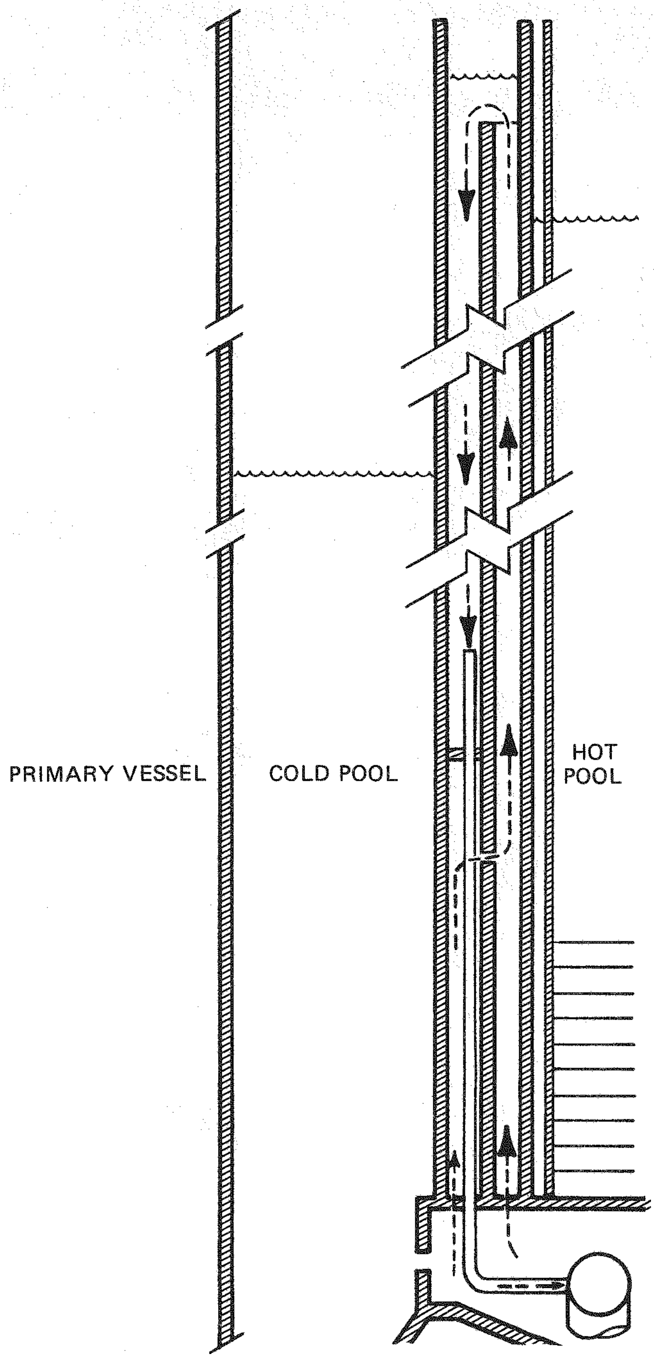


FIGURE 10. ACTIVE THERMAL BARRIER

A desirable feature of this design is that the upper 7 feet of the barrier, extending above the cold pool and contacting only the hot pool, has no sharp temperature difference with respect to the lower part of the barrier.

The cold sodium supplied to the thermal barrier is taken from the cold pool and returned to the cold pool, the flow being derived from the primary pumps. In this manner the total heat load transferred from the hot pool to the cold pool by the vertical thermal barriers amounts to 16 MWt and raises the cold pool temperature by about 2°F. Because this heat is not dissipated externally it has minimal impact on plant efficiency.

Cold sodium is supplied to the barrier by the primary sodium pumps which would normally be kept running, at least at significant partial flow, during the few hour decay heat removal period following a reactor scram. After the decay heat removal period, pony motor flow would be established. Even if the primary pumps were to be tripped to pony motor speeds upon scram the vessel wall temperature would not rise more than 20°F. If all steam generators were not available for heat removal, a happening not expected to occur in the plant lifetime, the burden of vessel wall temperature control would be shifted over to the reactor auxiliary cooling system.

<u>Reactor Auxiliary Cooling System (RACS)</u>

In case the normal heat removal paths through the steam generators is not available, the RACS must act to prevent excessive core and vessel temperatures from decay heat following a reactor shutdown. The RACS is intended to be used mainly as a backup safety system to assure core cooling and events leading to use of the RACS are few in number. The RACS would be used, for instance, in case all off-site power were lost and, after a few hours, stored water supplies feeding the steam generators were depleted. Other cases requiring the RACS include loss of feedwater to the steam generators or a main steam line break. All told, it is expected that the RACS will be needed about once every two years.

Because the RACS is a safety system it must be highly reliable. Following EPRI guidelines, this is accomplished in the reference plant design by having two independent, diverse, redundant systems. One of these systems removes heat directly from the primary sodium by means of heat exchangers located in the hot sodium pool and is referred to as a primary reactor auxiliary cooling system (PRACS). The other system also removes heat directly from the primary sodium but has its heat exchangers located in the upper parts of the IHXs and is referred to as an intermediate reactor auxiliary cooling system (IRACS). Both systems are illustrated in Figure 11. While the three-loop PRACS is an active system which uses forced circulation of a liquid sodium-potassium mixture to transport heat from the hot sodium pool to the outside air, the six-loop IRACS is a completely passive system. The IRACS uses a pool-plant feature where a natural convection path exists when hot-pool sodium is cooled in the top of the IHX and falls by gravity through the IHX into the cold sodium pool.

Either five 6.8 MW IRACS loops or two 17 MW PRACS loops are each separately capable of keeping the vessel wall below 1100°F upon immediate loss of all steam generator water inventories, and below 900°F if a four hour supply of stored water is available to only three of the six steam generators. The higher temperature limit is important because it is judged that a critical requirement for licensing will be a vessel wall temperature never exceeding 1100°F under any credible set of circumstances. The 900°F temperature limit is important from an economic point of view because above this temperature it may be necessary to requalify the vessel for further use.

Operability

A plant based on the pool concept offers unique flexibility when operating with components out of service. Essentially, this flexibility arises because the primary sodium pumps are not linked in a one-to-one correspondence with the intermediate sodium loops. This allows, for example, all six intermediate loops to operate while only three of the four primary sodium pumps are running, and all four primary pumps can run while only five intermediate loops are available. With five intermediate loops available, the core power is about 83% of full power, a value expected by simple proportions. But three out of four primary pumps in operation provides 94% of full power, a result arising from the fact that reduced flow resistance allows the three remaining pumps to "runout" and operate at 15% above their normal flow.

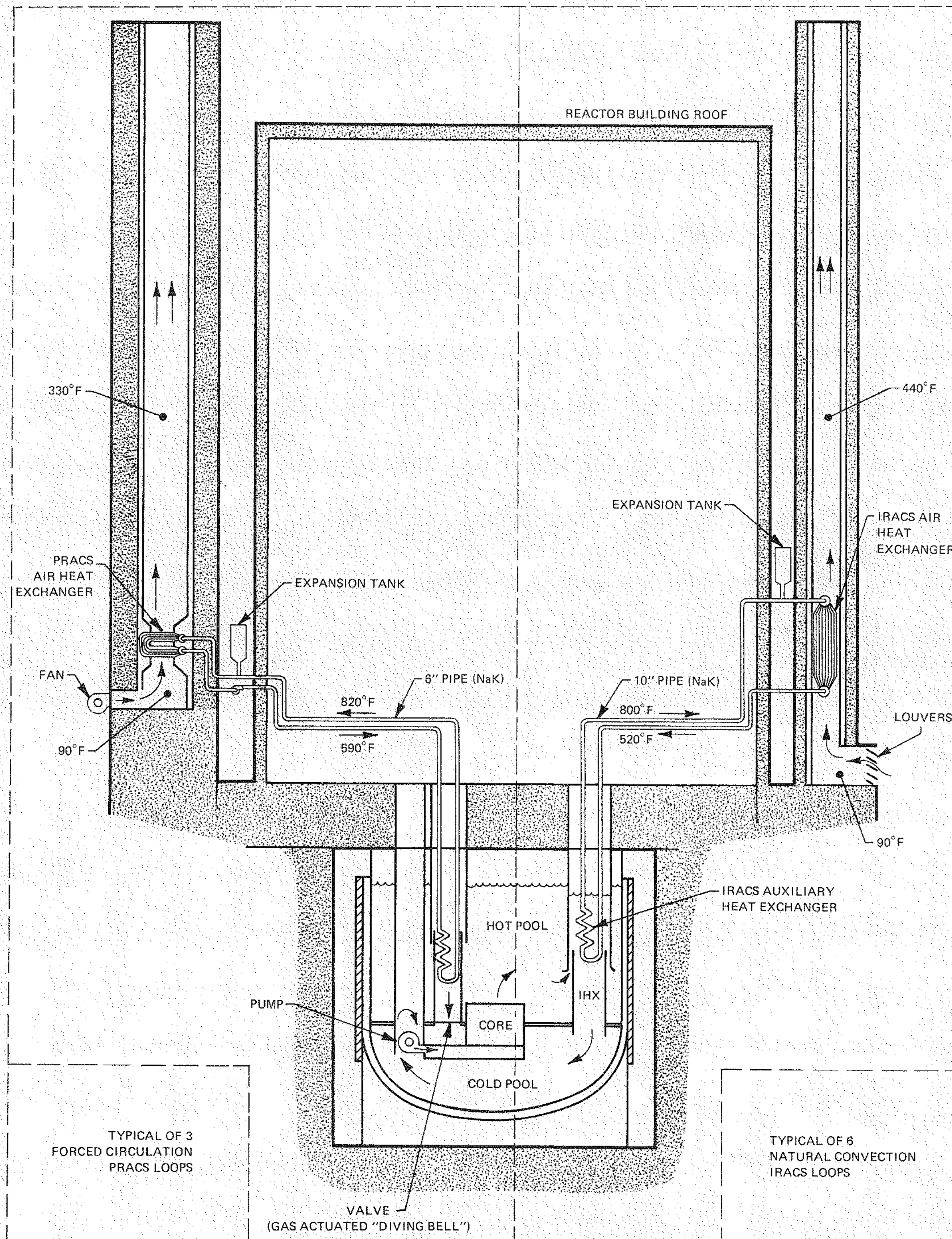


FIGURE 11. REACTOR AUXILIARY COOLING SYSTEMS

Maintainability and Inspectability

All welds in the lower part of the deck structure can be inspected in the three-foot high crawl space used for the nitrogen cooling system. The upper eight feet of deck can be made accessible by the use of removable shielding. In addition, there are access ports in the deck perimeter for inspecting primary vessel and guard vessel welds. The plant design also provides for syphon-draining of the primary system sodium if necessary for inspection or maintenance of the primary pool vessel internals.

The pool concept has the primary coolant system pumps and the IHX's located in the primary vessel deck in close proximity to each other and to equipment and control rod drives mounted on the rotating plugs. Therefore, the pool concept involves unique accessibility considerations with respect to the removability and relaceability of the pumps and IHX's, and also with respect to major in-vessel components which should be replaceable if necessary (i.e., the UIS and the core barrel structure).

A primary objective of the preliminary studies of the maintenance procedures and equipment is to provide initial assurance that the design arrangements and spacing of deck mounted equipment does provide adequate accessibility for anticipated replacement operations on both of the deck mounted equipment and the major removable in-vessel structural components.

The removal of a major component is performed using a transfer cask to maintain the component in an inert atmosphere and at the same time provide radiation shielding for maintenance personnel. It is this operation that determines the reactor containment building crane height and load requirements when the largest and heaviest component, the IHX, and its cask are removed. As shown in Figure 12 the IHX is first removed vertically, then horizontally as it passes alongside the reactor in the lower level of the reactor building. A primary pump, however, can be removed in a vertical position at all times.

The compact arrangement of equipment in the head access area of the pool concept should not be a serious impediment for maintenance operations that have been properly anticipated in the design process, but it does limit capability for later accommodation of un-anticipated space-consuming situations. It may also incur some penalty in time and cost relative to the loop concept,

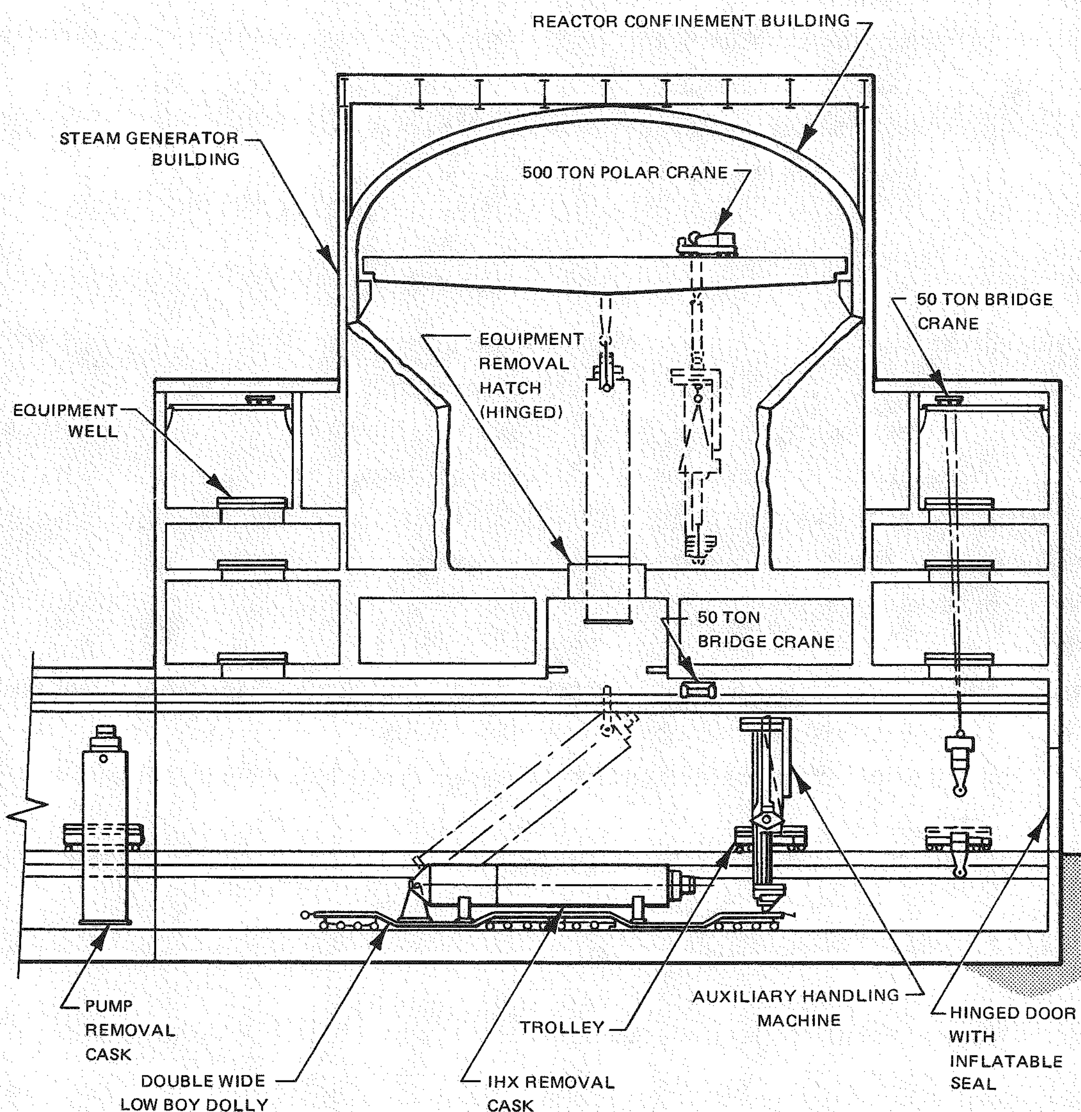


FIGURE 12. COMPONENT REMOVAL

especially if it prevents over-lapping of re-fueling operations and replacement maintenance operations on major primary coolant system equipment items that would be possible in a loop concept plant. Further thorough and early verification of this aspect of the plant design, including use of large-scale head access area mock-ups and critical feature testing of major maintenance equipment items, should be pursued with high priority as the design effort proceeds.

Fabricability and Constructibility

A construction schedule study by Bechtel has determined that there would be significant advantages in using commercially available, high-load lift equipment to install the guard tank and reactor pool vessel as single units. CBI Nuclear (CBIN) has confirmed that the reactor vessel and some of its core support internals, and half sections of the reactor deck can be fabricated on site, outside the reactor building. After placing the guard vessel in the cavity with a single lift, the reactor vessel would be lowered into the guard vessel, onto temporary supports as illustrated in Figure 13. Each deck half would then be positioned and welded together, and the reactor vessel welded to the deck. The temporary supports would then be removed. Final alignment machining of penetrations in the deck and core support platform would be performed. All operations would employ existing technology and presently available techniques and equipment.

Seismic Response

In a reactor assembly as large as that in a pool plant, possibly twice the diameter of a loop plant reactor assembly, limiting the seismic motion of the control rods relative to the core is a design requirement of major concern. The horizontal ground motion accelerations are assumed to be .15g and .30g for the operating basis earthquake and safe shutdown earthquake respectively. The shear wave velocity was assumed to be 2500 fps which reasonably approximates conditions at many potential sites. After amplification by building structures, these ground motions produce accelerations of 1.3g and 2.2g at the reactor support, with a frequency spectrum centered near 3 Hz. The rigid core support structure platform combined with the very rigid deck structure adequately limits relative motion of control rods and generates stresses in the reactor assembly which are fully acceptable. Although the vessel wall is only 1 inch thick it is structurally stable because of its large diameter. It has been determined that sodium sloshing during a seismic event is adequately accommodated by the reactor assembly.

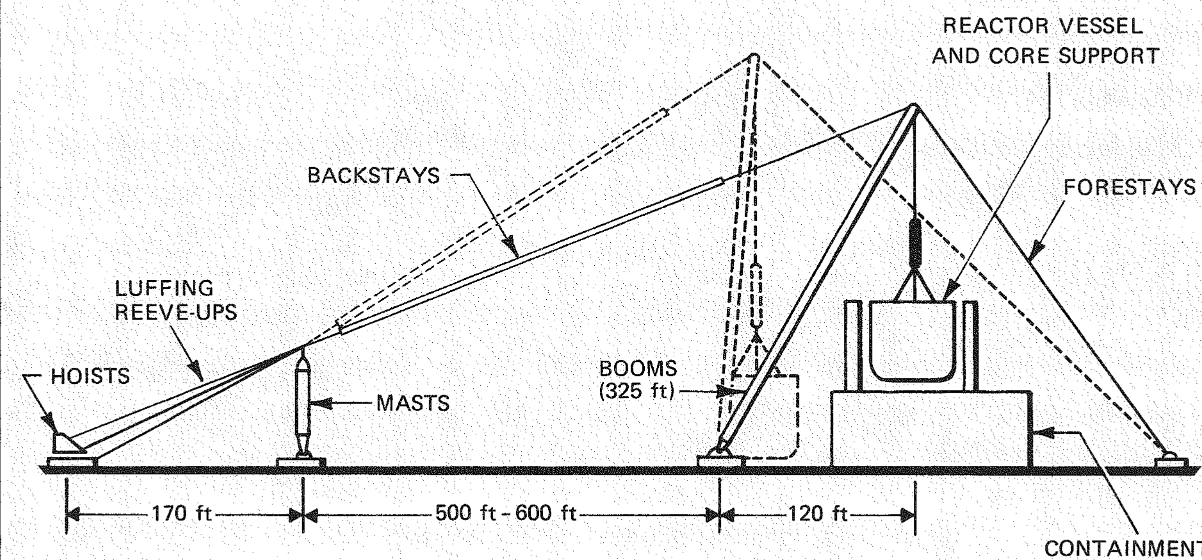


FIGURE 13. REACTOR VESSEL RIGGING SYSTEM

4. DEVELOPMENT AREAS

While there are no major issues which might invalidate the reference plant design, development areas have become evident which should be investigated. The outstanding areas of concern are directly or indirectly associated with the hot sodium pool and with structures affected by the hot pool. The most direct concern involves the mixing of cold sodium, leaving the core after a scram, with hot-pool sodium above the core. In addition, more information is needed about the vertical thermal barriers containing the hot-pool, and about the under-deck insulation in the gas environment above the hot pool. These concerns are addressed below.

Thermal-Hydraulics

In a pool plant the hot sodium pool is a large, relatively shallow configuration compared to, for example, the above-core volume in the Clinch River loop plant. Sodium enters the hot pool at the radial center and at a vertical location which depends on the upper internals structure (UIS) design. Sodium leaves the hot pool by gravity flow into the IHX entrances. The locations of the IHXs are such that only about half the hot-pool sodium is contained between their radial locations and the center of the hot pool, the other half being contained between the IHXs and the vertical thermal barriers.

Preliminary calculations indicate that the main part of the steady state flow pattern is best described as a "rolling donut" in the region between the UIS and the ring of IHXs as illustrated in Figure 14. Because this region accounts for only half the hot pool volume the total flow pattern is actually more complex, especially after a reactor scram when the temperature of the sodium entering the hot pool drops approximately 250°F in a few seconds.

The large diameter of the hot pool produces low velocities as the sodium flows outward, and the cooler, heavier post-scram sodium drops to the hot pool bottom before reaching the IHX. Thus less than half the hot pool sodium may be involved in the main flow pattern and uncertainties about the remaining, higher temperature volume become a major concern. These characteristics of steady state and post-scram sodium flow are illustrated in Figure 14.

While a pool plant has a large, hot-pool sodium volume which can mitigate

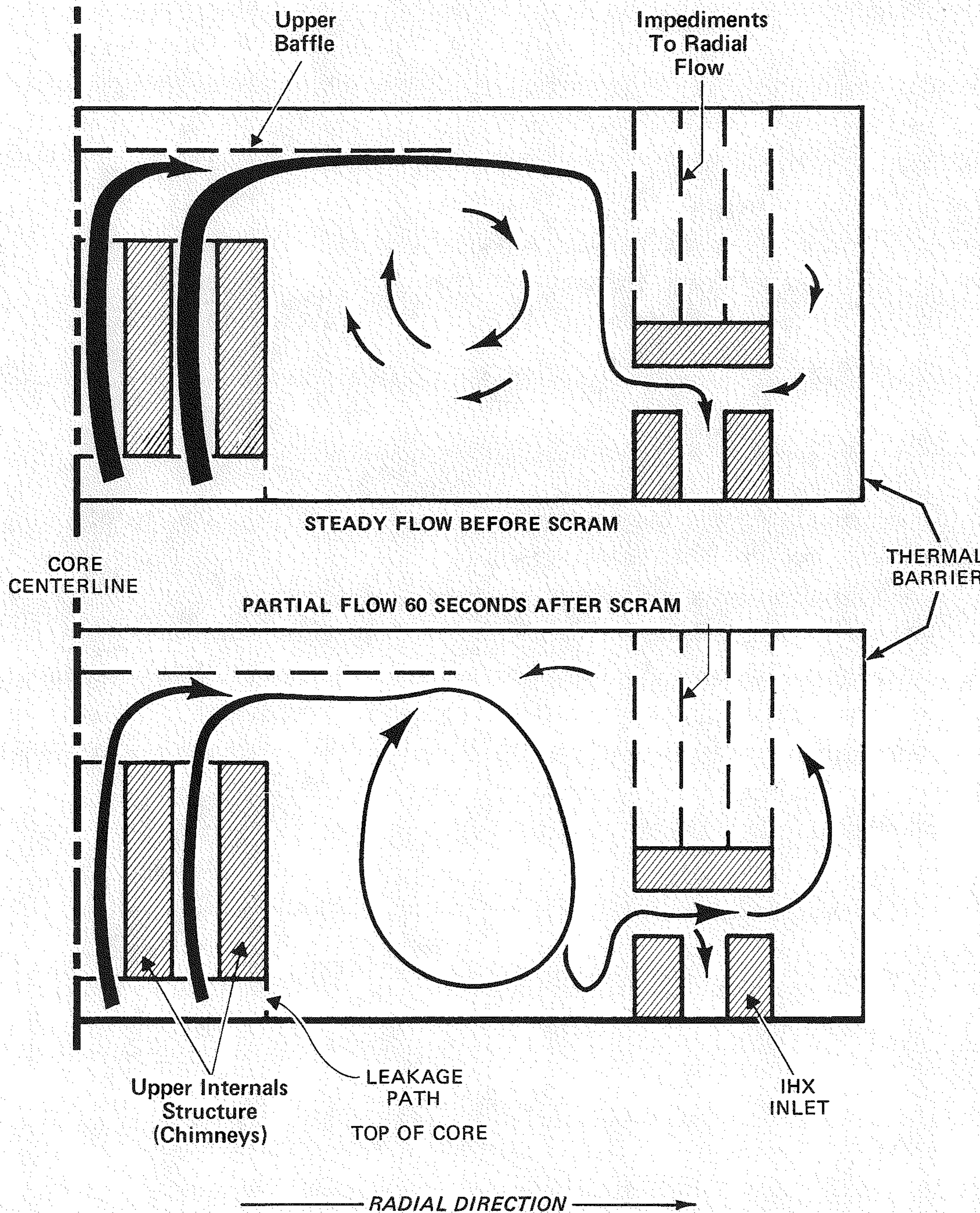


FIGURE 14. POSSIBLE HOT POOL FLOW PATTERNS

thermal transients following reactor scram, optimum use of the hot-pool sodium will require better analytical tools and more experimental knowledge about the mixing of hot and cold liquids in large pools.

Thermal Barriers

If active vertical thermal barriers are to be used in the reference plant design as previously discussed, then some key feature testing will be necessary to assure adequate performance. Heat transfer at low sodium flow, geometric effects, local circulation patterns and possible flow reversals at low flow are all areas that need experimental investigation. The experimental appartus need not be full size except in the vertical dimension (20 feet).

Deck Insulation

Although most of the European pool plant designs employ water cooled decks, such a method may present a formal licensing problem. Accordingly, EPRI has required that no water be used for deck cooling.

The volume beneath the deck is filled with reactor cover gas and sodium vapor at essentially atmospheric pressure and a temperature of 875°F at the interface with the hot sodium pool. This atmosphere must penetrate through passages between 27 reflecting plates and 10 inches of wire mesh before contacting the bottom surface of the deck. As mentioned previously, the deck's bottom surface is cooled from inside the deck by nitrogen. The nitrogen cooling system is designed for five times the amount of heat that can be carried to the deck bottom through the insulation. The major concern about this sytem is the adequacy of the factor of five in the calculated heat load. When consideration is given to possibilities for sodium frosting or other degradation over a 40 year lifetime, the conclusion is reached that testing of such insulation is desirable. Although such tests would be considerably shorter than the plant life, they would reduce the uncertainty in the calculated heat load.

5. CONCLUSIONS

General Electric and Bechtel have critically examined the major features specific to the pool concept. It has become evident that there are feasible solutions to all the various design requirements which have been identified. Major areas of the integrated plant have been developed to consistent levels and give a clear indication that a large pool-type LMFBR is well-suited for further development. Major conclusions are summarized below.

- A pool plant is a more compact arrangement than a loop plant because of the elimination of primary heat transport system loops, and associated inerted cells external to the reactor vessel.

- The stressed-skin, box-beam reactor deck, with internal gas cooling, has large safety margins and maintains small deflections and small temperature gradients under all operating conditions.

- Access to all deck structural welds is possible during plant shutdown by use of removable shielding.

- Thermal barriers separating hot and cold sodium zones of the pool have been investigated in considerable detail. Both active barriers, as employed in French designs, and passive barriers, as employed in British designs, are feasible for the pool concept. The current reference reactor assembly has active vertical barriers and passive horizontal barriers.

- The reactor, the deck and all other structures can be fabricated and erected with available techniques. Equipment is commercially available to lift the guard vessel, reactor vessel, and each of the preassembled reactor deck halves into the containment building as single units.

- Operation at reduced power level with either one primary pump, or one or two intermediate loops out of service is accommodated by the design.

- Adequate space is available in the head access area for all equipment and services and associated maintenance activities. Major components can be removed in a single vertical lift.

Pool-Type LMFBR Plant
1000 MWe Phase A-Extension-2 Design

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