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General Characteristics

by

A. Amorosi, E. Hutter, T. J. Marciniak,
H. O. Monson, R. W. Seidensticker,
and W. R. Simmons

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Pool Study Group

May 1976

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AN OVERVIEW OF POOL-TYPE LMFBRs

General Characteristics

by

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ABSTRACT

This report describes the results of a study conducted by a "Pool Study Group" organized at ANL in mid-1975 to examine the present state of the art of design of pool-type LMFBRs. The study concentrated on examination of the various design options used to date in the principal pool-type projects and design studies in this country and abroad, including the Phenix and Super-Phenix reactors (France), PFR and CFR (U.K.), BN-600 (U.S.S.R.), and EBR-II (U.S.A.). The objective of the report is to provide a step toward better understanding of the pool-type system and of the advantages and disadvantages of the various possible approaches to its design.

The report: notes the general desirable features of the pool-type LMFBR; attempts to identify and evaluate the principal technical considerations involved in making major design decisions for a commercial size pool-type LMFBR; addresses questions concerning the feasibility or adequacy of certain design features or characteristics of the system; and, describes the areas of design which appear to involve the greatest technical complexity or difficulty.

INTRODUCTION

The U. S. LMFBR Program is sharply focused on the development of the loop-type (or piped-type) reactor, as exemplified by the designs of the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor (CRBR). The development of a pool-type system has received little concerted U. S. effort since completion of EBR-II; consequently, there is comparatively little familiarity with the pool-type approach in this country. On the other hand, considerable effort has been devoted to pool-type LMFBRs in other countries.

To determine the current state of the art of pool-type systems, ANL performed a study for ERDA consisting of reviews and evaluations of the principal technical considerations involved in making major design decisions for a commercial size pool-type LMFBR. The effort was conducted by a special "Pool Study Group" organized for the purpose by the Laboratory in mid-1975. The study concentrated on examination of the various design options used in major pool-type projects and design studies in the U. S., France, the United Kingdom, and the U.S.S.R. The purpose of this report is to describe the results of the study.

The report hopefully represents a step toward a better understanding of the pool-type system and of the advantages and disadvantages of the various approaches to its design which have been proposed. The report is generic

in its treatment. It discusses general characteristics, features, and requirements of pool-type LMFBRs, with emphasis almost entirely on those aspects which are unique to, or of particularly great importance to, the pool type. Most of the discussion is concerned with the primary system, since it is in that area that most such aspects are manifested.

To aid in the study, general concept descriptions were developed by the Study Group of the five foreign pool-type systems that are in operation, under construction, or being designed: the French Phenix and SuperPhenix; the British PFR and CFR; and, the Russian BN-600. These descriptions--generated by study of the available literature and supplemented to some extent by travel contacts--are also concerned mainly with the primary system. Together with the existing complete descriptions of EBR-II (the only U. S. pool-type reactor), these descriptions collectively depict most of the various design approaches seriously considered to date, and were used in the study as a basis for comparison of specific design features.

To help make comparisons more meaningful, an attempt was made to conceptually scale up to 1200 MWe size in at least skeletal fashion those plants that are rated at less than that capacity (i.e., all plants except SuperPhenix and CFR, which are already sized for 1200 and 1300 MWe, respectively). In making the scale-up, the features of the smaller plants were reasonably faithfully maintained unless they appeared infeasible for the larger size.

This report should be considered as only a small beginning of the more extensive evaluation of the merits of the differing plant concepts that is needed before the building of a large-scale pool-type LMFBR in the U. S. can be seriously considered. The effort involved in the study and in preparation of the report was quite limited. There is no pretext that all areas of significance have been treated or that treatment is uniform. In particular, it should be noted that some of the areas needing evaluation which are not addressed at all include: the degree to which the existing R&D program for the loop-type reactor would have to be modified or augmented to encompass the needs of the pool-type system; economics and economic comparisons; and, the possible ramifications of utilization of foreign expertise.

Much of the information in the report was presented in preliminary form on September 9-10, 1975 to representatives of ERDA, the Electric Power Research Institute (EPRI), and reactor manufacturers and architect-engineers interested in LMFBR plants. Included were representatives of firms submitting proposals for developing concepts for prototype LMFBR plants under a program jointly funded by ERDA and EPRI.

I. GENERAL SUMMARY AND CONCLUSIONS

Commercial LMFBR power plants may, in general, be either of the pool type or the loop type. However, development of the pool-type system has received very little U. S. effort beyond the construction and operation of EBR-II; consequently, up-to-date knowledge of the pool-type LMFBR is lim-

ited in this country. On the other hand, foreign LMFBR experience in pool-type systems has been growing at an accelerating rate.

This report describes the results of an ANL study conducted for ERDA on the state of the art of pool-type LMFBR designs. In the study, attempt is made to identify and evaluate the principal technical considerations involved in making major design decisions for a commercial size pool-type LMFBR. The study is based on examination, within the limitations of the time and resources available, of the various general design approaches used to date in the major pool-type projects and design studies in the U.S.A., France, United Kingdom, and the U.S.S.R. The specific design options for principal systems and components found in these general approaches are then examined and their advantages and disadvantages discussed. Among the important design options are those relating to the primary coolant flow configuration, method of support of the main tank, method of support of main components, integration of main tank cover and shield deck, method of accommodating pipe and component thermal expansions, fuel handling arrangements, and shutdown cooling provisions. Treatment in general is limited to the primary system, as this is the principal characterizing area of the pool-type LMFBR.

In the pool-type LMFBR, the reactor and the entire radioactive primary cooling system are located in a single large vessel--the main tank. The tank is essentially filled with a pool of sodium. The principal major components in the tank are: the reactor core, blanket, and reflector; core support structure; instrument tree (upper internals); neutron shielding; primary pumps; intermediate heat exchangers (IHXs); all primary coolant piping (or ducting); and, the in-vessel fuel handling equipment. This arrangement results in: a single, simple containment for the primary system sodium coolant, with no penetrations in the tank wall; a surrounding environment of sodium common to all primary system components, which enables mitigation or elimination of pipe (duct) expansion stresses and minimizes effects of pipe leaks; a large volume of sodium coolant which, in addition to being used as part of the heat transport system, provides a large-capacity heat sink for emergency situations; and, a minimum number of required auxiliary structures, systems, and components.

Some of the principal technical questions which have arisen in connection with the pool-type system and which have been addressed in the study relate to: achievement of adequate earthquake resistance of the primary system, particularly of the main tank; fabrication and field erection of the main tank; fabrication and field erection of the shield deck located above the main tank; capability of the main tank and shield deck to withstand an HCDA of significant magnitude; employment of potentially incompatible and difficult-to-maintain shielding materials inside the main tank; and, component inspectability and maintainability.

The major conclusions of the study are:

1. It appears practical to achieve adequate earthquake resistance in pool-type systems, using main tanks with reasonable wall thicknesses. The

resistance achievable stems primarily from characteristically low length-to-diameter ratio and the relatively high natural frequencies of vibration of such tanks. Adequate freeboard above the sodium surface, and perhaps additional features, must be provided for accommodation of coolant sloshing.

2. Fabrication and field erection of the main tank--designed to resist dead loads, earthquake loads, thermal effects, and other loads--appear feasible for tank sizes up to 70 ft in diameter (and probably somewhat larger).

3. The largest field-erected component required is the shield deck. This is a stiffened-plate type of structure, 60-80 ft in diameter, as much as 10 ft thick, and weighing several hundred tons without its shielding concrete. It appears feasible to shop-fabricate this type of structure in sections and assemble it in the field.

4. From the standpoint of safety, the reactor coolant boundary (the main tank) can exhibit extremely high integrity against loss of coolant. This is primarily because of the tank simplicity, low stress level, relative freedom from thermal shock, and good inspectability. Scoping calculations provide preliminary indication that pool-type LMFBRs can be made capable of: (a) withstanding an HCDA energy release of significant magnitude; (b) providing sufficient core cooling to avoid coolant boiling following a primary system loss-of-piping-integrity accident at full power; and (c) maintaining natural-convection cooling of the reactor after shutdown, even with total loss of secondary system cooling. Also, provision for large scale core melt retention, if required, is simplified by the entire primary system being within a single tank and by the large coolant-submerged area available for an engineered retention device.

5. Knowledge gained since the earlier pool-type systems were constructed enables use of considerably simplified neutron shielding designs within the main tank. These designs, using only sodium and iron (steel), provide adequate shielding without the disadvantages of incorporating graphite.

6. In-service inspection requirements have not yet been firmly established for LMFBRs (particularly for the pool type), and comprehensive conclusions cannot be drawn in this area. However, based on estimation of likely requirements, brief review suggests that inspectability of the pool-type system should prove adequate. As examples: inspectability of the primary coolant boundary (the main tank) is reasonably good; inspection of the contained piping is difficult, or even impossible, but likely will not be required if piping failure ~~is~~ can be successfully demonstrated (as expected) to not represent a significant safety problem.

7. It appears that maintenance and servicing of primary system components can be accomplished satisfactorily. However, considerable care is required in design to assure this capability. Of particular importance are: layout and spacing of components penetrating the shield deck so as to provide adequate access for positioning of removal casks (or bags) and equipment; location of the above-deck cable runs, cooling ducts, and similar equipment

so as not to unduly interfere with component servicing; and, placement of as large a fraction of primary coolant system instrumentation as possible on the removable components of the system, with provision for remote replaceability (or sufficient redundancy) of the remaining critical instrumentation. Experience to date with primary systems of operating pool-type plants (EBR-II, Phenix, PFR) has not revealed excessive difficulty with maintenance.

8. The key decisions to be made in the design of pool-type primary systems are: (a) whether to use a hot pool or cold pool; (b) whether to provide the main-tank cover and shield deck as separate components or combine them (integral deck); and (c) what type of fuel handling concept (and rotating-plug size) to use.

9. The size and complexity of the rotating plugs--and, hence, their cost--are sensitive to the type of fuel handling system employed. There appear to be several approaches to fuel handling which might minimize the size of these plugs and enhance their design (and operational) simplicity, thus reducing their fabrication and installation costs. This observation applies both to pool-type and other LMFBRs.

10. The main tank and its support structure operate at and are designed for temperatures less than the maximum permitted (800°F) by the basic document of the ASME Code, Sec. III. Accordingly, use of high temperature design criteria--such as for creep and fatigue--are not required. However, it should be noted that because official requirements for the pool-type LMFBRs have not been established explicitly, various aspects of design, construction, and in-service inspection need to be more fully explored with appropriate code and regulatory authorities.

11. Among the areas of design of commercial pool-type systems which appear to present the greatest technical complexity or difficulty, or which would require the most research and development, are: (a) structural design of the shield deck; (b) design of the cooling system to maintain adequately low temperature of the deck (in the integral deck design approach); (c) design of the thermal insulation on the underside of the shield deck (integral deck approach); (d) design of low pressure drop IHXs; (e) provisions for maintaining low and uniform main-tank wall temperature (in the hot-pool concept); (f) provisions for assuring appropriate temperature distribution in the main-tank cylindrical support skirt, where used; (g) provisions for accommodation of sodium sloshing in seismic events; and, possibly, (h) provisions for assuring adequate access for convenient maintenance. Preliminary evaluations indicate that adequate design solutions for these problem areas can be accomplished or, in some cases, may already exist. Acquisition of further information on existing and proposed designs in these areas would be useful.

12. As a general conclusion, it appears that the pool-type concept has on balance sufficient attractive features, as noted above or described in the body of the report, to warrant consideration of this concept for possible commercial use in this country.

II. BRIEF SUMMARIES BY SUBJECT AREA

A. Primary-system Configuration

The major option in the design of a pool-type plant is the choice between a "hot" or a "cold" pool. The hot pool operates with the upper portion of the primary system and, in some cases, the shield deck exposed to the reactor outlet temperature (~1000°F). The cold pool operates with the main tank at the reactor inlet temperature (~700°F).

The "hot" pool is achieved by using an open duct between the reactor outlet and the IHXs--the so-called open plenum concept. Thus, open pools having free liquid levels are used to contain both the flow from the reactor to the IHX (at reactor outlet temperature) and from the IHX to the pump (at reactor inlet temperature). A flow barrier is used between the hot and cold regions of the primary system. This pool arrangement, shown in Fig. II.1(a), has been used in Phenix and PFR and is proposed for CFR and SuperPhenix. (The pool thus far has not been used for transport of coolant between the pump and the reactor inlet, because design of the primary tank for pump outlet pressure is considered impractical.)

A variation of the hot-pool concept is used in BN-600, as shown in Fig. II.1(b). In this approach the coolant going from the IHX to the pump, and from the pump to the reactor, is piped or constrained within enclosed ducts, thereby permitting the main pool to be hot. Bypass flow from the pump outlet is used to cool the inner surface of the main tank.

The cold-pool concept utilizes an enclosed reactor outlet plenum, and the flow from the reactor outlet is piped (or, possibly, ducted) to the IHX. The bulk sodium of the pool is at core inlet temperature and remains essentially constant with load. The concept as used in EBR-II is shown in Fig. II.1(c). This concept uses a reactor-vessel cover. This cover is located directly above the core in such a manner as to provide a relatively small outlet plenum and is lifted out of the way during refueling. The control-rod drive lines also must be raised a considerable distance to permit fuel handling.

A cold pool with a baffled reactor outlet plenum is another approach. It is shown in Fig. II.1(d). This system employs a small positive flow of cold sodium above the horizontal baffle(s) (a loose-fitting enclosure) to maintain a cold pool above the baffles. This variation, which has not been employed to date, was not addressed in detail in this study. However, it appears to have the attractive features of eliminating the routine lifting of a heavy reactor-vessel cover and elevating of control drive shafts for fuel handling, while maintaining the principal attractive aspects of the cold pool.

Other major characteristics, and the relative advantages and disadvantages, of the hot pool and cold pool approaches are summarized below.

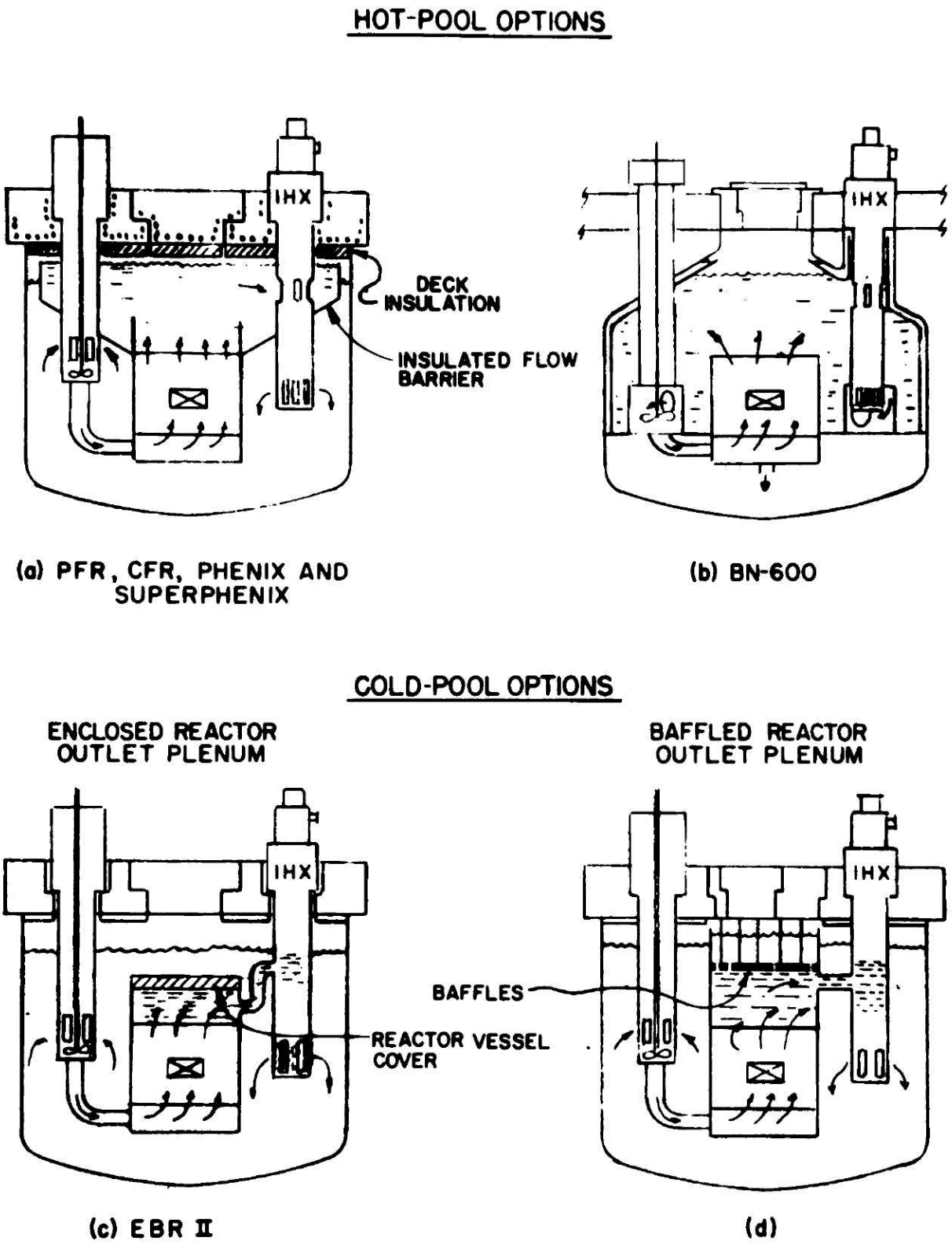


Fig. II.1. Configuration Options in Design of Pool-type LMFBR

A number of the topics are addressed in greater detail in subsequent subsections. Unless otherwise indicated, the BN-600 variation is included in the discussion of the hot pool.

1. Thermal Expansion and Insulation

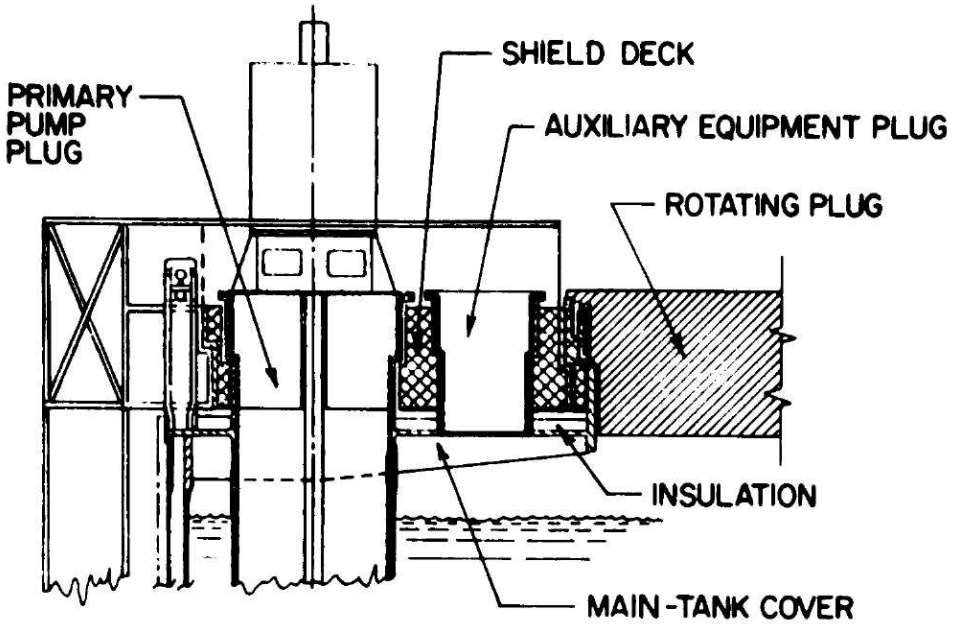
Thermal expansion has a great impact on design of the hot-pool system because so much equipment is subjected to the reactor outlet temperature, which is high and varies with load. Potentially troublesome areas are the shield deck, main tank, and in some cases, the main-tank cover, the primary pump casing, and the barrier between the hot and cold portions of the tank. Phenix, for example, has a main-tank cover separate from the room-temperature deck, but the cover follows the hot-pool temperature with too great a time delay. Therefore, the possibility of high stresses in the tank and cover must be considered in determining permissible operating maneuvers. The SuperPhenix design eliminates the main-tank cover and utilizes only an insulated, integral cover and shield deck, as shown in Fig. II.1(a). The integral cover and shield deck approach requires the main-tank wall (720°F) to be integrally attached to the cold (room-temperature) shield deck. Special provisions are required to maintain the temperature gradient in the tank wall within acceptable limits in respect to stress. Design features for accommodation of thermal movement of the pump casing relate strongly to the method of component support (which is discussed in Sec. II.A.3). The flow barrier has been developed in various ways in France and England. Detailed evaluation of foreign experience with this barrier and the related effects on thermal convection and/or stratification in the bulk sodium in the main tank is important to the development of the hot-pool concept.

For the cold-pool concept, the roof design can be either an integral structure (Fig. II.2, bottom) or two structures, as used in EBR-II (Fig. II.2, top). In the latter approach, a ring-girder type of main-tank cover is used integral with the main tank and separate from the shield deck. The main-tank cover carries the loads of all components that penetrate the cover, except for the rotating plugs. The cover is exposed only to the bulk sodium in the tank, which remains nearly constant at core inlet temperature, and requires no in-sodium or in-sodium-vapor insulation. Important design considerations of the cover are seismic resistance and containment of certain hypothetical accidents.

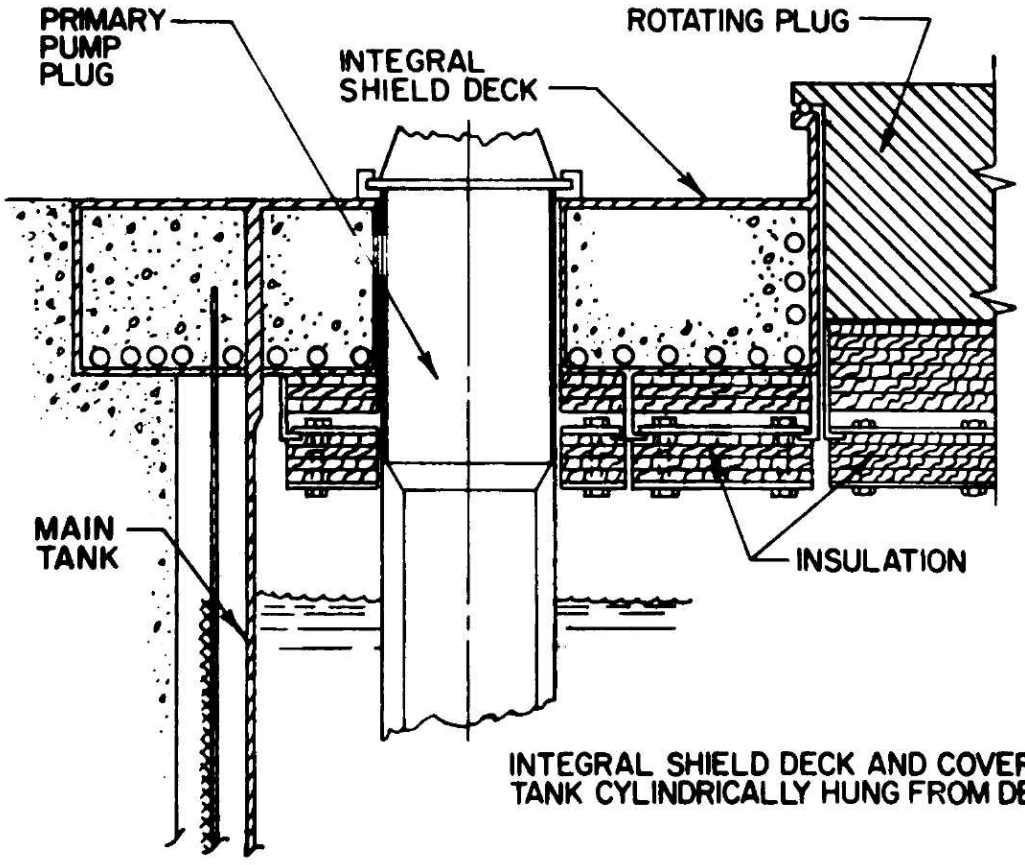
The other design approach--the integral shield deck coupled with the cold pool--may have some interesting attributes:

(1) The shield deck insulation operates at lower temperature.

(2) The shield deck insulation does not have to withstand rapid temperature changes; thus, it can be backed with a thick plate to protect against damage under earthquake (sodium-sloshing) conditions.



SEPARATE SHIELD DECK AND TANK COVER—
TANK HUNG ON DISCRETE HANGERS



INTEGRAL SHIELD DECK AND COVER—
TANK CYLINDRICALLY HUNG FROM DECK

Fig. II.2. Examples of Shield Deck Arrangements and Different Tank and Component Support Methods

(3) The maintenance of appropriate temperature distribution in the main-tank wall is simplified, since the sodium pool is at lower temperature and is essentially isothermal.

Design for thermal expansion in the cold-pool concept focuses primarily on the reactor outlet piping. A number of methods for accommodation of piping expansion have been suggested, but are not addressed in detail in this study. They include: (a) pipe expansion loop, (b) removable bellows, (c) slip joint, (d) inverted bell jar, (e) support of pipe and IHX inlet plenum from the hot-to-cold-pool barrier, (f) syphon pipe and (g) thermal baffles inside the pipe so that the pipe operates at nearly pool temperature. (Some of these methods are shown in Fig. II.3.)

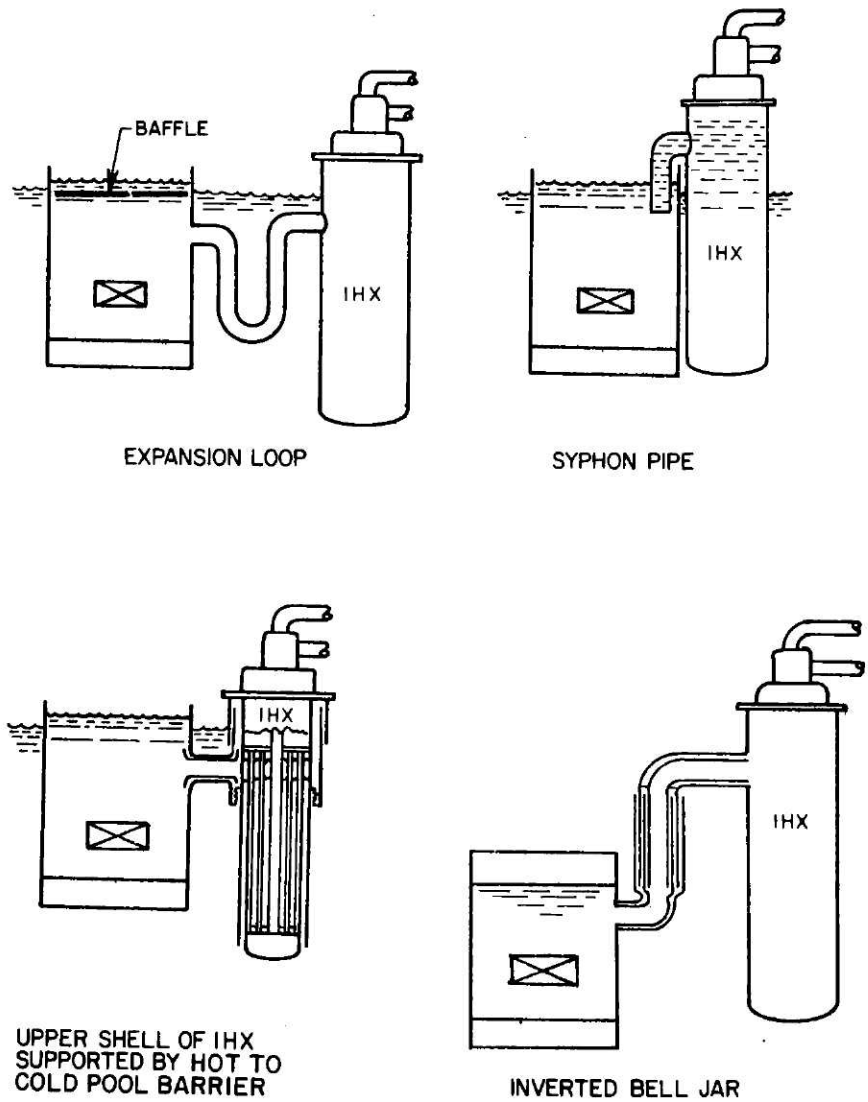


Fig. II.3. Some Methods of Accommodating Piping Thermal Expansion in Cold-pool Concept

2. Sodium Vapor Pressure

The sodium vapor pressure at the hot-pool temperature is 35 to 40 times higher than for the cold pool. The higher concentration of aerosols in the cover gas increases the potential for plugging the rotating plugs, control-rod drives, fuel handling equipment, and other devices exposed to the cover gas. Twelve years of experience with EBR-II have shown that equipment can operate successfully at 700°F. Phenix has operated at 1000°F for $1\frac{1}{2}$ years with no difficulties reported. However, not enough experience is available to date to draw firm conclusions for a plant life of 30-40 years.

3. Component Support

The choice between the hot pool and the cold pool influences the choice of component support. For example, one support method requires use of a separate main-tank cover, but such a cover can be used only with a cold pool. The choice of support for primary pumps and IHXs affects the type of interfaces and magnitude of thermal expansion between components and mating structures or between components within the main tank.

Four component support approaches have been used to date: (1) bolt the unit to the cold shield deck (PFR, SuperPhenix), (2) take the weight of the unit on the cold shield deck and provide for lateral movement relative to the deck (Phenix), (3) support the unit on a separate main-tank cover (EBR-II), and (4) support the unit on nozzles that carry the load to the core support structure (BN-600). Each type of component support has its own unique thermal differential expansion considerations and, thus, influences (a) the choice of the kind of joints between the pumps and piping, and between the IHX and related penetrations through the flow barrier or connections to piping; (b) the amount of leakage that may have to be accommodated through these joints; (c) the kinds of seals that may be applicable; (d) the need for large-diameter bellows; and (e) the motion of the IHX secondary sodium nozzles.

Innovative design choices have been made in SuperPhenix to handle differential thermal expansion requirements resulting from cold shield deck support of components. An inverted bell jar is used as the joint between the IHX and the flow baffle. (This concept may have other applications--such as for reactor outlet piping, which is illustrated in Fig. II.3.) In SuperPhenix the pump is aligned in the hot position, i.e., for operating conditions. A flexible joint is used between pump and inlet piping, and the piping is supported from the core support structure. As a result, thermal expansion considerations appear to be largely overcome, with the principal area of focus now being the pump/pipe connection.

4. Heat Capacity of Primary-system Sodium

Pool-type systems for 1200-MWe plants are likely to have 3500-5000 tons of primary-system sodium within the main tank. The heat

capacity of all this sodium is available to absorb heat following a postulated loss of IHX cooling. Consequently, the pool concept has a large thermal inertia to change in temperature. This provides time to activate emergency cooling systems and greater assurance of recovery from loss of normal heat-removal systems. The relative thermal capacities of the various kinds of pool-system concepts are related to the quantity of sodium in the main tank and the fraction of sodium at the reactor inlet temperature. The cold-pool concept has the most capacity, the BN-600 variation the least, and the CFR/SuperPhenix concept is in between.

5. IHX Pressure Drop and Flow

The allowable pressure drop in the IHX depends upon the system flow configuration. The configuration with open pools depends upon a small difference in liquid level between the hot and cold pool to provide a driving head of 2 to 6 ft for the IHX. For the BN-600 variation, the allowable pressure drop is determined by the submerged head of the pump less its required net positive suction head (NPSH) and can be somewhat higher. For the cold-pool system, it is possible to have a still higher pressure drop if the resulting reduction in IHX size is sufficiently attractive.

The cold-pool system provides more predictable flow patterns and sodium temperatures at the IHX inlet. These parameters aid in estimating the response of the coolant system during transition from forced- to natural-convection flow. Scoping calculations of natural-convection cooling conducted for this study indicate that both the hot- and cold-pool systems can have satisfactory capability for natural-convection cooling, even after severe hypothetical failures such as total loss of the intermediate cooling system. Two parameters were briefly studied: (1) the elevation of the IHX exit above the core, and (2) the primary-pump coastdown rate. Results suggested that convection flow rates may be fairly insensitive to the elevation of the IHX exit. Longer coastdown rates generally give lower peak core temperatures and larger core transients. For the hot-pool arrangement, the resultant transfer of a large amount of sodium from the hot to the cold region raises the reactor inlet temperature considerably.

6. Pump and IHX Design

Pool-system pumps are quite similar to loop-system pumps with similar flow rate and head requirements, an important difference being that the suction flow for the former is usually taken from an open pool with a nearly constant temperature.

All pool systems have been designed with cold-leg pumps. For a design like BN-600, where the flow between the IHX and pump is essentially piped, a hot-leg pump could be used. However, a cold-leg pump was chosen--with a reduction in inlet head to the pump equal to the pressure drop in the IHX. This approach accepts the penalty of ~~a relatively low pressure drop in the IHX~~. having to provide an IHX with a relatively low pressure drop.

An isolation valve is required for each IHX to permit operation with one loop shut down. The various methods of isolating the IHX are shown in Fig. II.4.

To obtain high effectiveness in the transfer of heat in the IHX, counterflow is used--in most cases with the primary sodium on the shell side. Uniform flow distribution with low pressure drop through the bundle is important. In Phenix and SuperPhenix, the tubes are arranged with constant radial pitch as shown in the left view of Fig. II.5. This arrangement minimizes the pressure losses across the bundle and tends to provide uniform flow distribution in the bundle. In EBR-II the flow paths are balanced by the use of orifices within the tube-support plates. To minimize the pressure drop along the tube bundle, tube supports need to be designed to offer minimum flow restriction. In BN-600 the tubes are supported by lacing as shown in the right view of Fig. II.5.

LMFBR IHXs have an excellent performance history. No tube failures have been recorded in the IHX for EBR-II (12 years of operation), Fermi, Phenix, and PFR. The performance records of pumps have also been good, with no major breakdowns reported after initial operating difficulties were overcome.

B. Structural Design and Construction

The structural design and construction of the main structural components--main tank, guard tank, component supports, shield deck, and shielding--of a 1200-MWe pool-type LMFBR are generally within present technology. Prudent design and careful construction practices will be required to derive maximum benefit from the pool concept's simplicity of containment and resistance to seismic events. A number of design choices (and specific precautions) have been identified during this study. As previously discussed, these choices relate to tank and component support and to accommodation of thermal expansion.

The natural periods of vibrations expected for the very large-diameter main and guard tanks are around 20 cps (empty). (This is reduced to about 13 cps when the main tank is filled with sodium.) This response is due primarily to the low ratio of height to diameter resulting in a high stiffness of these tanks. As a result, the response of the vessels to earthquakes is not significantly further aggravated by an amplification of the acceleration found in vessels of less stiffness. There is, of course, an amplification of the base ground motion through the building response. This effect is probably higher if the tank is supported at the top instead of the bottom, although many other factors--containment embedment, soil conditions, etc.--are very important. The forces and moments resulting from a free-field ground acceleration of 0.3 g (amplified by a factor of 2 at the operating-floor level) appear to require material thicknesses that can be fabricated and erected with present technology.

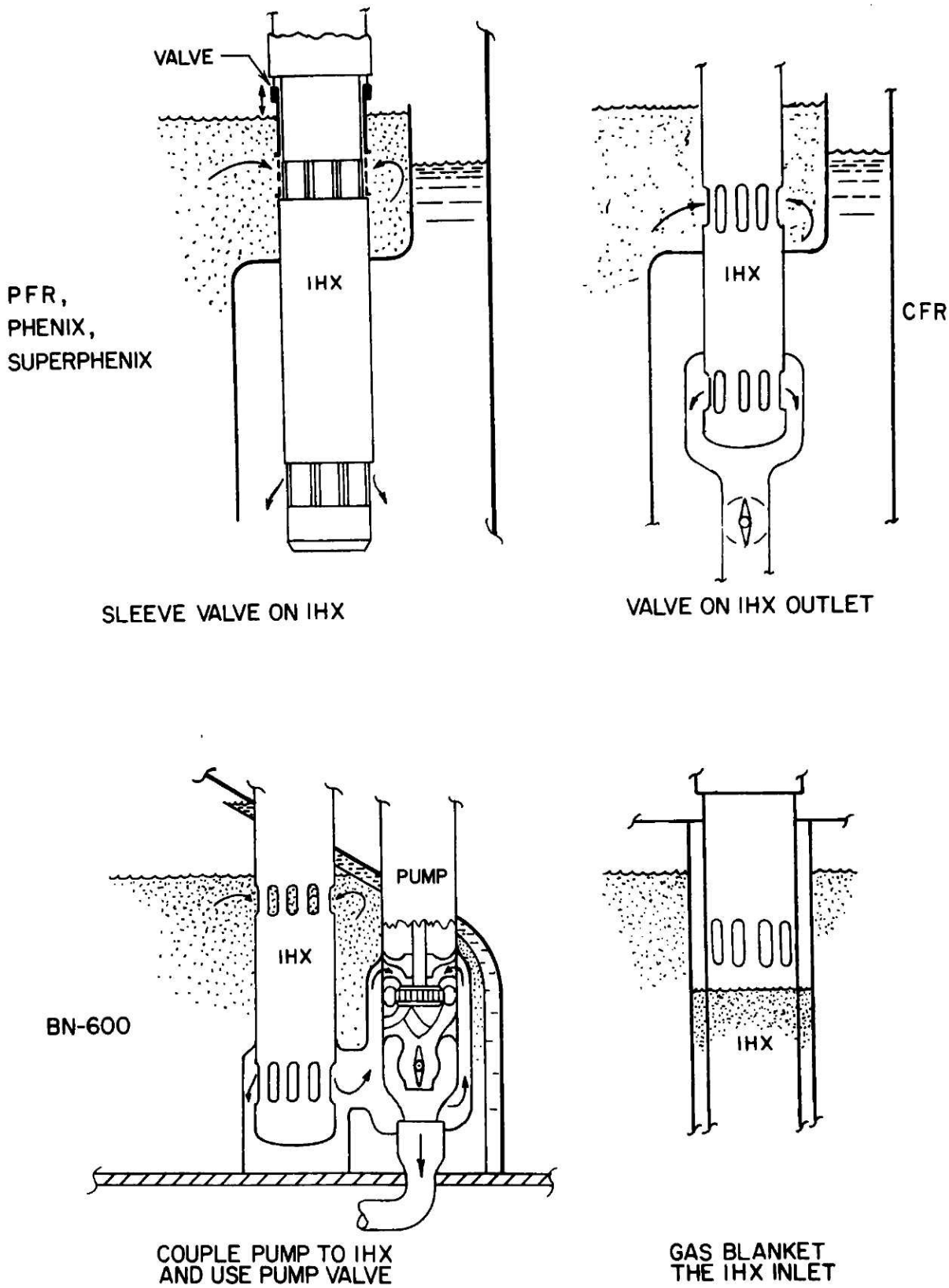


Fig. II.4. Conceptual Methods of Isolating the IHX

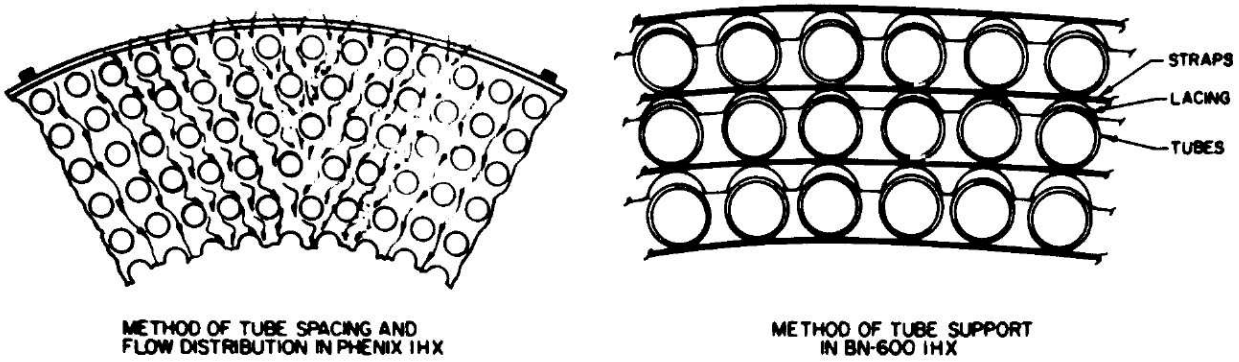


Fig. II.5. Tube Spacing for Phenix IHX, and Tube Support for BN-600 IHX

If major components such as pumps and IHXs are enclosed in stiff shrouds fitted fairly close to the components, there may be significant damping of the equipment motion, especially if the equipment and shrouds are attached firmly to the shield deck. The calculations performed in this study were not extensive, and the results not conclusive. The phenomenon does, nevertheless, seem to deserve further analysis. Conversely, the interactions between the equipment and main tank deserve more study.

If the pool system selected omits the hot main-tank cover, the suspension system for the main tank must be a continuous welded skirt to complete the leaktight enclosure with the shield deck. If a hot main-tank cover is used, the designer may select a skirt, though it need not be continuous around the tank periphery, i.e., leaktight. He may decide to employ discrete hanger systems (such as in EBR-II or Phenix) that transfer loads via rollers or laminated steel/neoprene bearing pads, especially in low seismic areas. Preliminary calculations indicate that the required number of such hangers is reasonable for a set of design conditions which include modest seismic requirements.

Although a main tank suspension system of several discrete hangers can be designed to provide a reasonable level of seismic resistance, the use of a continuous skirt support extending above the main tank provides the most efficient structural means to transfer earthquake loads to the system support. Pin-connected, pendulum-type hangers probably have little value in resisting seismic forces.

When the designer chooses a skirt support he must be aware of a design-requirement conflict. First, the flexibility needed to lower stresses from radial thermal expansion requires minimizing tank wall thickness and increasing skirt length. On the other hand, seismic resistance is generally enhanced by increasing tank wall thickness (in the skirt region) and keeping overall tank height to a minimum.

Preliminary seismic studies have shown that this conflict may not be serious, as first believed, since the studies indicate that the higher

seismic stresses are occurring in the lower regions of the tanks (due to fluid/internal structure/tank interaction). Even for relatively strong earthquakes (0.6 g at support level), the net pressure on the tank wall and shield deck is only about 8 ft of sodium.

The sloshing of sodium following an earthquake results in unbalanced hydrostatic pressures being exerted on the main tank. The resulting nonsymmetrical forces require special analysis and design not normally encountered in gravity loads. It appears that for modest earthquakes, the sloshing effect can be virtually eliminated by providing sufficient freeboard (cover-gas space) to avoid excessive hydrodynamic forces on both the tank walls and the top closure head.

C. Fuel Handling

The fuel handling process for pool-type reactors is divided into two steps: (1) loading and unloading fuel assemblies into and out of the reactor with a gripper mechanism, and (2) operations between the gripper mechanism and the exit port of the main tank. For the first step, combinations of grippers and rotating plugs (from zero to three) are considered. Gripper types include pantographs, fixed-offset grippers, and straight-pull grippers. For the second step, possible concepts include fixed and rotating-plug exit ports. Open cell transfer by removing a large shield plug and operating in a shielded cell accomplish both steps simultaneously.

The latest generation of pool-type LMFBRs (SuperPhenix, CFR, BN-600) all use a fuel-unloading concept involving the straight-pull gripper with two rotating plugs. This choice was primarily based on (a) the greater simplicity (and therefore greater reliability) of a straight-pull gripper compared to a pantograph or fixed-offset gripper, and (b) the possible need to exert large pulling forces due to fuel-assembly swelling. With a large instrument tree--covering the core and part of the blanket, as in CFR and SuperPhenix--little reduction in the size of the rotating plug results from the use of an offset gripper instead of a straight-pull gripper. The plug sizes of SuperPhenix and CFR are large--38 and 33 ft, respectively. These sizes are of concern from the standpoints of operational reliability, method of shipment, and cost.

All LMFBR reactors except EBR-II use rotating plugs exposed to reactor-outlet-temperature sodium (~1000°F). In EBR-II, the plugs are exposed only to inlet-temperature sodium. To what degree there will be fouling problems with the large plugs at the higher temperatures has not been established. These problems are common to pool-type and other types of LMFBRs.

Because of these findings, decreasing the size of rotating plugs and reducing the temperature at the free surface of the sodium were identified

in this study as appropriate developments for improving the fuel handling concept. BN-600 is an example of a method of achieving smaller plugs. It has double rotating plugs with two direct-pull grippers, one at the edge of the inner plug, and the other farther in. The 10.5-ft-dia removable-assembly area can be handled with a 13-ft-dia plug. However, the fuel is moved to an intermediate storage location with one gripper and, after decay, to the exit port by the other gripper. Other potential ways to reduce plug size include: (1) using an off-center rotating plug in relation to core; (2) using an off-center core in relation to reactor vessel; (3) using an off-center pattern for control rods in relation to core; (4) reducing the instrument-tree area, or scalloping one side; (5) dividing the instrument tree and control-rod pattern between two rotating plugs.

The method of moving the fuel from the edge of the core to the exit port may differ depending on the region from which the fuel is removed from the tank. If unloading is from the hot-pool region, the fuel assemblies can be transferred directly into the inclined exit port, as in BN-600, Phenix, and SuperPhenix. If from the cold-pool region, the fuel assemblies must go through the hot-to-cold barrier, and an extra step is required to do this.

In-tank interim storage of fuel is a potentially worthwhile design option to decrease the decay heat to be handled outside the tank and to reduce reactor downtime during refueling if difficulties with the transfer equipment through the tank exit port are encountered. Many factors that have not been fully explored are involved in whether or not to provide in-tank storage. No foreign pool reactors use in-tank storage.

D. Safety

Several inherent characteristics of the pool-type system are especially advantageous in respect to safety. In a pool-type LMFBR the entire primary system is contained within a large, "main" tank. This results in: a single, simple containment for the reactor coolant, with no tank-wall penetrations; a common sodium environment for all primary system components, which can be of great benefit in minimizing expansion stresses and which virtually eliminates any safety significance of pipe leaks (since no loss of sodium from the primary system results); a large sodium volume that is part of the primary heat transport system and provides a major heat sink for emergencies; and, a minimum of auxiliary structures, systems, and components.

In the study, preliminary containment calculations were made which indicate that the pool-type system with integral shield deck can accommodate significant HCDA energy releases without failure of the main tank or massive breach of the deck. This result is in great part due to the large cross-sectional area of the tank and the substantial cover gas volume, which enable dissipation of the accident sodium-slug energy over a large area and the development of a low final equilibrium pressure. In illustration, peak plug loads from an HCDA

were calculated to be about a factor of three lower for a tank 70 feet in diameter than for one 35 feet in diameter. As for all LMFBRs, capability for in-reactor retention of a limited core melt also is considered achievable. Provision for large scale core melt retention, if required, is simplified by the entire primary system being within a single tank and by the large coolant-submerged area available for an engineered retention device.

Calculations indicate that for a pool-type LMFBR with appropriate design of reactor inlet piping, a hypothesized double-ended instantaneous rupture of any inlet pipe (even at the inlet plenum nozzle and either with or without reactor scram) while at full reactor power would not raise coolant temperature within the reactor (peak local) to the boiling temperature. For a reactor outlet pipe break accident, the effect would in general be of even less significance. Thus, loss-of-piping-integrity (LOPI) accidents in pool-type systems of proper design do not appear to constitute a significant safety problem.

Scoping calculations conducted in the study of primary system natural-convection cooling of the reactor indicate that both the hot-pool and cold-pool systems can have satisfactory capability for natural convection cooling, even after severe failures such as total loss of the intermediate cooling system. To remove and dissipate to the atmosphere the heat from the primary system, the pool-type system can readily accommodate an auxiliary decay heat removal system (or shutdown cooling system) which also operates by natural convection and ensures heat removal even if the intermediate system is completely inoperable. This decay heat removal system usually employs sodium-to-NaK heat exchangers in the main tank and NaK-to-air heat exchangers outside the reactor building.

E. Inspection and Maintenance

In-service inspection requirements have not yet been firmly established for LMFBRs (particularly for the pool type), and comprehensive conclusions cannot now be drawn in this area. Nevertheless, a brief examination was made in the study of the principal in-service inspection considerations related to the major pool-type primary system components, and a preliminary assessment made. The components considered were: the main tank and guard tank; tank support structure; main tank cover and shield deck; reactor support structure; and, the primary system piping (or ducting). For each component, three key aspects were evaluated: accessibility for inspection; importance to safety (likelihood of requiring inspection); and, type of examination useable. The results of the review suggest that inspectability of the pool-type system, if given proper attention during design, should prove adequate. As examples: inspection of the primary coolant boundary (the main tank), which likely would be required, appears clearly feasible; inspection of the contained primary piping is difficult, or even impossible, but likely would not be required if piping failure is successfully demonstrated (as expected) to not represent a significant safety problem. In general, it appears that visual inspection would be the predominant method of examination used.

Although the Pool Study Group work scope did not permit a concerted study of the subject, a brief appraisal of the maintainability of pool-type systems also was conducted. As with in-service inspection, consideration was limited to components within the main tank, with emphasis on aspects for which maintenance considerations potentially may be unique. It appears that maintenance and servicing of primary system components can be accomplished satisfactorily. However, considerable care is required in design to assure this capability. Of particular importance are: layout and spacing of components penetrating the shield deck so as to provide adequate access for positioning of removal casks (or bags) and equipment; location of the above-deck cable runs, cooling ducts, and similar equipment so as not to unduly interfere with component servicing; and, placement of as large a fraction of primary coolant system instrumentation as possible on the removable components of the system, with provision for remote replaceability (or sufficient redundancy) of the remaining critical instrumentation. A few primary system components, such as the fixed portions of the reactor inlet piping, are not removable from the main tank for maintenance and, because they are submerged in sodium within the tank, could be serviced only with great difficulty; accordingly, it is necessary to design and construct such components in a manner that makes negligible the probability of their failure during service life. Experience to date with primary systems of operating pool-type plants (EBR-II, Phenix, PFR) has not revealed excessive difficulty with maintenance.

F. Shielding

Study has shown that use of considerably simplified neutron shielding designs within the main tank are feasible. Such designs, using only sodium and iron (steel), provide adequate shielding without the disadvantages of incorporating graphite (carbon). Permanently placed steel shielding of rugged and simple design, supplemented by the large volume of sodium present, can be readily arranged to reduce reactor neutron flux to acceptable levels with respect to: activation of repairable or replaceable components; activation of intermediate system sodium; radiation damage to structural materials; and, meeting appropriate dose rate criteria for operating and maintenance personnel with reasonable thicknesses of biological shielding. Only a relatively small amount of shielding (reflector) located within the reactor itself undergoes sufficient irradiation damage to require replacement during service life, and this is easily accomplished through the fuel handling system. Calculations indicate that an 18-in. thickness of steel shielding surrounding the reactor in the radial direction (assuming a depleted uranium oxide radial blanket of 13-in. thickness and a steel reflector of 12 inches in the reactor) is adequate for a typical pool-type system. In the vertical direction, a 21-in. thickness of steel above the reactor (assuming an axial blanket of 13-in. thickness and a steel reflector of 12 inches in the reactor) appears adequate; this can be provided either by incorporation within the reactor assemblies or within a reactor vessel cover. Such an arrangement is calculated to require a biological shielding thickness (surrounding the guard tank, and in the shield deck) of about 5 ft of 240 lb/ft^3 concrete or 30 in. of steel, primarily for gamma shielding.

III. GENERAL ASSUMPTIONS AND A CONCEPTUAL REFERENCE REACTOR

Early in the study it appeared that, in order to understand as well as possible the significance of the differences in the various pool-type design approaches that have been taken to date (such as the approach taken in PFR, in EBR-II, BN-600, Phenix, etc.), it might be advisable to try to normalize each of these designs to a common plant size, or power level. A power level of approximately 1200 MWe was selected. It also appeared useful to try to employ in each of these normalized pool concepts exactly the same reactor design. Accordingly, a conceptual "reference reactor" was developed for this purpose. It is the intent in this section first to discuss briefly some general assumptions and considerations applicable to the normalized 1200-MWe concepts and then to describe the conceptual reference reactor design adopted.

During the course of the study, it became evident that the normalized plant concepts could not be developed to the level of detail that had been planned. This was partly due to the time restraint, and partly to the lack of sufficiently detailed information in the open literature on the LMFBR designs that were to be extrapolated. In many areas design details and, frequently, dimensions were not available. As a result, this aspect of the study was de-emphasized, and only rather cursory representations of normalized 1200-MWe concepts eventually were achieved; a partial exception is the somewhat firmer and more detailed representation of an enlarged EBR-II, which was possible primarily because full information on EBR-II was available. Despite the de-emphasis, it is thought to be of some usefulness to note and comment here upon the assumptions that had been intended for use in connection with the normalized concepts.

A. General Assumptions

To make most meaningful the comparisons of the general arrangements and design features of the various normalized 1200-MWe concepts, a set of assumptions and considerations intended to be applied to all the concepts was established. The set is shown in Table III.1. Brief comments on each of the items listed are presented here. More detailed comments on a number of them are made in other sections of the report.

1. 1200-MWe Power Level

The first assumption is that the normalized plants should be of nominal 1200-MWe size, as indicated earlier. No specific study was undertaken to arrive at this power level; it was arbitrarily selected as a reasonable figure appropriate to early commercial LMFBRs. This level is comparable to that of the largest LWRs now planned, the same as that proposed for Super-Phenix, and the same as that used in the most recent General Electric LMFBR study.

TABLE III.1. General Assumptions and Considerations

1. Reactor of commercial size (~1200 MWe)
2. Fuel and reactor design reasonably state-of-the-art

   Fuel: mixed oxide
   Structure: stainless steel
   Appropriate allowance for swelling and creep
   Simple, controlled-bowing core restraint system
   Replaceable modules for support of assemblies
   Hydraulic holddown of assemblies
   Backup mechanical holddown over all of core and blanket
   Instrument tree extending through outer row of blanket

3. Shielding (in-tank) of high reliability; permanent or replaceable by FHS
4. Use of 3 primary pumps and 6 IHXs or 4 primary pumps and 8 IHXs
5. Plant operable with 1 primary pump or 2 IHXs failed (after shutdown)
6. System temperature state-of-the-art
7. Fuel handling system

   Infrequent refueling (annual, 1/3 core and 1/4 blanket)
   Reasonably optimistic assumption re burnup
   Consider in-tank fuel storage, and transfer of assemblies to tank exterior during power operation
   High reliability; consider redundancy of components

8. Consideration of natural convection cooling of reactor under shutdown transient and steady-state conditions
9. Consideration of seismic (SSE) effects
10. Consideration of HCDA, including PAHRS

2. Design State-of-the-Art

The second criterion is that the design of the reactor to be employed should be reasonably state-of-the-art. The purpose of this is to help assure realism and prevent inclusion of questionable "to be developed" features that might unrealistically distort the apparent interaction of the reactor features with the remainder of the system. Accordingly, a conventional configuration of unspoiled geometry is used, with a central cylindrical core and surrounding upper, lower, and radial blankets. Mixed-oxide fuel is used. All structural material is stainless steel. Provision for an adequate allowance for stainless steel swelling and creep is stipulated, primarily because this must be reflected in the magnitude of clearances between fuel assemblies and therefore can affect somewhat important reactor dimensions, such as core and blanket diameters, control-rod-circle diameter, etc.

Stipulation of a specific type of core restraint system appeared desirable because the spatial needs required by some types to accommodate

special components, such as peripheral core formers or movable core clamps, could represent major design perturbations. Use of a controlled-bowing (modified EBR-II type) core restraint system that does not require such components is assumed.

Incorporation of replaceable "modules" for support of the reactor assemblies is assumed required to enable replacement of damaged seats and to permit in-service modification of coolant flow distribution. This is of importance in the study because the use of modules can affect significantly the reactor height and, therefore, the required main tank depth.

Hydraulic holddown of assemblies, together with backup mechanical holddown for all assemblies of the core and blanket, is provided. Similarly, instrumentation at assembly outlets is considered to extend out through the outer row of the blanket. These provisions are of significance in the study, because they effectively establish the diameter of the instrument tree (or holddown tree) required, which in turn can greatly affect design of the fuel handling system and, in some cases, even the location of certain components within the main tank. Instrumentation or backup holddown coverage to the outer boundary of the radial blanket is perhaps overly conservative, but this assumption tends to accentuate the effects on other systems and was chosen for that reason.

More detailed considerations concerning the reactor design and its specific characteristics are presented in Sec. III.B below.

3. Shielding Reliable and Permanent or Easily Replaced

The third general criterion is adopted in recognition of the fact that in-tank shielding in a commercial LMFBR will have to exhibit extremely high reliability. Because the shielding is submerged in the primary system sodium, failure of shielding elements of certain designs (such as graphite encapsulated in steel) potentially can unacceptably contaminate the coolant or, possibly, even produce direct mechanical damage to the system; at the same time, shield-system layout normally tends to make very difficult the location and replacement of any such defective elements. The assumption therefore is made that in-tank shielding must either be of a "permanent" type--that is, known not to be subject to defection during plant life (an example being solid steel elements in locations of moderate fluence)--or be reliably and readily removable for inspection and replacement, as by means of the regular fuel handling system. Considerable investigation of shielding provisions was accomplished in the course of the study and is reported in Appendix A.

4. Three Pumps with Six IHXs or Four with Eight

The optimum numbers of primary pumps and intermediate heat exchangers (IHXs) for a large pool-type LMFBR are not definitely known and

may be different for different pool and fuel handling configurations. Almost all designs utilized or proposed to date for pool-type systems in the range of 250-1200 MWe use either three pumps with six IHXs or four pumps with eight IHXs. For purposes of this study, it was arbitrarily decided to require use of one of these combinations in the development of the normalized 1200-MWe concepts (and, if time permitted, to develop a version of each). This assumption is of significance because the general sizes of the pumps, IHXs, and reactor outlet piping (if any) are established by it, and these sizes directly affect the primary system layout and the required main tank diameter.

5. Plant Operable with One Pump or Two IHXs Down

The fifth item listed indicates that the plant should be operable at substantial partial power with one pump, one IHX, or one intermediate heat transport system failed (the last being the equivalent of two IHXs failed, assuming the usual arrangement wherein each system serves a pair of IHXs). The assumption is of significance because it requires incorporation of valves in the primary coolant system, and the valves can affect the dimensions and layout of the pump assemblies and IHX assemblies (particularly the latter) and, consequently, of the main tank.

6. Temperature State-of-the-Art

The criterion is adopted that system temperatures in the normalized concepts should be state-of-the-art. Again, this is for the sake of realism. The specific temperatures assumed, which are presented and discussed further in Sec. III.B, are of some significance in this study in that they can affect the pump and IHX sizes, the allowances made for swelling and creep (and therefore the reactor dimensions), the achievable fuel burnup assumed, the severity of thermal transients that must be considered, etc.

7. Fuel Handling Infrequent, Fast, Reliable

The seventh general assumption concerns provisions for fuel handling. For commerical plants it is necessary that relatively infrequent refueling and adequately high fuel burnup be achieved in order to realize acceptable plant capacity factor and fuel cost. Accordingly, annual refueling involving replacement of one-third the core and one-fourth the blanket is assumed, and a reasonably optimistic view of attainable burnup (core average burnup of about 80,000 MWd/T) is taken. Details concerning burnup and related core composition and operating conditions are given in Sec. III.B. The desirability of in-tank fuel storage capability (probably in conjuction with capability for transferring assemblies to and from the tank exterior during power operation) appears debatable. However, in view of the possible advantages--increased fuel handling system flexibility and possibly shortened refueling time--it is felt that incorporation of such capability should be at least considered. Similarly, it is assumed that to aid in achieving high overall

system reliability, redundancy of the principal components of the system should be considered. These provisions are of importance to the study because, in addition to affecting the layout of the fuel handling system and the characteristics of its components, they directly affect the primary system layout and tank dimensions.

8. Cooling by Natural Convection

Cooling of the reactor by natural convection in the primary system under shutdown transient conditions (such as following a loss of primary pumps with reactor scram), as well as under shutdown steady-state conditions, is a complex process requiring great attention during the design stages with respect to detailed analysis, selection of components, relative location of components, characteristics of components, etc. Consequently, it is assumed that in the development of the normalized concept, consideration must be given to design features for providing such natural-convection cooling. Some specific design aspects of the various concepts that could be affected are the difference in elevation between IHX outlets and reactor core, incorporation or omission of shutdown cooling coils at IHX inlets, system hydraulic resistances, etc.

9. Seismic Effects Considered

The ninth general consideration is concerned with possible effects of seismic events. All pool-type LMFBR designs employ a very large cylindrical tank containing a massive amount of liquid sodium, the reactor, and all the primary coolant pumps and IHXs. The tank has a collective weight of several thousand tons and is sometimes supported from the bottom, but is more frequently supported from the top by suspension from an overhead structure. It is apparent that seismic considerations could have profound effects on the system design approach adopted. Among the principal design features that could be affected are: the tank support system; support methods for the reactor, pumps, and IHXs; design of the main tank cover and/or shield deck; permissible depth of tanks; tank wall thickness; clearance between concentric tanks; depth of cover-gas space; internal pipe-suspension provisions; etc. Accordingly, it is assumed that safe shutdown earthquake (SSE) effects must be considered in the development of the normalized concepts. Most of the results of the study effort in this area are reported in Sec. IV.

10. HCDA Effects Considered

A final general assumption is made that consideration should be given in the normalized concepts to the possible effects of a hypothetical core disruptive accident (HCDA) and to the possible significance of incorporation of a post accident heat removal system (PAHRS). The principal design features that could be affected are much the same as those noted in the preceding paragraph. Most of the results of the study effort in this area are reported in Sec. VII.

B. Reference Reactor

The purpose of this section is to describe the conceptual "reference reactor" developed for use in the normalized 1200-MWe pool-type concepts of the study. This reference reactor was used without modification in the enlarged EBR-II concept; in the other 1200-MWe concepts, deviations were effected as thought desirable to minimize study effort or to reflect more specifically certain characteristics of the LMFBR design being extrapolated.

It should be noted that there is no essential difference between the design of a reactor for use in a pool-type LMFBR and one for use in a loop type. Hence, the task here was primarily one of selecting the design details required from technology already developed that meet the applicable assumptions and considerations listed in Sec. III.A, and specifying them in a manner usable in the study. In the process of determining the design data thought important, considerable additional data became available which, while not needed for the study, were nevertheless of some interest. For the purpose of presenting a more complete and comprehensive picture of the reactor, some of these additional details are included in the description.

It also should be emphasized that the reference reactor described is conceptual only; it is not meant to represent a detailed design, fully substantiated. Certainly there is no pretext that it has been optimized. What is important for the purposes of this study, however, is only that the concept be such that most designers would accept it as fairly realistic, "in the right ballpark" in respect to most of its major features, and therefore reasonable for use as a reference. It is believed that this objective is adequately met.

For the purpose of this study, the reference reactor is considered to consist of the required reactor assemblies (fuel, blanket, control, and reflector), the core support structure, and the inlet plenum. Some related components discussed in this section, but arbitrarily considered not to be part of the reference reactor, include: the core barrel, reactor vessel, radial shielding, and instrument tree.

1. Reactor-Assembly Array and Operating Conditions

The plan of the reference reactor is shown in Fig. III.1. It is an essentially circular array of hexagonal assemblies, with a two-zone core. The inner zone is composed of 228 fuel assemblies and the outer zone of 138, for a total of 366. There are 31 control rods spread over the two zones, arranged in two approximate circles with a single central rod. The radial blanket is two rows thick and consists of 162 assemblies. The reflector region (or shield) is also two rows thick, consisting of 186 assemblies. The total number of assemblies is 745. The total number of rows of assemblies is 15-17, depending upon azimuth. As indicated in the figure, if a core barrel is employed surrounding the assembly array, an inner diameter of about 16 ft 9 in. is required.

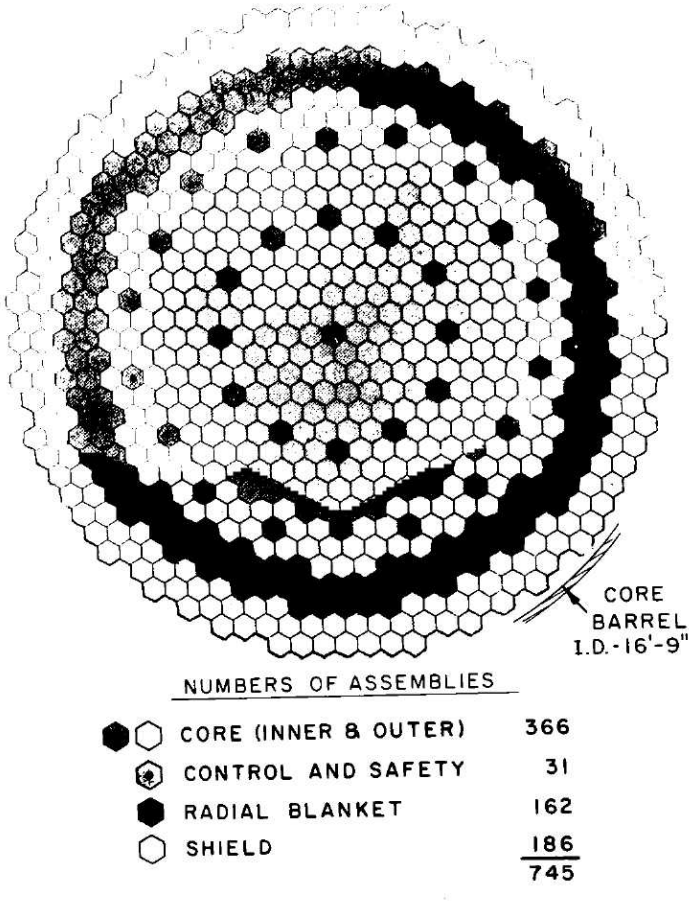


Fig. III.1. Plan of 1200-MWe Reference Reactor

Table III.2 lists some of the principal design and operating data for the reactor. The thermal power level is 3019 MW, and all related data are based on this number. It should be mentioned that no attempt was made to adjust this power level to correspond precisely to a net electrical power output of exactly 1200 MWe; however, based on the operating temperatures employed (discussed later), it is believed to be close. Accordingly, the 1200-MWe capacity should be viewed as a nominal figure.

As shown, the primary system operating temperatures were selected as: reactor outlet, 1000°F; reactor inlet, 720°F; reactor ΔT, 280°F. No detailed study was undertaken in an attempt to optimize these temperatures. They were quite arbitrarily selected as being state-of-the-art, practical, and reasonably consistent with those of existing and proposed LMFBRs of both pool and loop types. Table III.3 compares the approximate temperatures in six such LMFBRs and the reference reactor. These temperatures are referred to as approximate because in many cases more than one value (sometimes several) appears in the literature.

TABLE III.2. Design and Operating Data for Reference Reactor

Reactor power	3019 MWt
Fuel	Mixed oxide
Axial and radial blanket material	Depleted UO_2
Cladding, wire-wrap, and duct material	CW 316 SS
Core volume	10,340 liters
Core equivalent diameter	11.81 ft
Core height	40 in.
Core L/D	0.28
Reactor outlet temperature	1000°F
Reactor inlet temperature	720°F
Reactor ΔT	280°F
Peak linear heat generation rate	13.5 kW/ft
Radial power peaking factor	1.23
Axial power peaking factor	1.26
Maximum coolant velocity in core	~24 fps
Total reactor coolant flow rate	281,900 gpm (720°F) (121.4×10^6 lb/hr)
ΔP through fuel assembly	~66 psi
ΔP, reactor inlet to outlet (closed plenum)	~78 psi

TABLE III.3. Approximate LMFBR Operating Temperatures (°F)

	LMFBR Designation						
	PFR	BN-600	Phenix	Phenix-450	Super-Phenix	CRBR	Reference Reactor
Reactor outlet	1044	1022	1040	1004	986	995	1000
Reactor inlet	752	716	752	734	716	730	720
Reactor ΔT	292	306	288	270	270	265	280
Secondary IHX outlet	990	968	1022	-	-	936	950
Secondary IHX inlet	698	608	662	-	-	651	660
LMTD	54	-	43	-	-	68	55
Steam temp	955	941	943	-	914	900	~910

Calculations indicate that 95.4% of total reactor power is generated in the core and about 2.4% in the axial blanket. The peak linear heat generation rate is 13.5 kW/ft, as indicated; this corresponds to radial and axial peaking factors of 1.23 and 1.26, respectively, the highest values calculated to occur at any time during either the initial cycle or the equilibrium cycle. The average linear heat generation rate for the core is 8.7 kW/ft. Peak total neutron flux is 6.32×10^{15} n/cm^2-sec. All neutronics calculations, including those of burnup as described later, were made with the REBUS-2 (Ref. 1) code and the assumption of all rods full out.

Consistent with the reactor ΔT cited, total reactor coolant flow rate is 281,900 gpm (at 720°F). Maximum coolant velocity in the core is approximately 24 fps, which results in a maximum pressure drop of about 66 psi from fuel assembly inlet (entrance to lower adapter) to fuel assembly outlet. For a pool concept in which a reactor vessel cover is provided to form a closed reactor outlet plenum, and which uses the specific core support structure and inlet plenum configuration adopted for the reference reactor (as described later), the pressure drop from reactor inlet to reactor outlet is calculated to be approximately 78 psi.

Table III.4 shows the principal diameters of importance in the reference reactor. As earlier indicated, the equivalent diameter of the core is 11.81 ft; that of the radial blanket is 14.01 ft, and that of the reflector (shield) is 16.18 ft. Thus, the radial thickness of the blanket as well as that of the reflector is approximately 13 in.

TABLE III.4. Principal Diameters (ft) in Reference Reactor

Equivalent diameters	
Core (inner zone)	9.20
Core (outer zone)	11.81
Radial blanket	14.01
Reflector (shield)	16.18
Center-to-Center distances	
Outermost control-rod positions	10.35
Outermost core assemblies	11.79
Outermost first-row blanket assemblies	12.82
Outermost second-row blanket assemblies	13.87
Outermost reflector (shield) assemblies	16.01
Major diameters (based on unit cell)	
Core (outer zone)	12.44
First row of radial blanket	13.47
Second row of radial blanket	14.52
Reflector (shield)	16.64

The diametral distance between the centers of opposite outermost control rods is 10.35 ft. This is of significance in the normalized pool concepts because, depending upon the fuel handling scheme used, the in-tank volume swept out by the rod drive lines during fuel handling can be of importance (as use of that volume for any other component is denied) and is directly dependent upon this distance. In the concept employing a closed outlet plenum (the EBR-II approach), the magnitude of this distance also can make the difference between having to raise the drive lines for fuel handling a matter of several feet--to clear the plenum walls and surrounding shielding--and having to raise them only a few inches.

The diametral distance between outermost second-row blanket assemblies is 13.87 ft. This dimension is of particular importance because, in conformance with the assumption made that the instrument (and holddown) tree must extend to the second row, the span of the tree must equal or exceed this distance. Thus, the instrument tree will have a diameter of about 14 ft, representing a substantial design problem in view of the coolant velocities, temperature profile, temperature transients, neutron fluence, and long life for which it must be designed. In addition, the in-tank sweepout area of the tree during fuel handling can be significant for the same reason as mentioned above.

The diametral distance between the centers of the outermost reflector (shield) assemblies is 16.01 ft. These assemblies may accrue sufficient neutron fluence during the life of the plant to undergo sensible radiation damage. In accordance with the assumptions made in Sec. III.A.3. concerning shielding, these assemblies must be included within the scope of the fuel handling system capability. Thus, the diametral throw of the fuel handling system must be slightly more than 16 ft, and this dimension is reflected directly in fuel handling system component sizes, including those of the rotating plug(s).

Figure III.2 shows another plan view of the reactor. Here a schematic representation of the necessary radial neutron and gamma shielding surrounding the reactor is included. As effective contributors to the shielding,

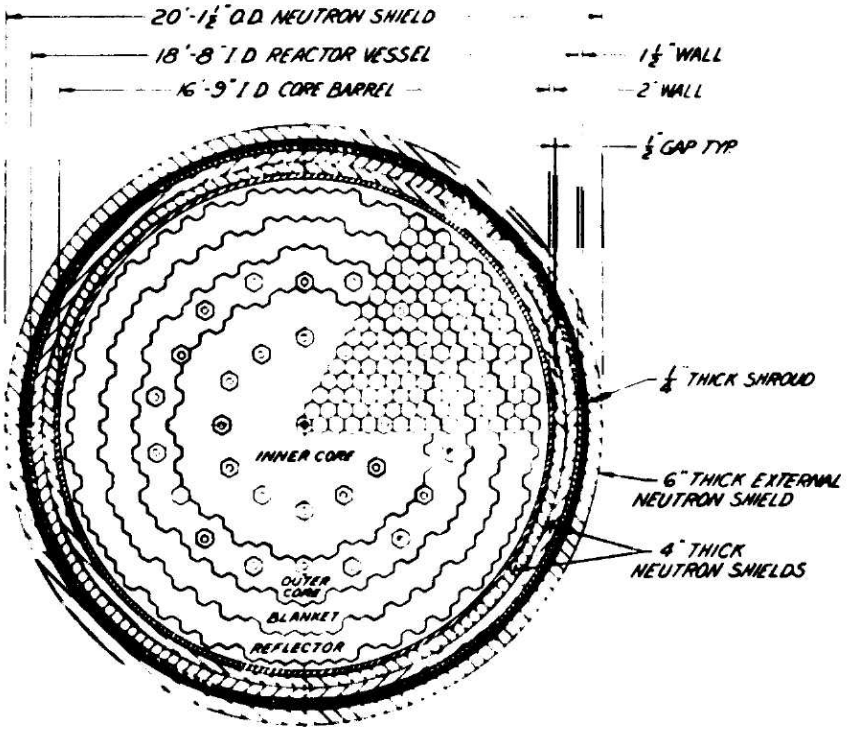


Fig. III.2. Plan of Reference Reactor with Schematic Representation of Radial Shielding

a core barrel, reactor vessel, and reactor vessel shroud are also depicted. To assure that the shielding is of "permanent" character--i.e., that it need not be removed for examination or replacement during the life of the plant--the only material used is stainless steel (a portion of it borated). It consists of two cylindrical sections each 4 in. thick and one section 6 in. thick; the former are located between the core barrel and the reactor vessel, with the latter being outside the vessel, and borated (1/2 wt %). Maximum diameter of the shield system is 20 ft $1\frac{1}{2}$ in. Including an assumed core barrel thickness of 2 in., a reactor vessel wall of $1\frac{1}{2}$ in., and a shroud 1/4 in. thick around the the vessel (to channel and control natural-convection sodium flow past the vessel surfaces), an effective total shielding thickness of $17\frac{3}{4}$ in. is realized with this arrangement. Calculations indicate that this thickness of radial shielding surrounding the reactor is probably adequate to meet all requirements; details of the assumed criteria and calculated results are described in Appendix A.

Figure III.3 depicts a fuel assembly for the reference reactor.* Overall length of the assembly is 15 ft. Each assembly contains 271 fuel elements, is 6.458 in. across flats, and has a duct wall thickness of 0.160 in. The fuel elements are spaced by helical wire wrap. The core section is 40 in. in height, as indicated earlier. Above and below the core is a 13-in.-long axial blanket section and a 3-in.-long reflector. The gas plenum in each element is bottom-located, 40 in. long, and supplemented by a top-located, 3-in.-long plenum accommodating the fuel column holddown spring. Not shown in Fig. III.3 is the length (1 in.) of the top and bottom closures of the fuel elements. Total length of a fuel element, including end closures, is 117 in. A short (9-in.) shield section is incorporated near the top of the assembly to reduce neutron fluence on the instrument tree to an acceptable level.

Assembly spacer pads are provided approximately 8 in. above the top of the core. These are in the form of a raised circular dimple in the center of each flat, projecting 0.155 in. above the surface to effect a nominal cold clearance of about 0.007 in. between pads of adjacent assemblies.

A transition section and orifice plate for flow adjustment are provided at the bottom of the assembly body. A truncated-sphere seating surface, which mates with a conical seat in the support module, is provided at the lower end of the transition section. The distance from the seating line to the bottom end of the assembly lower adapter, or inlet nozzle, is 36 in. The assembly is free to pivot at the seating line; with an assumed 14-mil diametral clearance between nozzle (near its bottom end) and structure, the lateral position of a single, straight assembly is defined to within about 32 mils of nominal at the top end and about 25 mils at the spacer pad plane. The nozzle is designed to

*This figure shows the "short" version of the assembly, which would be employed in a closed-plenum (EBR-II type) normalized plant concept. For all other normalized concepts, because a reactor vessel cover filled with neutron shielding would not be employed, the axial shield section near the top of the assembly would have to be augmented by about an additional 30 in. in length. Thus, for those versions, the overall assembly length is $17\frac{1}{2}$ ft.

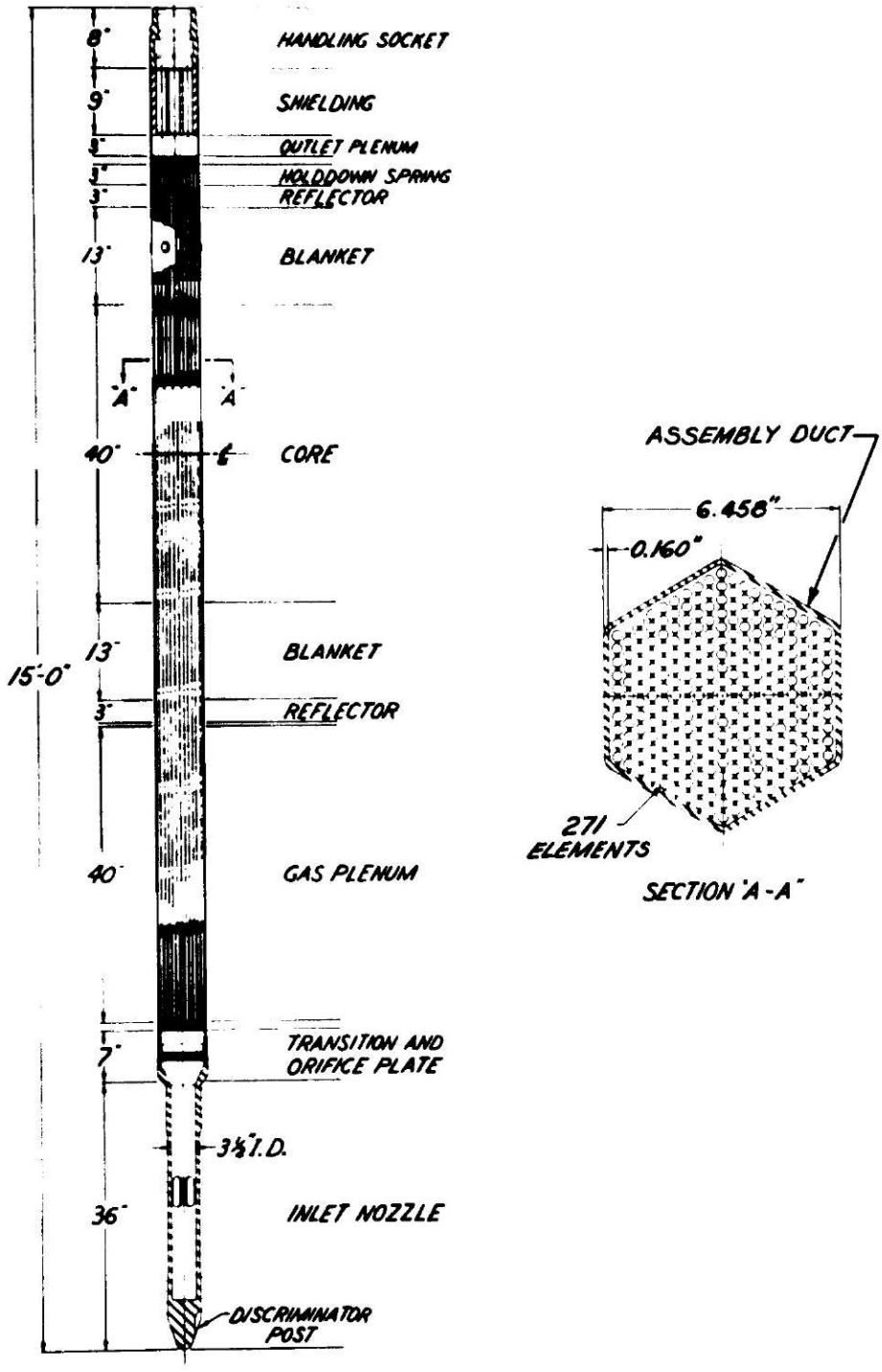


Fig. III.3. Fuel Assembly for Reference Reactor

have a low moment of inertia compared to that of the assembly duct and can accommodate considerable displacement of the assembly near its upper end by nozzle bending.

Table III.5 lists some additional fuel assembly design data. The assembly pitch is seen to be 6.774 in., resulting in a nominal gap of 0.316 in. between assemblies of a virgin core. This magnitude of gap is calculated to be adequate for the peak neutron fluence (E > 0.1 MeV) expected with three-year (900 effective full power days) residence time. In this calculation, the effects of swelling and radiation-enhanced creep (accounting for pressure differential across the duct wall) plus a nominal clearance for fuel handling are included, as indicated in Table III.6. The temperatures shown and employed in the calculation are approximate average coolant temperatures rather than duct temperatures, and the core-centerline fluence values indicated are assumed to obtain throughout the core; both of these assumptions are somewhat conservative.*

The core restraint system is generally similar to that of EBR-II and Phenix, with some modifications. The bottom location employed for the

TABLE III.5. Some Additional Design Data for Fuel Assembly of 1200-MWe Reference Reactor

Fuel element OD	0.300 in.
Fuel cladding thickness	0.018 in.
Fuel/Cladding gap (diametral)	0.005 in.
Fuel pellet density (fraction of theoretical)	92%
Smeared fuel density (fraction of theoretical)	87.9%
Fuel element pitch/diameter ratio	1.220
Spacer wire diameter	0.066 in.
Unit cell area	39.73 in.^{2}
Assembly pitch	6.774 in.
Distance across duct flats (outside)	6.458 in.
Gap between assemblies	0.316 in.
Duct wall thickness	0.160 in.
Peak 3-yr (900-EFPD) fluence (E > 0.1 MeV)	2.66×10^{23} n/cm^2

*However, the relation used for calculation of the duct rounding due to creep has since been found to underpredict this effect. If the relation in the current Nuclear Systems Materials Handbook[2] is used, the calculated value is significantly larger and, in effect, reduces correspondingly the effective handling allowance from the value shown. Nevertheless, it is thought that the total gap magnitude indicated, even with this larger creep effect, is quite probably adequate.

TABLE III.6. Determination of Assembly Gap Size for 1200-MWe Reference Reactor[a]

Location	Fast Fluence, nvt	Temperature, °F	Swelling (ΔV/V), %	Dilation due to Swelling, in.	Duct Rounding due to Creep,[b] in.	Handling Allowance,[b] in.	Total Gap, in.
Core inlet	1.12×10^{23}	720	0.31	0.007	0.033	0.085	0.125
Core midplane	2.66×10^{23}	860	8.38	0.179	0.052	0.085	0.316
Core outlet	1.12×10^{23}	1000	7.59	0.162	0.017	0.085	0.264

[a]Residence time assumed to be 900 EFPDs.
[b]See text for comments.

fuel element gas plenum tends to minimize potential assembly duct displacements due to bowing. Also, the spacer pads are positioned above the core where fluence is low, full reactor ΔT is available for duct thermal expansion, and (partly because of bottom location of the gas plenum) the distance from the pads to the top of the assembly is very short (32 in.). Besides, the radial blanket and reflector assemblies are assumed to have a reduced-diameter section at the top (above the elevation of the core fuel element bundles), which further aids in mitigating potential mechanical interactions at top ends of assemblies due to bowing. Only the single spacer pad plane described is employed, and no core former rings are provided; principal radial restraint derives from the designed stiffness of the reflector assemblies. Although a detailed analysis of the restraint system has not been made for the conceptual design, and seismic response of the assembly array would require futher investigation, it is believed that a detailed design using the general system arrangement depicted can be successfully engineered.

The weight of a fuel assembly for the reference reactor is about 790 lb. Approximate weights for blanket, reflector, and control assemblies are as shown in Table III.7. Total weight of all assemblies is about 360 tons.

Table III.8 lists the reactor volume fractions. Fuel volume fraction in the core is 0.358; coolant volume fraction is 0.407. In the radial blanket, the corresponding values are 0.597 and 0.241, respectively, these being arbitrarily adopted from CRBR (without development of a specific blanket assembly design, an effort thought unnecessary for purposes of the study).

Table III.9 indicates the heavy metal composition of the inner and outer core zones, and of the radial and axial blankets, at the time of charging fuel into the reactor. Core composition is based on use of plutonium from reprocessed light-water-reactor fuel. Enrichment (based on total plutonium) is 15.47% in the core inner zone and 22.51% in the outer zone. Total heavy metal in the core at beginning of equilibrium cycle is 30,279 kg. Table III.10 gives the calculated average compositions of fuel and blanket material discharged at the end of the equilibrium cycle.

TABLE III.7. Approximate Weights of Assemblies for 1200-MWe Reference Reactor (lb)

Fuel assembly	790
Radial blanket assembly	1,250
Reflector assembly	1,180
Control assembly (estimated)	500
Total (745 assemblies)	725,000

TABLE III.8. Reactor Volume Fractions for 1200-MWe Reference Reactor

Fuel Assembly		
Fuel		0.358
Coolant: Within assembly	0.316	
In gap between assemblies	0.091	
Total		0.407
Structural: Cladding	0.109	
Wire wrap	0.023	
Duct	0.088	
Total		0.220
Void (fuel/cladding gap)		0.015
Radial Blanket Assembly		
Fuel		0.597
Coolant		0.241
Structural		0.161

TABLE III.9. Charge Fuel Composition for 1200-MWe Reference Reactor

Isotope	Inner Core	Outer Core	Radial Blanket	Axial Blanket
		Charge Fuel Composition, wt %		
^{238}U	84.365	77.333	99.800	99.800
^{235}U	0.167	0.153	0.200	0.200
^{239}Pu	10.494	15.274	0.0	0.0
^{240}Pu	3.007	4.377	0.0	0.0
^{241}Pu	1.587	2.309	0.0	0.0
^{242}Pu	0.380	0.554	0.0	0.0

Total Heavy Metal BOEC,[a] kg	
Core	30,279
Radial blanket	38,035
Axial blanket	19,575

Enrichment (based on total Pu), %	
Inner Core	15.47
Outer Core	22.51

[a]Beginning of equilibrium cycle.

TABLE III.10. Discharge Fuel Composition for 1200-MWe Reference Reactor (wt %)[a]

Isotope	Inner Core	Outer Core	Radial Blanket	Axial Blanket
^{238}U	75.95	72.59	96.92	97.72
^{235}U	0.07	0.09	0.15	0.16
^{239}Pu	10.48	13.10	2.46	1.94
^{240}Pu	4.05	5.16	0.09	0.05
^{241}Pu	0.84	1.40	0.00	0.00
^{242}Pu	0.47	0.65	0.00	0.00
Fission products	8.15	7.02	0.38	0.14

[a]Compositions are based on 3-yr residence in reactor.

Upon discharge, average burnup in the fuel assemblies removed is about 81,600 MWd/T. Calculated average burnup for assemblies removed from each "row" of the core, as well as that of blanket assemblies removed, is shown in Table III.11. Row, in this case, is equivalent row. The radial blanket, actually distributed over four physical rows, is treated as comprising two equivalent rows. Specific power, averaged over the equilibrium cycle, also is shown in the table. The breeding ratio (integrated over the equilibrium cycle, and including ^{241}Pu decay in the fissile loss) is calculated to be 1.19. As indicated in Sec. III.A.7, all calculations are based on annual refueling with one-third of the core and one-fourth of the radial blanket being replaced.

TABLE III.11. Calculated Discharge Burnups for 1200-MWe Reference Reactor[a]

Region	Average Burnup, MWd/T	Specific Power, MW/kg (fiss)
Inner Core		
Row 2	97,300	0.925
Row 3	97,000	0.923
Row 4	95,200	0.906
Row 5	91,800	0.871
Row 6	88,600	0.840
Row 7	86,000	0.813
Row 8	84,100	0.794
Row 9	80,700	0.763
Row 10	76,500	0.723
Average	85,500	0.857
Outer Core		
Row 11	83,200	0.588
Row 12	65,800	0.453
Average	74,100	0.517
Core Average	81,600	0.740
Radial Blanket	6,280	0.130

[a]Based on annual refueling with replacement of one-third of core and one-fourth of radial blanket.

2. Other Features of the Reactor

The preceding material is devoted primarily to description of the reactor plan, zone configurations, assembly and fuel element designs and compositions, reactor operating conditions and performance data, etc. In what follows, attention is given to the vertical arrangement of the reactor assembly, including the core support structure and inlet plenum.

Figure III.4 shows a vertical section of a possible reactor arrangement using the reactor assembly array already described. In this version, the present CRBR design of replaceable "modules," core support structure, and inlet plenum was used as a basis. The dimensions were modified, however, to accommodate the larger reference reactor assemblies, provide essentially the same coolant velocities within the modules and liners, support adequately the larger and heavier array of assemblies, and reduce somewhat the depth of the plenum. It was also necessary to alter slightly the reference assembly nozzle design to make it compatible with the CRBR type of module design. The radial shielding described in Sec. III.B.1, above, is included in the figure, as is a schematic representation of the instrument (and holddown) tree immediately above the tops of the assemblies and spanning the 14-ft diameter of the radial blanket. The support for the tree is not shown, because its design depends upon the pool design approach employed; in the closed-plenum (EBR-II type) approach, the tree is supported from the reactor vessel cover only a few feet above, but in other approaches it is suspended from the rotating shield plugs some 20 ft above.

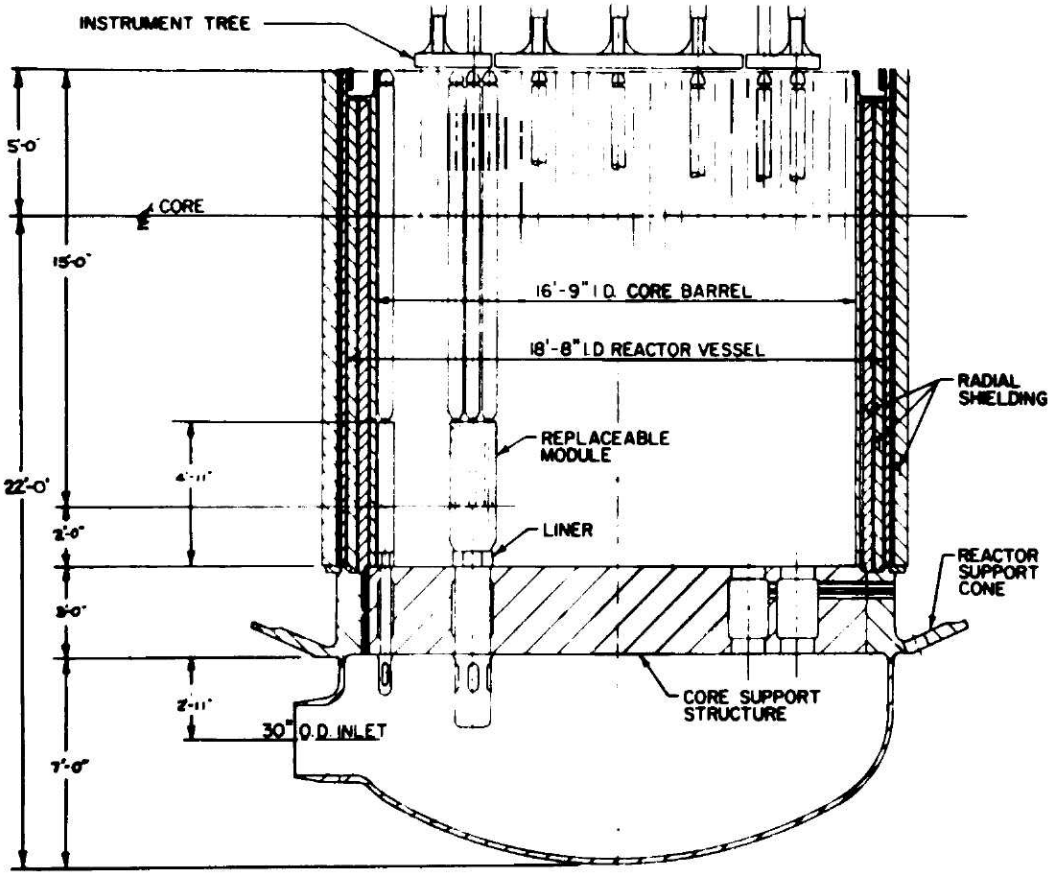


Fig. III.4. A Possible Reactor Arrangement (vertical section)

It is immediately apparent that the distance from the center of the reactor core to the bottom of the inlet plenum in this version is very great (22 ft). As the dimensions of the modules, core support structure, and plenum

were arrived at rather cursorily, it is probable that a detailed design effort could reduce this distance somewhat, but not by a large amount if the same basic features are retained. A dimension of this order of magnitude is undesirably, and probably unnecessarily, large for use with the pool-type system. That is, it is probable that the elevation at which the core will be positioned in a pool-type system will be dictated primarily by the needs of accommodating the fuel handling function and, possibly, by considerations for assuring adequate natural-convection cooling. As a result, because the reactor will be the primary system component at lowest elevation in the system, the distance from the core to the bottom of the reactor will tend to establish the depth of the main tank. In view of the cost of each added foot of tank depth (which is reflected also in tank-cavity and building depth), and because of the effect upon seismic response, minimization of tank depth is of importance. To minimize tank depth it is essential to minimize the distance from core to reactor bottom. Accordingly, an alternative vertical arrangement of the reactor was investigated, which is described next.

This alternative arrangement was adopted as the reference reactor concept for the study. It is shown in Fig. III.5. Radial shielding provisions, a core barrel, a reactor vessel, and an instrument tree are also depicted in Fig. III.5. In this arrangement, the EBR-II design was used as a basis, with dimensions modified as appropriate and replaceable modules incorporated. The principal effect sought in this approach is the utilization of a single horizontal zone for accommodation of the module, assembly support, and inlet plenum provisions (for which three vertically consecutive zones are used in the preceding version). With this arrangement, the three functions are accomplished within a height of about $4\frac{1}{2}$ ft, and the distance of core to reactor bottom becomes less than 12 ft. Compared to the earlier version, the required

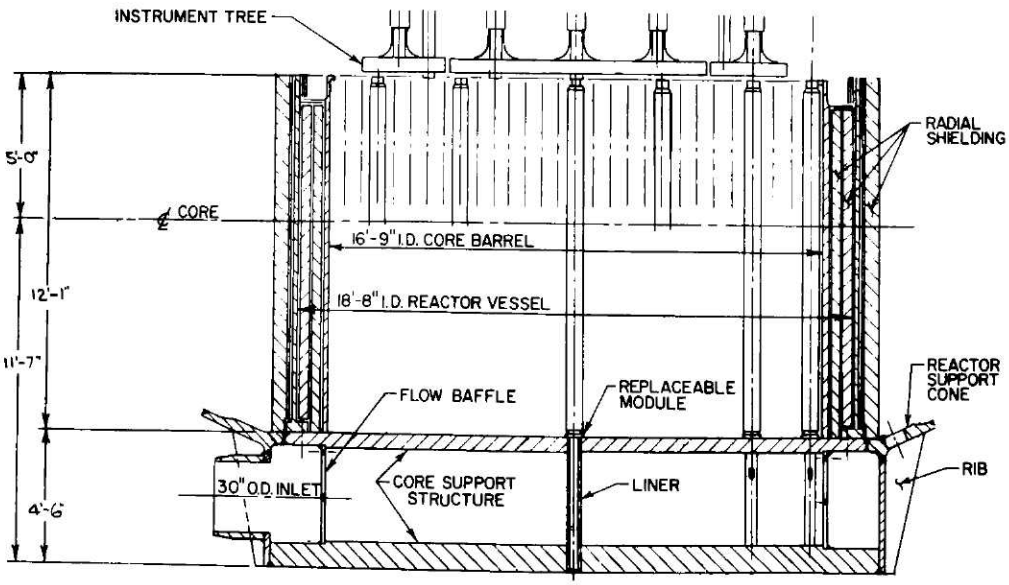


Fig. III.5. Vertical Section of 1200-MWe Reference Reactor (with radial shielding, core barrel, reactor vessel, and instrument tree added)

main tank depth is therefore less by approximately 9 ft (1 ft of the apparent gain being lost because of the flat, instead of rounded, reactor bottom). It is stated again that the arrangement shown is viewed as a concept, not an actual design; it is possible that detailed design study would show that a greater reactor height is the minimum achievable or, possibly, that this general approach would need major modification. What is important here, however, is to note the fact that for a pool-type LMFBR, minimization of reactor height is a significant design consideration.

The core support structure of the reference reactor consists of two heavy, circular grid plates tied together by a cylindrical shell at the periphery, a cylindrical flow baffle plate at somewhat shorter radius, and a series of interconnecting tubes. Each grid plate is perforated with a hole at the location of each reactor assembly (fuel, blanket, control, and reflector). A tube, or "liner," is inserted through both plates at each of these 745 locations and welded into the upper grid plate at its top end and into the lower grid plate at its bottom end. Thus, a forest of tubes spaced at the reactor assembly pitch interconnects the two grid plates. The outer cylindrical shell is fitted with six 30-in.-dia coolant inlet nozzles (or eight smaller nozzles if four primary pumps instead of three are used). Radial stiffening ribs (54) are added to this shell between nozzles, as shown, to provide a degree of additional fixity at the assembly boundary. The structure is supported by a conical shell, shown truncated in the sketch, attached at the edge of the upper grid plate.

The outer cylindrical shell has an OD of 21 ft and a wall thickness of 2 in. Thickness of the flow baffle plate is $1\frac{1}{2}$ in. Radial thickness of the annular plenum (between shell and baffle) is approximately 22 in. Depth of the central plenum is 36 in. Thicknesses of the upper and lower grid plates are $6\frac{1}{2}$ and $11\frac{1}{2}$ in., respectively. Preliminary calculations made using these dimensions (and the liner dimensions referred to later) indicate acceptable stress levels and grid plate deflections for design conditions of 750°F and 140 psi plenum pressure. Operating conditions are 720°F and 97 psi or less. No calculations of seismic or HCDA loading effects were made, however, and more comprehensive calculations might well show that somewhat heavier grid plates would be needed.

As indicated in Fig. III.6, each liner of the core support structure is equipped with a replaceable module in the form of a tube fitted with adapters at both top and bottom ends. The top adapter sits on the top of the upper grid plate, and its upper surface is provided with a conical seat for support of a fuel (or other type) assembly. The bottom adapter provides a locking mechanism of the bayonet type to prevent upward movement of the module; this locking arrangement is of the same design used in EBR-II for locking control assembly thimbles in position. The bottom adapter also provides an assembly orientation (angular) feature and a discriminator feature to preclude undetected insertion of the wrong type of assembly. In this scheme, the module is inserted into position and then rotated 60° to lock it. With an assembly inserted in the

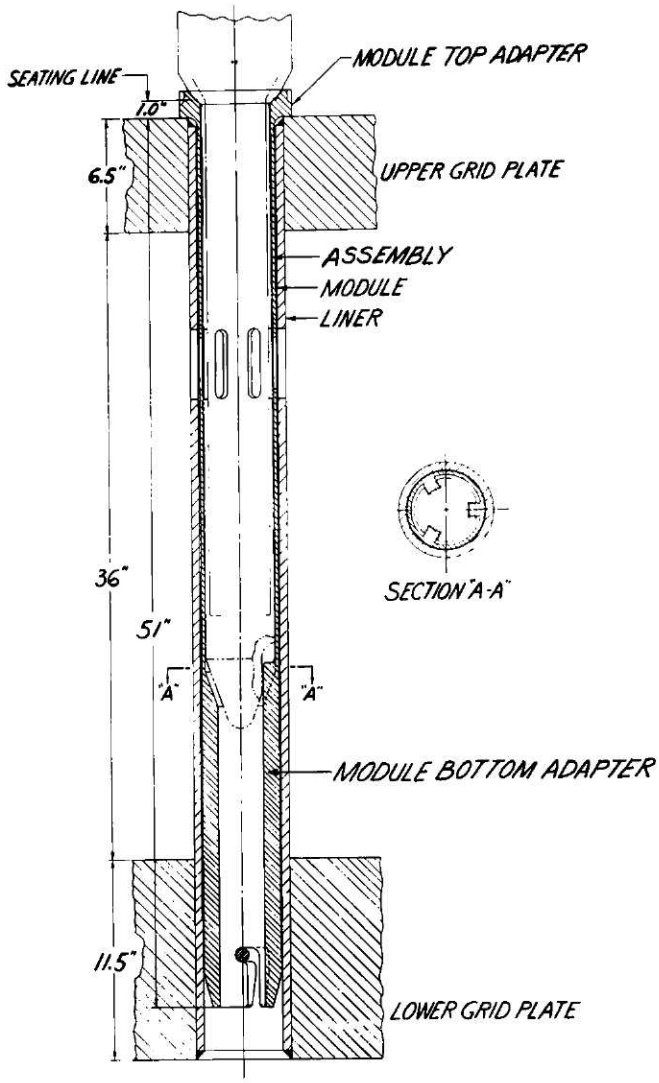


Fig. III.6. Reference Reactor Module Arrangement

module and automatically engaging the module bottom adapter as shown, the module cannot rotate to the unlocked position because the assembly cannot rotate (as long as one or more adjacent assemblies is in the reactor). For module replacement, the assembly is withdrawn, the module rotated to unlock it, and the module then withdrawn. Module replacement may be desired during service either because of damage to the conical seating surface or to change the reactor coolant flow distribution by altering the orificing slots in the module walls.

Diameters and thicknesses of the liner tubes are selected to accommodate the various appropriate sizes of assembly nozzles (and modules) required, provide adequate structural strength, and present the maximum feasible widths of gap between adjacent liners for coolant flow. In the conceptual design, four different sizes of liners are used, with diameters ranging from about $5\frac{1}{2}$ to $4\frac{1}{4}$ in. and flow gap widths from about $1\frac{1}{4}$ to $2\frac{1}{8}$ in. With dimensions of this order, plenum coolant velocities and pressure drops are acceptably low. Adequate hydraulic holddown of the assembly within the module is also achievable; pressure beneath the assembly nozzle is maintained at essentially pool pressure by leakoff of coolant through the small clearance between nozzle and module, downward through the central hole in the bottom adapter of the module, and then to the pool.

The reactor coolant flow path is through the inlet nozzles into the annular plenum between the assembly shell and flow baffle, radially through holes in the baffle, and then radially inward through the forest of tubes (through the gaps between tubes) toward the center of the inlet plenum. Slots in the wall of each liner enable admission of flow through the liner, through orificing slots or holes in the module, and into entry holes in the wall of the assembly nozzle. The desired number of reactor flow zones is established by suitably

dimensioning the orificing holes in the modules. To effect very small flow rates, such as are required for the reflector assemblies, a slightly different arrangement is used in which much of the pressure drop is still taken across the module wall, but a significant drop also is taken across the assembly nozzle wall. A conceptual arrangement employing 54 holes through the liner. with four small holes through the module and two through the assembly nozzle, all at staggered elevations, appears adequate to throttle flow to the desired rate (about 10 gpm) and still assure against excessive plugging of holes, or flow blockage. Total pressure drop from inlet nozzles to center of plenum, including a $1\frac{1}{2}$-psi drop across the flow baffle, is calculated to be approximately 9 psi. Coolant mixing in the relatively shallow plenum is believed not to be a problem, since in a pool-type system (if advantage is taken of a reasonable fraction of the pool volume to mitigate temperature transients between outlets of IHXs and inlets to pumps) any temperature changes at reactor inlet will be very slow and of limited magnitude, thus greatly reducing the requirement for plenum mixing.

This second reactor arrangement, as described above, was adopted as the reference reactor for the study.

Figure III.7 is a sketch of the reference reactor as it might be employed in a closed-plenum (EBR-II) type of pool design. A reactor vessel cover 3 ft thick and filled with the equivalent of 2 ft of steel for shielding purposes is provided. The cover is, of course, removable for fuel handling

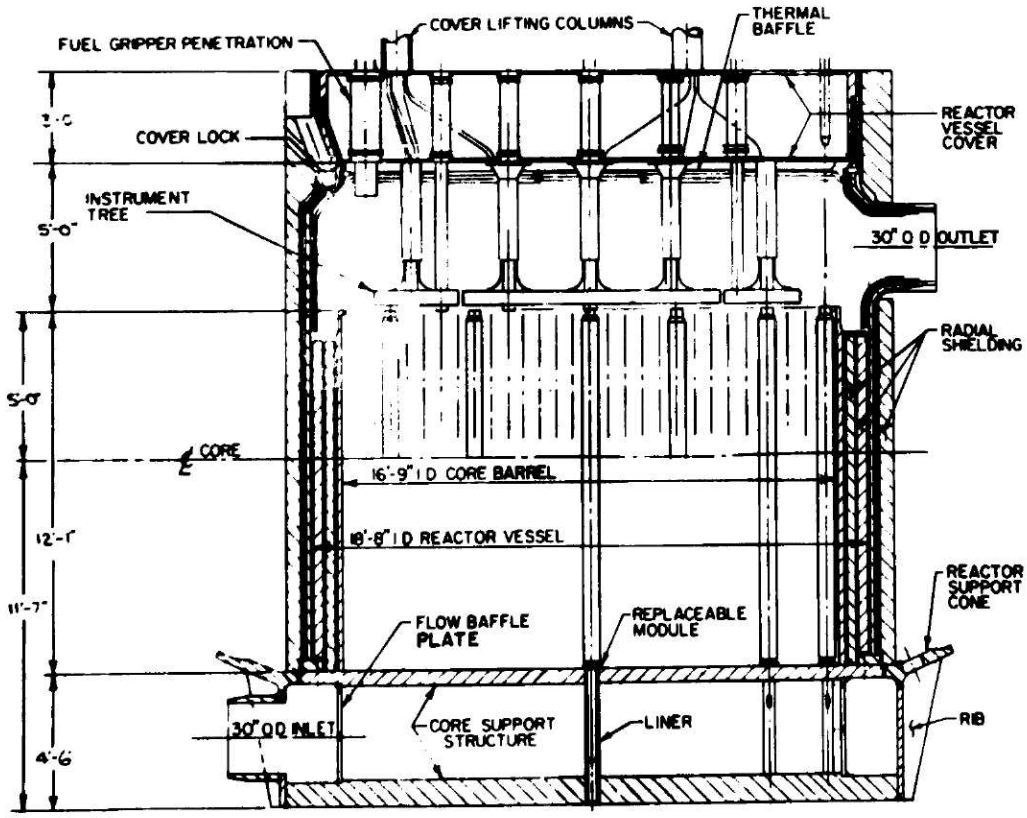


Fig. III.7. Reference Reactor as Used in a Closed Outlet Plenum Design

purposes. Penetrations through the cover are provided for accommodation of the 31 control rod drive lines, two fuel handling grippers, etc. Some of these are illustrated in the figure, the penetration for a fuel handling gripper mechanism (at the left) being rotated somewhat out of position. Locking mechanisms are required to securely lock the cover to the reactor vessel during normal operation. Two sets of three locks each are assumed to be used; one of the locks is illustrated at the left side of the sketch. The instrument (and holddown) tree is suspended from the bottom plate of the cover; in this conceptual design of the system, some of the suspension columns also shroud control rod drive lines. Instrument lines from the tree are brought up into the cover and are there distributed to the cover lifting columns, as indicated, within which they run upward to emerge above the rotating shield plugs. Three cover lifting columns are employed; only the bottom ends of two are represented in the sketch.

For several reasons it is desirable to minimize the diameter of the reactor vessel cover. One is to limit the dead weight of the cover that must be raised and lowered for fuel handling and must be adequately (for seismic considerations) secured to the rotating plug when the cover is in the raised position. Structural design of the cover and of the locking devices also is eased as diameter is reduced; this is significant because the cover and locks may sustain an appreciable pressure difference between outlet plenum pressure and ambient pool pressure--perhaps as much as 20 psi, depending upon the design of the outlet piping and heat exchangers. The cover diameter also directly affects the main tank volume (or sweepout-area diameter) swept out by the cover as the rotating plugs, to one of which the cover is secured, are rotated during fuel handling. Accordingly, the reactor vessel diameter (which is considerably larger than the core barrel diameter, primarily to provide a reactor outlet plenum of adequately large size) is reduced substantially near the vessel top to enable use of a smaller diameter cover. In this conceptual arrangement, the outside diameter of the cover is approximately 18 ft 3 in., and the cover weight including shielding is about 90 tons (in sodium).

The outlet plenum is 5 ft in depth and about 18 ft 8 in. in diameter. The relatively small depth is desirable from the standpoint that it: reduces the required main tank depth; eases the problems of design of the instrument tree, since it reduces the distance between the tree and its point of support (the cover bottom plate); reduces the required length of operating mechanisms such as control rod drive lines and fuel handling gripper; and enables the cover to provide positive lateral location of control rod drive lines a short distance above the control assemblies. However, it is undesirable in that it also reduces the coolant mixing achieved before the coolant enters the outlet nozzles, pipes, and heat exchangers, and thus increases the severity of thermal transients (in particular, upon reactor scram) in these components. As illustrated schematically in the sketch, adequate thermal baffles are required to protect the vessel cover bottom plate, vessel walls, and nozzles.

Note that the radial neutron and gamma shielding surrounding the reactor is extended upward so that it projects above the top of the reactor vessel top flange and to the top of the reactor vessel cover. The shielding calculations reported in Appendix A are based on this configuration.

IV. TANK DESIGN AND SUPPORT METHODS

The main tank, guard tank, core support structure, the major primary system components, and the associated supporting structure comprise a complex, integrated structural system. The design of much of this system is strongly influenced by--and, in turn, strongly affects--operational needs, seismic-response loadings, fabrication and erection methods, and in-service inspectability.

This section describes the nature of these interrelated structures to gain an understanding of the structural problems and solutions associated with key features of existing and planned pool-type systems. Because of its importance to the large, massive tanks and weights involved, special attention is devoted to the effects of earthquake loadings, particularly on the thin-walled main and guard tanks and their support. The material presented addresses the generic aspects of various key structural features. Specific systems in existence are used generally for illustrative purposes.

Three general areas are discussed: (1) structural design; (2) effects of seismic loads; and (3) codes and standards. The discussion in the first area will encompass the following structural elements:

- main tank
- guard tank
- main-tank cover
- main-tank support
- reactor and neutron shielding support
- major component support
- rotating plug support
- thermal baffles
- shield deck

Each of these areas is discussed in the light of the variety of design approaches in use. Included are evaluations of advantages and disadvantages of selected features with regard to structural integrity, resistance to earthquake loads, fabricability, and other related factors.

The effects of moderate to fairly severe earthquakes on pool-type LMFBR primary systems are considered to be of special significance. Therefore, this matter is discussed separately in the second part of this section, concentrating on those features of existing or proposed designs which appear sensitive to seismic loadings. Evaluations are carried out using a base ground acceleration of sufficient magnitude to cover a large number of potentially useful sites in the U. S. for nuclear reactor plants.

Current codes, standards, and NRC Regulatory Guides are evaluated--in a general way--as to their applicability to the structural aspects of the LMFBR pool system, and are discussed in various parts of this section. Emphasis is placed on design, fabrication, erection, and acceptance testing. In-service inspection and surveillance is discussed in Sec. IX, and construction considerations are discussed in Sec. VIII.

Three limitations of the study reported in this section should be emphasized: (1) the study concentrated on selected features; (2) the actual analysis performed must be considered only as preliminary; and (3) the data needed for even preliminary analysis was either unavailable or questionable. Each of the structural areas examined needs far more detailed study and analysis. It is believed, however, that the essential features of each structural area studied have been identified and understood. Thus, the results of further analysis hopefully would affect only the details of the structural features--not the underlying structural principles and concepts. Wherever pertinent data on foreign reactors were lacking, judgments were made on structural arrangement, physical dimensions, and related matters.

A. Structural Design

Major structural features of each of the various pool-type systems studied were characterized generically, thus providing a convenient reference framework for describing the designs and their evaluations. Included in the evaluations were the following considerations:

1. Static loads in the system, including those for tanks, main-tank cover, shielding, supports, and shield decks.

2. Qualitative analysis of the thermal transients.

3. Evaluation of thermal movements or distortions as they might affect structural or geometrical stability.

4. Effect of steady-state temperatures on stress levels.

5. Certain dynamic loadings such as those from a hypothetical core disruptive accident (HCDA) and earthquakes.

For more meaningful comparisons of the various structural schemes in use, the size of the reactors was normalized (or "upsized") as far as practicable to a power level of 1200 MWe. For a system like SuperPhenix, this required no change in physical size, since 1200 MWe is its design power level. For systems such as Phenix, EBR-II, and PFR, all of which operate at lower power levels, the overall sizes, material thicknesses, need for certain modifications required for the larger system, and other items had to be estimated.*

*As discussed in Sec. III of this report, lack of time and design information precluded full use of this reference reactor approach. Therefore, many physical sizes such as tank diameter, overall height, etc., shown in this section are only subjective estimates, and must be regarded as such.

The reference main tank selected for most of the analyses is a double-walled austenitic stainless-steel vessel with an ID of the main tank of 68 ft 0 in. Overall depths of the main tank varied from about 48 to 66 ft. Table IV.1 shows some dimensions of the main tank for existing (or planned) pool-type LMFBRs.

TABLE IV.1. Selected Dimensions for Main Tanks of Existing and Planned Pool LMFBRs

Reactor	ID of Main Tank, ft	Overall Tank Height, ft	Thickness of Main-tank Wall, in.
EBR-II	26.0	30	1/2
Phenix	38.7	33.2	1
PFR	40.1	a	1/2
BN-600	42	41.3	$1\frac{1}{4}$
SuperPhenix	~69	60	a
CFR	~80	a	a

[a]Data not available.

1. Important Structural Elements

Regardless of the size or overall arrangement of any of the pool systems in use, there are certain fundamental functions that the various structural elements must perform. A listing of each of these elements follows, including those functions which must be considered for each of the elements. Figure IV.1 shows a cross section of a general LMFBR pool system and identifies key structural elements.

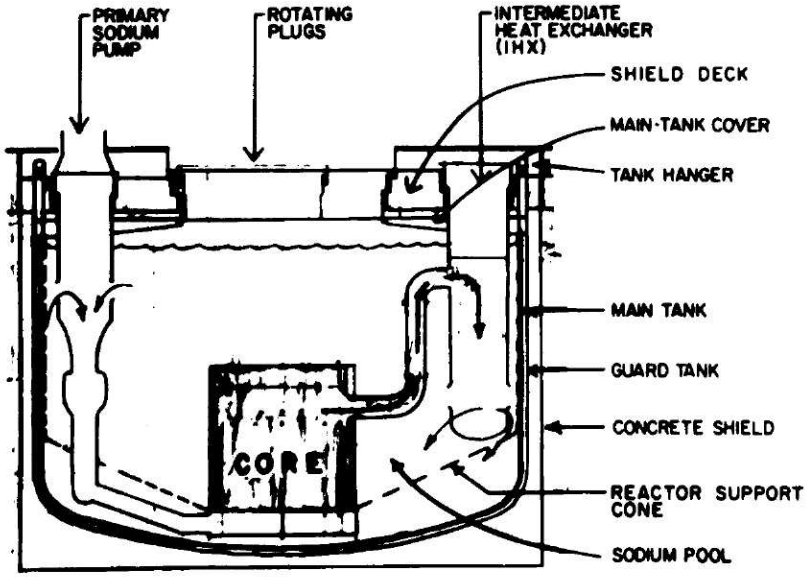


Fig. IV.1. Cross Section of Pool LMFBR, Showing Major Structural Elements

670 - derate
(see loop report)

a. Main Tank and Guard Tank

The primary function of the main tank is to contain the large amount of liquid sodium at a temperature of about 720°F. The sodium may weigh as much as 4000 to 5000 tons. The main tank usually also provides support for the nuclear reactor, the reactor vessel, and surrounding neutron shielding. These components may be supported directly on the bottom head of the main tank (as in SuperPhenix, and the existing EBR-II) or on an inverted conical shell (reactor support cone) welded to the main tank either where that conical shell intersects the main tank (as in Phenix) or above the liquid level (PFR and the "upsized" EBR-II). The main tank may also be used to absorb some of the work energy involved in the HCDA analysis (see Sec. VII of this report).

The major function of the guard tank is to contain the bulk primary sodium in the event of leakage from the main tank. To minimize the drop in bulk sodium level, should a leak in the main tank occur, the annulus between the main tank and guard tank should be as small as possible. Other factors, however, also bear on the size of the annulus. These are: (1) access for in-service inspection (by remote control); (2) erection tolerances; (3) sufficient clearance to avoid "hammering" of the tanks during earthquakes; and (4) clearance to permit expansion of the main tank under an HCDA without incurring significant loads on the guard tank. A gap of about 9 in. was used for the reference design evaluations in this section. An interesting design for the guard tank is that under consideration for the CFR, in which a steel-lined prestressed-concrete vessel is used to back up an independently suspended steel vessel (the main tank).

Generally, the main tank and guard tank are constructed of austenitic stainless steel (Type 304 or 316), although in at least one case--the PFR--the guard vessel is of a low-alloy steel ($2\frac{1}{4}$ Cr-1 Mo). Fabrication and erection are discussed in Sec. VIII of this report. It appears that representative main tank diameters and thicknesses may be of the order of 60 to 70 ft and $1\frac{1}{2}$ to 2 in., respectively, indicating that there will be extensive field welding. The use of good design details and appropriate fabrication and quality control techniques should result in high reliability of these tanks. This has already been demonstrated on the smaller vessels already built.

A noteworthy feature of most, if not all, main and guard tanks is that they are designed to operate at temperatures below 750°F. This is an important feature with regard to anticipated present (ASME) code coverage, which requires use of special analyses and consideration for systems above 800°F. Also, strength properties of the materials are higher at temperatures below 800°F, and problems of creep are minimized.

Finally, all designs in use completely avoid penetrations through either the main-tank or guard-tank walls.* In some cases, significant

*This requires that we define the top conical closure shell on BN-600 as the main-tank cover; i.e., not the "wall."

discontinuity effects do occur, such as the welded attachment of the reactor core support structure to the main-tank wall. In view of the primary benefits to be derived from the pool concept--by virtue of a highly reliable primary coolant boundary--the use of such weldments should be minimized. In no case should a through-the-wall penetration be permitted below the liquid sodium level.

b. Main-tank Cover

In some designs--EBR-II and Phenix--the main tank and guard tank are attached to a main-tank cover or a vessel closure head structure (see Figs. IV.2 and IV.3). This closure serves several purposes, the most important of which are to effect a leak-tight barrier for the inert cover gas over the sodium and to separate the hot structural region of the system from the "cold" shield deck, e.g., the hanger/roller support in EBR-II.

The main-tank cover can also serve to support equipment penetrating into the bulk sodium, as was done on EBR-II. If such use is contemplated, the cover must either have appropriate beam/rib stiffeners or employ a domed shell-like plate (e.g., a torus-shaped form).

The EBR-II main-tank cover also was designed to support all equipment entering the main tank (except the rotating plugs) and to carry a mixture of steel balls and grit about 2 ft deep. This latter feature reduced the shielding thickness required in the shield deck. If shielding is placed in the cover, however, it can probably be beneficial in keeping the overall system depth to a minimum only if the shielding is contained within the depth of the cover beams or ribs. This latter detail requires placement of the gas-tight plate in the main-tank cover below the top of the ribs, thereby resulting in thermal bowing of the cover (due to the ΔT across cover) or thermal bending stresses, or a combination (trade-off). On EBR-II, this gas-tight plate was

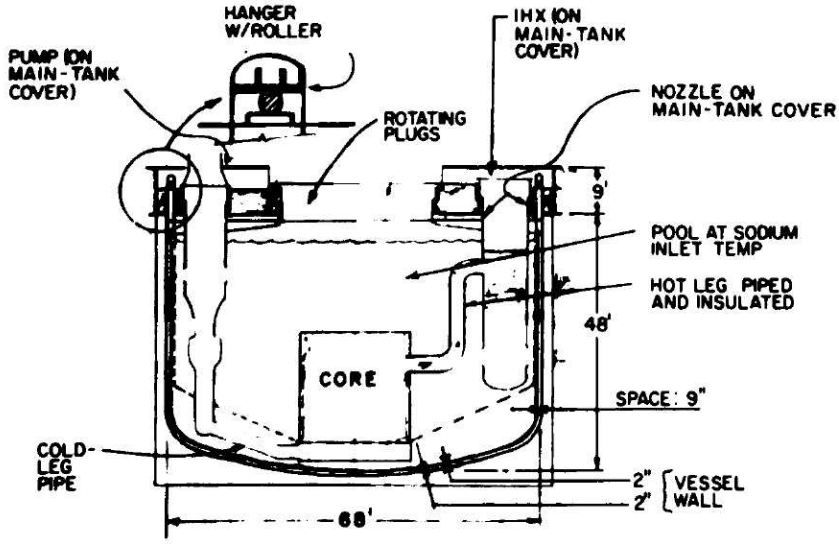


Fig. IV.2. Upsized (1200-MWe) EBR-II Type System

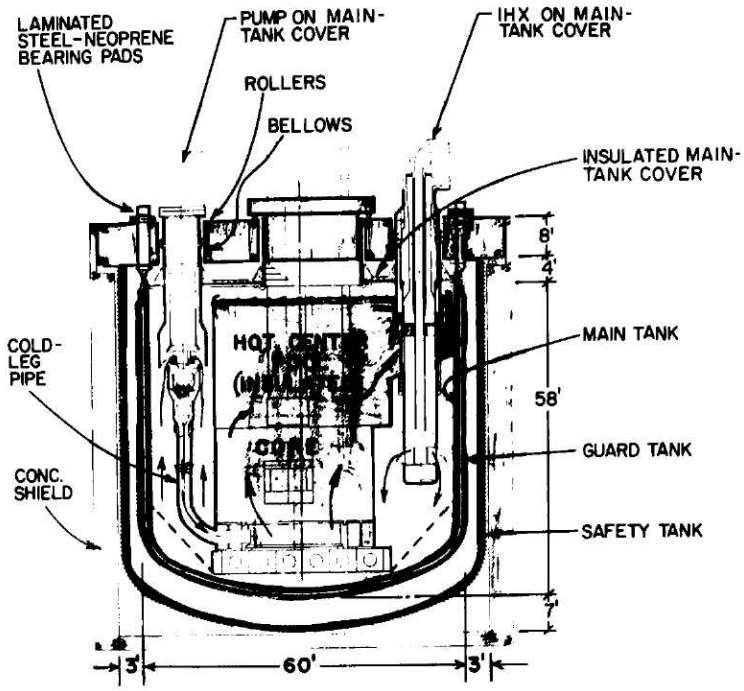


Fig. IV.3. Upsized (1200-MWe) Phenix System

below the ribs, and some bowing occurred. In the upsized (1200 MWe) EBR-II this plate is placed above the ribs and beams.

When the main-tank cover is used to support equipment, such support can be derived from vertical nozzles welded only to the cover structure and free from restraint by the shield deck (see Fig. IV.2). The distinct advantage of this arrangement is that it drastically reduces the temperature differentials within the system and the resulting differential displacements that must be accommodated by either bellows, slip joints, or piping flexibility. (This feature is discussed in greater detail in Sec. VI of this report.)

c. Main-tank Support

The main tank may be top-supported or bottom-supported. Except for BN-600, all present pool designs use the top-support system. (The bottom-supported BN-600 design is described in Sec. IV.A.2, below). For a top support, several approaches have been used: (1) continuous welded skirts (PFR, SuperPhenix); (2) discrete hangers rigidly welded to the main-tank cover and bearing on rollers (EBR-II); and (3) discrete hangers rigidly attached to the main-tank cover and bearing on "bridge-type" pads of laminated steel plates and neoprene. Some versions of concepts reported in the late 1960s indicated pin-connected suspension rods. This scheme has not been put to use, probably because of its poor resistance to lateral forces (such as seismic).

A very important effect of a decision to omit the hot main-tank cover (i.e., use the shield deck as the tank cover) is that the skirt suspension

system which must be used with this scheme needs to be continuously welded to effect a cover-gas seal. This is quite important, since the use of a skirt to accommodate radial thermal growth requires minimizing skirt thickness and maximizing skirt length. Skirts in use at present are carried up into the shield deck and are designed to provide adequate length for thermal stresses (see Fig. IV.4). The depth of the shield deck is reduced at the tank periphery to provide greater skirt length without adding to the overall system height. The welded skirt is the most reliable support system for the transfer of boundary reactions and the assurance of shell action, especially for lateral loads, such as seismic.

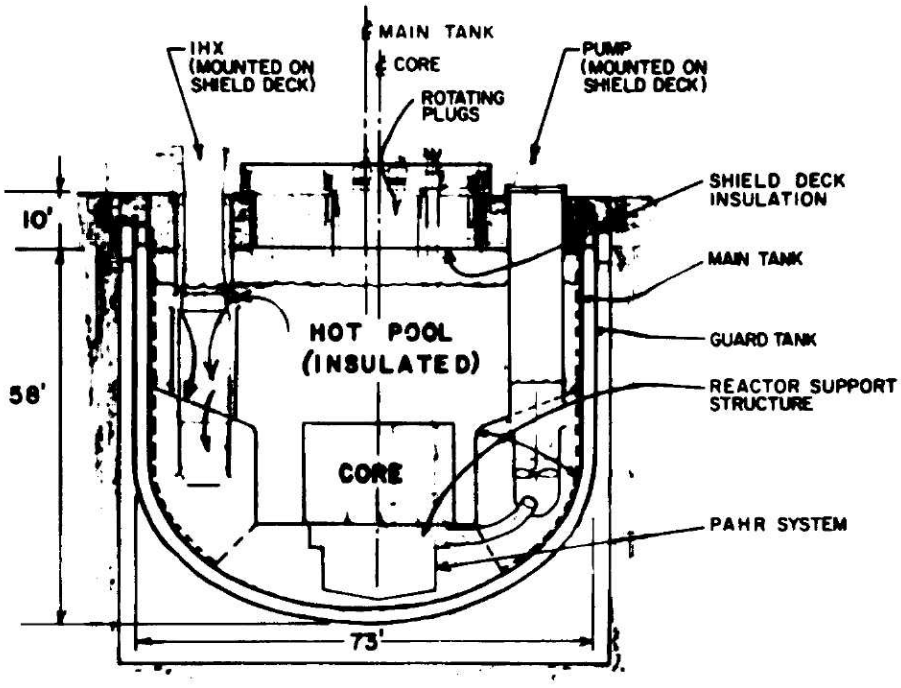


Fig. IV.4. SuperPhenix System (1200 MWe)

The disadvantage of a welded skirt is its rigidity which resists radial thermal growth of the main tank. In other words, the axial temperature gradient in the main tank between the sodium level (or top of the main-tank cover) and the shield deck must be absorbed by the welded skirt in flexure.

Pendulum-type (i.e., pin-connected) hangers are not suitable as a support system whenever seismic forces are involved, no matter whether the hangers are vertical or inclined. It is obvious that these types of hangers are incapable of transferring tangential shear stresses and cannot absorb effectively reactions from moments due to seismic forces.

Hanger-type support systems could possibly be modified to make them applicable for certain seismic conditions. Such a scheme, shown in Fig. IV.5, provides rigidity in the tangential direction, but allows freedom of movement in the radial direction.

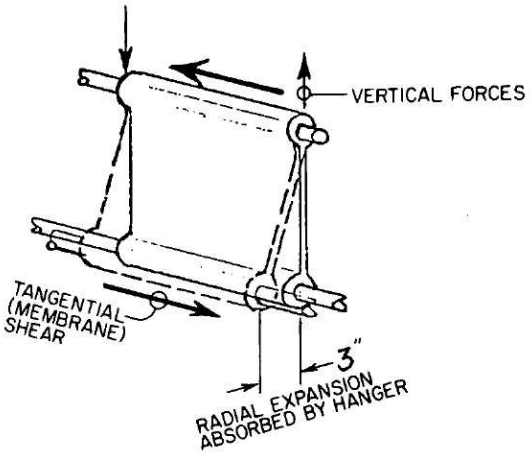


Fig. IV.5

Schematic of Pin-connected Hanger with Tangential Shear Resistance

A support system consisting of discrete hangers rigidly welded to the main-tank cover (and bearing on either rollers or pads) offers several advantages. It can easily be developed for radially free movement and still provide some tangential rigidity. It provides easy access for inspection and maintenance from the top of the concrete. However, such a system must be developed to take uplift forces which may overcome the downward gravity forces under seismic effects (overturning due to horizontal and vertical ground motion combined).

It should be noted that for systems using skirt supports, the designer will be required to reach a compromise between a thin, long skirt to accommodate radial expansion and a thick, short skirt desirable to resist seismic loads. This problem is alleviated considerably if the skirt is used in conjunction with a hot main-tank cover. In this arrangement, the skirt need not be gas-tight and can be slotted vertically to reduce thermal stresses.

d. Reactor Support Structure

The reactor support structure serves primarily to provide adequate support and positioning for the reactor core and blanket subassemblies, the reactor vessel and support grid, and the neutron shielding immediately surrounding these components. This support can be provided by connecting directly to the lower head of the main tank such as in the original EBR-II, in BN-600 (see Fig. IV.6), and in SuperPhenix. Support can also be provided by an inverted conical shell attached on its inner edge to the reactor-vessel grid and on its outer edge either directly to the main-tank wall (Phenix) or to an internal cylindrical-like shell that carries the weight to the main tank above the liquid level (PFR).

In designs employing a hot center pool, the core support structure also provides support for the structures forming the reactor outlet "duct" to the IHXs. This arrangement presents some design problems associated with seismic-load effects; this subject is discussed in some detail in Appendix B.

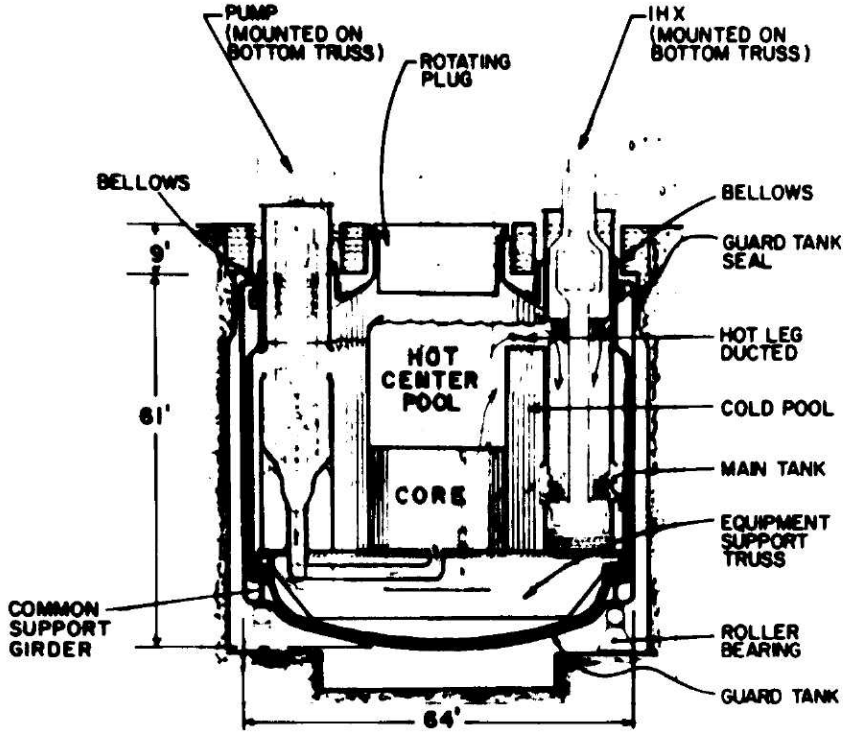


Fig. IV.6. Upsized (1200-MWe) BN-600-type System

e. Major-component Support

In addition to the reactor and shielding discussed above, the structural system must provide support for various heavy pieces of equipment--IHXs, pumps, the rotating plug(s)--and various (lighter) items such as instrument thimbles. Alternate schemes for accomplishing this are described in detail in Table VI.5 of this report. It should be emphasized that latitude in selecting various component support methods is reduced if a separate hot main tank cover is not used.

f. Shield Deck

The main functions of the shield deck include: support of the main tank, the rotating plugs, and (in some designs) the major components; provision of adequate shielding above the sodium surface; and provision of structural resistance to the effects of an HCDA. If no hot main-tank cover is used, the shield deck--properly insulated--must also provide a leak-tight barrier for the inert cover gas.

Usually, the depth of the shield deck required to provide adequate structural stiffness provides ample depth for the shielding material (whether steel or concrete).

For top-supported main tanks, the usual configuration for the shield deck is a "spider" or "spoked" structure comprised of a central ring,

several radial beams, and a bottom plate.* The layout of the radial beams must be carefully chosen to allow ample room for component penetrations and yet provide reasonable symmetry of the structure. For spans upward of 80 ft, the depth of the steel spider would be about 9-10 ft. The radial beams can extend outward to bear on either supporting columns or bearing plates.

When the shield deck serves also as the main-tank cover, its average temperature may be high enough to preclude direct welding of a rigid joint, or the deck may require rollers or rocker-type support pads (at its periphery) to avoid excessive thermal stresses. On the other hand, the temperatures can be controlled to obviate the need for such a detail. Indeed, for CFR, use of a prestressed-concrete guard tank integral with the shield deck is being proposed. In that case, the shield deck and biological shield will form an inherent rigid connection.

Except for BN-600, the shield deck also supports the rotating plugs. On some designs (SuperPhenix, for example), it appears that the OD of the plugs will be quite large (over 35 ft). This results in a shorter span of the shield deck between its outer support and the ring at the plugs. Although the shorter span should simplify design of the shield deck, the structural design of the plugs would obviously be more difficult.

As pointed out earlier in this section, use of the shield deck as the main-tank cover raises problems as to which code requirements, if any, would govern the design. It is interesting to note that the CFR prestressed-concrete guard tank and shield deck would probably be covered under Division 2, Sec. III of the ASME Boiler and Pressure Vessel Code, if the guard tank were the main primary-coolant boundary.

One possible code interpretation could be that the ASME Code covers the main tank only up to, and including, the connecting welds in the shield deck. How acceptance testing would be accomplished is, of course, uncertain. In fact, the exact nature of the acceptance tests for the primary tank is also uncertain. Since the shield deck (or the hot main-tank cover) is normally not under pressure, it is not at all clear what criteria can be applied for acceptability of the design and construction. If, for example, the design of the system stipulates that the shield deck plus the guard tank constitute secondary containment, acceptance tests possibly could parallel those now used for more conventional containment.

It is clear that code requirements must be clarified. It has been suggested, and with considerable merit, that the pool-system tanks and structures serve such a unique function that a more fundamental code approach be taken. In any event, explicit code requirements for pool systems, per se, are lacking.

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*On the proposed CFR, it appears that the shield deck is a thick prestressed-concrete slab.

2. Bottom-supported Main Tanks

The only pool system at present that uses a bottom support is the BN-600, now being built. Figure IV.6 shows the general cross section of the BN-600 upsized to 1200 MWe. Note that the main tank and guard tank are supported from a ring girder (common support girder) that subsequently transfers the weight to a series of roller bearings. The reactor core support structure (equipment support truss), which supports all major components, is also supported by this ring girder. Another unique feature is that the cover of the main tank is an inverted conical shell; because of its inherent strength, it is used to support the rotating plugs.

Before proceeding with specific discussion, it should be noted that time restrictions prevented any but a cursory review of bottom-support schemes. Any conclusions expressed or implied should be evaluated with this in mind.

Although there is no evidence that the Soviets were influenced by earthquake effects, this particular bottom-support scheme seems to offer some advantage in resisting seismic forces. There are several nonstructural difficulties associated with bottom support. One is accommodating all the vertical expansion of the equipment and the main tank in the vicinity of the operating floor level and thus complicating the design of that area. This is covered in some detail in Sec. VI.

Aside from these difficulties, however, there is one major undesirable structural disadvantage to bottom support: The common structural connection between the main and guard tanks (i.e., the common support girder in BN-600) occurs below the sodium free surface. This seriously compromises the reliability inherent in the simplicity of the suspended main-tank and guard-tank design, since a common failure of both tanks could occur at the support joint (see Fig. IV.7). In fact, cursory examination of generic-type bottom supports seems to indicate that this joint will always represent a relatively rigid junction, thus resulting in high discontinuity stresses. Use of sliding, lubricated plates, shown in Fig. IV.8, could alleviate this condition. However, even

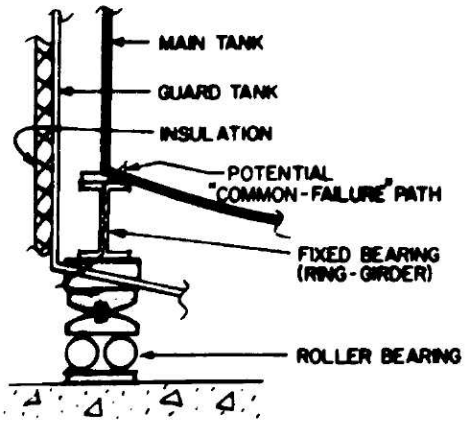


Fig. IV.7

Bottom Support with Common Joint for Main Tank and Guard Tank

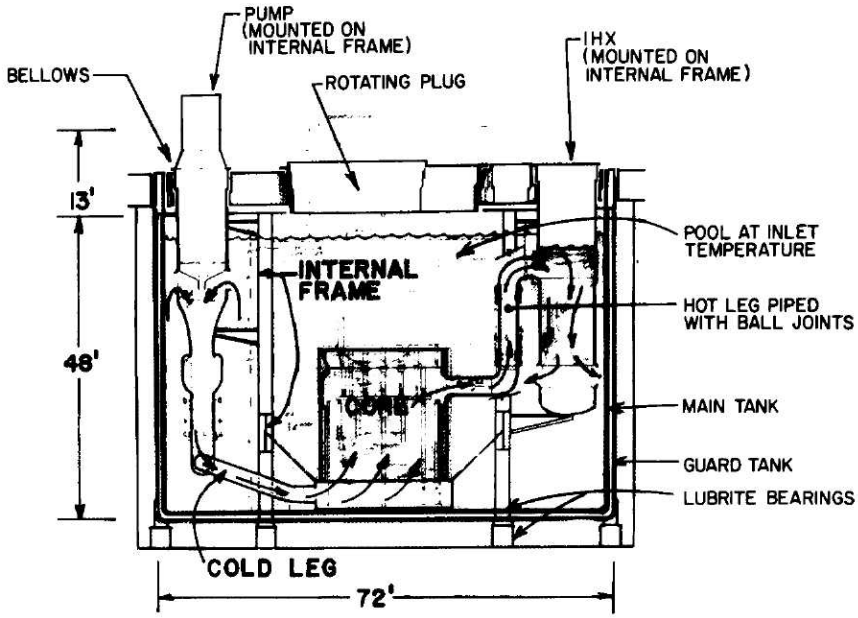


Fig. IV.8. All-bottom-supported Pool LMFBR (1200 MWe), Using Sliding Bearing Plates

though this support scheme could be designed to resist lateral and vertical seismic forces, it does require that these plates be placed between the two tanks (i.e., in the annular gap). Although this would complicate in-service inspection, the main objection is that these plates most likely are permanent and could be replaced only with great difficulty, if at all.

Two other points on bottom-supported systems should be noted: (1) the use of a ring stiffener (or cover) at the top of each tank is of great importance to enhance buckling stability and general resistance to deformation; and (2) as will be shown later, the tensile stress in suspended tanks increases the shear-buckling stability of these tanks. With bottom support, the gravity loads cause no such vertical tension in the cylindrical walls.

Apart from structural considerations per se, one advantage of the bottom support that could accrue is in the erection of the two vessels. The permanent support could provide construction support for the vessels, at least the main tank. This would simplify erection and lessen requirements on lifting equipment.

3. Some Typical Structural Calculations

During the course of this study, it became necessary to perform certain preliminary-type calculations on various structural aspects of the pool system. These calculations were not done in great detail or depth in many cases, but it is believed that they capture the essence of the stress problems. Here is a summary of some of the more important areas reviewed. Work done on seismic effects is presented in Sec. IV.B, below.

a. Characteristics of Main Tank Chosen

For the purposes of the scoping calculations done, we assumed a main tank with the following characteristics:

- Inside diameter of main tank, 68 ft.
- Outside diameter of guard tank, 70 ft.
- Overall depth (from bottom of shield deck), 59 ft.
- Reactor core support structure transmits its load to the main tank above the liquid level.

Gravity Loads, tons

a.	Main tank	600
b.	Guard tank	450
c.	Liquid sodium	5000
d.	8 IHXs, each at 200	1600
e.	4 pumps, each at 200	800
f.	Reactor and neutron shielding	800
g.	Miscellaneous internals	200
	Approx total weight	9450

Seismic Loads

Horizontal and vertical acceleration: 0.3 g.

Thermal Conditions

The maximum temperature of the main tank (and guard tank) was assumed to be 720°F. The main tank wall temperature in the support skirt region was assumed to vary axially from 720°F at the sodium surface to about 160°F at the juncture with the bottom of the shield deck. (Various skirt temperature distributions were assumed, as described later.)

b. Analysis Results and Design Recommendations

(1) Main Tank and Guard Tank

Scoping calculations gave a maximum wall thickness of 2 in. for the main tank. The average wall thickness is $h_{AV} = 1\frac{3}{4}$ in. The minimum wall thickness at points of low stresses is $h_{MIN} = 1\frac{1}{2}$ in. The guard tank requires wall thicknesses of $1\frac{1}{2}$ in. or less. Based on these parameters, the following selected results were obtained for the main tank:

Membrane normal stresses due to gravity loads

Hoop	4.4 ksi
Longitudinal	3.9 ksi

Shear membrane stress due to seismic loads

τ_{max} = 2.2 ksi

Membrane normal stresses due to seismic loads

Hoop	0.7 ksi
Longitudinal	2.9 ksi

(2) Support Systems

The thermal stresses occurring in the main-tank support skirt due to the axial temperature drop across the length of the skirt were estimated using two assumed temperature profiles: (1) a linear temperature distribution; and (2) a cosine temperature distribution. The resulting calculated thermal stresses for a 9 ft-long skirt, 2 in. thick, were 31 ksi and 53 ksi for the cosine and linear temperature distributions, respectively. Since the larger of these stresses plus stresses from other loads exceeds the allowable (ASME) value of 45 ksi it might be necessary to increase the skirt thickness locally at its juncture with the shield deck. Alternatively, the effective skirt length could be increased or appropriate use made of insulation and cooling provisions to improve the temperature profile to reduce the stress levels. Finally, it should be noted that these are steady state conditions; the effects of possible thermal transients--beyond the scope of this study--are important and should be included in any further studies.

Preliminary calculations were made to determine design feasibility of the use of several discrete hangers and to see if the number of such hangers would be reasonable. This set of calculations assumes a total suspended system weight of 10,000 tons. Figure IV.9 is a plan view of the general hanger configuration. Assuming 15 individual hangers, based on convenient spacing for, say, the upsized EBR-II, analysis yields results that indicating that use of discrete hangers is generally feasible and would provide a fair degree of seismic resistance. However, the calculated contact stresses for the hanger pin to bearing block were very high and would need to be significantly reduced.

To reduce pin contact stress, four options are open (which may be used singly or in combination): (1) increase number of hangers; (2) increase pin diameter; (3) increase thickness of hanger lugs to afford longer contact length; and (4) use more than one roller per hanger.

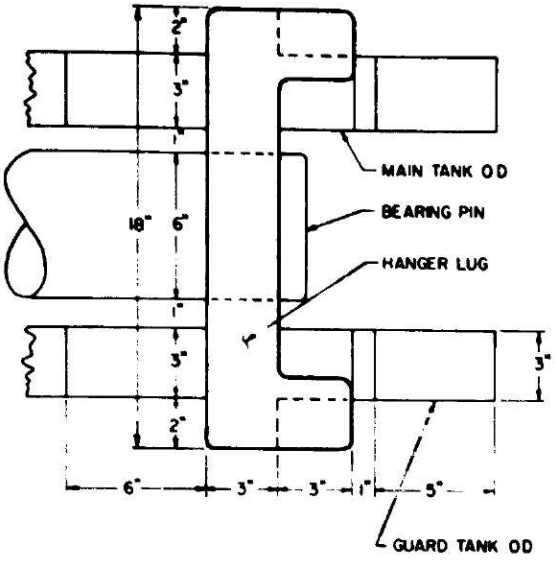


Fig. IV.9

Plan View of Attaching Main-tank Hangers to Main and Guard Tanks for Upsized (1200-MWe) EBR-II Type System

(3) Shield Deck

The shield deck provides shielding above the liquid level of sodium, direct support of major components (in some cases), and support for the rotating plugs. The main load reacting members assumed for this analysis are 15 radial beams which are attached in a spoke-like configuration to an inner circular ring, which supports the rotating plugs, and an outer circular ring, which transfers all loads to the foundation.* The radial beams are assumed to be 9 ft deep and attached to a 2-in. steel floor plate, and are conservatively assumed to act as cantilevers supporting a uniformly distributed load (concrete shielding) and a superimposed end load (rotating plugs).** No credit is given to the structural capacity of the 5-ft-deep concrete slabs.

Conservatively, this welded honeycomb-like structure of 9-ft I-beams, 2-in. steel floor plate, and many integrally welded steel plate compartments can be analyzed in such a way that the 9-ft I-beams take all the load as shown in Fig. IV.10.

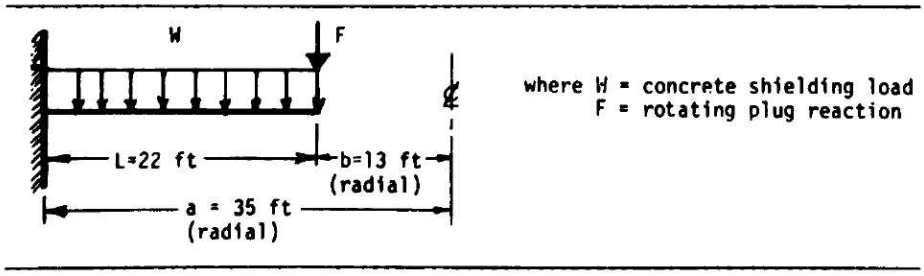


Fig. IV.10. Assumed Loading Model for Typical Radial I-Beam

*Similar to that used on EBR-II and PFR.

**It is assumed here that the 10,000-ton primary system weight is delivered to the outer biological shield independently of these radial beams. This could be done by a skirt or hangers transferring this load to short cantilevers emanating from the biological shield.

Figure IV.11 shows the cross section of the radial beams used in the analysis. The maximum bending stress for gravity loads was calculated to be 12.3 ksi, which could easily be accommodated by a structural-steel grade of plate material. Shear stresses calculated are also reasonable.

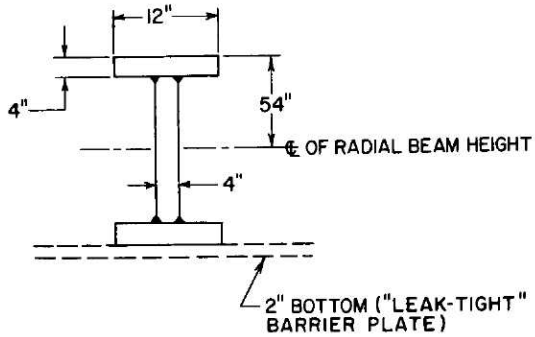


Fig. IV.11

Typical Cross Section of Radial Beam in Shield Deck

Very preliminary calculations of HCDA effects on a structural system similar to the above structure indicate that this design approach is probably feasible (see Sec. VII).

B. Seismic Loads

As is the case for all the structural analyses made in this study, the calculations for seismic response and stresses must be considered preliminary. Especially for early work in this study on seismicity, the intent was to identify key areas, define the magnitude of the design problems, and explore the need for further work to be done. Subsequent to the very preliminary work performed up to November 1975, ANL engaged URS/John A. Blume and Associates--a firm well-known for its seismic work--to both confirm earlier analyses and pursue in greater depth certain key critical aspects of the pool system. This latter work is covered in Appendix B.

1. Reference Earthquake

To evaluate the response to earthquakes for certain critical features of the various pool systems, a reference earthquake had to be selected. For this study, we decided on an earthquake with a maximum ground acceleration of 0.3 g (both horizontal and vertical). This 0.3 g corresponds to the safe shutdown earthquake (SSE) defined by the Nuclear Regulatory Commission (NRC). The operating basis earthquake (OBE) was selected as 0.15 g. The design response spectrum contained in NRC Regulatory Guide 1.60 was used in all work under this study. Figure IV.12 shows the horizontal design response spectra given in that Regulatory Guide. A damping value of 2% of critical damping was assumed for the very preliminary studies performed.

The value of 0.3 g was chosen after careful examination of current positions taken by commercial reactor vendors for the design of standardized nuclear power plants and other factors. In one paper which described seismic

tank (x). When $h \leq 1.5x$, the tank is considered shallow; when $h > 1.5x$, the tank is classified as a tall vessel. Most pool-type systems are considered as shallow (i.e., $h \approx 50$ ft, and $1.5x = 1.5 \times 35 = 52.5$ ft). In some designs, it is possible that the model would call for a tall vessel category; however, for this study, we assumed shallow tanks.

When a tank containing liquid of weight W is accelerated in a horizontal direction, an impulsive force, P_1, and a convective force, P_2, are exerted on the walls of the tank. Dynamic pressures are also exerted on the bottom of the tank due to oscillations of the liquid. The mathematical model of a shallow tank is shown in Fig. IV.14, in which W_1 and W_2 represent the unsprung (non-oscillating) and sprung weights of the liquid, respectively. W_t is the weight of the tank, and heights h_1 and h_3 locate the forces P_1 and P_2. Heights h_1 and h_3, which account for the dynamic pressures on the bottom of the container, are used for computing the overturning moment, OTM_{max}, on a horizontal plane just above the support of the main tank.

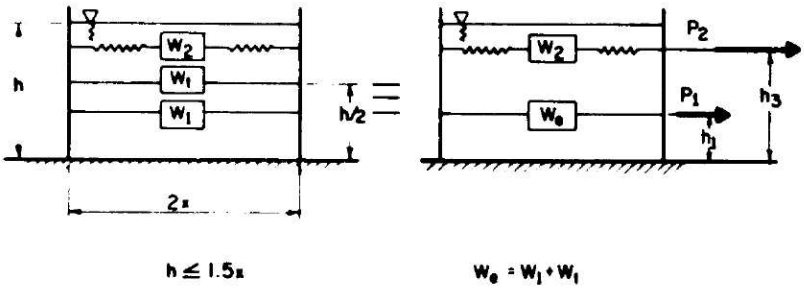


Fig. IV.14. Mathematical Model of a Shallow Tank on Ground or Floor[5] (courtesy of J. A. Blume and Associates)

In the calculation of the forces on tanks, a modified weight, W_e, is used. For shallow tanks, W_e includes the weight of the tank, W_t, and the unsprung weight of the liquid, W_1. In most cases, the weight of the tank is small in comparison with the weight of the liquid.

The analytical relationships used in the seismic analysis and design of shallow (or tall) tanks supported directly on the ground or the building floor have been calculated and are available in the literature. With these relationships, it is possible to determine the maximum liquid surface displacement, the maximum overturning moment on a horizontal plane just below the base (or above the support for a suspended vessel), the maximum bending moment on a horizontal plane just below the support, and the maximum shear at the upper edge of the tank. The wave height, d, is a nonlinear (hyperbolic) function of the input ground acceleration:

$$d = \frac{C_1 A_0}{C_2 - C_3 A_0},$$

where A_0 = input ground acceleration and C_1, C_2, and C_3 are constants determined from charts.

For this study, very preliminary calculations indicated that the wave height, d, would be about 3.5 ft for a support input acceleration of 0.3 g, and about 6 ft for a value of 0.5 g. More detailed calculations performed later in the study indicate that an input acceleration of 0.6 g at the top of the skirt will result in a maximum wave height of about 13 ft. (Those later calculations used 0.6 g for the SSE; for the OBE, the wave height would probably be about 5 to 6 ft.)

If the main-tank cover (or shield deck) is high enough above the liquid level, it will not entrap the sloshing liquid and, hence, will not be subject to any hydrodynamic pressures, as shown in Fig. IV.15. If sufficient freeboard is not provided to accommodate the calculated sloshing, the top of the tank and the main-tank cover (or shield deck) will be subjected to some hydrodynamic loading (see Fig. IV.16). Analysis of such loads indicates relatively low stresses. Furthermore, there seems to be evidence that the wave (or sloshing) effect occurs after the major earthquake disturbance subsides; thus the sloshing stresses may not even have to be added to other seismic stresses. Nevertheless, the designer should verify by analysis that the effect of hydrodynamics will not control the design. If a design should prevent most of the sodium surface from free vibration during the earthquake, then the lateral loads on the tank and structure would increase. This results because the sodium, under those conditions, tends to act as a solid mass.

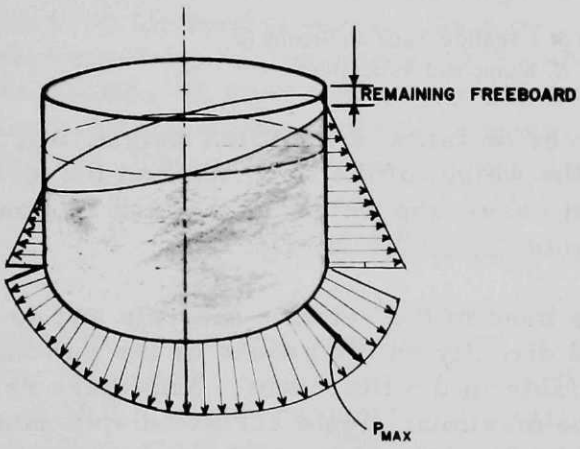


Fig. IV.15. Distribution of Hydrostatic Pressure for Sloshing Fluid When Freeboard Is Greater than Wave Height

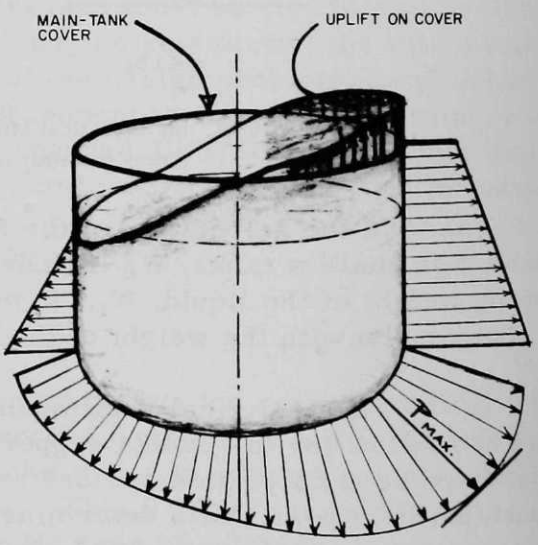


Fig. IV.16. Distribution of Hydrostatic Pressure for Sloshing Fluid When Wave Height Is Greater than Freeboard

The effect of the sloshing on the major components (pumps, IHXs, etc.) may also affect their design and should be included in the appropriate load requirements.

3. Membrane Shear Stresses and Overturning Moments

Seismic response of the main tank and the guard tank is primarily a function of their natural frequencies. Because the response to an idealized earthquake spectrum is an approximation in itself, it was considered sufficient to treat the main tank and the guard tank as single degree of freedom systems. On that basis, they exhibit a natural frequency of vibration of about 20 cycles per second (cps) without any sodium. This value (in the aggregate) becomes about 13 cps when the sodium interaction effects are included. The amplification factor, conservatively assuming zero damping, thereby becomes 1.5 to 2.0 for the acceleration and practically zero for the relative displacement. It appeared reasonable and conservative, therefore, to base the design evaluations on a response acceleration of twice the horizontal ground acceleration. Of course, there will be damping of the order of 2 to 4% of critical damping, thereby mitigating the intensity of the resulting shear stresses and normal stresses. Figure IV.17 shows a greatly simplified seismic response spectrum, and one can see that the large vessels generally will not be in the resonance zone.*

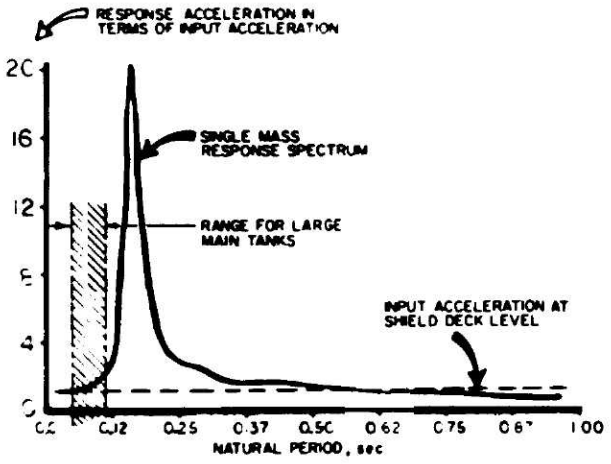


Fig. IV.17

Simplified Typical Seismic Response Spectrum

Gravity loads and thermal loads cause an essentially axisymmetrical state of stress in the tanks. This can readily be analyzed by conventional shell analysis methods. Because of rigid boundaries, a predominantly membrane state of stress prevails. Secondary stresses (bending) can also readily be determined at boundary disturbances, at discontinuities, and at points of load concentration (line loads).

Shear stresses and unsymmetrical normal stresses, however, are a function of the seismic response and of the inertia forces and unbalanced hydrostatic pressure which are associated with it. No precise mathematical solution in closed form is available for these loading conditions. However, several approximate methods allow estimates concerning the magnitude and the distribution of the stresses involved. The final estimates are made by using finite

*A more complete discussion of this is contained in Appendix B.

element solutions. Lateral seismic forces (as well as vertical seismic forces) result in tangential membrane shear stresses and membrane normal stress as shown in Fig. IV.18. As seen from this figure, a skirt-type support is the most efficient, since it transmits these stresses directly to such a skirt support. If discrete hangers are used, there will be concentration of the shell forces, although preliminary calculations indicate a fairly high level of seismic resistance of a properly designed hanger system.

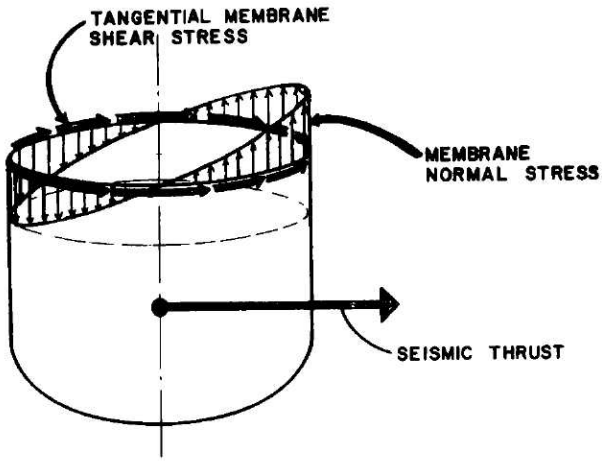


Fig. IV.18

Boundary Stresses (at top of main tank) due to Seismic Forces

The main tank and the guard tank are independent shells with an annular space between them of 9 to 12 in. They must be designed for rigidity to prevent collision under seismic excitation. Since one cannot assume the two vessels to vibrate in harmony, the sum of their relative displacements with respect to ground must be less than the annular space provided. Analysis seems to indicate very low values of relative horizontal displacement; values of only about 0.25 to 0.50 in. are the maximum displacements at any tank location.

An important factor in the design of the main tank and guard tank concerns their buckling stability when subjected to seismic loads. The value calculated for seismic shear membrane stresses in the skirt areas when compared to calculated critical buckling stresses seems to show considerable safety margin. If it proved necessary, the upper part of the vessel (or support skirt) could be thickened, and with an appropriate transition piece, the welding problems would not be affected significantly. It does appear, however, that a thickness of about 2 in. is sufficient to resist the seismic loads assumed.

Some very preliminary calculations on seismic resistance of several discrete hangers show that with proper attention to detail, significant resistance could be developed. One of the limiting factors when rollers are used seems to be the increased contact stresses caused by the vertical earthquake motion and the overturning moment from the horizontal acceleration.

4. Support of Major Components

Probably the most efficient component support design (from a seismic point of view) is to fix the component securely to the shield deck. With the

component supported this way--and possibly if it were supported on a hot main-tank cover--another design feature that may alleviate the seismic design problems can be added. That feature is separate shrouds, or sleeves, surrounding the equipment. If the major internal components such as pumps and heat exchangers are enclosed in fairly rigid shrouds, seismic excitation will result in deflection of the components as well as their shrouds. To prevent collision between the component and shroud, one could, for example, separate them by leaf-spring-type buffers filling the relatively small gap between them.

Another possibility for buffering vibrations of the equipment is to utilize the confined sodium between component and shroud. It may be that under high-frequency acceleration, this annulus of sodium will act as a buffer and, hence, reduce significantly the seismic response of the enclosed equipment (see Fig. IV.19). A limited analysis done under this study is not conclusive; the results indicate a damping factor of perhaps 10% of critical damping. However, 10% damping could be significant and prompts a recommendation for further study in this area.

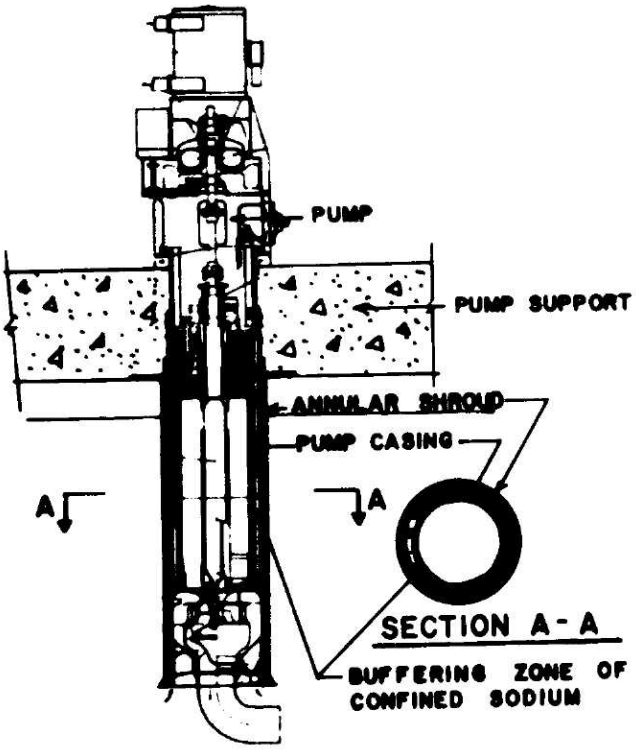


Fig. IV.19

Buffering Effect of Sodium Confined in Annular Space

V. FUEL HANDLING METHODS

This section deals with fuel handling systems of pool-type LMFBRs, as applied to 1200-MWe commercial or near-commercial plants, and is divided into the following topics:

A. Design requirements due to the 1200-MWe size

B. Basic, or generic, fuel handling schemes

C. General fuel handling considerations

D. Optimizations and trade-offs

E. Conclusions

F. R&D suggestions

G. Illustrations of actual arrangements

A. Design Requirements due to the 1200-MWe Size

A multitude of fuel handling schemes and mechanisms exist, each serving a specific reactor. The generating capacity of most of these reactor plants lies in the range of 15-600 MWe. Therefore it seems prudent to consider the most important changes in fuel handling requirements as the reactor plant size increases to 1200 MWe and the commercial aspects of the plant become more dominant.

To be able to compare the benefits (and short-comings) of various fuel handling systems, one has to consider the requirements of the 1200-MWe reference design as described in Sec. III. Some of those requirements are listed below. However, Sec. V.D describes various optimizations and trade-offs that suggest departures from them.

1. The reactor loading area that contains replaceable assemblies will be about 16 ft in diameter.

2. The length of each fuel assembly will be about 15 to 17 ft.

3. Because of the increased fuel assembly size, each fuel assembly will weigh about 800 lb, and each blanket assembly will weigh about 1200 lb.

4. The larger cross sectional area of the assemblies, approximately 36.6 in.^{2}, will contribute to greater stiffness of the assemblies, creating greater friction forces during loading and unloading.

5. Higher burnup limits will lengthen the residence periods of fuel assemblies in the reactor, causing greater swelling and distortion, and further adding to the friction forces.

6. The instrument tree will cover a large portion of the reactor area, approximately 14 ft in diameter. This affects the rotating plug size; Sec. V.D shows some departures from this criterion.

7. The area over which the control rods will be dispersed will be 11 ft in diameter. This, too, affects the rotating plug size and is dealt with in Sec. V.D.

8. The spatial alignment of various mechanisms may have to be adjustable to prevent misalignment caused by Δt's between the main-tank cover and the main-tank internals during fuel handling. The magnitude of this "shift" may be significant because of the increased size of the main tank.

9. The heat generation of each spent fuel assembly may be of the order of 50 kW 24 hours after reactor shutdown. If removal of spent fuel assemblies from the main-tank sodium shortly after reactor shutdown is contemplated, cooling will have to be assured.

10. Containment of the fuel handling mechanisms that extend into the main tank may have to be maintained during fuel handling operations. This may require a "top hat" (an additional metal containment shell) over a part of the main tank.

11. Minimizing reactor downtime will become a major objective. Therefore, the speed of fuel handling operations will be of great importance. One exchange per hour seems to be a desirable goal. This must be coupled with high reliability of the mechanisms. Therefore, simplicity of mechanisms, redundancy wherever possible, and a minimum of fuel handling operations per fuel assembly during reactor shutdown are most important goals.

The preceding requirements refer only to some of the changes in requirements as the reactor size increases. Considerations of safety of personnel and equipment remain, of course, paramount among the requirements.

B. Basic, or Generic, Fuel Handling Schemes

In general, fuel handling is performed by a number of closely interrelated mechanisms or systems, each with its own characteristics and design problems. However, because of their close interrelationship, the versatility of each mechanism affects the complexity of the others.

To illustrate the large number of variations of fuel handling modes, Table V.1 lists the major fuel handling components and their differing versions. Although this table is far from complete, one can derive from it over one thousand reasonable variations. If one adds a few more options, such as off-center cores, versions of rotating plug seals, etc., one can find several thousand variations. Thus it becomes evident that judgment and experience are required to select the most promising combinations.

Fortunately, it is possible to group the fuel handling components into several basic or generic schemes. Some of these schemes have been used in now-decommissioned reactors, some are being used in existing reactors, and others are being considered for planned pool-type LMFBRs.

TABLE V.1. Major Fuel Handling Components

Component	Variation or Versions	Illustration
Rotating shield plug	None One Two Three	
Gripper or holddown	Straight-pull Fixed-offset (1, 2 or 3 each) Pantograph	Fig. V.1
Indexing mechanism	None One	
Reactor vessel cover lift	None One	
Instrument tree	One structure Several structures	
Transfer arm	None One Two	
Fuel assembly storage	Inside reactor structure In rotating storage basket (1 or 2) Outside primary tank	Fig. V.2
Transfer location	None One Two Several	
Fuel assembly handling method	Bare Within a thimble	
Transfer to and from main tank	Casket supported by crane Unloading machine on rails A-frame transfer system Tilted elevator and unloading machine	 Fig. V.3 Fig. V.4 Fig. V.5
Reactor plant traffic	Via well opening Via shielded cell Via airlock, etc.	

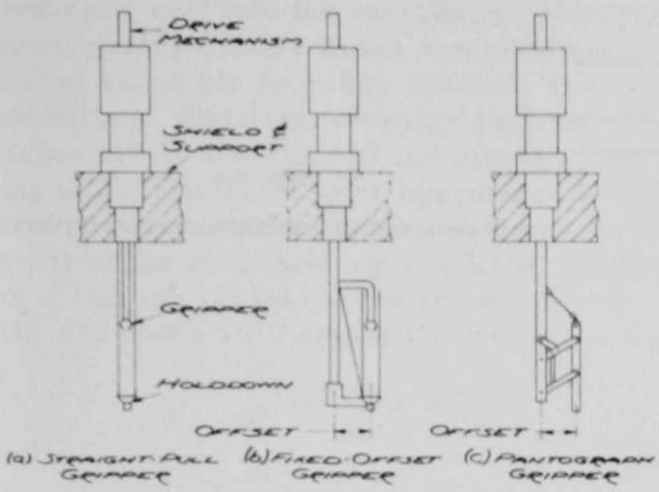
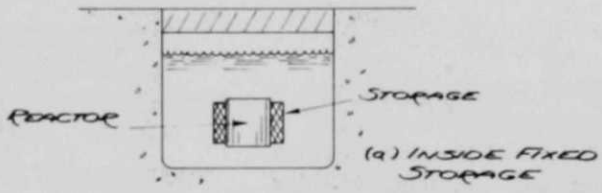


Fig. V.1

Grippers and Holddowns for Fuel Assemblies



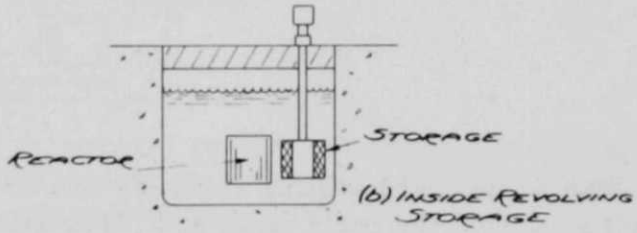
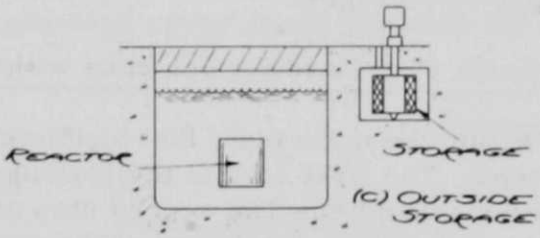


Fig. V.2

Storage of Fuel Assemblies



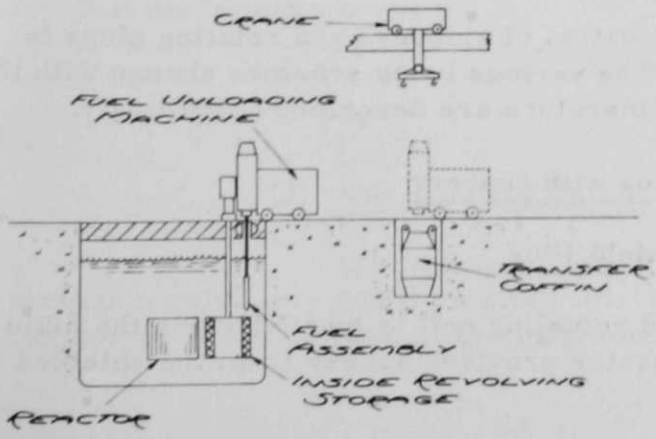


Fig. V.3

Transfer of Fuel Assemblies: Unloading Machine

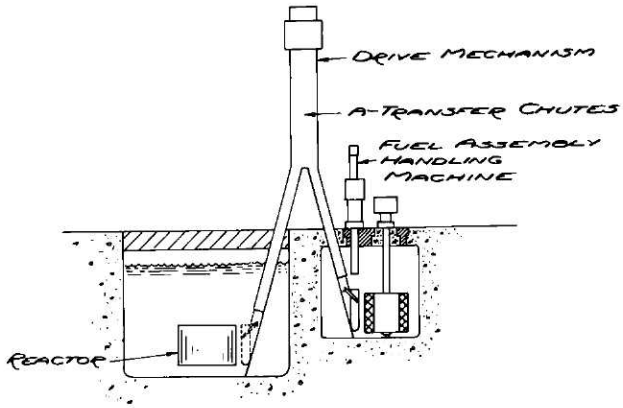


Fig. V.4

Transfer of Fuel Assemblies: A-frame System

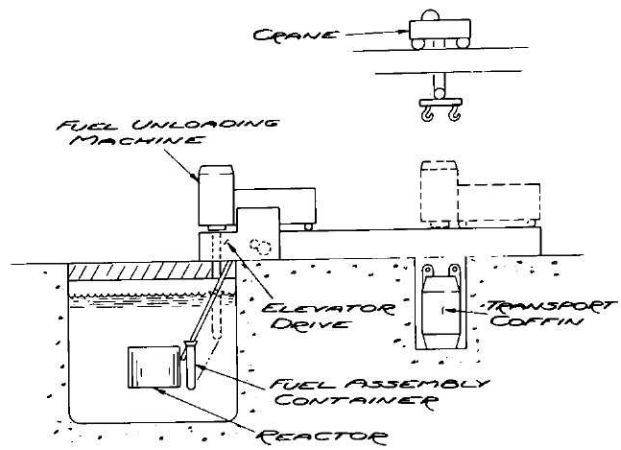


Fig. V.5

Transfer of Fuel Assemblies: Tilted Elevator and Unloading Machine

The basic fuel handling schemes can be divided into two categories: (1) handling within the main tank, and (2) handling outside the main tank (in-plant transport or storage).

1. Basic Fuel Handling Schemes within the Main Tank

Within the main tank, fuel handling concerns itself primarily with two major steps. The first step is the loading or unloading of fuel assemblies with a gripper mechanism. The second step covers operations between the gripper mechanism and the exit port of the main tank.

In the first step a combination of grippers and rotating plugs is needed to reach the reactor area. The various basic schemes change with the number of rotating plugs used, and therefore are described accordingly.

a. Handling Assemblies with Gripper

(1) No Rotating Shield Plug

(a) A shielded refueling cell is located above the main tank. A large opening above the reactor provides access from the shielded

refueling cell into the main tank. Manipulators (grippers) are suspended from gantry cranes in the refueling cell. They dip into the sodium, extract a fuel assembly from the reactor, and deposit it into a submerged transfer container. This transfer container is then removed from the main tank and deposited in a sodium-filled storage tank that is also located in the refueling cell. During reactor operations a shield plug, which supports the control drives and the instrument tree, seals the main-tank opening. At the start of the fuel handling sequence the reactor closure is lifted and moved to a storage pit inside the refueling cell. This scheme was planned for the GE 350-MWe PDP and is illustrated in Fig. V.6.

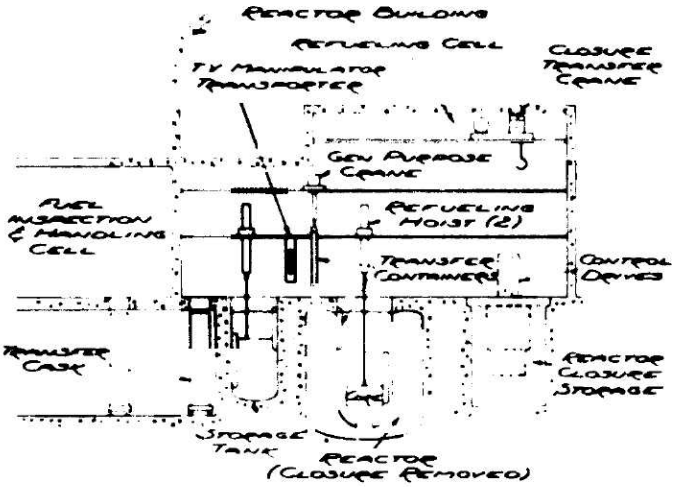


Fig. V.6

Fuel Handling for GE 350-MWe PDP. (Courtesy of General Electric Co.)

(b) One or several pantographs protrude through the main-tank cover. After the instrument tree and the control-drive extensions are lifted, the pantographs reach any location of the reactor area (see Fig. V.7). The number of pantographs depends on the size of the pantograph offset. Direct vertical access through the main-tank cover is not possible for most reactor locations.

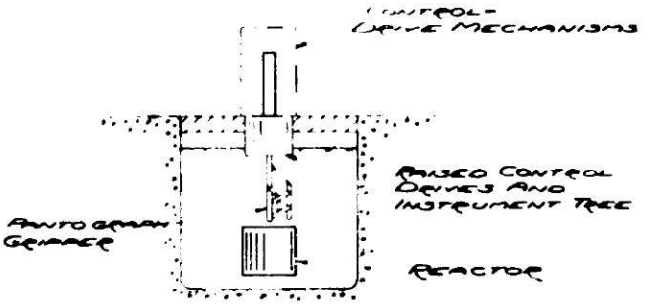


Fig. V.7

Fuel Handling without Rotating Plug

(2) <u>One Rotating Shield Plug</u>

(a) A fixed-offset gripper protruding eccentrically through a relatively small rotating shield plug reaches any location of the reactor area after lifting of the instrument tree, control-drive extensions, etc.

The lift of these components has to substantially exceed the length of a fuel assembly in order to allow for the sweep of the fixed-offset gripper (see Fig. V.8).

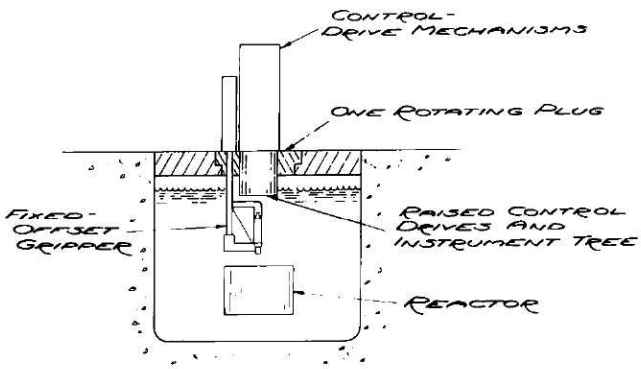


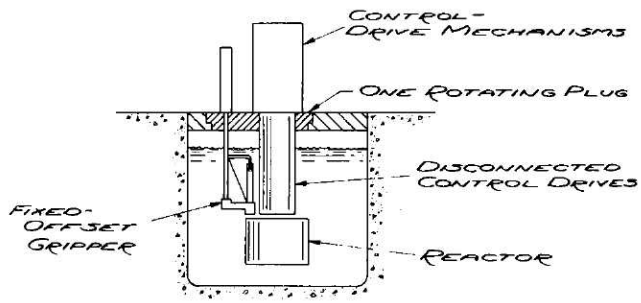
Fig. V.8

Fuel Handling with Fixed-offset Gripper--One Concentric Plug

(b) A fixed-offset gripper protrudes eccentrically through a rotating shield plug that is substantially larger than the one needed for the preceding scheme. However, the larger plug obviates the need for raising the instrument tree, the control-drive extensions, etc., more than a nominal distance above their operating elevation (see Fig. V.9). This general scheme was used in the Fermi I reactor (Fermi I had a small, triangular instrument tree), and is being used in the Phenix-250 MWe plant.

Fig. V.9

Fuel Handling with Fixed-offset Gripper--One Eccentric Plug



(c) A pantograph protrudes eccentrically through a rotating shield plug and operates in a radial "alley" of the instrument tree. The alley path extends from the reactor center to the periphery of the instrument tree. A combination of plug rotation and pantograph movement enables the gripper of the pantograph to reach any location of the reactor area without the need for raising the instrument tree and the control-drive extensions beyond a nominal distance. During reactor operation the pantograph can be replaced with a fold-out instrument section that restores instrumentation in the alley of the instrument tree. This scheme is used in the PFR 250-MWe plant and is illustrated in Figs. V.10 and V.11.

(3) Two Rotating Shield Plugs

(a) A fixed-offset gripper, protruding eccentrically through the smaller rotating shield plug, not only reaches any location of the

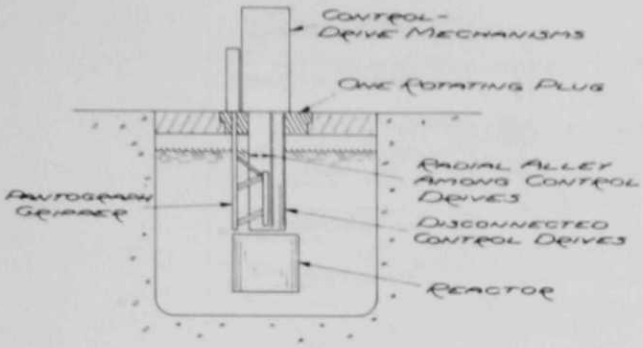


Fig. V.10

Fuel Handling Pantograph Gripper--
One Concentric Plug

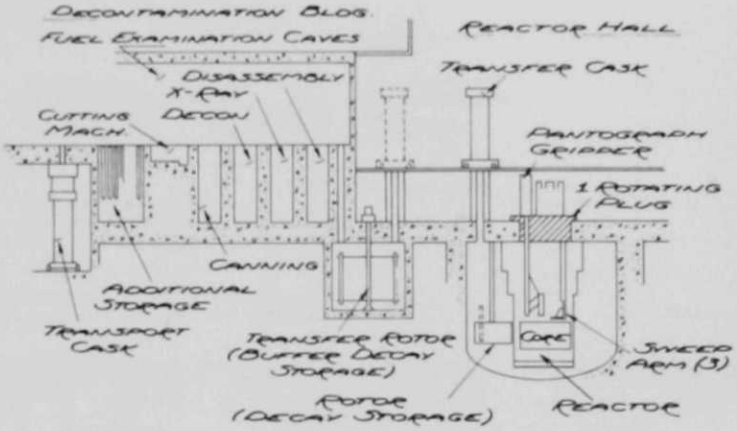


Fig. V.11

Fuel Handling for PFR 250-MWe Plant

reactor area but also provides the extra "swing" to reach an ex-reactor transfer location. The two rotating plugs reduce the interference of the instrument tree, etc., and therefore reduce the overall rotating-shield-plug size required in Scheme 2.b (see Fig. V.12).

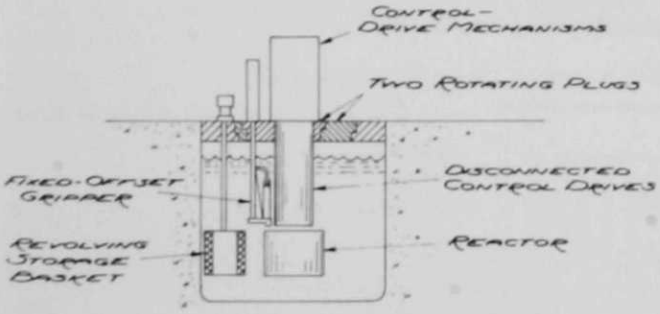


Fig. V.12

Fuel Handling with Fixed-offset Gripper--
Two Rotating Plugs and No Transfer Arm

(b) A straight-pull gripper protruding eccentrically through the smaller rotating shield plug causes the least interference problems with the instrument tree, etc. It is also the simplest and therefore most reliable gripper mechanism. However, in order to reach an external-reactor storage or transfer position, a transfer arm is needed, or somewhat enlarged rotating shield plugs are necessary, or the centerlines of the large shield plug and the reactor must be offset. This scheme is used in EBR-II, BN-350, and BN-600 and is being considered for the SuperPhenix 1200-MWe plant and CFR (see Figs. V.13-V.16).

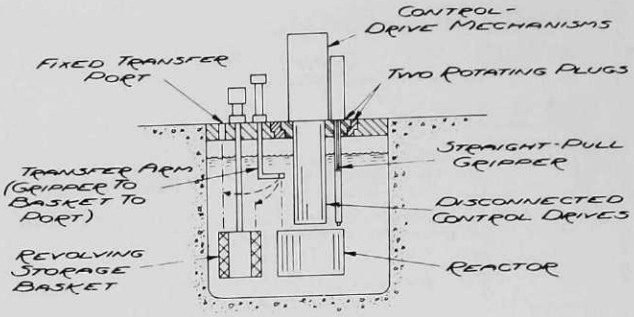


Fig. V.13. Fuel Handling with Straight-pull Gripper--
Two Rotating Plugs and Transfer Arm

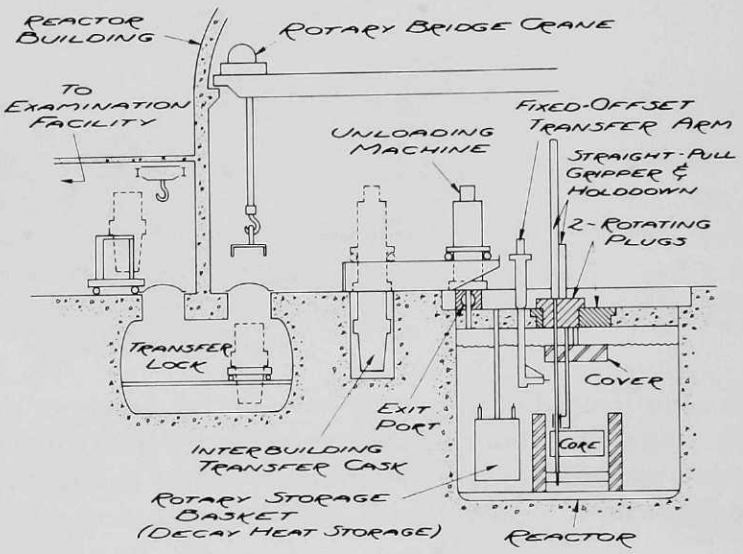


Fig. V.14. Fuel Handling for EBR-II

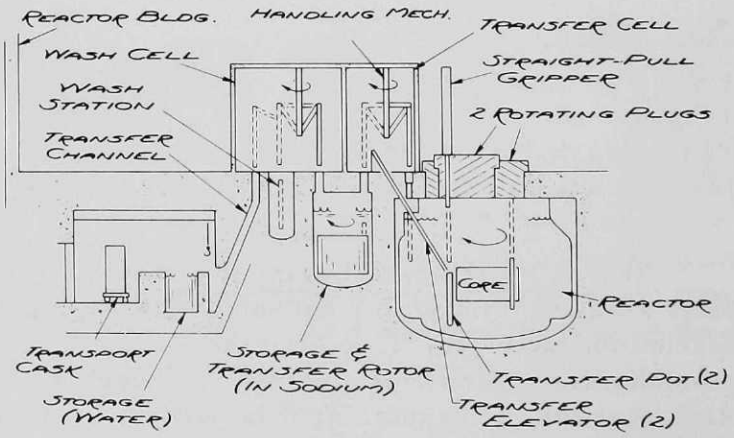


Fig. V.15. Fuel Handling for BN-600

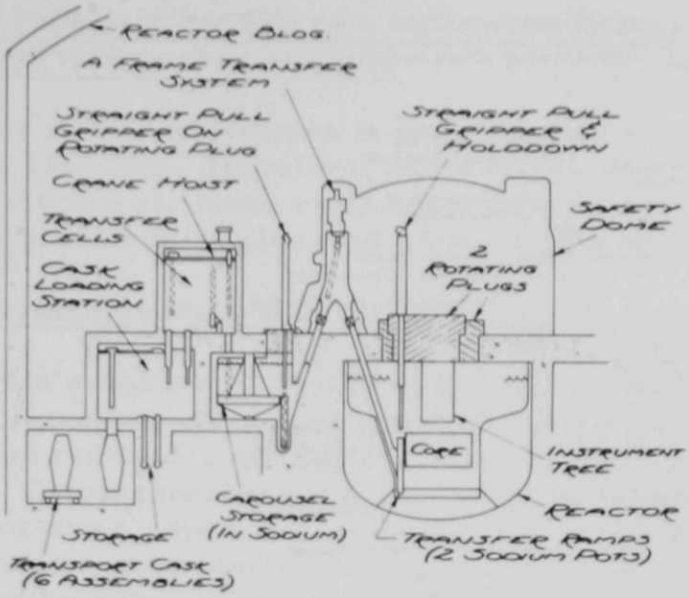


Fig. V.16. Fuel Handling for SuperPhenix 1200-MWe Plant

(4) Three Rotating Shield Plugs

A straight-pull gripper protruding eccentrically through the smallest plug is sufficient. A third plug provides the "swing" to reach an external-reactor transfer location, as provided in Scheme 3.a by the fixed-offset gripper. A large instrument tree area requires relatively large rotating shield plugs. The third plug does not increase these sizes; it only adds to the complexity. This scheme is used in CRBR, the B&W 1000-MWe plant, and SNR-300 and is shown in Figs. V.17 and V.18.

b. Handling between Gripper and Exit Port

The second step covers the remaining fuel handling operations within the main tank.

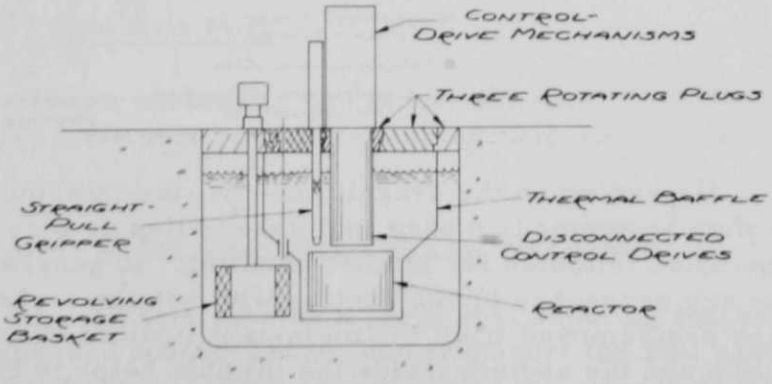


Fig. V.17. Fuel Handling with Straight-pull Gripper--
Three Rotating Plugs and No Transfer Arm

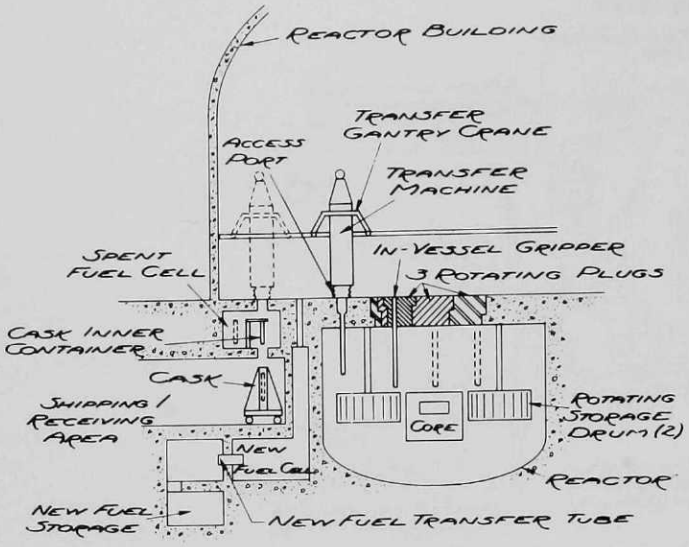


Fig. V.18

Fuel Handling for B&W 1000-MWe Plant

In all schemes an intermediate deposition place or transfer location is provided. This may be (a) a rotating storage basket(s) as in EBR-II, B&W 1000-MWe, and PFR 250-MWe (Figs. V.11, V.14, and V.18); (b) fixed storage locations on the periphery of the reactor as in CRBR, SNR-300, Phenix 250-MWe, BN-350, and BN-600 (Figs. V.15 and V.19); or (c) a transfer location, as presently planned for SuperPhenix (Fig. V.16).

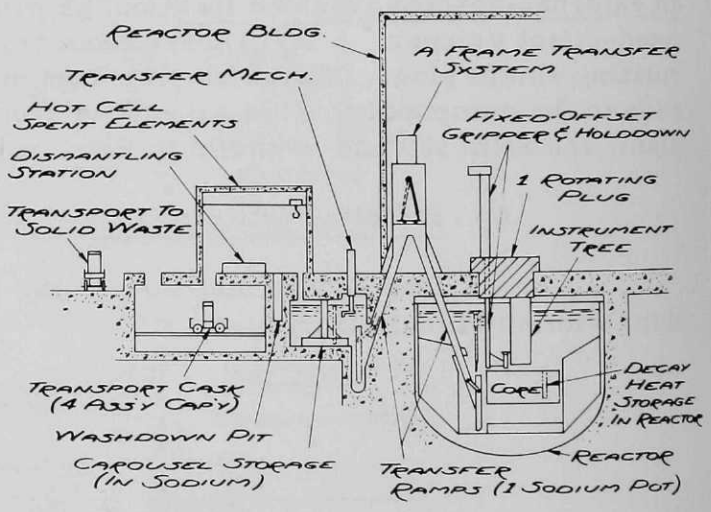


Fig. V.19

Fuel Handling for Phenix 250-MWe Plant

Depending on the overall scheme, the intermediate deposition locations may accommodate bare fuel assemblies or may provide individual, sodium-filled thimbles for further handling. In general the sodium-filled thimbles are necessary in conjunction with schemes where the spent fuel assemblies are removed from the main-tank sodium shortly after reactor shutdown and the sodium inside the thimble helps to disperse the decay heat once the fuel assemblies leave the main tank. Thus the thimbles act in conjunction with the gas cooling provisions of the transfer systems.

2. Basic Fuel Handling Schemes outside the Primary Tank (In-plant transport or storage)

A great variety of schemes is possible, applying equally to pool-type and loop-type LMFBRs. Therefore, only the pool-type-related features are dealt with in this report. Some of them are: (a) removable main tank closure, (b) exit port in rotating plug, and (c) fixed exit port.

a. Removable Main-tank Closure

This scheme requires a large shielded transfer cell above the main tank and extending over a fuel assembly storage tank and over a storage area for the removable main-tank closure (plug). No rotating shield plugs are needed. Gripper mechanisms are positioned by gantry cranes over specific reactor locations. Spent fuel assemblies are transferred in sodium-filled thimbles from the main tank to the storage tank. These operations can be performed by relatively simple mechanisms. However, access into the shielded cell for maintenance or repair of the fuel handling mechanism and control drives is difficult and may become a serious problem when fission products spread through the system. Fuel handling into and out of the main tank can be performed only when the reactor is shut down. This system was considered for the GE 350-MWe PDP, as was shown in Fig. V.6.

b. Exit Port in Rotating Plug

This scheme requires a large fuel handling machine mounted on or supported from a gantry. Handling can be performed only when the reactor is shut down; therefore, fuel assemblies must be transferred in sodium-filled thimbles, and cooling inside the fuel handling machine becomes very important. However, the mobility of the fuel handling machine makes the transport of fuel elements over relatively large distances possible, e.g., to a storage tank inside or outside the reactor building. This scheme is used in CRBR.

c. Fixed Exit Port

Three variations can be considered: (1) fixed exit port and transfer cask, (2) fixed exit port and shielded cell, and (3) fixed exit port and A-frame.

(1) Fixed Exit Port and Transfer Cask

In this scheme a large fuel handling machine mounted on rails, or suspended from a crane, can transport the fuel assemblies over a relatively large distance. This is similar to the preceding Scheme 2.b with two important exceptions. The location of the fixed exit port can be far enough from the rotating plugs and their superstructures so as not to impose severe

space limitations on the fuel handling machine or its support rails. Also, the fixed exit port may be located over a rotating storage basket, allowing this stage of fuel handling to take place while the reactor is operating. This advantage not only saves reactor downtime but also reduces the cooling problems in the fuel handling machine, because spent fuel assemblies may be stored in the main tank long enough to reduce their decay heat generation. This scheme is used in EBR-II, PFR-250, and B&W-1000 and is illustrated in Figs. V.11, V.14, and V.18.

(2) Fixed Exit Port and Shielded Cell

In this scheme a small shielded transfer cell extends over a part of the main tank including the fixed exit port. It is compatible with a permanent system containment during fuel handling and is used in BN-350 and BN-600 in conjunction with tilted transfer elevators (see Fig. V.15). If such a system were to be coupled with a storage basket in the main tank, this stage of fuel handling could be performed while the reactor is operating, thereby providing the beneficial effects on reactor downtime and on cooling requirements as mentioned in the preceding Scheme 2.c(1).

(3) Fixed Exit Port and A-frame

The A-frame transfer system also uses a fixed exit port and is compatible with a permanent system containment. It is used in Phenix 250 and planned for the "stretch" Phenix (450 MWe) and SuperPhenix (as shown in Fig. V.16). If it were to be coupled with a storage basket in the main tank, the advantages would be as already described in Schemes 2.c(1) and 2.c(2) except that the carousel storage (outside the main tank) must be relatively close to the main tank since it is dependent on the spread of the A-frame legs.

C. General Fuel Handling Considerations

The figures presented so far in this section are simplified, stylized sketches of the basic or generic characteristics. To give a better appreciation of how the fuel handling schemes actually look, Sec. V.G presents drawings of EBR-II, Phenix, SuperPhenix, PFR, CFR, and BN-600, showing the overall fuel handling schemes in conjunction with other plant components.

Based on the above-listed requirements for 1200-MWe pool-type LMFBRs and on the generic fuel handling schemes just described, it is possible to list several general considerations:

1. The pantograph needs the smallest rotating shield plug (14 ft in diameter) and has a very beneficial impact on the other fuel handling mechanisms. However, it is very complex and relatively untried at present. It may have a diminished pull capacity, and because of its variable offset capacity

it may also have reduced "feel." In case of its malfunction, removal of the pantograph may cause problems if it cannot be collapsed properly. Because of the versatility of the pantograph, fewer (or no) rotating plugs and other access holes are needed. This may be a disadvantage if some auxiliary handling tools have to be used.

2. Most other schemes need rotating shield plugs of 27-29 ft diameter, and in some cases have sweep areas of 29-34 ft. The number of plugs (one to three) does not affect the size of the large plug significantly; however, if a transfer point outside the reactor area has to be reached, the large plug may have to be bigger in some schemes. In support of these statements, Figs. V.20-V.27 depict various arrangements of fuel handling mechanisms and rotating shield plugs, superimposed on a 16-ft-dia reactor layout. The resulting plug diameters refer to the bottom dimensions of the stepped plug designs.

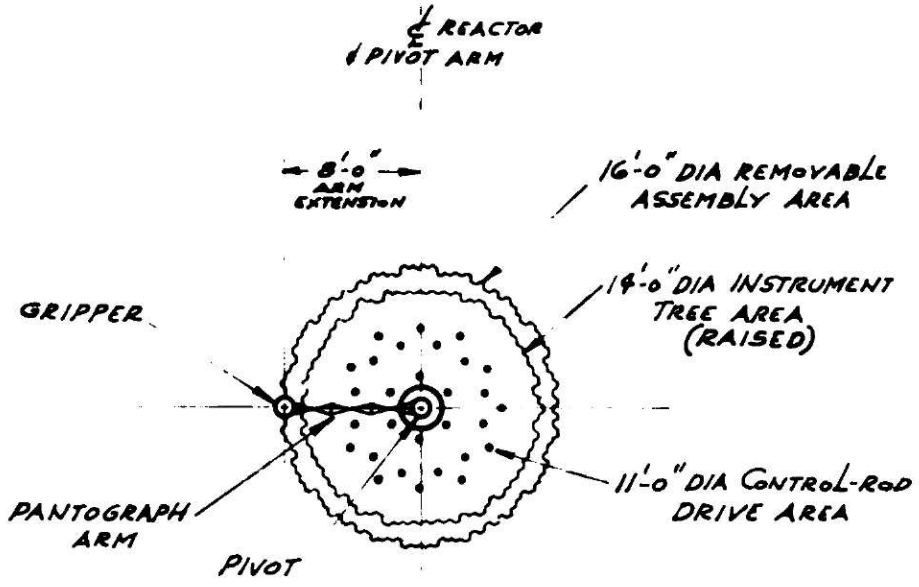


Fig. V.20. 1200-MWe Fuel Handling Scheme—No Rotating Plug and Pantograph Gripper

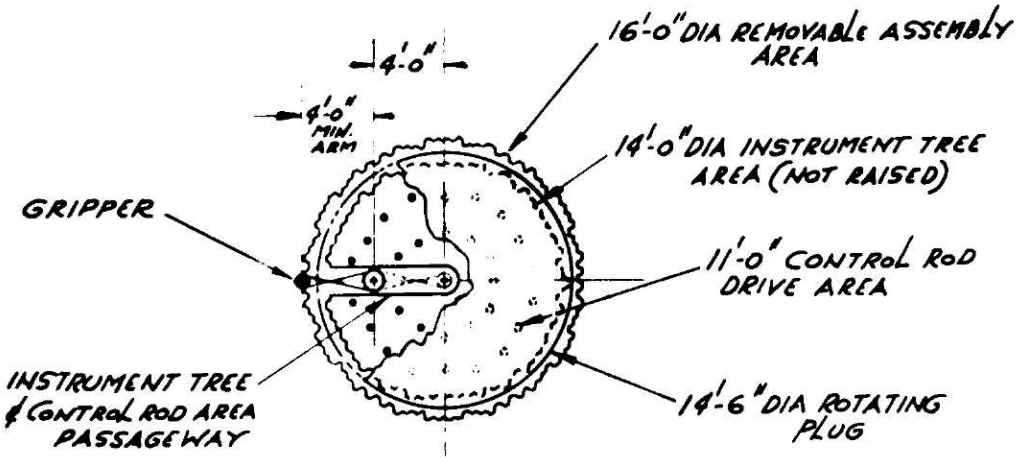


Fig. V.21. 1200-MWe Fuel Handling Scheme--One Rotating Plug and One Pantograph Gripper

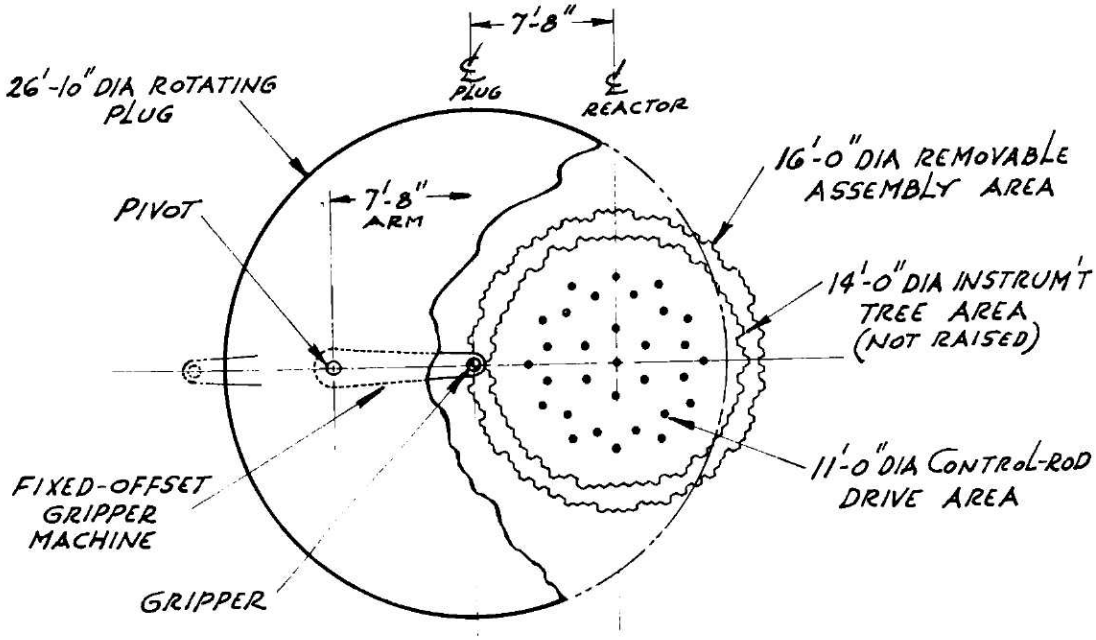


Fig. V.22. 1200-MWe Fuel Handling Scheme--One Rotating Plug and One Fixed-offset Gripper (Fermi-Phenix type)

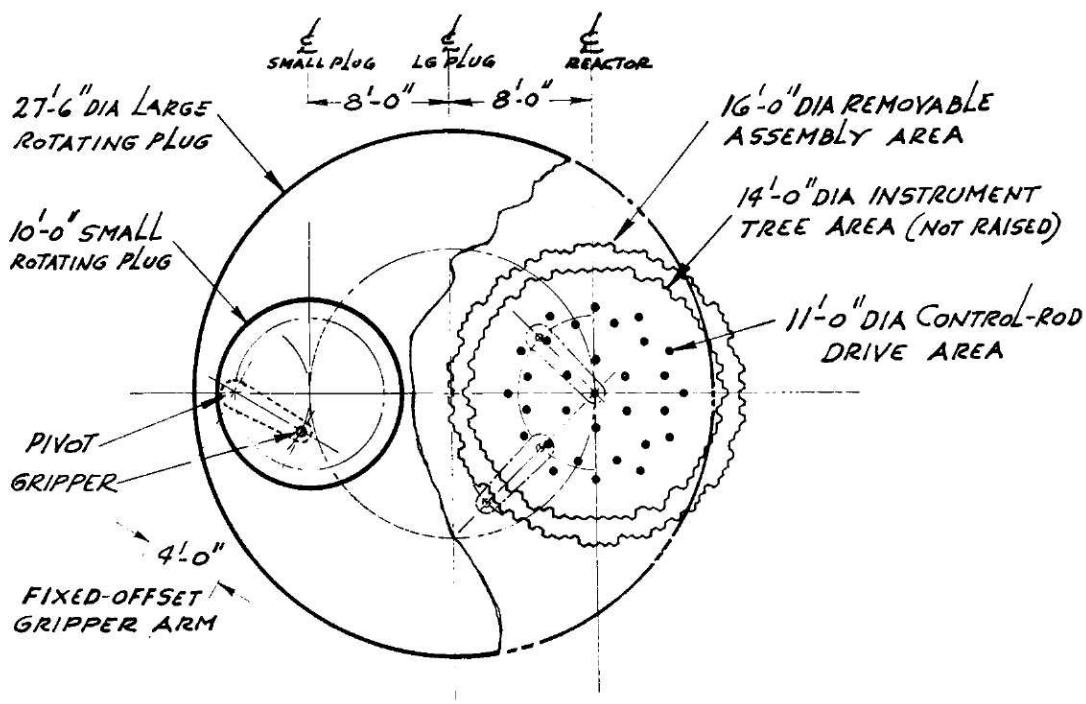


Fig. V.23. 1200-MWe Fuel Handling Scheme--Two Rotating Plugs and One Fixed-offset Gripper

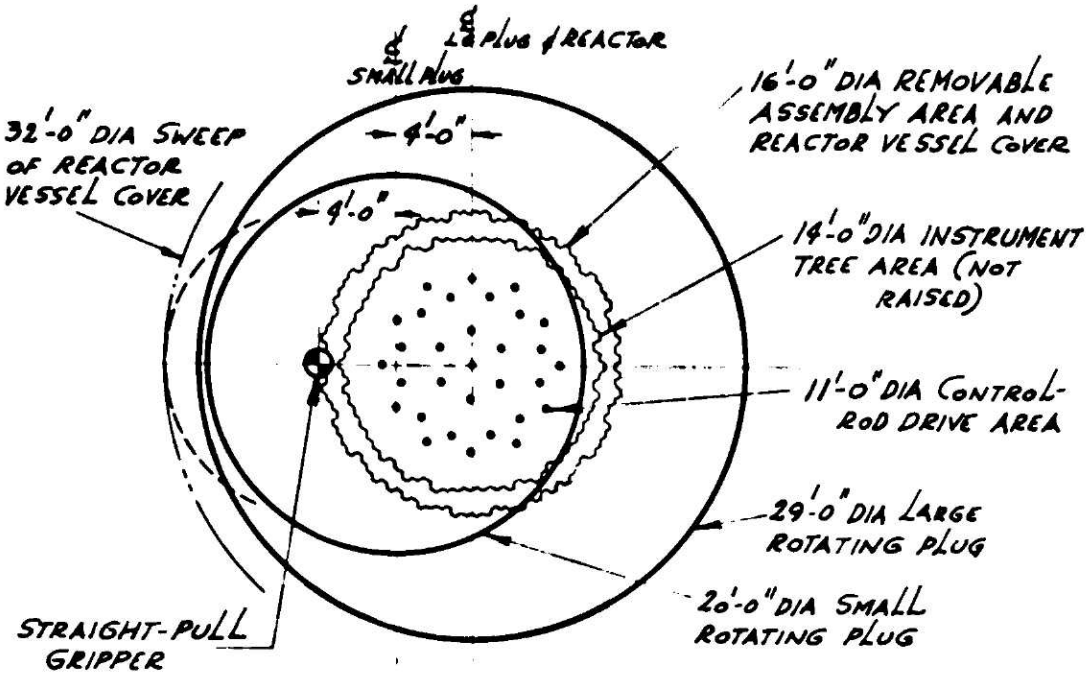


Fig. V.24. 1200-MWe Fuel Handling Scheme—Two Rotating Plugs and One Straight-pull Gripper (EBR-II type)

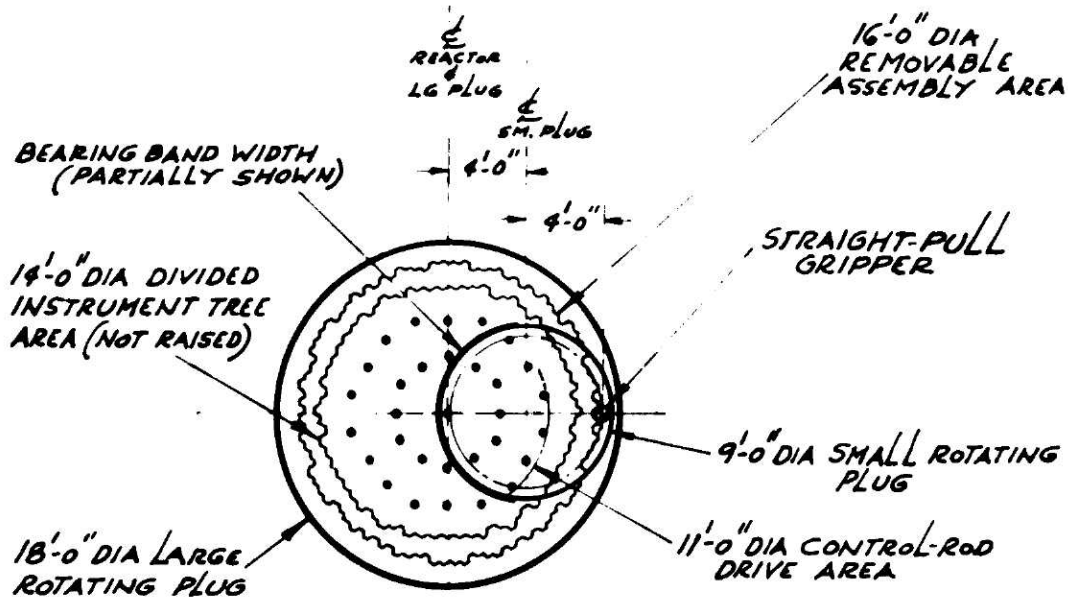
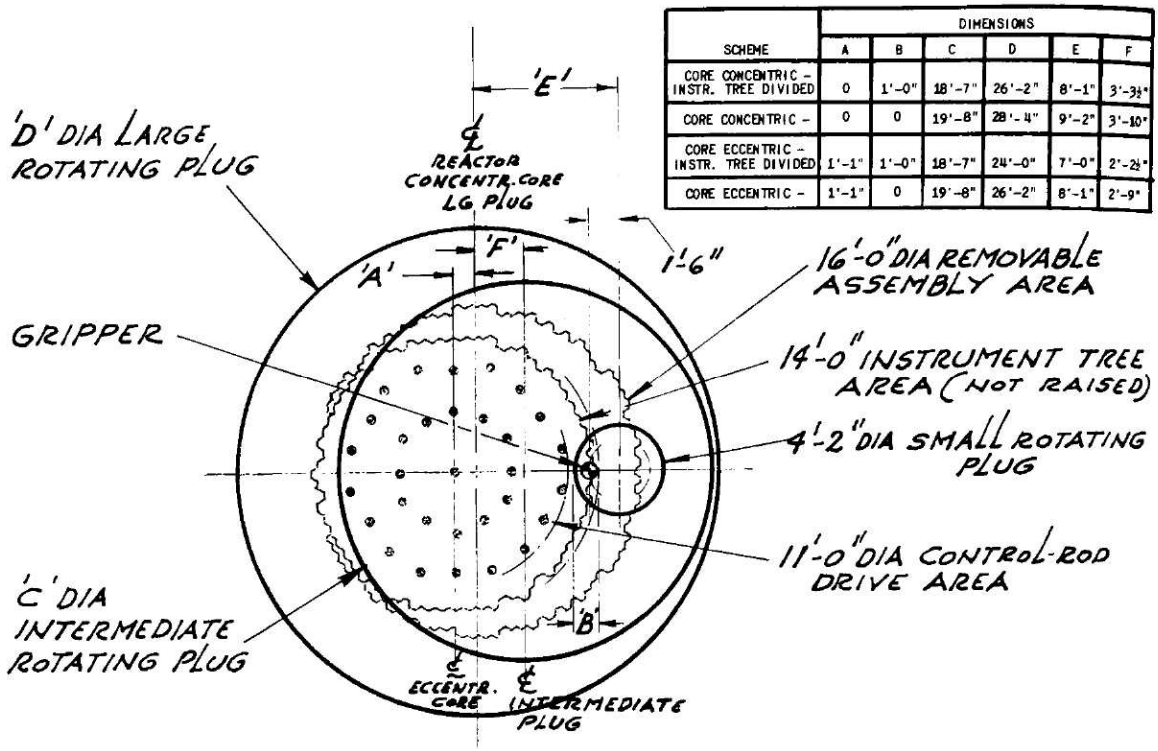
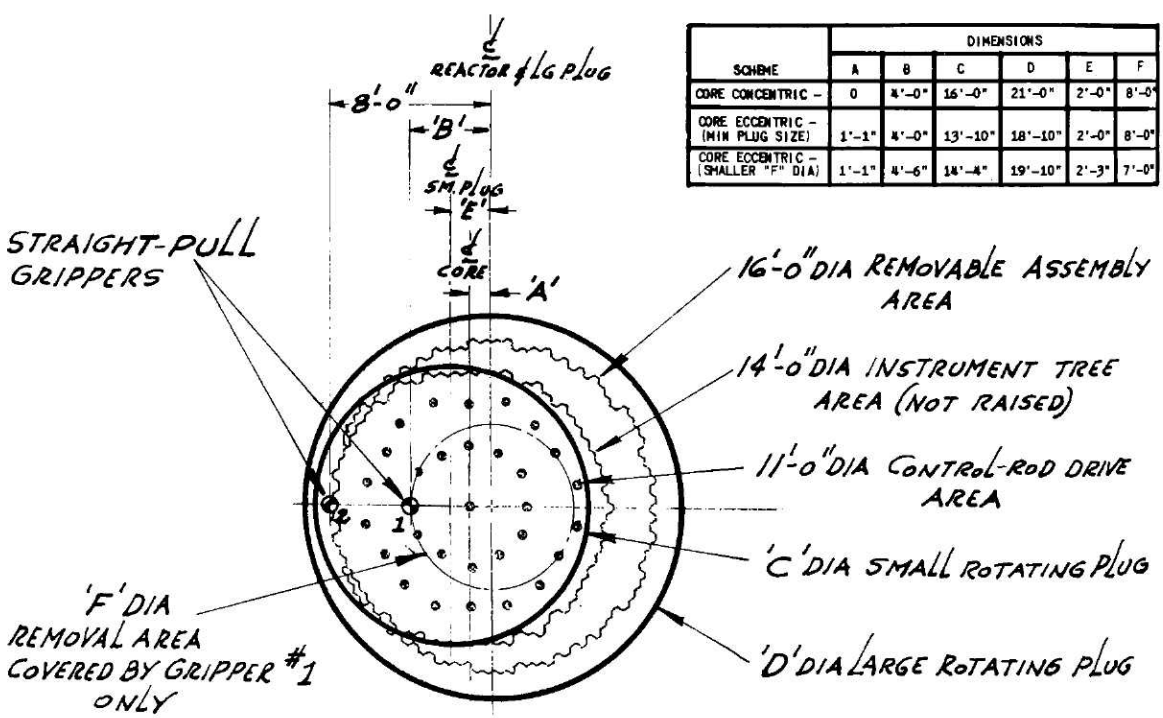


Fig. V.25. 1200-MWe Fuel Handling Scheme—Two Rotating Plugs and One Straight-pull Gripper: Instrument Tree Divided



SCHEME	DIMENSIONS					
	A	B	C	D	E	F
CORE CONCENTRIC – INSTR. TREE DIVIDED	0	1'-0"	18'-7"	26'-2"	8'-1"	3'-3½"
CORE CONCENTRIC –	0	0	19'-8"	28'-4"	9'-2"	3'-10"
CORE ECCENTRIC – INSTR. TREE DIVIDED	1'-1"	1'-0"	18'-7"	24'-0"	7'-0"	2'-2½"
CORE ECCENTRIC –	1'-1"	0	19'-8"	26'-2"	8'-1"	2'-9"

Fig. V.26. 1200-MWe Fuel Handling Scheme--Three Rotating Plugs and One Straight-pull Gripper



SCHEME	DIMENSIONS					
	A	B	C	D	E	F
CORE CONCENTRIC –	0	4'-0"	16'-0"	21'-0"	2'-0"	8'-0"
CORE ECCENTRIC – (MIN PLUG SIZE)	1'-1"	4'-0"	13'-10"	18'-10"	2'-0"	8'-0"
CORE ECCENTRIC – (SMALLER "F" DIA)	1'-1"	4'-6"	14'-4"	19'-10"	2'-3"	7'-0"

Fig. V.27. 1200-MWe Fuel Handling Scheme--Two Rotating Plugs and Two Straight-pull Grippers

3. There are several ways of reducing the large plug diameter by varying degrees. To do this, some of the 1200-MWe reference design requirements have to be relaxed. This is described in Sec. V.D.

4. Due to the lack of sufficient operating experience with pantographs and the relative lack of experience with fixed-offset grippers, the straight-pull gripper seems at present to deserve most serious consideration because of its relative simplicity and excellent operating experience in EBR-II.

5. High fuel burnup limits make heat dissipation of spent fuel assemblies important. This makes prolonged storage in the main tank desirable (as long as it is independent of reactor operating cycles), because it simplifies cooling provisions in subsequent components.

6. Fuel handling has a direct impact on the length of reactor shutdown; therefore its speed (rate) and reliability are of the utmost importance. Thus, when the reactor is shut down, the chain of fuel handling operations that must be performed in series should be as short as possible, in time as well as in the number of individual operations. This further favors the in-main-tank storage scheme, as long as the storage can be depleted while the reactor is operating and as long as the storage space does not affect the diameter of the main tank.

7. Fuel handling components which add to the length of the handling operations chain during reactor shutdown diminish the reliability of the fuel handling scheme, even if they do not affect the overall speed of fuel handling.

8. In order to minimize unexpected fuel handling delays, redundancy of components is recommended whenever possible. For example, there should be two gripper mechanisms, two transfer arms, two storage baskets, two transfer ports, etc.

9. Redundancy of some components also increases the fuel handling speed. For example, two straight-pull grippers in conjunction with two transfer systems save time because one gripper mechanism can divest itself of a spent fuel assembly while the other gripper mechanism simultaneously receives a new fuel assembly, etc.

10. Long fuel handling intervals (10-15 months) will necessitate the loading and unloading of many fuel assemblies during each reactor shutdown. Two storage baskets would therefore not only provide redundancy but would also accommodate a large number (~150) of fuel assemblies scheduled for transfer.

11. The size of the large rotating shield plug has a significant effect on plant costs. Fabrication and shipping problems also increase very significantly as the plug size grows.

12. Main-tank sodium temperatures affect many components. The surface of the main-tank sodium may be "hot" (~1000°F) if the reactor outlet plenum is open to the soffit of the main tank cover, or it may be "cool" (~700°F) if the reactor outlet plenum is topped by a cover or by thermal baffles.

Hot-sodium-surface schemes eliminate the need for a reactor vessel cover, its elevating mechanism, several reactor vessel cover-locking devices, etc. On the other hand, the 1000°F sodium surface temperature poses more problems than a 700°F surface temperature. The most serious is the significant increase in formation of sodium vapor and aerosol formation. This effect causes much faster plugging of clearance gaps on the circumference of the rotating plugs and of the clearances around the shafts of fuel-handling mechanisms. Various cleaning methods will have to be developed to prevent the "crudding" from becoming a problem. The hotter sodium surface also requires more insulation on the bottom of the rotating plugs (and on the main-tank cover), making the plugs deeper and heavier. This in turn lengthens other mechanisms which are penetrating the rotating plugs, such as the fuel gripper, control drives, etc. Thus, the "hot" and the "cool" sodium pools have certain advantages and disadvantages whose evaluations are important.

D. Optimizations and Trade-offs

This section is intended to illustrate some of the benefits that can be gained in one area by penalizing engineering or physics aspects in other areas. Since this chapter deals with fuel handling, it will be shown how to enhance it; but, alas, something else has to suffer.

There are two areas where improvement would be especially beneficial. The first is a decrease in the size of the rotating shield plug(s), and the second is a decrease in the sodium surface temperature in the main tank.

1. Effect of Decreasing Size of Rotating Shield Plug(s)

The decrease in plug size beneficially affects many areas of the reactor project. Fabricability, constructibility, and accuracy of the rotating plugs can be significantly enhanced. The main-tank cover and support structure can be simplified. The main-tank diameter might be reduced, bringing about a large cost reduction.

Plug size can be reduced by any one or a combination of the following changes from the reference design of a 1200-MWe pool-type reactor assembly.

a. Displacement of the vertical centerlines of the large rotating plug and the reactor core (i.e., use of an off-center rotating plug); nonconcentricity of the reactor core within the reactor area (i.e., the core is moved to one side); and nonuniformity of the control-drive pattern in regard to the reactor core (i.e., control drives are not centered within the core).

Depending on the type of fuel handling system, the above-listed "nonconcentricities" reduce the rotating plug size in some measure. The off-center rotating plug can be provided by either locating the reactor

eccentrically in the main tank, locating the rotating plug eccentrically in the main-tank cover, or by a combination of both. The problems caused are an eccentric reactor or rotating plug support and its ramifications, nonuniform reactor piping, and other nonuniformities. The eccentric core and control-drive patterns may create physics problems, make uniform coolant distribution more difficult, and affect many other design features. Naturally the magnitude of the problems depends on the size of the off-sets.

b. Reduction of the instrument tree area, or scalloping of one side of it. This provides room for the swing of an off-set gripper and reduces the rotating plug size. However, the instrument tree will not quite cover the area size as stipulated in the reference design (some fuel-assembly exit temperatures may not be read directly).

c. "Splitting" the instrument tree and the control-drive pattern between two rotating plugs. This has the most pronounced effect on the rotating plug size (if the straight-pull gripper and two-plug scheme is used). Its major problem is that it divides the instrument tree into two separate structures of less stability and creates a nominal gap between them. It also creates a wider semicircular band along the periphery of the smaller plug, so that the regular control-drive pattern has to be disrupted by moving the control-rod locations to either side of the band (see Fig. V.28).

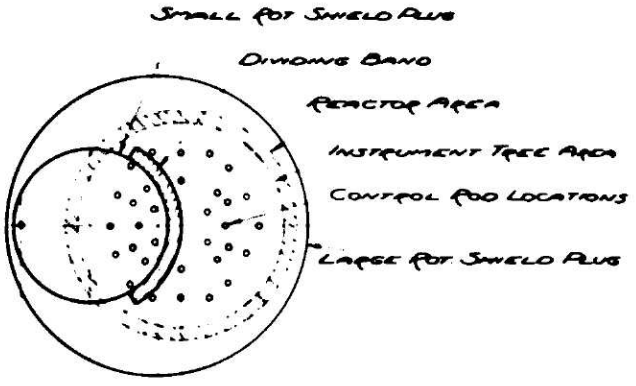


Fig. V.28

Scheme Showing Split Instrument Tree and Control-drive Areas

d. Various, largely safety-related, core configurations are being considered from the physics point of view. In some of them the most centrally located fuel assemblies would be replaced by unfueled assemblies that would have much longer reactor residence times than the fueled assemblies and thus would have to be replaced less frequently. Therefore, a fuel handling system that is more cumbersome at these center locations might suffice. In the two- or three-plug schemes, a substantial decrease in the diameter of the large rotating plug could be achieved if access to the center region would be acceptable in a two-step operation. Because of the relatively small number of transfers that would be needed in that region, the time delay would not be significant when measured against the desirability of smaller plug diameters (see Fig. V.27).

2. Effect of Decreasing Main-tank Sodium Surface Temperature

Because of the long-term effects of sodium aerosol formation (frosting), and because of the Δt across the main-tank cover and across the rotating shield plugs, it is desirable to keep the main-tank sodium surface at reactor inlet temperatures rather than at reactor outlet temperatures. This has been achieved in EBR-II by providing a reactor vessel cover 18 in. above the tops of the fuel assemblies and piping from the hot plenum directly into the IHX. Other pool-type LMFBRs eliminate the cover and much of the piping, but have to accept a sodium surface temperature close to the reactor outlet temperature. Figure V.29 illustrates a compromise between these two versions. It provides a hot plenum of about 15- to 18-ft depth so that the thermal baffle (internal tank) and the IHX couplings are similar to the no-cover designs. However, the top of the hot plenum is formed by a thermal baffle still several feet below the main-tank sodium surface. A bypass stream of cool sodium is pumped directly into the area above the thermal baffles, to keep the sodium surface near the inlet temperature. The thermal baffles are supported from the rotating shield plugs and are compatible with the "split" design described in Sec. V.D.1.c. The associated problems are also similar.

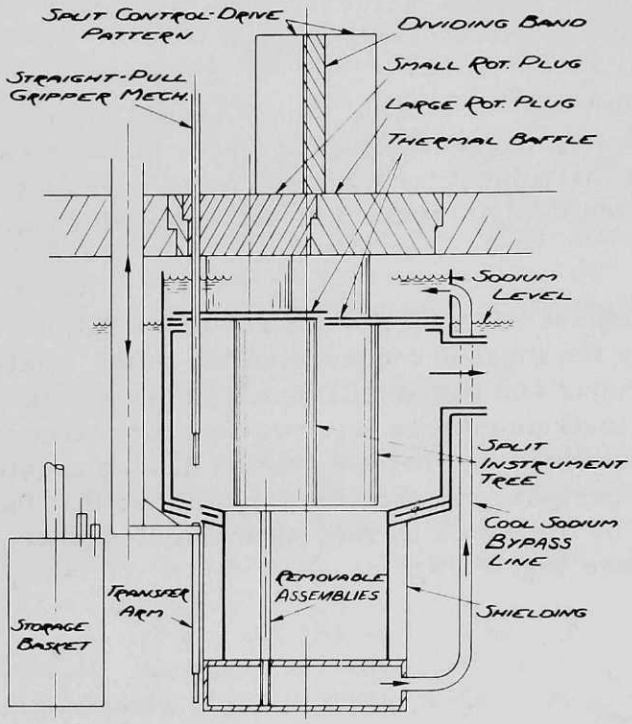


Fig. V.29. Scheme Combining Cool-top, Open-plenum, and Split-instrument-tree Concepts

why is aerosol formation not a problem in loop designs

E. Conclusions

Many dissimilar fuel handling schemes have been studied. Depending on the various specific reactor designs, certain schemes have represented the best solution at the time of their inception. Some have had an excellent performance record but would have to be substantially modified to fit a 1200-MWe LMFBR scheme (EBR-II, Phenix-250 MWe, etc.). Other fuel handling schemes are more compatible sizewise, but it may be many years before a meaningful performance evaluation will be possible (CRBR, PFR-250, etc.). However, the present state-of-the-art makes a prudent scheme selection possible if the fuel handling requirements are integrated early enough into the overall LMFBR design.

It appears that schemes that include straight-pull grippers and two rotating shield plugs incorporate more desirable features and fewer undesirable ones than alternate schemes. Planned large pool-type LMFBRs (Super-Phenix, CFR, BN-600) are using such schemes. The primary reasons are: the greater simplicity and therefore greater reliability of a straight-pull gripper when compared to a pantograph or fixed-offset gripper; the possible need to exert large pulling forces because of fuel assembly swelling; and the relatively small amount of experience with the more complex gripper mechanism.

Reduction of the large diameters of the rotating plugs becomes of major importance, resulting in cost savings, easier fabricability, greater accuracy, fewer transportation problems, and significant impact on other reactor systems. Several approaches have been presented in Sec. V.D.1 and illustrated in Figs. V.28 and V.29. Included are nonconcentric reactor and plug arrangements, reduced or scalloped instrument trees, instrument trees and control-drive areas "split" between two rotating plugs, two-stage fuel handling for a part of the reactor area, etc.

The impact on the overall fuel handling system of a "hot" pool (reactor outlet temperatures of ~1000°F) versus a "cool" pool (reactor inlet temperatures of ~700°F) has been considered. "Hot" pool schemes eliminate several components such as the reactor vessel cover or thermal baffles, the cover lifting mechanism, several cover locks, etc. On the other hand the "hot" pool significantly increases the sodium vapor and aerosol formation and deposition rates. This effect causes faster plugging of the ammuli around the rotating plugs and of the clearances around other component shafts. Also, to accommodate the higher sodium surface temperature, more insulation is needed on the bottom of the rotating plugs, making them thicker and heavier. This in turn lengthens the components which must penetrate the plugs, such as the gripper mechanisms, the control-drive shafts, etc., making them less accurate. Thus the "hot" and "cool" sodium pools each have many advantages and disadvantages whose evaluation in respect to each other is very important.

Pool-type LMFBRs can accommodate storage space for fuel assemblies within the primary sodium pool, yet be located outside of the reactor assembly. This provides the possibility of "under-sodium" fuel handling shortly after reactor shutdown without having to worry about the loss of decay-heat dissipation systems. In-tank storage facilities may also allow the simplest, and therefore most reliable, fuel handling operations while the reactor is shut down. Only transfers of fuel assemblies between the reactor and the in-tank storage facility have to be performed during the critical time period when the plant is shut down. Transfers between the in-tank storage facility and destinations outside the tank can be performed while the reactor is operating and can be scheduled as the demand or supply dictate.

It is hoped that these conclusions emphasize the necessity to consider fuel handling as an important part of the reactor scheme--one that has to be considered in the earliest reactor concepts and designs.

F. R&D Suggestions

Research and development in several areas would greatly aid in engineering efforts pertaining to fuel handling. Some of these areas are listed below. It should be noted that the suggested R&D would provide equal benefits for pool- and loop-type LMFBRs.

1. Establishment of quantitative guide lines for the width of annuli around rotating and stationary plugs that are exposed to sodium vapors and aerosols for long periods of time. Some of the parameters are listed in Table V.2. This work could be a followup on the "Sodium Fuel Technology" report TC-25 Rev. 1, dated July 1974, by HEDL.

TABLE V.2. Parameters for Determining Annuli Widths around Plugs

Diameter of rotating plugs (Realistic fabricating tolerances are also to be considered)	15-35 ft
Diameter of stationary plugs	2-10 ft
Height of annulus	5-8 ft
Temperature of sodium	700-1000°F
Quality (purity) of sodium	LMFBR grade
Cover gas composition	Argon (perhaps helium)
Length of exposures	1, 10, and 30 years
Effectiveness of various seals (e.g., dip ring, gaskets, etc.)	
Effectiveness of vapor barriers (e.g., flexible metal ring, metal-to-metal contact, etc.,) near the annulus entrance	

2. Development of simple tools for cleaning the annuli of large rotating plugs. These devices could view, scrape, heat, etc., and would be inserted in specially provided locations on or around the plugs. Such tools should be used before plug rotation is impeded. Indeed, plug rotation may be required to effectively use such tools. Timely development of such devices would influence the design of the rotating plugs, so as to assure the most effective locations and access provisions.

3. Development of improved "dip ring" seals for rotating plugs. Long-term effects of sodium aerosols and of cover gases should be studied to assist in the selection of the proper sealing liquid. Reliable methods for purification and replacement of the sealing liquid should be developed.

4. Development of improved seals of shafts which not only move vertically but also rotate (gripper mechanisms), to minimize cover gas leakage which might contain fission products.

5. Development of gripper mechanism features that minimize the effects of thermal expansion on mechanical actuators of jaws and sensors. As the long and concentric shafts move into and out of the sodium, they experience various nonisothermal cycles that may affect their operating responses. "Soaking time" corrects these effects but prolongs fuel handling times.

6. Development of improved reactor grid-plate bearing surfaces and of compatible materials for the fuel assembly lower adapters would decrease fuel handling friction forces and might decrease fabrication costs.

7. Studies of solubility of helium in sodium at various temperatures and surface agitation levels would provide more insight into the option of using helium as the cover gas. The use of helium may be under consideration if spent fuel assemblies are extracted from the sodium shortly after reactor shutdown while they still have relatively high decay heat emissions of 20 to 60 kW. High solubility of helium in sodium might provide a reactor cooling hazard.

8. Development of improved cleaning methods for full-sized fuel assemblies that contain fuel elements with breached cladding.

9. Continuation of development of "universal tools" that could "grasp" and "feel" as aides if fuel assembly retrieval would be required below the sodium surface.

10. Testing of various models of fuel handling mechanisms to provide simple, reliable, and economical components.

11. Determination of minimum requirements in areas that have significant impact on the size or complexity of fuel handling components. Such areas are: degree of instrumentation (instrument tree) needed above the fuel assemblies, requirements of auxiliary mechanical holddown of fuel assemblies, coolant needs for decay heat of spent fuel assemblies, etc.

12. Determination of size limits for practical fabrication and construction of rotating shield plugs and ball-bearing races, as well as economics and realistic tolerance requirements for these components.

13. Development of a practical fuel handling system using rotating shield plugs in the 20-ft-dia range.

14. Study of the effects of elevated temperatures (hot pool) on designs of rotating shield plugs.

G. Illustrations of Actual Arrangements

Figures V.30-V.36 depict the actual physical arrangement of fuel handling components and other plant components in EBR-II, Phenix, Super-Phenix, PFR, CFR, and BN-600.

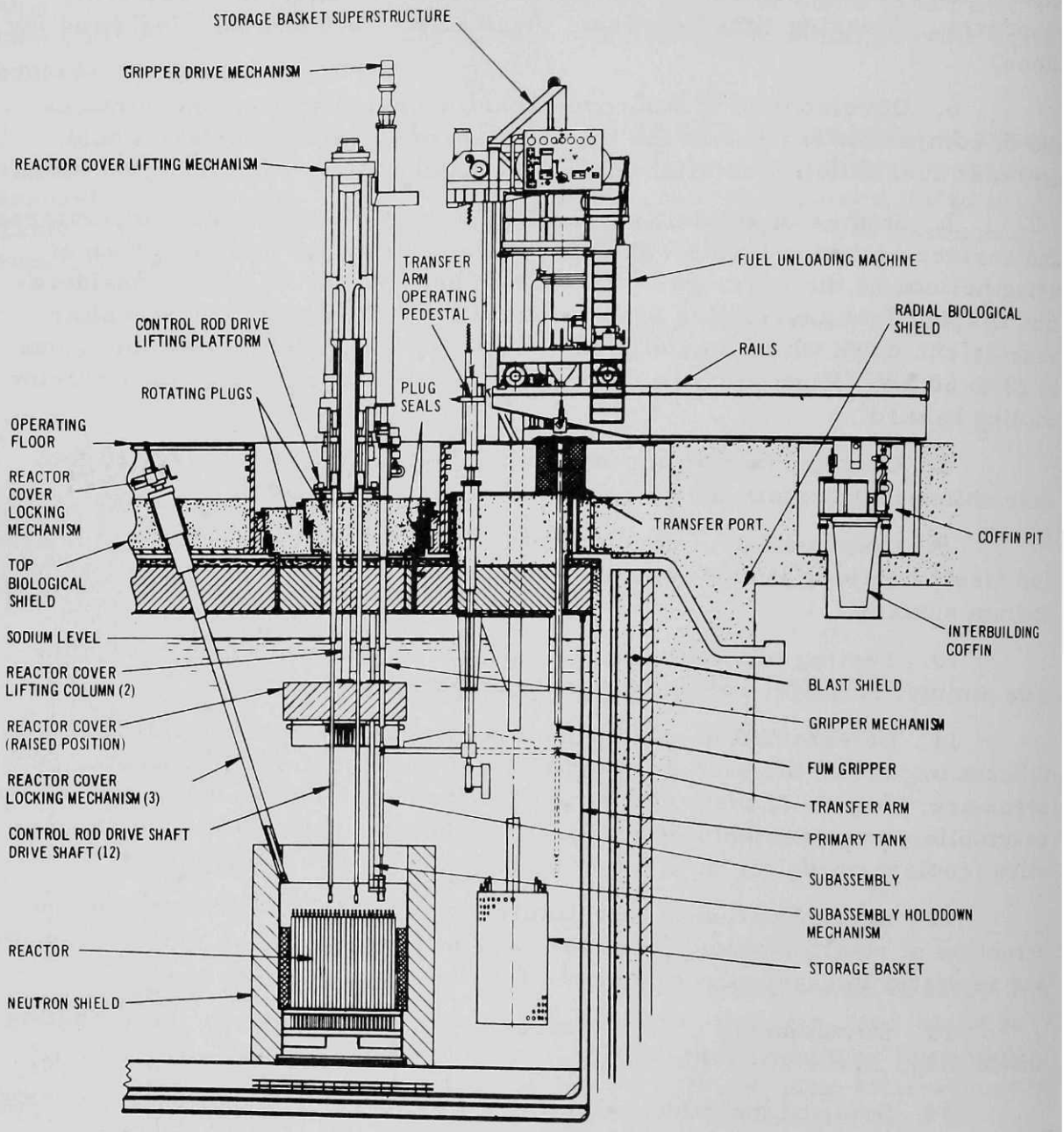


Fig. V.30. Composite of EBR-II Fuel Handling System

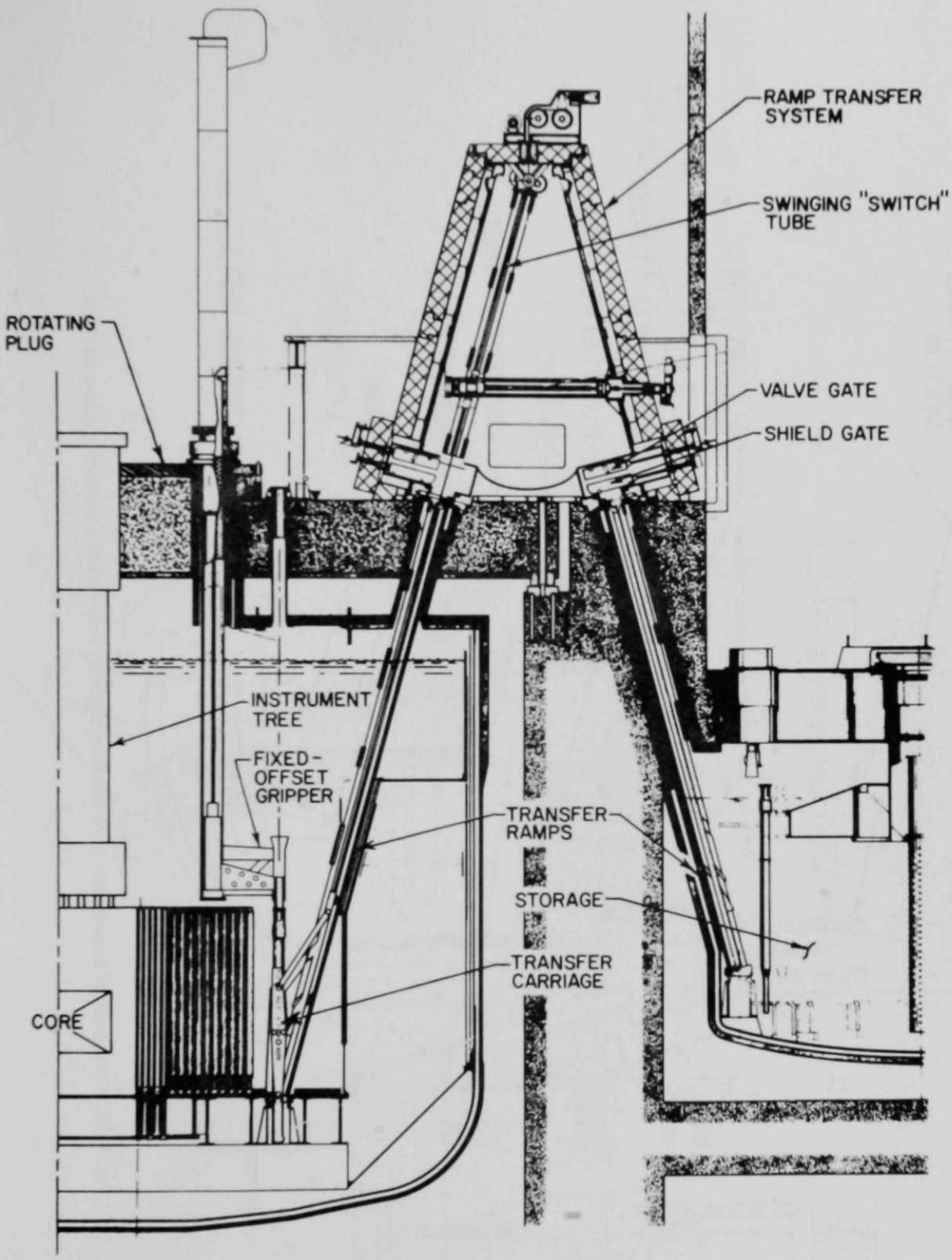


Fig. V.31. Composite of Phenix Fuel Handling System[6] (courtesy of American Nuclear Society)

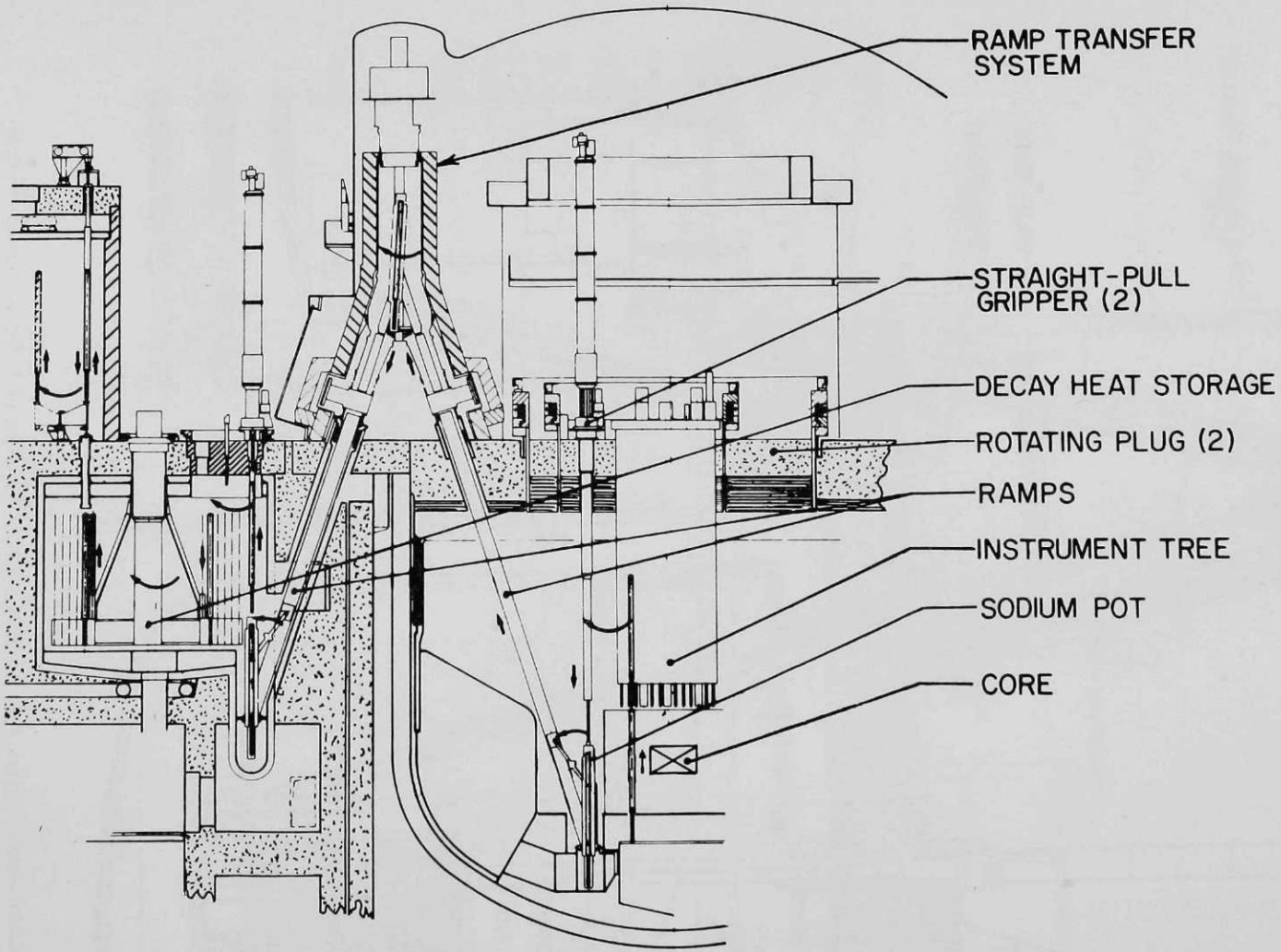


Fig. V.32. Composite of SuperPhenix Fuel Handling System[7] (courtesy of American Nuclear Society)

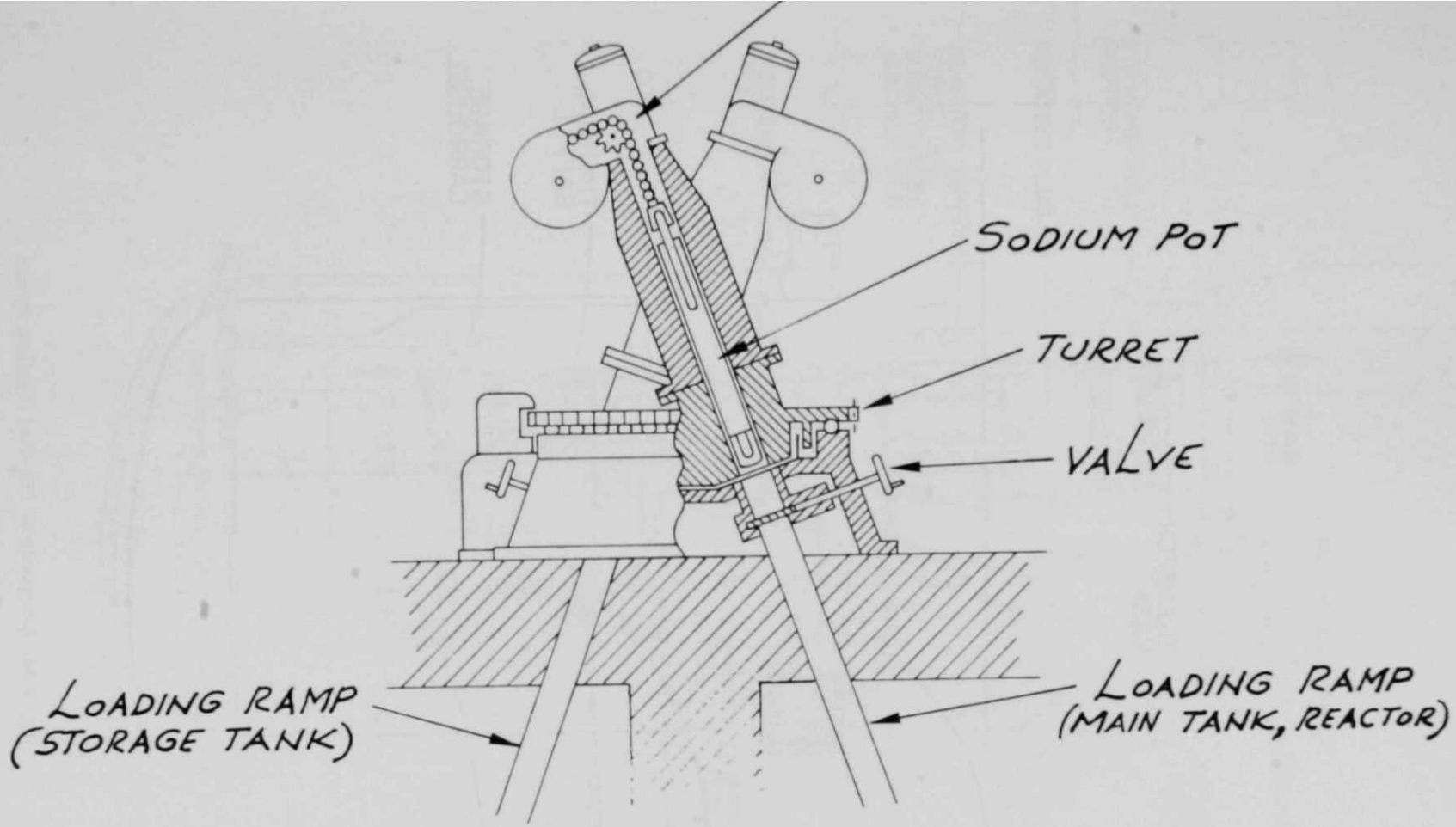


Fig. V.33. Alternate Concept of SuperPhenix Ramp Transfer System (rotating)

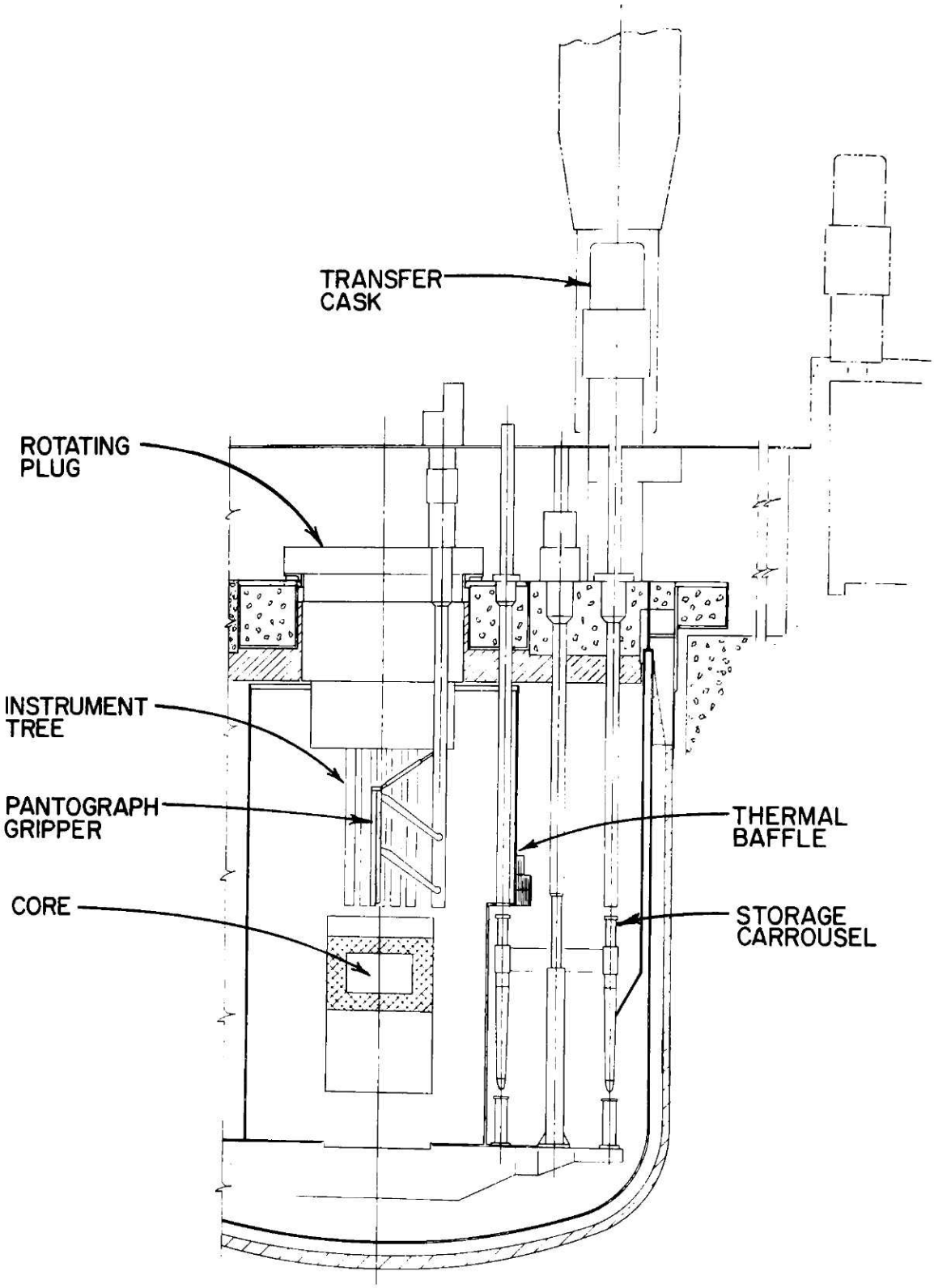


Fig. V.34. Composite of PFR Fuel Handling System

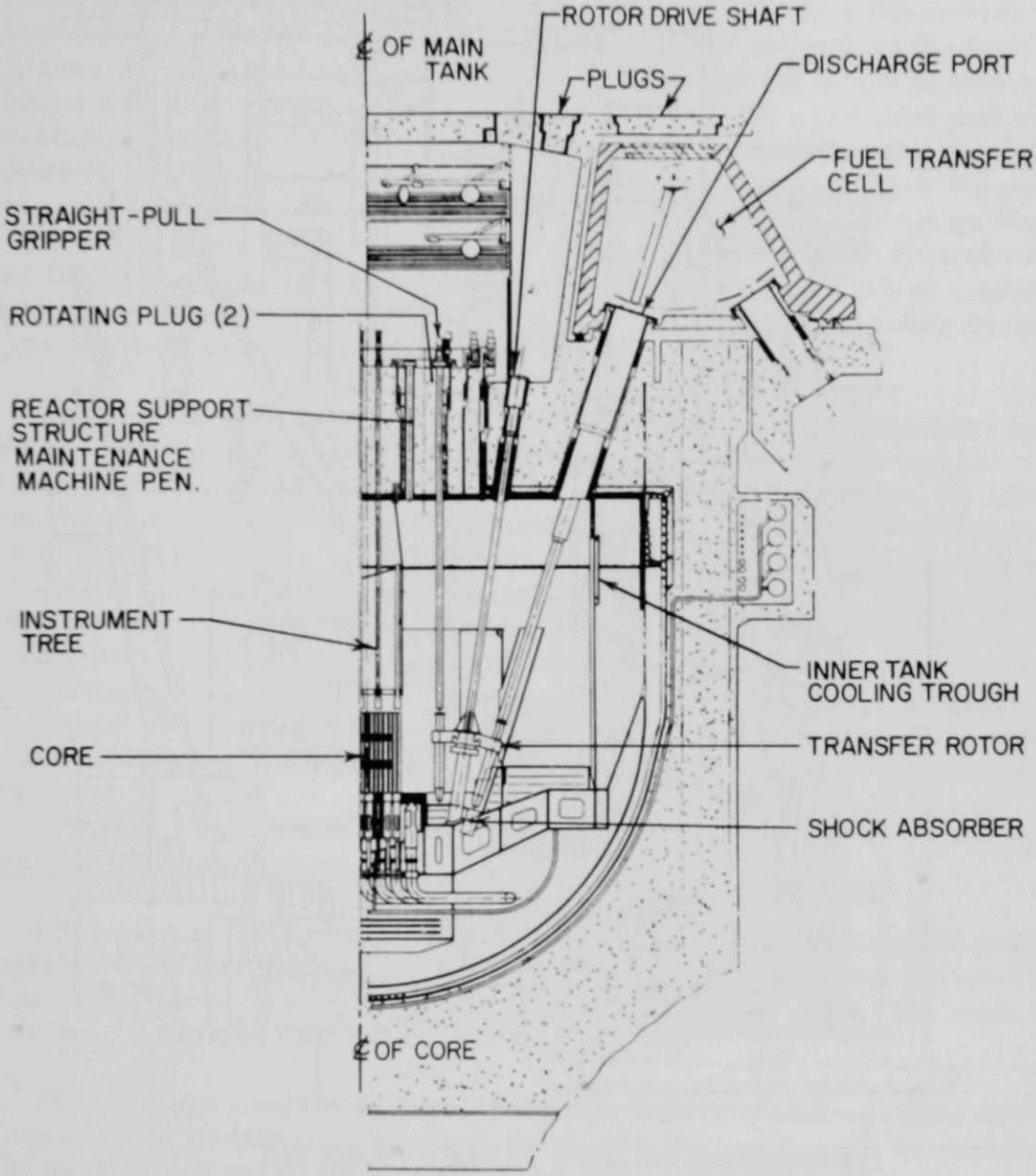
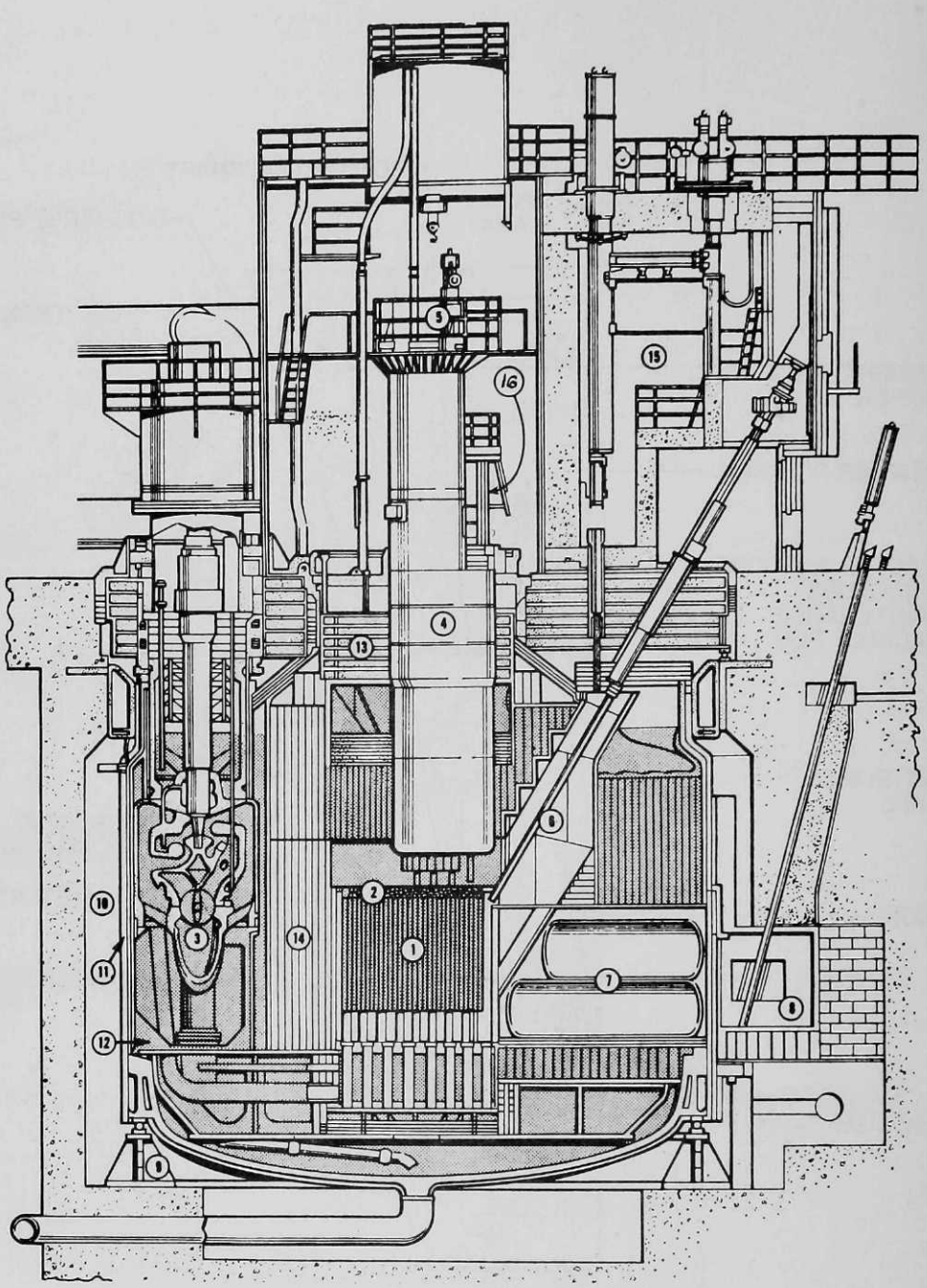


Fig. V.35. Composite of CFR Fuel Handling System[8]
(courtesy of Nuclear Power Company Limited)



COMPONENTS RELATED TO FUEL HANDLING:

1-REACTOR CORE
4-INSTRUMENT TREE
6-TRANSFER ELEVATOR
13-ROTATING PLUG (2)
15-TRANSFER CELL
16-STRAIGHT-PULL GRIPPER (2)

Fig. V.36. Composite of BN-600 Fuel Handling System[9] (courtesy of Nuclear Engineering International)

VI. PRIMARY-SYSTEM CONFIGURATIONS AND OPTIONS

This section reviews the various configurations of primary coolant systems of the pool-type LMFBR. Key design choices are identified, and the important consequences of these choices are discussed. The discussion focuses on those aspects of primary coolant systems that are generic to the pool system; issues not generic to the pool system are beyond the scope of this review. Consistent with the reference core described in Sec. III, core inlet and outlet temperatures of 720 and 1000°F, respectively, are used throughout this section as generally applicable reference temperatures for typical 1200-MWe pool systems. Results are presented of comparative calculations to scope the effects of the elevation of the IHX primary-coolant outlet and the time of primary-coolant flow coastdown on the natural-convection flow capability of a system under extreme conditions. Systems currently in use for dissipating decay heat from the primary system are discussed briefly.

The primary coolant system consists of the array of components, equipment, and related appurtenances and structures needed to cool the reactor and to contain and handle the reactor coolant and cover gas. In the pool concept the primary coolant system is contained largely within the main tank. The principal components of the system, which are, in general, common to most LMFBR systems, consist of the primary coolant pumps; the piping from the pump discharge to the reactor inlet plenum; the core and blanket, including inlet and outlet plenums; the coolant conduit from the reactor outlet to the intermediate heat exchanger (IHX); the coolant conduit from the IHX outlet to pump suction; and related components such as valves, instruments, flex joints and seals, insulation, and, for some designs, the outlet-plenum cover and latches.

A. General Characteristics

Pool-type primary coolant systems share many characteristics which, in general, relate intrinsically to the submerged aspect of these systems and to the potential for a simple coolant containment boundary, i.e., the main tank. Some of the more important characteristics follow.

The primary coolant system is located mainly within one shielded area. Thus, there is no multiplicity of steel-lined, shielded cells with related services of heating and ventilating, electricity, leak and smoke detection, special-entry considerations, accident-leakage confinement barriers, etc.

The primary-coolant boundary is the main tank, a structure that can be of simple geometry if suspended from the top. The simple geometry of the tank contributes to its structural reliability, enhances safety of the facility, aids fabricability, and improves capability for periodic nondestructive examination. The small L/D of one or less enhances capability to withstand seismic events. Bottom-supported tanks give rise to design, fabrication, and periodic-examination difficulties in the area of the bottom support.

Biological shielding is limited primarily to around the main tank. Shielding within the shield deck over the main tank can be complex owing to the intricate nature of the deck and the rotating-plug structures.

The submerged nature of the primary coolant system permits small leakages; i.e., zero leakage from system components within the primary tank is not a requirement. The use of nonleaktight connections between removable components and permanently installed equipment is permitted and eases accommodation of differential thermal expansions. Pipe-heating and leak-detection systems are not required.

The configuration of the primary coolant system is simple. Figure VI.1 is a schematic of an EBR-II-type pool system illustrating certain aspects of the primary coolant system common to most concepts. Not shown are the IHX, the reactor outlet plenum, or the coolant conduit between the IHX and the reactor outlet plenum, because the configuration of the part of the system containing those items is one of the principal differences in current designs and is discussed in detail in later paragraphs. Most main tanks are top-supported, with components supported similarly. The major components, such as pumps and IHXs, are supported from the shield deck, the main-tank cover, or the reactor support structure. These components join to the permanently installed equipment in the tank via non-leaktight connections. The pump is usually submerged in sodium at reactor inlet temperature. The piping between the pump and the reactor inlet plenum

essentially ΔP: NPSH + P_{abs} + SUBMER

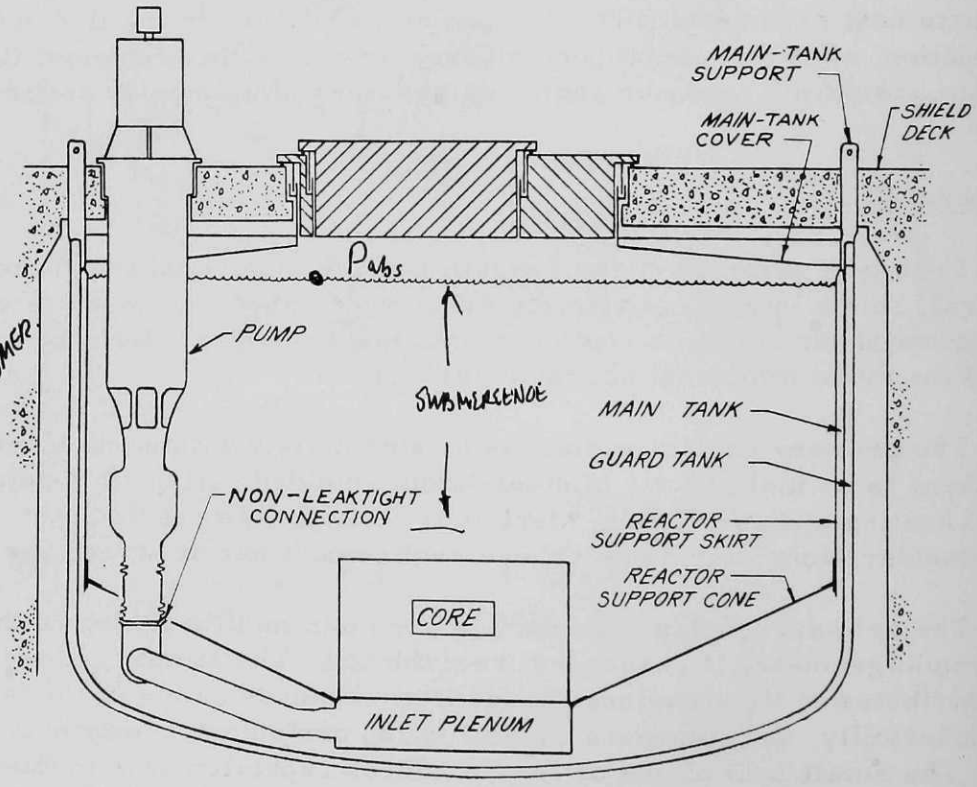


Fig. VI.1. EBR-II-type LMFBR Pool System--General Schematic (without reactor outlet plenum, outlet coolant piping, and IHX)

operates isothermally with no need for insulation. The supports for the inlet lines must accommodate high-thrust loads since these lines are not hard-coupled to the pumps. Differential thermal expansion between pumps and inlet lines depends upon the type of pump support chosen. The reactor is located on a conical structure that is either supported from a separate reactor support skirt (as shown in Fig. VI.1) or connected directly to the wall or bottom of the main tank. Check valves and process instrumentation, such as inlet coolant flowmeters, thermocouples, and pressure gauges, are included most usually in the removable pump unit or in the instrument tree (not shown in Fig. VI.1) over the core.

The simple coolant-system configuration obviates the need for a number of systems found on other concepts. For example, the coolant system is completely within the main tank and cannot be approached readily for in-place, hands-on maintenance. Consequently, there is no strong need for a primary-coolant drain system with its complexity of piping, tanks, pumps, heaters, controls, etc., and associated shielded space. A single cover-gas space over the primary-coolant liquid level(s) eliminates the need for cover-gas systems to vent high points or to provide cover gas for liquid levels such as can exist at IHXs, pumps, expansion tanks, dump tanks, and high points of piping. There is no need for overflow systems, including piping, pumps, IHXs, tanks, instrumentation, etc., to maintain and/or control liquid levels. The potential exists for locating the radioactive portion of the primary-sodium purification system within the main tank, reducing further the need for shielded areas outside the shielded cavity of the main tank.

Pool-type LMFBRs have a large thermal inertia and an operating state of low stored thermodynamic energy, i.e., there is a large margin available between operating temperature(s) and sodium boiling point. Moreover, operating pressure is low (atmospheric). These characteristics provide important safety advantages. Pool systems of the 1200-MWe size will have primary systems that contain 3000-5000 tons of sodium, giving an adiabatic rise characteristic of the order of 1°F per full-power second.

The primary coolant pumps operate at reactor inlet temperature and have good suction head conditions, usually in the range of 45-55 ft of sodium; low gas-seal pressures of a few inches of water; shaft lengths of 30 ft or less; and no significant thermal transients.

Differential thermal movements between components must be acommodated within the limited space of the main tank. Frequently, design innovations are required that are unique to this application. The piping (if used) between the reactor outlet plenum and each IHX is an example.

Insulation within the primary system is an important design feature. Design requirements focus primarily on the following: insulating those parts of the system that separate the 1000°F (reactor outlet) sodium from the 720°F (reactor inlet) sodium; insulating the main/guard tank complex; establishing

acceptable temperature gradients in the upper wall of the main tank; and thermally protecting the shield deck of the system. As a rule, insulation within the main tank or main-tank vault must operate reliably, without maintenance, for the life of the plant.

Pool systems may have the potential for limiting Class 1 requirements to the primary system and its immediate subsystems. This requires, as a minimum, the capability of reliably cooling the reactor in the event all secondary and/or steam systems become inoperative.

B. Primary-coolant-system Choices of Key Importance

The key choice in the selection of a pool-type primary coolant system is the kind of reactor outlet plenum to be employed. Other considerations of importance are the method of support for the primary pumps and IHXs, attention to design details to exploit the good natural-convection characteristics of these systems, and the method of removing decay heat from the main tank.

Either an open or closed outlet plenum may be selected.

The Phenix, PFR, and BN-600 systems use the open-plenum approach. Figure VI.2 illustrates the Phenix concept. In this case, the coolant exits the

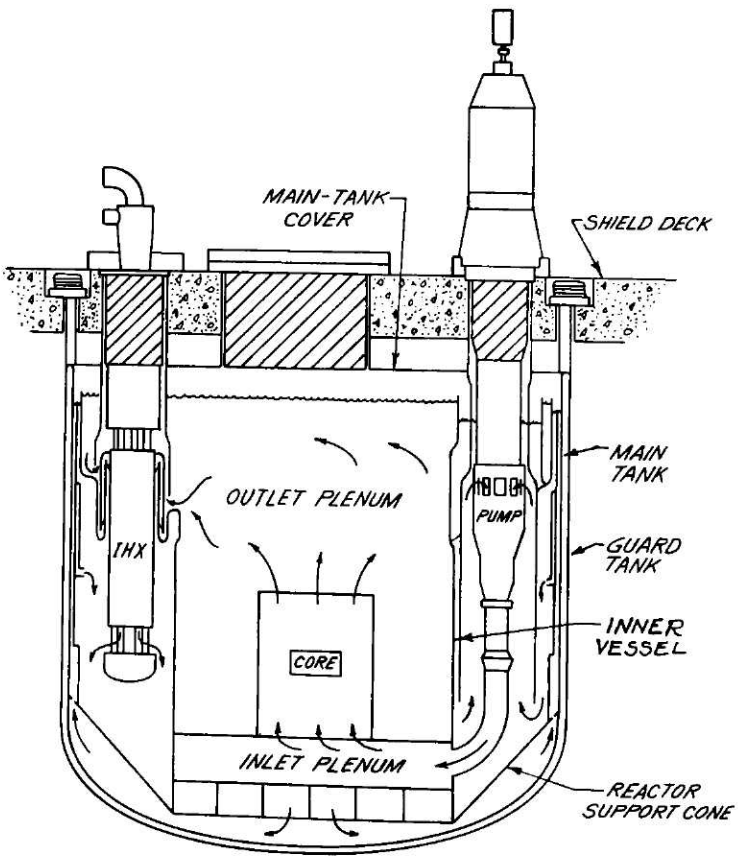


Fig. VI.2. Open-outlet-plenum Pool System (Phenix type)

core and enters the outlet plenum, a region of the main tank that operates at the reactor outlet temperature. The sodium in this region is separated from the cooler, pump-suction portion of the tank by an inner vessel. In general, different liquid levels exist in the hot and cold portions of the tank, and the liquid-level difference manifests the hydraulic loss through the IHX.

The EBR-II approach (Fig. VI.3) uses a closed plenum, wherein the coolant exiting from the reactor is confined to a small pressurized volume consisting of outlet plenum, outlet piping, and IHX. The hot sodium exits the outlet plenum and flows to the IHX via piping. The bulk of the main tank is isothermal at the reactor inlet temperature. Only one liquid level exists.

reactor cover

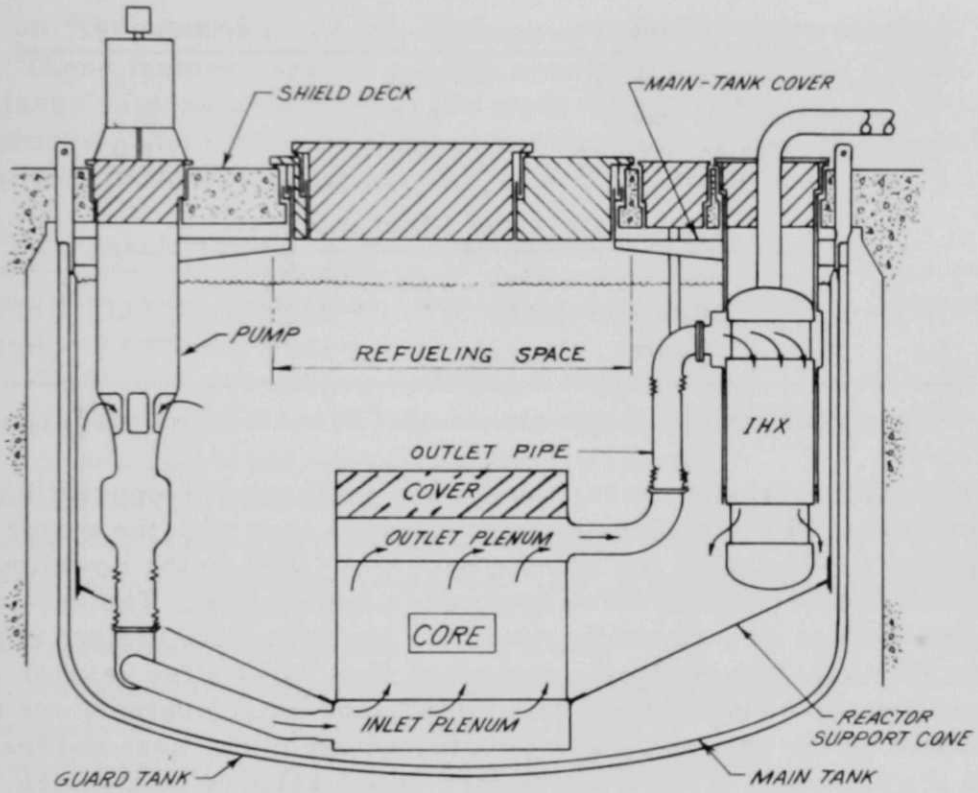


Fig. VI.3. Closed-outlet-plenum (isothermal) Pool System (EBR-II type)

Primary pumps and heat exchangers may be supported in a variety of ways. They may be fixed to the cold shield deck above the main tank (as in PFR); supported on the cold shield deck on members permitting lateral motion parallel to the deck (Phenix); fixed to the hot cover of the main tank (EBR-II); or supported on nozzles that carry the load to a bottom support structure (BN-600).

An important consideration in selecting a component support method is the thermal movement to be accommodated, which is dependent on system temperatures. As shown in Table VI.1, there are three system temperatures of greatest interest, namely the reactor outlet and inlet temperatures and

room temperature. Consider an IHX supported on the room-temperature shield deck above the main tank and interfacing with the 1000°F coolant outlet conduit from the reactor. Over a radial span of 30 ft (typical center-of-IHX to center-of-core distance) a differential expansion of over three inches obtains. The method of accommodation of these thermal differential expansions is an important element of system design.

TABLE VI.1. Typical Values for Differential Thermal Expansion

Expansion of Typical 30-ft Structure		
Temperature Difference	ΔT, °F	Expansion,[a] in.
Reactor outlet to room	900	3.2
Reactor inlet to room	620	2.2
Reactor inlet to outlet	280	1.0

Temperatures of Chief Interest	
Reactor outlet	~1000°F
Reactor inlet	~720°F
Room	~100°F

[a]Expansion (in.) = length (in.) x ΔT (°F) x 10^{-5} (in./in.-°F).

The coolant-system arrangement has good natural-convection cooling characteristics. The system is usually a simple loop with the major heat source (core) in the upflow leg and the heat sink (IHX) in the downflow leg. The hydraulic resistance of these systems is usually low. The vertical elevations and system ΔTs, in conjunction with low hydraulic system resistance, appear to give adequate thermal-convection flow rates after reactor scram, when quasi-steady conditions prevail. Important considerations are the reliance placed on the secondary systems to remove decay heat and the resultant effect on the elevation of the IHX primary outlet relative to the core, the coast-down characteristic of the primary pump to cope with the tendency for a power/flow mismatch during the early phase of an LOF (loss-of-flow) with scram transient, the ability to cope with other events such as overcooling of the core with a delayed loss of all pumps, and a detailed knowledge of the thermal hydraulic conditions in the system under all conceivable conditions.

In most pool systems emergency cooling of the main tank is effectively decoupled from emergency core cooling due to the large volume of sodium in the tank and the resulting large thermal inertia and capacitance of the system. This lack of important interface between these two systems greatly simplifies design. Key considerations here are (1) the degree of dependence placed on the secondary systems (and IHXs) for dissipation of decay heat from the main tank and whether an independent, redundant backup system(s) is needed, and (2) if so, the design approach to be taken.

C. Considerations in Choosing between Open- versus Closed-outlet-plenum Concepts

The choice of an outlet-plenum concept will determine, in large measure, the nature and magnitude of the technical issues that will be encountered throughout the primary system. The key engineering considerations that are generic to the selection are discussed for the open and closed concepts as well as the open concept with separated coolant loops at the reactor inlet (BN-600 type).

1. Open-outlet-plenum Concept

This concept has two distinctive features which strongly influence design. These features are (1) a large open pool of hot (1000°F) sodium and (2) two large regions of sodium in the main tank at substantially different temperatures. The influence of these two features of the open-plenum concept are reflected in the key design issues, which are listed in Table VI.2.

TABLE VI.2. Key Design Considerations for the Open-plenum Concept

No closed outlet plenum, plenum cover, cover-lifting and latch mechanisms, or outlet piping
Instrument tree required
Space for outlet piping not required, potentially leads to smaller tank
High sodium vapor pressure: ~35 times greater than vapor pressure at 720°F
Shield deck and top part of tank exposed to hot (1000°F) sodium pool
Insulation and cooling of shield deck a special concern
Inner vessel separates hot (1000°F) and cold (720°F) portions of pool
Heat losses and resultant free-convection flow currents on cold side of inner vessel creates difficulties; structures and equipment must be protected
Insulation and cooling of main-tank wall a special concern
Flow baffles required at core outlet to obtain a quiet free sodium surface
Gas entrainment at free surface or at IHX inlet to be avoided
Differential level between hot and cold zones provides flow through IHX
Liquid levels change as a function of total coolant flow rate through IHX and flow through core
Low differential liquid levels require low-ΔP IHX: 1-2 psi
Coolant flow pattern and temperature to IHX less predictable
Pump casing subject to nonuniform temperature distributions and differential-expansion pipe loads
Thermal inertia of outlet plenum greater, inertia of inlet zone less, and total inertia less than isothermal pool ✓
Thermal-inertia mismatch between hot pool and main-tank cover may limit operational flexibility

The open reactor-outlet plenum eliminates both the outlet piping to each IHX (a major design problem) and the pressurized outlet plenum. The latter, in the case of an EBR-II design, voids the need for a plenum cover,

cover-lifting structure and operators, plenum-cover latch mechanism, and space within the main tank to rotate the cover during refueling. Since there is no need for space for IHX piping and for plenum-cover movements, primary pumps and IHXs can be located as close to the rotating plugs as the design of the shield deck and other design requirements permit. This should help hold down the diameter of the main tank.

The vertical lift of the fuel assembly is reduced (thus reducing elevation space requirements) because the fuel assembly need not be lifted over the upper edge of a 5-ft-high outlet plenum as in the closed-plenum approach. However, the gain is partially offset, since the shielding provided by the plenum cover is likely to be located in the upper portion of the fuel assembly, making for a longer assembly.

With either the open- or the closed-plenum approach, the need exists for instrumentation at the reactor outlet. With an open plenum, a separate instrument tree is required. This component will be large and unwieldy, and will have severe and demanding design requirements.

As noted previously, the principal design difficulties with the open-plenum approach stem from: exposure of the shield deck or main-tank cover and upper wall of the main tank to high-temperature sodium from the reactor; and effects of two large volumes of coolant existing in the main tank at substantially different temperatures, namely reactor outlet (1000°F) and reactor inlet (720°F) temperature. The exposure of the main-tank cover or shield deck to the core outlet temperature imposes severe requirements on the design of these structures. Clearly, the shield deck must be kept cool, have low thermal gradients, and maintain structural stability under all conditions. High sodium temperatures make these requirements more difficult to meet.

Explain why this not a problem w/ loops?

If a main-tank cover is used that is not insulated on the sodium side, as in Phenix, the cover operates at a temperature greater than the lower wall of the tank, which is in contact with sodium at reactor inlet temperature. Temperature differences as large as the mean ΔT between the hot and cold sodium pools can exist. The magnitudes of these temperature differences will be influenced by the heat flux across the various parts of the tank. The differential expansion between the cover and the walls of the tank must be accommodated by deformation of the structure, mainly the wall of the tank. In addition, axial thermal gradients in the upper region of the tank wall must be controlled to avoid unacceptable stresses. This requires thermal insulation or other means for controlling the temperature gradient. The insulation if exposed to the sodium and/or sodium vapor and cover gas is not accessible for maintenance.

The use of such a "hot" cover introduces a further design consideration--the thermal inertia of the cover relative to other parts of the

system, such as the wall of the tank. If the tank and cover do not respond uniformly to system transients, undesirable stresses can occur. This situation was found to exist in Phenix. Early "trials" indicate that the cover changes temperature more slowly than the tank wall, which promptly adjusts to the temperature of the sodium at the pump suction. The resulting temperature difference can become excessive under certain startup and shutdown conditions, particularly those where low system temperatures obtain, e.g., 250°C (482°F) for refueling. Figure VI.4 shows the problem encountered at Phenix. A maximum temperature difference of about 110°C (230°F) is indicated from the data given by the figure. The effect of this situation has been to limit operational flexibility.

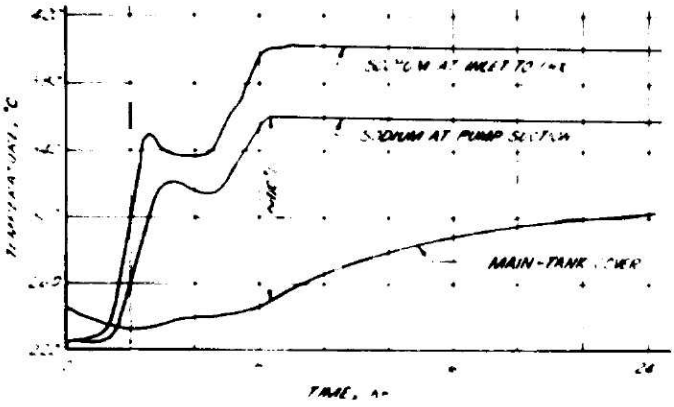


Fig. VI.4. Main-tank Cover and Coolant Temperatures during a Cold Startup in Phenix[10]

If the shield deck serves also as a cover for the main tank, as in PFR and SuperPhenix, insulation of the underside (sodium side) of the shield deck is required and is a paramount design problem. Details of the PFR insulation are uncertain; however, the insulation is believed to consist of layers of thin stainless steel separated by spaces which communicate freely with the cover gas. A problem recognized early by the British is the possible buildup of sodium deposits between the layers of stainless steel. Bridging of sodium deposits can significantly change the thermal resistance of the insulation and can lead to local hot spots in the shield deck, with resultant undesirable thermal stresses. Furthermore, the wall of the main tank connects directly to the cold shield deck. The rim of the tank is fixed at approximately room temperature, and the tank wall (at the sodium liquid level) is at the reactor inlet temperature (~720°F). Heat radiates from the 1000°F sodium surface to the tank wall above the sodium level. The design must accommodate the differential expansion between the cold, room-temperature rim of the tank and the 720°F or higher tank wall as well as assure reasonable temperature gradients axially along the tank wall from the high-temperature region to the cold rim.

No details appear to be available of the insulation for the underside of the SuperPhenix shield deck. It is known that the lack of adequate insulation was one of the reasons that the Phenix design incorporated the hot main-tank cover. Clearly, however, the French believe that suitable insulation technology exists now. integral deck

The effects of two large volumes of sodium in the tank at substantially different temperatures create a number of important design considerations, as listed below.

- Need for an inner vessel to separate the hot (1000°F) and cold (720°F) sodium.
- Structural-support, vibration, and insulation requirements of the inner vessel.
- Penetration of the inner vessel by pumps, IHXs, refueling equipment, etc.
- Sodium flow patterns and free-surface effects in the hot region.
- Free-convection and thermal-gradient considerations in the cold-sodium region.
- Effects of high (1000°F) temperature upon the main-tank wall, main-tank cover or shield deck, and components that penetrate through the shield deck into the sodium system.

With an open outlet plenum the hot (1000°F) coolant is separated from the cold (720°F) coolant by an inner vessel. Various approaches have been taken in the configuration of these vessels, as manifested by the PFR and Phenix designs.

The PFR inner vessel (reactor jacket by PFR nomenclature) is shown in Fig. VI.5. The inner vessel consists of a cylindrical portion that extends from the reactor support structure to the top tubesheet level of the IHX.

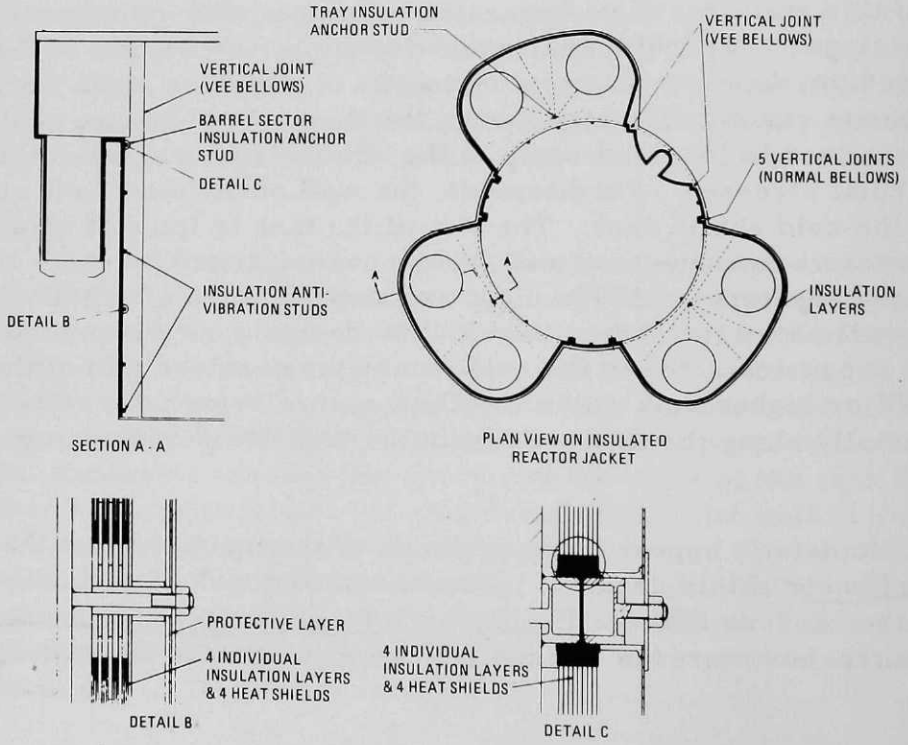


Fig. VI.5. PFR Inner-vessel (reactor jacket) Insulation[11] (courtesy of American Nuclear Society)

At this level, horizontal, cloverleaf-like trays are provided for IHX penetrations. Above the tray level the vertical wall of the vessel follows the contour of the cloverleaf pattern. Provisions for fuel assemblies to pass through the vessel during refueling further complicate the design. The vertical wall of the inner vessel is insulated by four layers of insulation. Each layer consists of two thin sheets of stainless steel which are seam-welded together at 18-in. intervals in a quilt-like pattern. An inert gas is encapsulated between the two sheets of stainless steel during the welding process. Vertical differential expansion joints and antivibration studs are provided.

The Phenix inner vessel consists of upper and lower cylindrical sections with a conical section between (Fig. VI.6). The lower section is somewhat larger than the outermost diameter of the radial shield of the reactor, and the upper section is slightly smaller than the diameter of the main tank. The inner vessel is supported on the reactor support structure. Vertical penetrations through the vessel are made in the conical region.

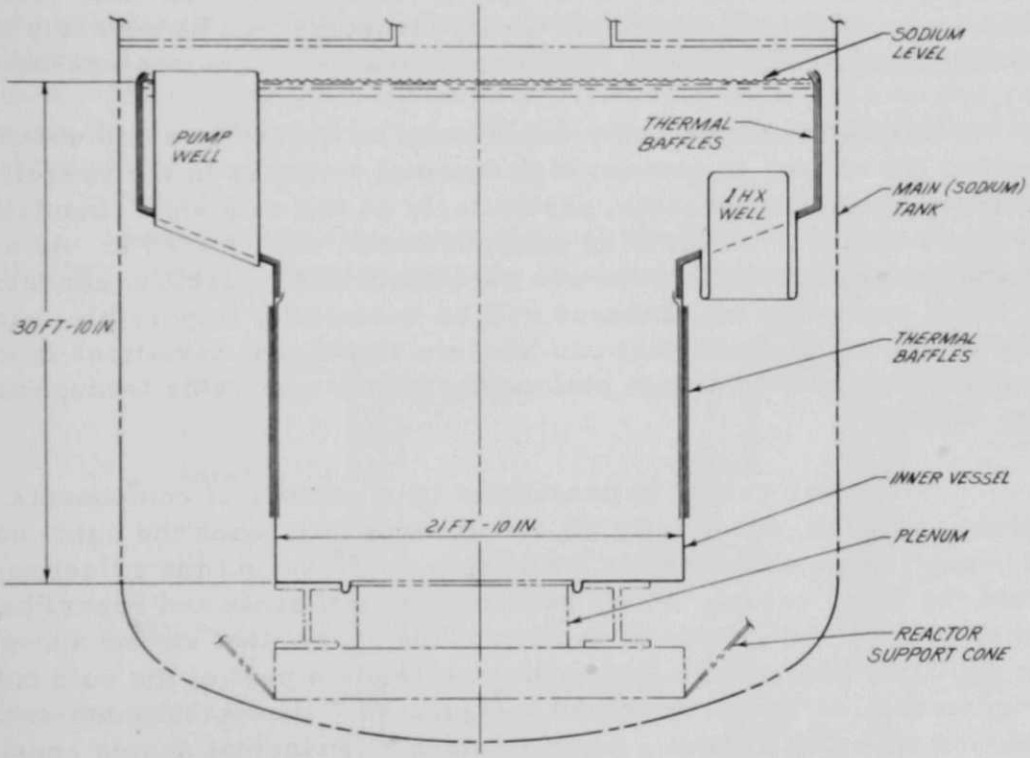


Fig. VI.6. Phenix Inner Vessel*

Each of these inner-vessel configurations introduces unique design considerations. The PFR vessel has a more complex configuration than the Phenix vessel. This undoubtedly aggravates differential expansion, causes design and fabrication difficulties, and elevates costs. The complexity of the

*Recently it has been learned that the thermal baffles indicated were omitted in the final design; Phenix employs no insulation (thermal baffles) on the inner vessel.

vessel and the use of flexible expansion members contribute to the concern for long-term reliability. The design permits location of the pumps outside the inner vessel, in the lower-temperature bulk sodium region. In this region the pump is less subject to heat-conduction effects of the hot sodium. It is, however, subject to asymmetric heat radiation from the surface of the 1000°F-plus sodium and possibly free-convection streams from the inner vessel during low-flow conditions. Nonuniform thermal effects on the pump casing must be avoided, or distortion of the casing and binding of the pump can occur. In Phenix, the pumps as well as the IHXs penetrate the inner vessel. As a consequence, thermal protection of the pump from both heat radiation and heat conduction through the sodium is a consideration. Uniformly distributed sodium circulation within the pump well should be provided to avoid casing distortion. The pump hydrostatic-bearing flow can be the source of this sodium. The close proximity of the inner vessel to the main tank creates a likely region for thermal stratification. A symmetrically shaped inner vessel at the sodium liquid level has the advantage of providing a more uniform distribution of radiant heat to the walls and shield deck or main-tank cover. Maintenance capability of the inner-vessel structure will be severely limited; consequently, the design should emphasize a maintenance-free approach.

Insulation of the inner vessel may be needed to avoid undue loss of heat across the vessel, to prevent high thermal stresses in the vessel, and to reduce free convection currents, particularly on the cold side. Insulation is clearly more difficult for complex configurations, e.g., for PFR. As a result, design and fabrication difficulties are escalated, and reliability concerns develop. Since insulation maintenance will be essentially impossible, passive systems of stark simplicity that can tolerate significant deviations from design should be sought--a design philosophy that is applicable throughout the primary system.

The inner vessel is penetrated by a number of components. In the Phenix approach, essentially all components that reach the cold sodium region penetrate the vessel. For PFR, only the IHXs and the refueling port penetrate the inner vessel. Pump penetrations in Phenix and SuperPhenix consist of nozzles in the conical section of the vessel that extend above the sodium surfaces; the sodium inside the nozzles is a part of the cold bulk sodium region outside the inner vessel. Figure VI.7 shows the pump well for SuperPhenix with and without a pump in place.* Principal design considerations are avoidance of nonuniform temperature patterns (and thus distortion) in the pump casing, provision for adequate flow arrangements for the pump suction, and suction arrangement to promote good mixing of bulk sodium.

Penetrations for the IHXs impose additional design requirements. These penetrations must accommodate radial and axial differential expansion between the IHX and the inner vessel and must limit sodium leakage through

*The sodium temperatures for SuperPhenix as shown in Fig. VI.7 and subsequent references to SuperPhenix may be high and should be considered as approximate since various values appear in the literature.

the penetration to reasonable values. As noted in Table VI.1, radial differential expansions can be 3 in. or larger, depending upon the manner of component support that is chosen, outlet temperature of the reactor, and radial distance of the IHX from the center of the system. Figure VI.8 shows the system planned for SuperPhenix (similar to Phenix), with and without the IHX in place. Leakage can occur through the long, narrow annulus, as shown in the enlarged detail of the penetration. For large radial differential expansions of the kind expected in 1200-MWe-size systems, some form of seal or flow-restriction device is required to limit leakage to suitably low values.

We believe that both Phenix and SuperPhenix employ a penetration that eliminates all physical interaction between the IHX and the inner vessel. A gas space is maintained in the upper region of the penetration, thereby forming a seal against sodium leakage through the joint. The differential sodium level established within the penetration is equal to the differential sodium level in the main tank. Operation must assure that the seal is maintained. Loss of the gas space will permit large sodium-leakage rates and the undesirable effects of a hot sodium stream entering the cold side of the system. Loss of sodium in the seal will permit gas to escape into the main tank, with the attendant concern that it might be swept into the reactor. Otherwise, loss of the seal is not critical, since it presumably could be reestablished.

Leakage may also occur through the IHX during those occasions when the IHX is out of service with the inlet valve closed. Inlet valves of the

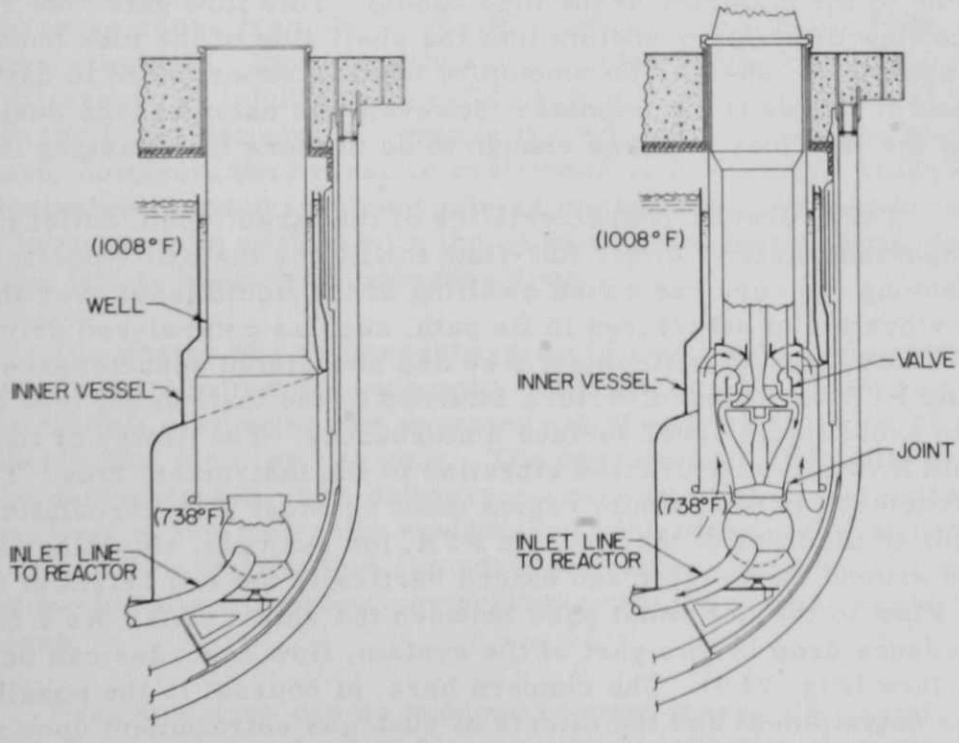


Fig. VI.7. SuperPhenix Pump Well with and without Pump in Place

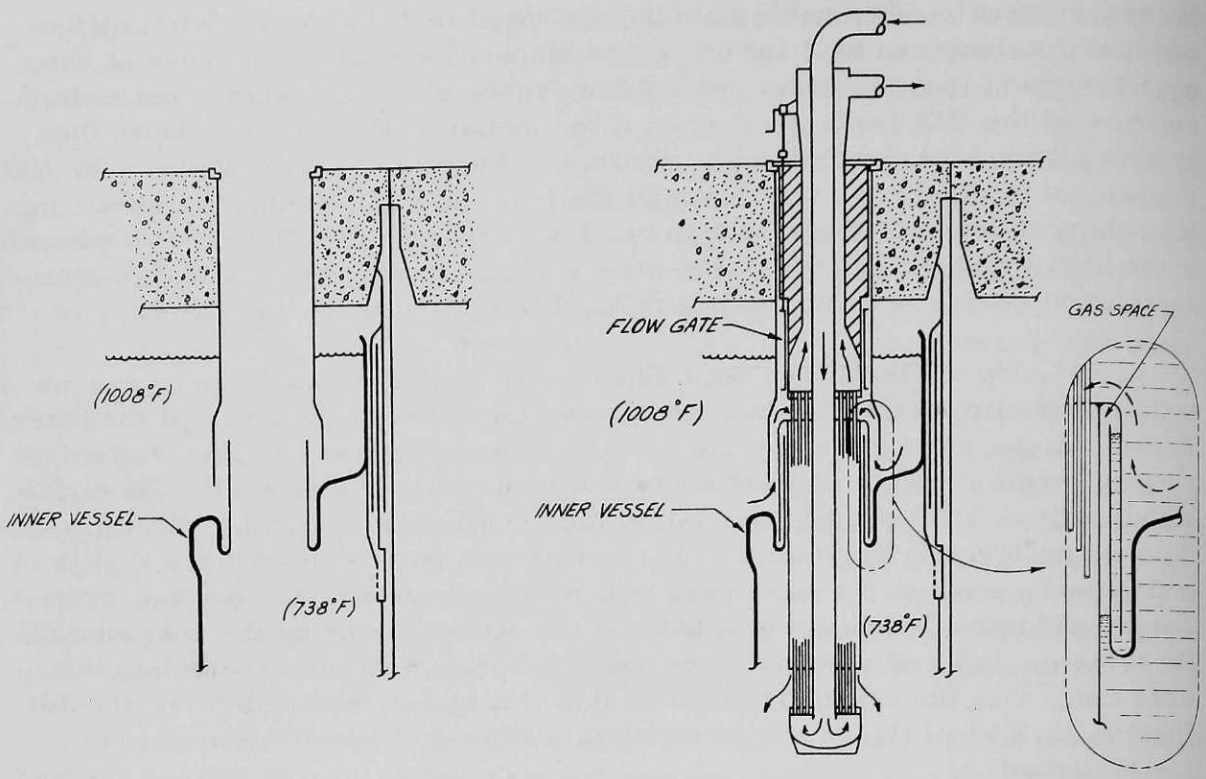


Fig. VI.8. SuperPhenix IHX Well with and without IHX in Place

type used for PFR and Phenix are large cylindrical structures of a diameter comparable to the diameter of the tube bundle. This flow gate (see Fig. VI.8) blocks the flow of primary sodium into the shell side of the tube bundle when lowered against the shell. The amount of leakage experienced to date with these types of valves is not reported; however, the nature of the design suggests that the rate may be large enough to be of more than passing interest.

The hydraulic characteristics of the large, open, outlet plenum are an important factor. Under full-flow conditions the exit velocity of the coolant leaving the core can cause swelling of the liquid level over the core area and vibration of structures in its path, such as control-rod drives, refueling devices, and the instrument tree and associated appurtenances. Both Phenix and PFR have flow diverters attached to the instrument tree over the reactor to avoid liquid-level surface disturbances. The design of the diverter must avoid hydraulically created vibration of the instrument tree. The location of structures in the plenum region must consider flow circulation from the reactor to the inlet of the IHX. In PFR, for example, shielding (shield rods) is located around the reactor and extend vertically the full height of the outlet plenum. Flow to the IHX must pass between the shield rods. As a consequence of the pressure drop in this part of the system, flow cascades can occur at the liquid surface (Fig. VI.9). The concern here, of course, is the possibility of cover-gas entrainment and the effects of such gas entrainment upon reactor operation. A uniform flow pattern into the IHX is also important. The flow

pattern is affected by the features previously discussed as well as the structures immediately in the vicinity of the IHX. Failure to obtain a uniform flow into the IHX can create differential strains in the unit that exceed design values.

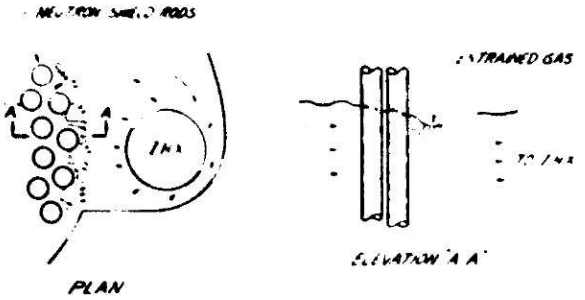


Fig. VI.9

Gas Entrainment in Sodium Exiting around PFR Neutron Shield Rods[12] (courtesy of IAEA)

The IHX primary-coolant inlet must have sufficient submergence to avoid vortex formations and consequent gas entrainment, flow starvation of the IHX, and transient effects on the IHX from the transient vortex conditions. The well for the Phenix IHX, for example, extends well below the liquid level. This not only protects against vortex actions and induces a more uniform flow pattern, but provides the IHX with a heat shield against radiant energy from the 1000°F-plus sodium which surrounds the upper part of the unit. Submergence of the IHX primary-coolant inlet plenum must also be sufficient to permit flow through the unit in the event of failure of the main tank and resultant depression of the sodium liquid level.

The liquid levels in the hot and cold regions reflect the hydraulic resistance of the IHX flow circuit to the flow through the IHXs. Current liquid-level differences range from 26 in. for Phenix[13] to 5 or 6 ft for PFR; the latter is not known precisely. Each facility has about the same full-power flow rate; thus, the difference in liquid levels reflects design choices affecting flow resistance in the hydraulic circuit between the hot and cold regions of the tank. In each case, however, the hydraulic resistance is low. A low differential level is desirable to avoid transient effects on the main tank, inner vessel, and other structures in contact with the sodium liquid level. This, of course, requires the IHX to have a low pressure drop.

The design of low-pressure-drop (<3 psi) heat exchangers for LMFBR use is not uncommon; such units are used in EBR-II, Phenix, and PFR. The EBR-II exchanger has operated satisfactorily for over 10 years, and the Phenix unit for over one year. The operation of PFR units at power is yet to be demonstrated. The primary concern with a low-pressure-drop IHX is the effect of low hydraulic resistance of the tube bundle on flow distribution. Poor flow distribution can adversely affect operational reliability by causing temperature distributions and resultant thermal stresses that exceed design values.

Pressure drop can be reduced in many ways, e.g., small L/D bundle ratio, large inlet and outlet plenum, use of tube-side flow with large-diameter tubes, large tube pitch, circular rather than triangular pitch, etc.

Figure VI.10 shows the relationship between bundle diameter, bundle length, hydraulic resistance, and rating for an EBR-II type of IHX using a 0.625-in.-OD tube and a 0.0625-ft triangular pitch. A low ratio of pitch to tube diameter (1.20) was used to accentuate pressure drop and minimize bundle diameter for use in the 3000-MWt-size EBR-II concept described in Sec. III. The surface area in the inlet- and outlet-plenum region is assumed 50% effective. For the example shown in Fig. VI.10 (left), a unit with a 20-ft-long bundle has a 7.4-ft diameter and a ΔH of 22 ft of sodium. Similarly, an IHX with a 25-ft-long bundle has a diameter of 6.7 ft and a ΔH of 34 ft of sodium--a rather large increase in pressure drop for a small change in bundle diameter. The effect on ΔP would be less severe for a larger pitch-to-tube-diameter ratio.

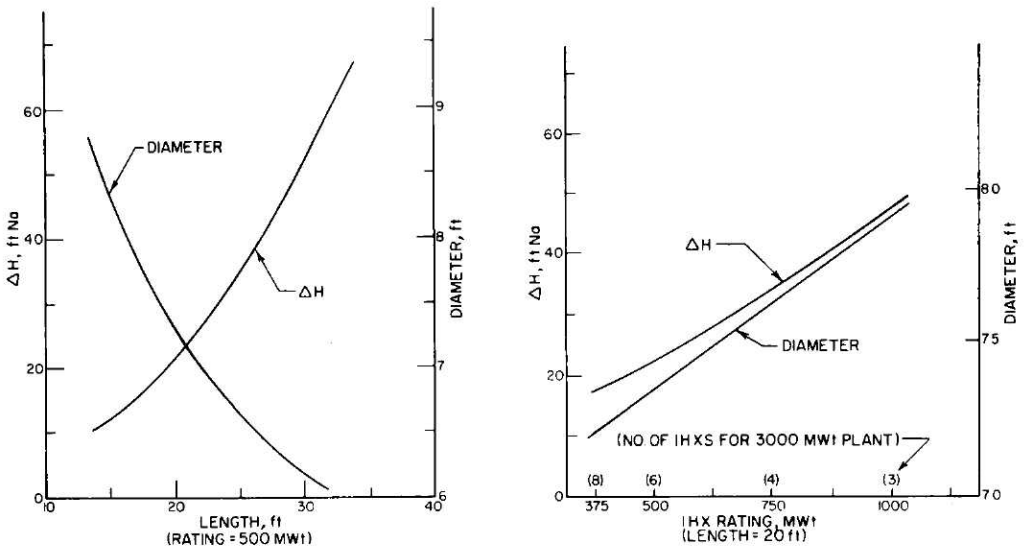


Fig. VI.10. Relationship between Bundle Diameter and Length, Hydraulic Resistance (ΔH), and Rating for EBR-II Type of IHX Using 0.625-in.-dia Tube and 0.0625-ft Triangular Pitch

Figure VI.10 (right) shows the effect of a change in thermal rating on the bundle diameter and ΔH for constant bundle-internals geometry and length. Consider the case of one or two heat exchangers per secondary loop for a three-loop plant. With one IHX per loop, each IHX has a rating of 1000 MWt. For a bundle length of 20 ft, the IHX has a bundle diameter of 7.9 ft and a ΔH of 47 ft of sodium. With two IHXs per secondary loop (each IHX rated at 500 MWt), the bundle diameter is reduced to 7.4 ft, and the ΔH is 22 ft of sodium. Selecting four secondary loops rather than three, each with two IHXs, further reduces the bundle diameter and ΔH to 7.2 ft and 17 ft of sodium, respectively.

Figure VI.11 illustrates the effect of increasing the pitch in the triangular tube lattice on hydraulic resistance and bundle diameter. With a pitch of 0.0677 ft for a 20-ft-long unit, a diameter of 7.9 ft and a ΔH of 9.4 ft of sodium are calculated as compared to a 7.3-ft diameter and 22-ft ΔH for a 0.0625-ft pitch.

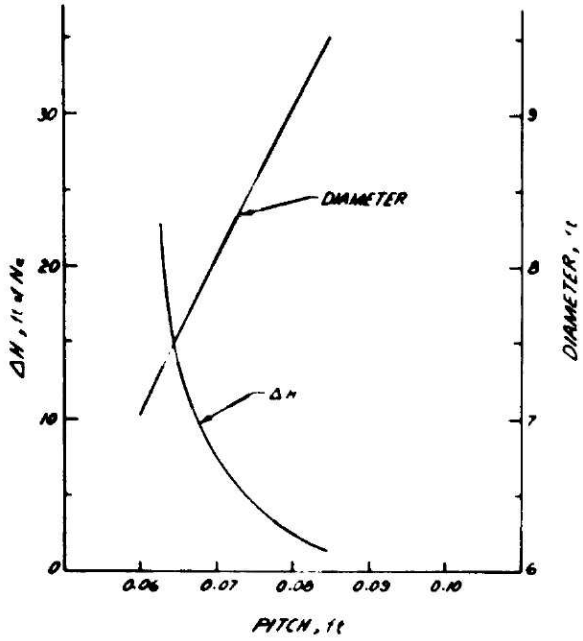


Fig. VI.11

Variation of Bundle Diameter and Hydraulic Resistance (ΔH) with Tube Pitch for a 500-MWt, 20-ft-long Pool-type IHX

The use of a circular pitch has a number of advantages. First, the tubes are arranged in a circumferential pattern. This provides a more uniform fit of the bundle with the shell and, therefore, less of a problem with sodium bypass of the bundle and overheating or overcooling of the edge tubes. Second, the transverse pitch between tubes along the circumference can be different from the radial (lateral) pitch between circumferential tube rows. This permits a more generous circumferential pitch to reduce hydraulic resistance to cross flow and a tighter radial pitch to maintain axial velocities within the tube bundle. Third, the flow across a circular tube pattern should approximate that of an in-line tube arrangement and, therefore, should have cross-flow friction factors better estimated by in-line than staggered tube correlations. As noted in McAdams,[14] the correlation for in-line arrangements gives lower friction factors than for equivalently spaced staggered tubes in a triangular pitch. For example, compare an in-line 0.625-in.-tube arrangement having a circumferential (transverse) pitch (P_c) of 0.0677 ft and a radial (lateral) pitch (P_l) of 0.0677 ft with a triangular 0.625-in.-tube arrangement having a pitch of 0.0677 ft. The calculated friction factors (f") would be

$$f'' = 0.54\left(\frac{D_0 G_{max}}{\mu}\right)^{-0.15} \quad \text{(in-line tubes)},$$

and

$$f'' = 0.63\left(\frac{D_0 G_{max}}{\mu}\right)^{-0.15} \quad \text{(staggered tubes)},$$

where

D_0 = outside diameter of the tube,

G_{max} = maximum mass velocity through the minimum cross section,

and

μ = absolute viscosity of the liquid.

Low pressure drop, however, does not ensure good flow distribution. On the contrary, it is more likely to aggravate the situation, causing poor flow distribution. An important element of acquiring good flow distribution on the shell side is the provision for a relatively low hydraulic resistance in the cross-flow region at the inlet and outlet of the shell side as compared to the axial resistance along the tube bundle. A circular tube arrangement and/or a liberal tube pitch is important. The number of tube rows through which the flow must cross should be small; the use of this criterion will increase bundle diameter faster than shown in Fig. VI.10 (right), as IHX ratings are increased, because the curves shown are based on constant secondary flow velocities in the secondary inlet line. Fluid velocities in the cross-flow region should be low relative to the axial velocity along the bundle--a requirement which increases the length of the heat exchanger and/or increases the length of inlet and outlet plenums relative to the length of the bundle.

Hydraulic problems in intermediate heat exchangers for the French Phenix and SuperPhenix are discussed in Reference 15, wherein flow distribution through the Phenix IHX, as determined by model tests, is shown. In addition, calculated isothermic curves along the length of an IHX for SuperPhenix are presented. The calculated temperature difference between fluid streams at the bottom exit is 158°F. This occurs from the lack of a uniform flow distribution and large inlet and outlet regions--about 30% of heat transfer surface in this area for the case presented in Reference 15. Because the vertical dimensions of the primary system change little as system capacity is raised, the increase in IHX capacity must be gained largely by an increase in IHX diameter. For a fixed tube length and tube geometry, the number of tubes increases as the square of the bundle diameter, the flow increases in proportion to the number of tubes, and the cross-flow areas in the inlet and outlet regions increase directly with bundle diameter. Accordingly, as IHXs increase in capacity, the height of the inlet and outlet regions must increase to avoid large hydraulic losses in this area and the resulting effect on flow distribution throughout the unit.

Improved primary-coolant flow distribution can be obtained with low primary-system hydraulic resistance by putting the primary coolant through the tube side, as in PFR (Fig. VI.12). The problem of flow distribution is then transferred to the secondary (shell side) of the unit. In this case full secondary coolant flow must enter at the inner (small) diameter of the tube bundle, with resulting cross-flow velocities that are higher than obtained for

the same flow rate entering the outer diameter of the bundle. Since high pressure drops can be more readily accommodated by the secondary system, flow baffles such as the disc-and-donut type could be considered to improve flow distribution.

tube maintenance problem compounded — now dealing with radioa. primary system

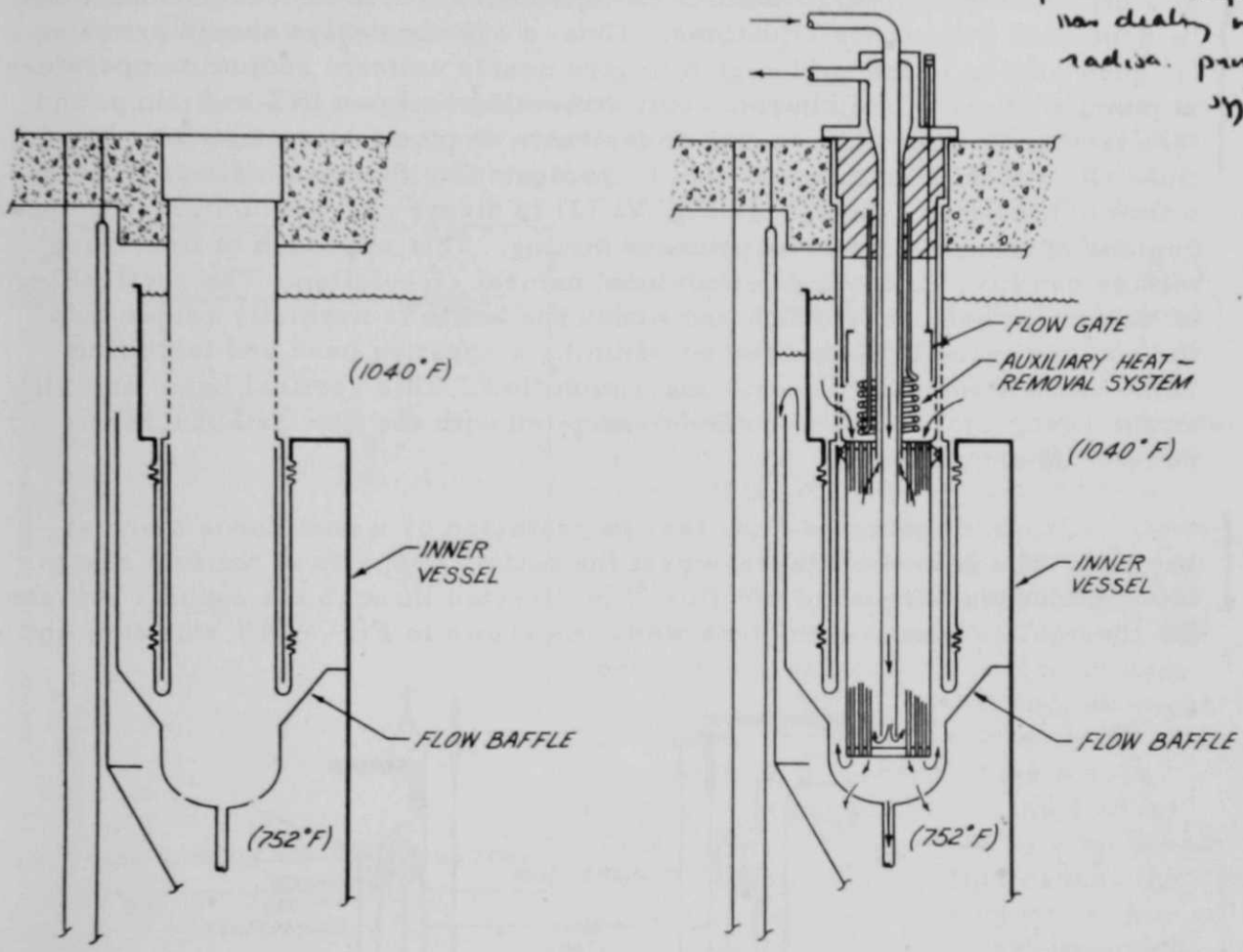


Fig. VI.12. PFR IHX Well with and without IHX in Place

The effects of free-convection circulation and thermal gradients in the cold-sodium region are important considerations. The large, hot surface of the inner vessel will cause thermal convection currents to form. These sodium currents can wash structural members such as the wall of the main tank, if unprotected, or the reactor support structure, causing local, transient temperature perturbations of an unknown magnitude. In confined areas, thermal stratification may occur, e.g., in the narrow annular region between the Phenix inner vessel and main tank. As mentioned earlier, leakage past IHX penetrations and "sneak" currents through out-of-service units are other sources of high-temperature sodium streams.

Also, structures in proximity to the primary coolant outlet of the IHX will be exposed to normal and transient temperature variations of the

coolant leaving the IHX. In particular, the close proximity of the pumps and IHXs is a concern. For example, a sudden loss of flow in a secondary circuit, such as might occur if a large water leak happened in a steam generator, can cause the primary coolant to leave an IHX at a high sodium temperature and at a high flow rate if no protective action occurs. Critical components must be protected from these situations. Thus, a system design should provide: (1) good mixing in the cold region to give nearly uniform sodium temperature at pump suction, (2) no short-circuit streaming between IHX and pump, and (3) circulation control to avoid undesirable thermal stratification. In addition, all critical structures should be protected by thermal baffles. PFR has a flow baffle around each IHX (Fig. VI.12) to divert cool sodium into the upper regions of the main tank and promote mixing. This approach to improving mixing can have an adverse effect upon natural circulation. The vertical leg of sodium external to the IHX and within the baffle is normally colder than the sodium in the IHX, thereby constituting a negative head and inhibiting rather than promoting thermal convection flow. This vertical head, as well as the reversals in flow direction associated with the flow path, increase pump head requirements.

In Phenix, the main tank is protected by a continuous thermal barrier. The downward leakage past the bottom supports of the fuel assemblies (about ten percent of full flow[16]) is directed through the annulus between the thermal barrier and the tank wall. As shown in Fig. VI.13, this flow spills

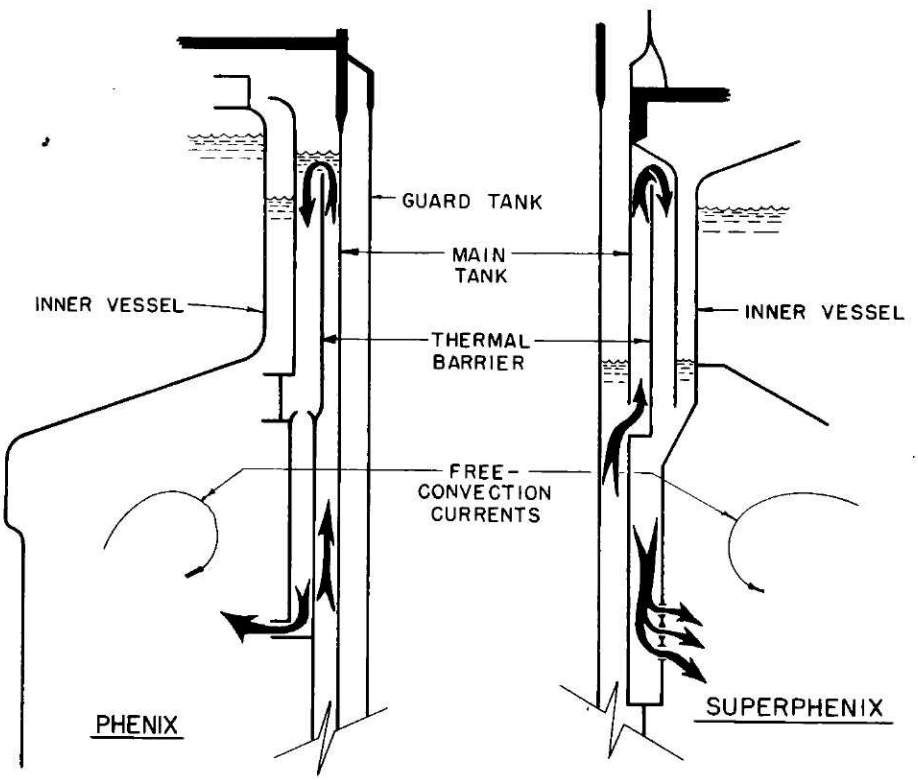


Fig. VI.13. Arrangement of Main-tank Cooling Baffles for Phenix and SuperPhenix

over the thermal barrier and into a second annular region from which it exits into the pump-suction region near the mid-elevation point of the tank. For PFR (Fig. VI.12), the flow from the IHX is redirected upwards to ensure that the upper volume of sodium in the cold region does not stratify. (The upward flow may adversely effect the development of thermal head for natural-convection-flow situations.)

In the open-plenum concept, as contrasted with the closed-plenum concept, the main-tank cover or shield deck and most of the components that penetrate these structures are exposed to 1000°F sodium. The transfer of radiant heat energy to these structures and components must be controlled. Insulation requirements to protect the shield deck are severe. The insulation must operate maintenance-free in a sodium-vapor, cover-gas environment for the life of the facility. Pump casings and the upper (cover-gas) portion of the tank wall must be protected. The designers of this protection must recognize that the sodium temperatures in this region can change rapidly, especially between the limits of the reactor inlet and outlet temperatures. The high sodium temperature brings a high sodium vapor pressure. In the region from 720°F (the temperature of sodium in the closed-plenum system) to 1000°F, the sodium vapor pressure increases by a factor of 35 to 40 (Fig. VI.14). This causes the concentration of sodium vapor in the cover gas to increase correspondingly. Sodium vapor will migrate into the spaces between rotating and fixed plugs and into annuli within components that open to the cover gas or high-temperature sodium, etc., and will deposit on cool surfaces. These deposits can cause bearing and seal problems and the binding of moving structures such as rotating plugs, pump shafts, control-rod drive shafts, and refueling devices.

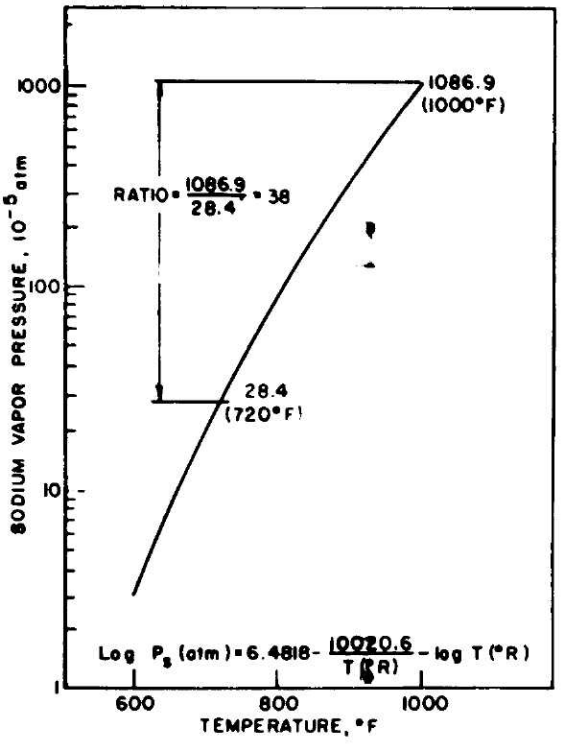


Fig. VI.14. Liquid-sodium Vapor Pressure (equation is from Reference 17)

2. Closed (piped)-outlet-plenum Concept

The principal feature and characteristic of the closed-plenum concept as used for EBR-II is the piped reactor-outlet flow arrangement and the isothermal nature of the pool. The key issues associated with this concept are listed in Table VI.3.

TABLE VI.3. Key Design Considerations for the Closed-plenum Concept

Reactor-outlet plenum and cover, cover-lifting and cover-latch mechanisms, and outlet piping required (cover doubles as instrument tree)
Outlet-piping space may increase tank diameter
Height of outlet plenum reflects upon height of tank, control-rod drives, and fuel handling mechanisms
Isothermal tank, i.e., coolant boundary
Reduced heat loads on shield deck
No severe temperature gradients in main tank
Reduced requirement for in-sodium insulation: outlet piping and plenum only
Low sodium vapor pressure; thus, a smaller aerosol concentration
Thermal transients confined to hot-coolant area
Large thermal inertia of bulk sodium; all sodium at reactor inlet temperature
Predictable flow pattern and temperature of coolant to IHX
ΔP of outlet piping and IHX relatively high
Favorable conditions for natural convection of primary coolant and dissipation of decay heat from main tank

tradeoff reality size of design of tank, deck vs. piping, cover, IHX differences

The closed-outlet-plenum concept requires a reactor vessel, an outlet plenum with outlet piping nozzles, a reactor-outlet plenum cover, and cover-lifting and redundant sets of cover-latch mechanisms. The plenum may operate at nominal pressures up to 20-30 psi. For a 3000-MWt reactor the diameter of the plenum is likely to be in the 14- to 18-ft range with a height of 4.5 to 5 ft. The cover will be of similar diameter. The thickness of the cover depends upon the amount of shielding that it contains. A 2-ft-thick cover probably will provide sufficient shielding to avoid the need for shielding in the upper part of the fuel assembly, thereby reducing the cost and height of the assembly. The reactor-outlet plenum cover also serves as an instrument tree, thereby eliminating the need for a separate structure. Latching the cover to the reactor vessel may avoid stability problems that can occur with an instrument tree having no lower support.

low thermal inertia of outlet pool.

The outlet plenum and associated piping are key design areas because these are the parts of the primary coolant system most exposed to thermal transients and because they interface with safety and control drives. The maximum temperature transients occur at the reactor outlet and are transmitted to the plenum with little change. Consequently, the reactor-outlet plenum, plenum cover, cover latches, piping and pipe nozzles, and subassembly holddown and/or reactor outlet instrumentation thimbles are directly exposed to these transients. Appropriate thermal shielding of critical areas is required. Thermal shields are subjected to fairly high fluid velocities and, therefore, must be supported in a fashion that permits thermal movement and avoids deleterious vibrations. The plenum operating pressure, and thus the maximum ΔP for piping and IHXs, will be limited by the maximum pressure capability of the plenum--probably associated with cover distortion or latch limitations. The latch limitations are particularly critical. More than one set

of independent latches will be needed, since failure of latches could possibly shift the cover and bind control-rod drive shafts which penetrate the cover. The plenum must be deep enough to accommodate the pipe nozzles. Code requirements will limit nozzle spacings circumferentially.

The design of suitable outlet piping between the reactor and the IHX is a major problem, since the piping configuration must not unduly increase the diameter of the main tank; minimizing the tank size will be a key design objective. Transient temperatures can be severe, however, since the small plenum volume does little to moderate the effect of reactor transients. Critical areas such as the plenum and plenum cover require thermal-protection features.

Accommodation of differential thermal expansion without unduly complex piping or resultant impact on tank size is difficult and is a major factor that has swayed some designers toward the open-plenum concept. The results of this study indicate that a 1200-MWe-size closed-plenum concept of the EBR-II type may require a 12-ft-dia core, an 18-ft-dia reactor vessel and vessel cover, and a free space of about 32 ft in diameter over the reactor area (Fig. VI.3) for maneuvering the cover for fuel handling. The piping will be about 30 in. in diameter and cannot encroach upon the cover-maneuvering space. The centers of the IHXs are likely to be 26-30 ft outboard of the core center for tank diameters of 65-70 ft. Nozzle (outlet plenum)-to-nozzle (IHX inlet) movements of 1-3 in. radially and 0.5-1 in. axially are to be expected. The magnitudes of the radial movements are strongly dependent upon the type of component support. The limited space available to accommodate differential motion amplifies the severity of the problem. Conversely, provision for additional space imposes economic as well as other design penalties.

Piping for pool systems has a number of advantages unique to the concept. A key advantage, of course, is the capability to permit some leakage at pipe joints. This characteristic broadens the designer's choice of acceptable piping arrangements. Moreover, piping designed for submerged sodium service does not have the support problems normally associated with sodium piping systems. The buoyancy effect of the pool cancels the weight of the sodium in the pipe and reduces the dead weight of the pipe. The piping cannot suddenly be drained of hot sodium. Therefore, the system does not have to withstand the sudden application of large pipe-hanger loads that can occur during such an occurrence. Distances between nozzles usually are small, thereby eliminating the need for an extensive system of pipe hangers, guides, supports, and anchors. Seismicity effects do not appear to be unduly severe; the problem focuses chiefly on the remote disconnect pipe joint, i.e., between the pump discharge line and the line to the reactor inlet plenum.

Some piping approaches that can be considered for pool systems are illustrated in Fig. VI.15. The ability to permit controlled leakage and the submerged aspect of the pool concept are evident in most of the ideas shown here.

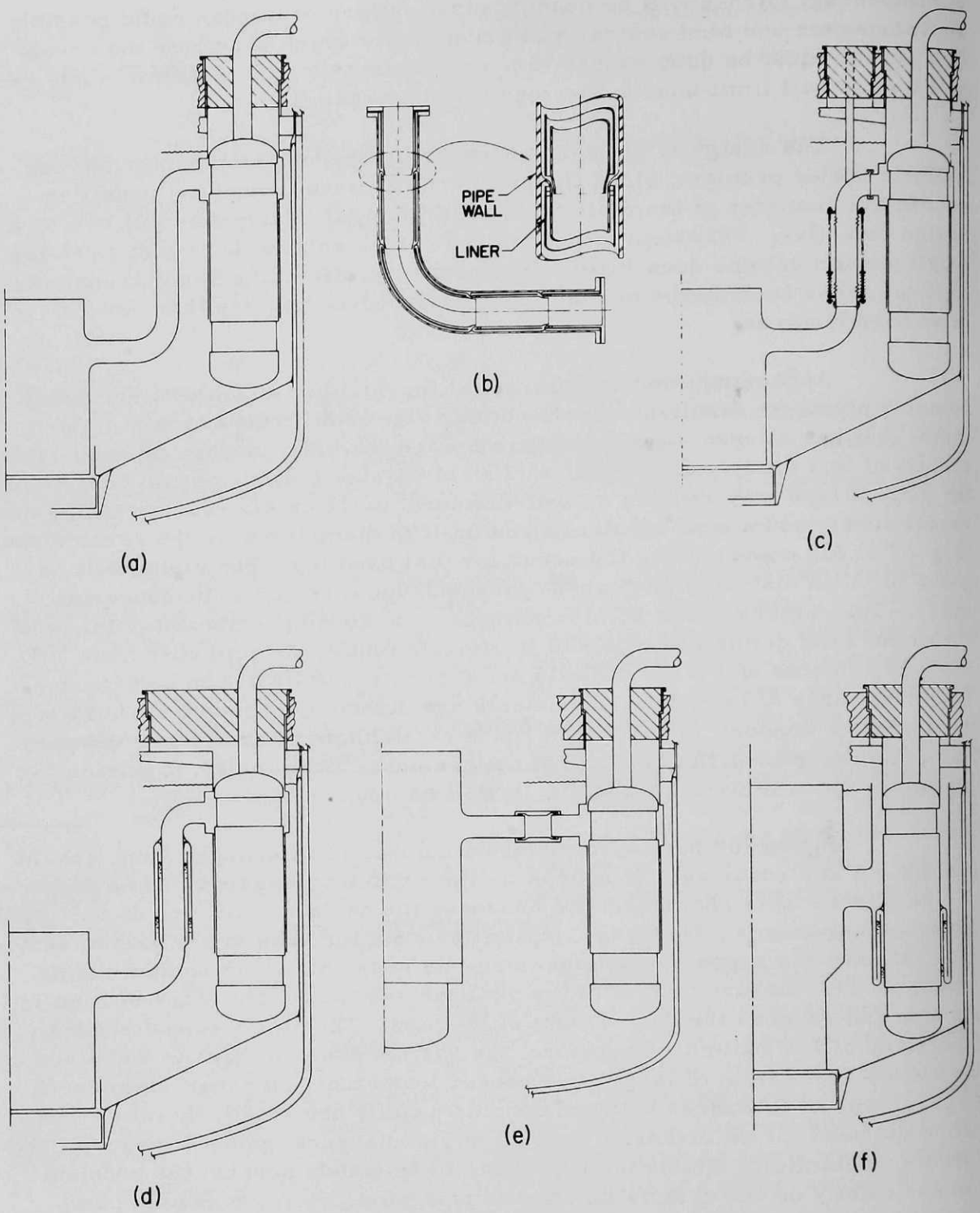


Fig. VI.15. Piping Approaches for LMFBR Pool Systems

Figure VI.15a shows a conventional, all-welded arrangement. Each pipe is welded to the reactor outlet plenum and IHX inlet plenum and is configured to provide the required flexibility. The configuration would be more complex

than the one illustrated. The challenge of design is to find a suitable configuration that has conservative stress levels, is reliable, provides for adequate reactor cooling, and has no intolerable effect on the size of the main tank. The piping must serve without maintenance or inspection for the life of the plant. This approach is used in EBR-II and has seen over 10 years of sodium service without incident.

The arrangement shown in Fig. VI.15a can use insulation (liners) inside the reactor outlet piping, as shown in Fig. VI.15b. This approach reduces the mean temperature of the fixed pipe, thereby reducing the amount of thermal differential expansion to be accommodated. The internal liner adds complexity to the system and increases the probability for maintenance.

Figure VI.15c illustrates the concept of removable piping. In this approach, a vertically oriented pipe section is supported from, and removable through, a plug in the roof of the main tank. This concept permits maintenance and inspection of the removable section of pipe. Consequently, flexible expansion members can be employed, reducing the difficulty of accommodation of differential movements and simplifying piping configuration. Considerations with this approach are (1) space for a shield plug over the removable pipe section, (2) remote connections of two joints, (3) the possibility to locate an IHX isolation valve, flowmeter, or other device in this part of the line, and (4) seismicity.

The piping concept shown in Fig. VI.15d eliminates the usual forces and loads created by differential thermal expansions. The piping is composed of two sections that are slipped together but are not attached. The upper pipe section is welded to the inlet plenum of the IHX and extends downward into the primary tank as far as possible. The downward section is shaped like an annular bell jar, i.e., an annulus closed at the top and open at the bottom. The lower piping is welded to the reactor outlet plenum; it extends upward as far as possible, and fits within the annular region of the upper pipe. Inert-gas pressure sufficient to prohibit flow through the seal is maintained in the annular region, thereby avoiding leakage from the outlet pipe. The differential liquid level in the two parts of the annular pipe joint will manifest the differential pressure between the pressure inside and outside of the piping. Some considerations relative to this approach are that (1) the available height of the annular pipe joint limits differential sodium pressure and thereby the available ΔP across the IHX; (2) sodium levels in the annular joint will change as a function of flow rate in the primary system; (3) changes in sodium levels will cause thermal transients in the pipe wall; and (4) long unsupported lengths of vertical piping may intensify seismicity design problems.

Figure VI.15e shows a concept with the reactor outlet plenum extended vertically upward so that each outlet pipe can run directly and horizontally to its IHX. Differential expansion is accommodated by a slip joint in the piping. This design focuses on the adequacy of the slip joint, the design

and function of the reactor plenum cover, and the thermal problems associated with the larger outlet plenum (similar in many respects to the thermal and hydraulic considerations of the open-plenum design).

Figure VI.15f extends the concept in Fig. VI.15e. Here the piping slip joint is eliminated, and the IHX well casing moves relative to the IHX without constraint. This approach avoids the problems of a slip joint in the piping, but requires the designer to provide an IHX well casing that can move freely and independently of the IHX and without undue effect on other parts of the primary system. A well casing cantilevered from, and supported by, the reactor outlet plenum via the connecting piping is the simplest form of this concept; feasibility of the configuration remains to be demonstrated.

The closed outlet plenum permits an isothermal main tank, i.e., one having a hot cover separate from the cold shield deck which supports the tank. The isothermal state of the main tank in conjunction with the relatively low temperature of the system, say 720°F, provides a number of attractive design considerations. Some of these are: no severe transients or thermal gradients in the bulk-sodium area of the tank; low sodium aerosol concentration in the cover gas; modest in-sodium insulation requirements; and large thermal inertia of the bulk sodium.

Confinement of the hot (1000°F) reactor-exit sodium to the small plenum and outlet piping greatly reduces the surface area for heat losses, the amount of free-convection circulation, and the amount of radiant heat transfer to the main-tank cover or shield deck. With an essentially isothermal tank, thermal stresses in the coolant boundary (the tank wall) are small. Should the tank be supported at the top by a skirt, as in PFR, rather than by hangers as in EBR-II or at the bottom like BN-600, thermal stresses in these areas will be important considerations. There are no large leakage currents such as occur at IHX penetrations for open-plenum concepts, since the differential-expansion space requirements between the IHX and its well casing are negligible. Critical structures in proximity to the IHX inlets, such as nozzle connections to the main-tank cover, and outlets must be protected from normal and transient effects of the coolant stream. Pump suctions and IHX exits must be located to assure hydraulic mixing and to avoid thermal stratification of the sodium in the tank.

The low vapor pressure (Fig. VI.14) associated with a 720°F bulk sodium temperature provides a low concentration of sodium aerosol. Nonetheless, aerosol deposition in cool, confined areas of structures and mechanisms that communicate with the primary sodium is a problem; sizable sodium depositions are now evident in EBR-II after 10 years of operation with a 700°F bulk sodium temperature. However, the lower the aerosol concentration (a function of sodium temperature), the smaller the deposition problem.

The in-sodium insulation requirements are less than in the open-plenum concept. If a main-tank cover is used, the requirement for insulation

in a sodium-aerosol environment is eliminated and exchanged for more conventional insulation in the gas space between the cover and shield deck. If no main-tank cover is used, the problems of roof insulation and protection of the upper portion of the tank wall are similar to, but of lesser magnitude than, those in the open-plenum approach.

The thermal inertia of any pool system is large. The closed plenum approach has the largest thermal inertia, as the mixed mean sodium temperature is lowest. For example, the closed-plenum concept has most of the sodium at reactor inlet temperature, whereas in the open-plenum approach about half the sodium is at reactor outlet temperature. Each contains about the same amount of sodium.

The piped outlet introduces an increased pressure drop in the system, permits a large ΔP through the IHX, and creates more predictable flow patterns and sodium temperatures from the reactor exit to the IHX inlet. The additional ΔP created by the piped system is readily handled by the pump design, since total system resistances are fairly low, usually less than 350 ft of sodium. The extra head available for the IHX permits the use of internal orifice plates or other devices to provide improved flow distribution. The ΔP from reactor exit to IHX exit creates a pressure in the outlet plenum. This pressure should be modest, probably not to exceed 20 to 30 psi, to avoid plenum-cover and latch design difficulties. More predictable flow patterns and sodium temperatures assist the designer in estimating the response of the coolant system in passing from forced- to free-convection flow.

3. Open-outlet-plenum Concept with Separated Coolant Loops at Reactor Inlet

The BN-600 utilizes the open-plenum concept with the primary coolant systems separated by the connection of the IHX outlet/pump-suction regions (Fig. VI.16). This approach is the opposite of the closed-outlet-plenum

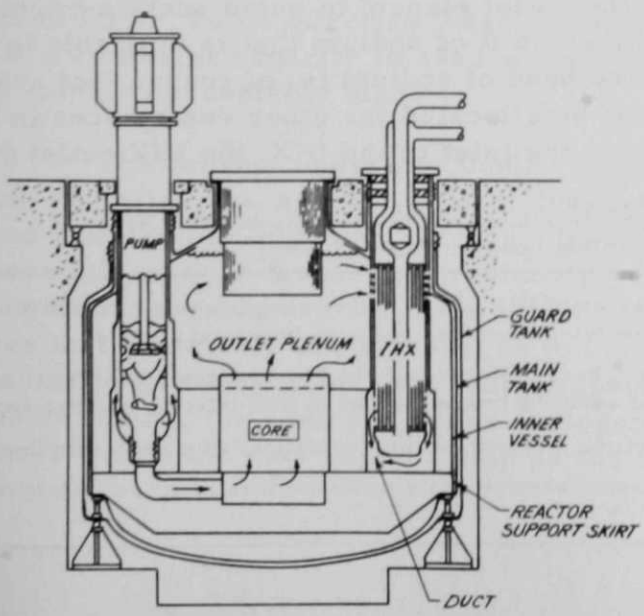


Fig. VI.16

Open-outlet-plenum Pool System with Connected Pump Suction and IHX Outlet (BN-600 type)

approach in the sense that the coolant from two IHX outlets is ducted to one pump suction (the two IHXs are common to one secondary loop) and all the IHXs draw from a common, open, outlet plenum. In this arrangement there is a small volume of sodium in the duct between the IHX outlets and the pump suction. No differential level exits between free-standing liquid levels. The only steady operating liquid level exists in the hot, outlet-plenum region. The main tank is cooled by sodium from either the pump discharge or inlet-plenum leakage, or both. The coolant flows in an annular region between the tank wall and an inner vessel and overflows to the outlet-plenum region (Fig. VI.17) in a manner ill-described in the literature.

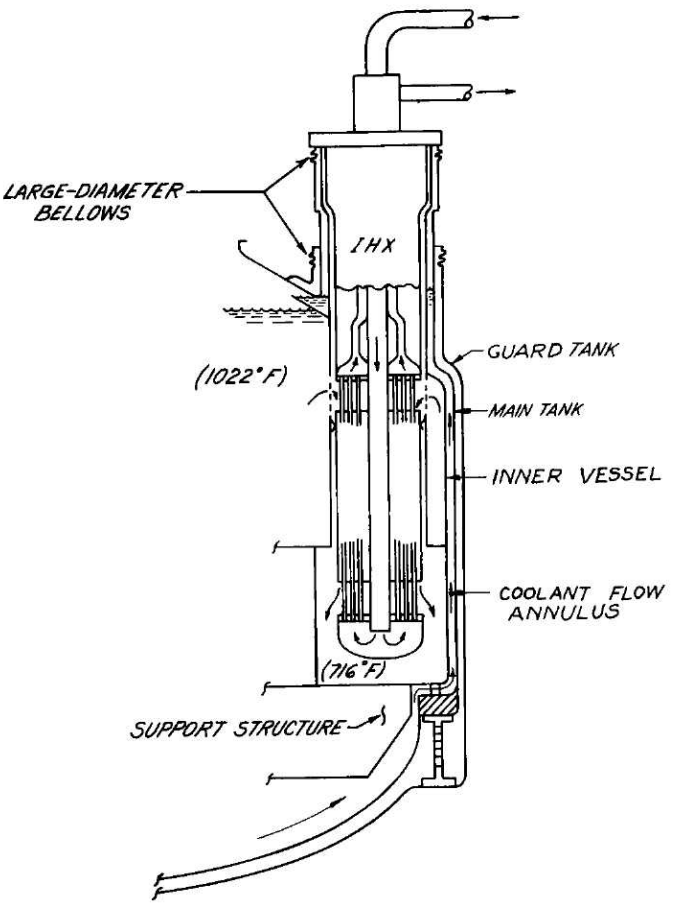


Fig. VI.17. Coolant Flow in BN-600 Main Tank

The use of one pump and a closed duct between the exit of each of two IHXs introduces the design and operating considerations listed in Table VI.4.

The hydraulic resistance of the IHX may be increased substantially (two to five times) over other plenum concepts by using a lower pump NPSH. At an NPSH of 30 ft of sodium, for example, the available head for coolant flow from the outlet plenum to pump suction might be 15 ft of sodium as contrasted to the 2-6 ft of sodium that is available in other open-plenum concepts. The entire head of sodium is, of course, not available for the IHX, since a fraction must be allocated for other resistances in this part of the coolant system, such as the inlet to the IHX, the IHX outlet duct, etc.

TABLE VI.4. Key Design Considerations for the Open-plenum Concept with Separated Coolant Loops at Reactor Inlet

Larger hydraulic resistance of primary side of IHX can be better accommodated than in those concepts having both pump-suction and outlet-plenum regions with free liquid levels.
Valve at pump suction serves to closs off back-flow through the two connected IHXs when associated secondary loop is out of service; no additional valve needed at IHX inlet as in other concepts.
Reactor inlet exposed to transients resulting from off-normal conditions of a secondary loop.
Natural-convection cooling of reactor cannot accommodate complete loss of all secondary-system heat-transport capability.

The larger available head across the IHX provides greater design flexibility, which can lead to a smaller-diameter unit and/or improved coolant flow distribution.

The use of a single valve which doubles both as the usual pump check and as a stop valve to isolate the system is an economic plus--no IHX valves are required. This arrangement reduces the flexibility of removing a pump from service and readily maintaining all IHXs in service. For example, the open-plenum systems such as those in Phenix and PFR could include an extra pump if pump outages were so frequent as to make that approach economic; or with one pump out, the additional capacity built into the remaining pumps could be used to operate all secondary systems at the maximum stretch capacity of the remaining pumps. Connecting the IHXs and pumps, as in BN-600, requires additional temporary equipment to obtain this flexibility.

The small sodium volume in the duct between the outlet of the IHX and the pump suction makes this concept susceptible to IHX temperature transients. Such a transient could occur from a sudden loss of secondary heat-transport capability (such as sudden drainage of the sodium system due to a water leak in the steam generator) coupled with failure to reduce flow through the reactor or, more specifically, failure to stop flow in the offended primary circuit. Such transients could be deleterious to the inlet-plenum structure.

Ideally, decay heat removal from the primary system should be, in the last resort, made independent of the secondary and steam systems. To do so requires that complete loss of these heat transport systems be tolerable. The concept under discussion is particularly sensitive to the loss of all secondary systems, because the volume of sodium between the outlet of the IHX and the inlet to the reactor is small. Failure to remove heat from all IHXs will heat the sodium in this region quickly and, thus, destroy the thermal head needed for natural-convection cooling of the reactor. It may be possible to avoid this problem by placing auxiliary cooling coils near the inlet to the IHX, such as in PFR. To be effective these auxiliary coils must operate continuously at a sufficient capacity to assure that the thermal center of the IHX always remains acceptably high.

The concept under discussion appears to avoid the free-convection, thermal-stratification, and hot-coolant-leakage problems associated with the PFR and Phenix open-plenum concepts. Insulation of the IHX outlet ducts are a concern for reasons discussed previously. In this instance, the complexity of the ducts raises concern for differential expansion and thermal stresses. Failures that allow only small leakages to occur would not necessarily be a serious safety problem unless the fault propagated and caused more serious failures resulting, for example, from impingement of the hot fluid on other structures within the duct region, such as the pumps or the reactor support core.

The main tank may be susceptible to loss of cooling from failure of all pumps. Under these conditions the upper portion of the main tank would increase in temperature, approaching the temperatures of the sodium in the upper plenum. Considering that differences may exist in thermal response of the various parts of the tank (such as the wall, IHX and pump nozzles, and rotating-plug nozzle), it seems possible that high stresses could result. The temperatures that may exist in the conical upper portion of the main tank are deserving of investigation. The conical section appears to be cooled at least partially by the main-tank-wall coolant system. Details of this coolant system are not known. For example, how much of the conical section is cooled; what is the temperature distribution, and what are the related thermal stresses in the section; how does it respond to thermal transients; what is the path of the sodium that cools the main tank, and how does it flow into and mix with the hot sodium in the outlet-plenum region?

Because a larger fraction of the sodium in the tank exists at reactor outlet temperature, the thermal inertia of the sodium in the main tank is less than that in other concepts previously discussed.

4. Concept Variations and Miscellaneous Considerations

a. Concept Variations

Numerous variations of the current closed- and open-plenum designs are discussed in the preceding Sec. VI.C.1-VI.C.3. Many have been considered and not developed further for one reason or another. Some of these design concepts are illustrated and discussed briefly in this section.

During the development of the SuperPhenix concept, the French considered a number of variations of the Phenix design. One important area of consideration was the arrangement of the outlet plenum and related pumps and IHXs. The numerous design considerations related to this part of the coolant system were discussed in Sec. VI.C.1. The decisions to eliminate the hot main-tank cover and to bolt the pumps and IHXs to the cold shield deck certainly must have been strong motivating factors in the search for improved designs to mitigate the inner vessel/pump/IHX interface problems. Some of this work is highlighted in Reference 18 and illustrated in Figs. VI.18-VI.21.

Figure VI.18 shows a combined pump-IHX concept. This approach is not novel and has been suggested by others, both here and abroad. This approach reduces the number of large equipment openings in the shield deck (by eliminating the pump opening) and increases the diameter of these openings. It is possible for the latter to have a detrimental impact by increasing the diameter of the main tank, depending upon other constraints upon system arrangement. The use of a combined pump-IHX arrangement probably would lead to as many pumps as IHXs and to minimum impeller diameters, in order to minimize the diameter of the unit. The pump must be free to move

axially, independently of the surrounding IHX. The pump should be removable with the IHX in place. A serious disadvantage of this approach is the exposure of the pump to all coolant temperature transients at the exit of the IHX. Little mitigation of these transients could be expected in the short distance from the IHX exit to the pump inlet. Moreover, even the normal maldistribution of temperature in the IHX exit coolant stream would be imposed on the pump--a highly undesirable situation, since significant differences in temperature can exist. (See Sec. VI.C.1.) A suitable location for the upper pump bearing and shaft seal may be a problem, considering the competition between these components and IHX secondary outlet nozzles for space at shield deck level. Increased shaft length and reduced availability of the bearing/seal area for maintenance may occur. Exposure of the main tank to 1000°F sodium and the related system transients are serious disadvantages.

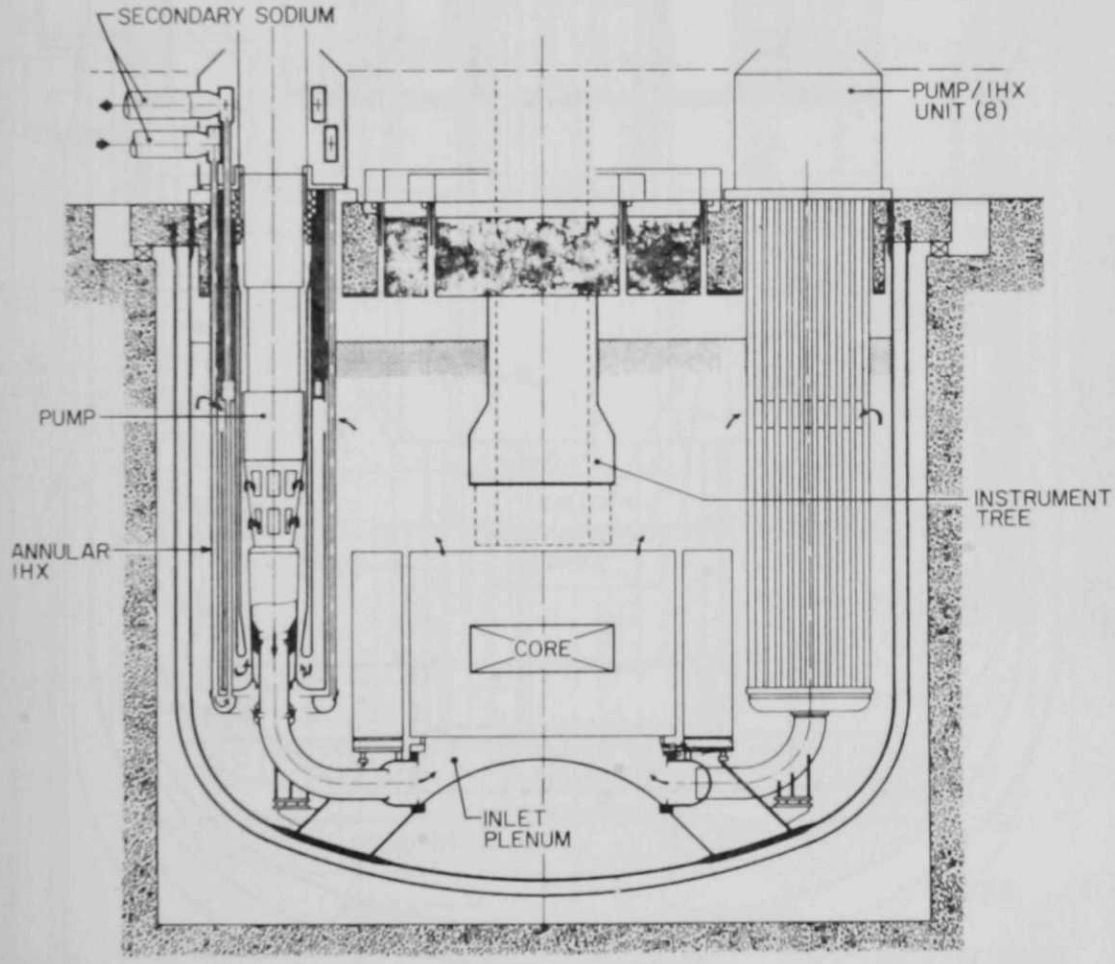


Fig. VI.18. SuperPhenix Concept with Combined Pump-IHX[18] (courtesy of Centrale des Revues)

Figure VI.19 shows the approach that was eventually chosen for SuperPhenix. It parallels the Phenix concept insofar as the inner vessel, pumps, and IHXs are concerned. The various aspects of this arrangement are discussed in Sec. VI.C.1.

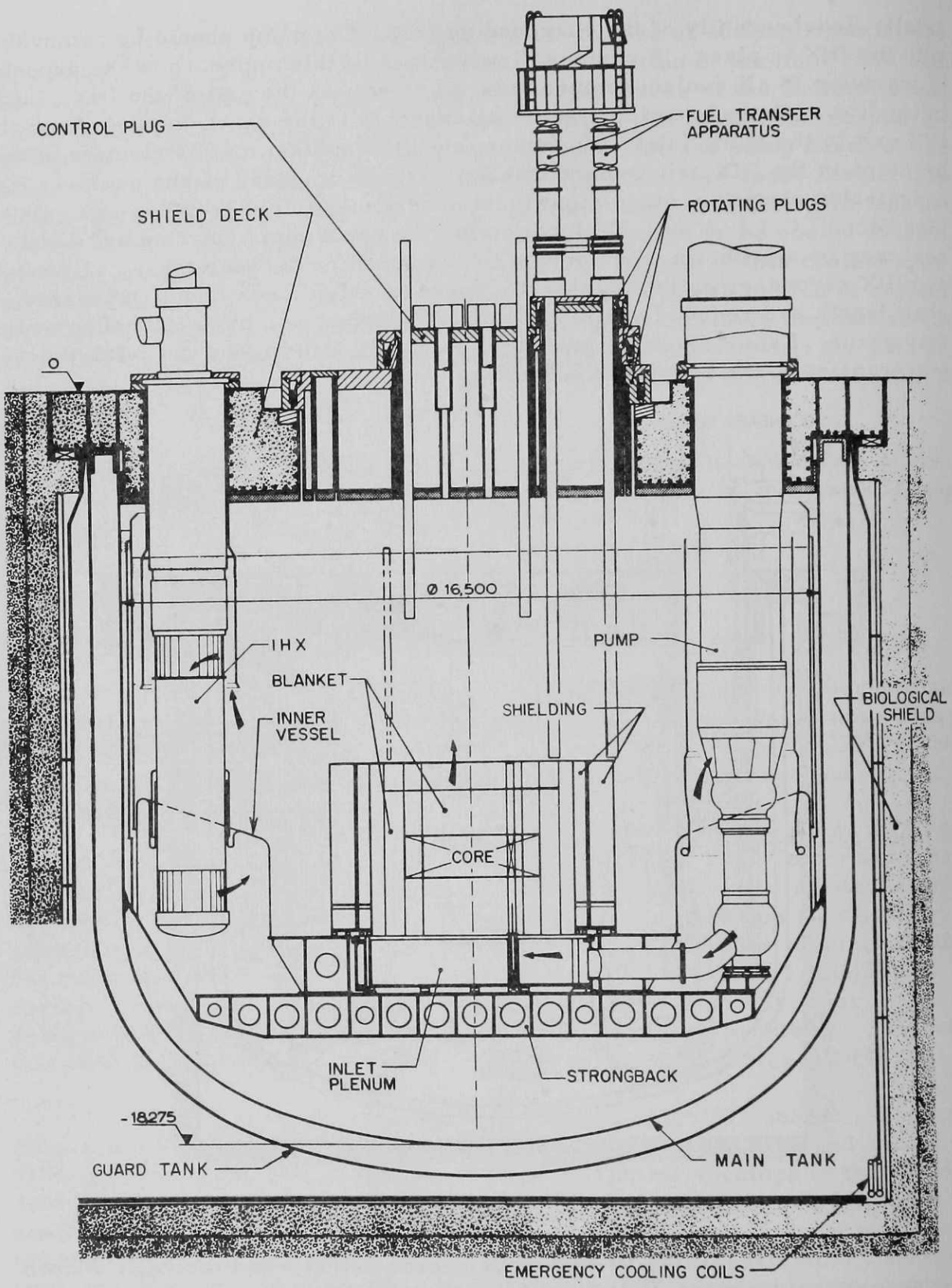


Fig. VI.19. SuperPhenix Concept with Phenix-type Inner Vessel[18] (courtesy of Centrale des Revues)

Figure VI.20 depicts an open-plenum concept with both the IHXs and the pumps external to the inner vessel. Each IHX is located within

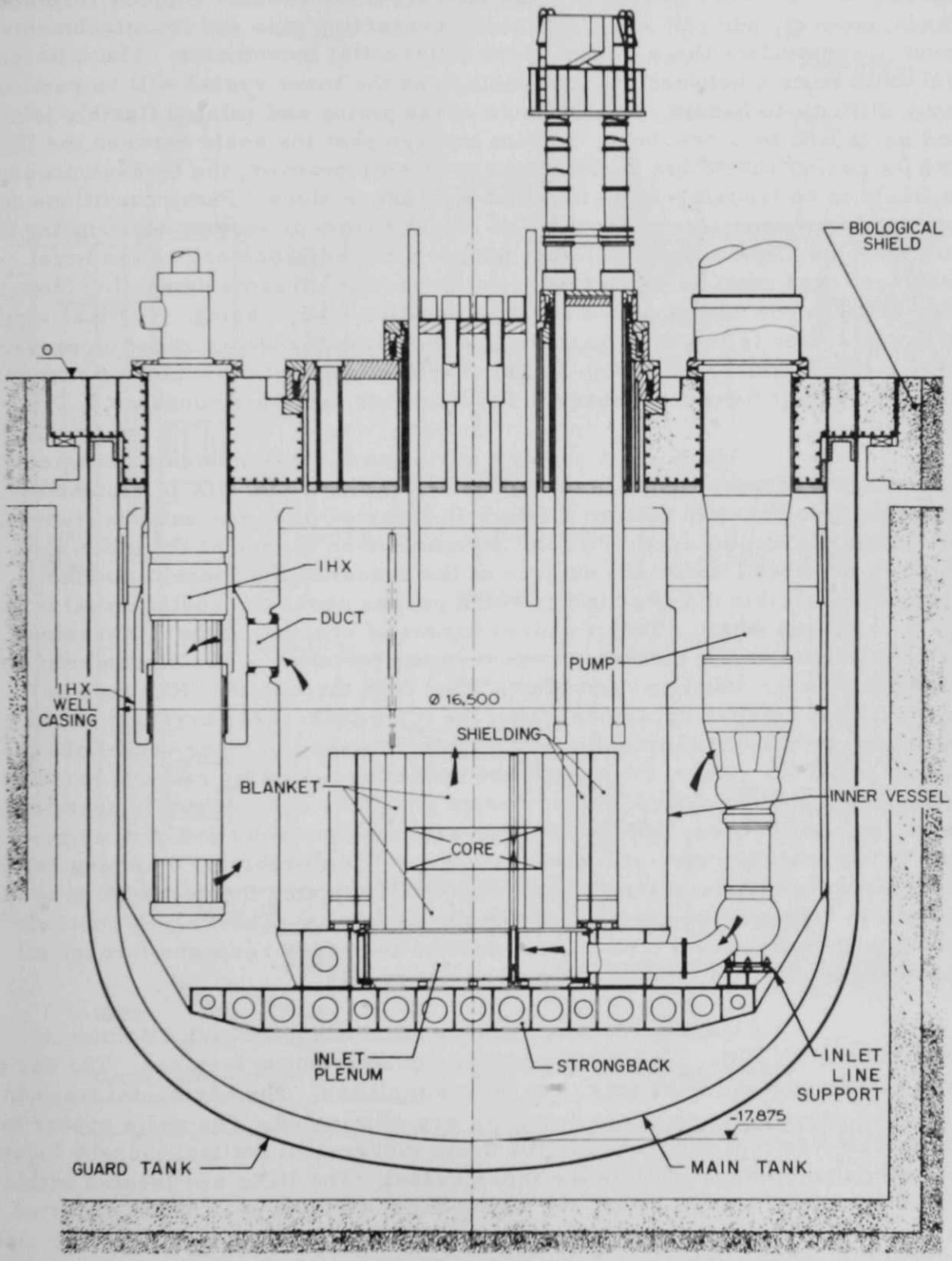


Fig. VI.20. SuperPhenix Concept with Mechanical Connection between IHX and Inner Vessel[18] (courtesy of Centrale des Revues)

a separate casing which is connected to the inner vessel with a short, large-diameter pipe. The IHX casing is attached to the cold shield deck, and expands downward. The inner vessel is supported from the reactor support core and expands upwardly and radially. The short connecting pipe and its attachments must accommodate these rather large differential movements. The differential axial motion between the IHX casing and the inner vessel will be particularly difficult to handle. Maintenance of the piping and related flexible joints and seals will be a problem. Coolant leakage past the seals between the IHX and its casing should not be a serious problem; however, the leakage problem is likely to be transferred to the inlet pipe connections. Pump conditions appear to improve in this concept. The large volume of sodium surrounding the pump should allow a more uniform temperature environment. Even here, however, care must be taken that free-convection streams emanating from the side of the inner vessel do not impinge upon the pump casing. Vertical support of the inlet line is required to withstand the pump discharge reaction pressure. This support is evident in Fig. VI.20. Vertical support must permit some differential radial movement between the inlet line and the strongback.

Figure VI.21 shows a variation that eliminates mechanical interactions between the IHXs and the inner vessel. The IHX is coupled hydraulically to the open plenum through an inverted pipe that extends from the top of the IHX casing (primary inlet plenum), over the top of the inner vessel, and to a point well under the surface of the lowest sodium level. Sodium is drawn through this dip-leg piping, which passes above the sodium level in the tank, by siphon action. This requires means of evacuating the IHX-casing volume to initiate the siphon and of assuming retention of the siphon action--especially under abnormal conditions when flow through the IHXs may be necessary to remove decay heat from the main tank. This arrangement is attractive because it eliminates interaction between the inner vessel and the pumps and IHXs. Thus, the design and operating activities relative to these components are decoupled, easing design problems and mitigating operating concerns. In addition, IHX-penetration seal requirements are greatly reduced, and leakage rates past these seals are not a problem. Pumping rates and system head losses are reduced appropriately, and the degradation of hot sodium by leakage into cold sodium is not a concern. The French conclude[18] that this system merits further examination to assess response through all operating modes and to study related safety issues.

Figure VI.22 is a cross section of the British Commercial Fast Reactor (CFR). There are a number of interesting features. The design of the inner vessel (inner tank) has been simplified. The horizontal trays in PFR that provide penetrations for IHXs are eliminated. The walls appear to be entirely vertical and to retain the basic clover-leaf outline, thereby locating the pumps completely outside the inner vessel. The IHXs are located within the inner vessel and penetrate the strongback, which serves as an insulated bottom to the inner vessel. The penetration appears as a round, tapered pipe or duct. A round extension of the IHX primary outlet plenum extends into the penetration. This exit pipe from the IHX can accommodate an exit valve, which,

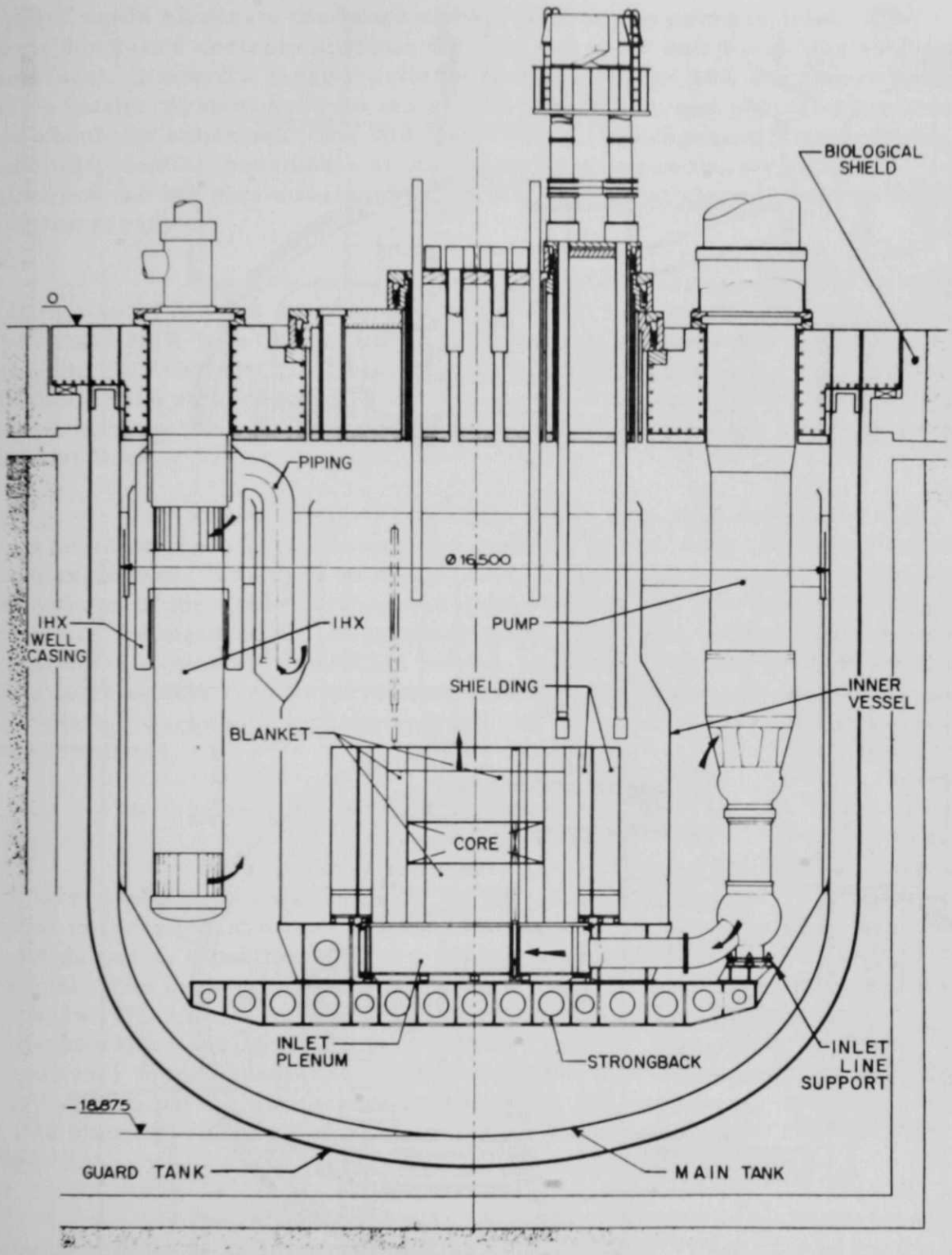
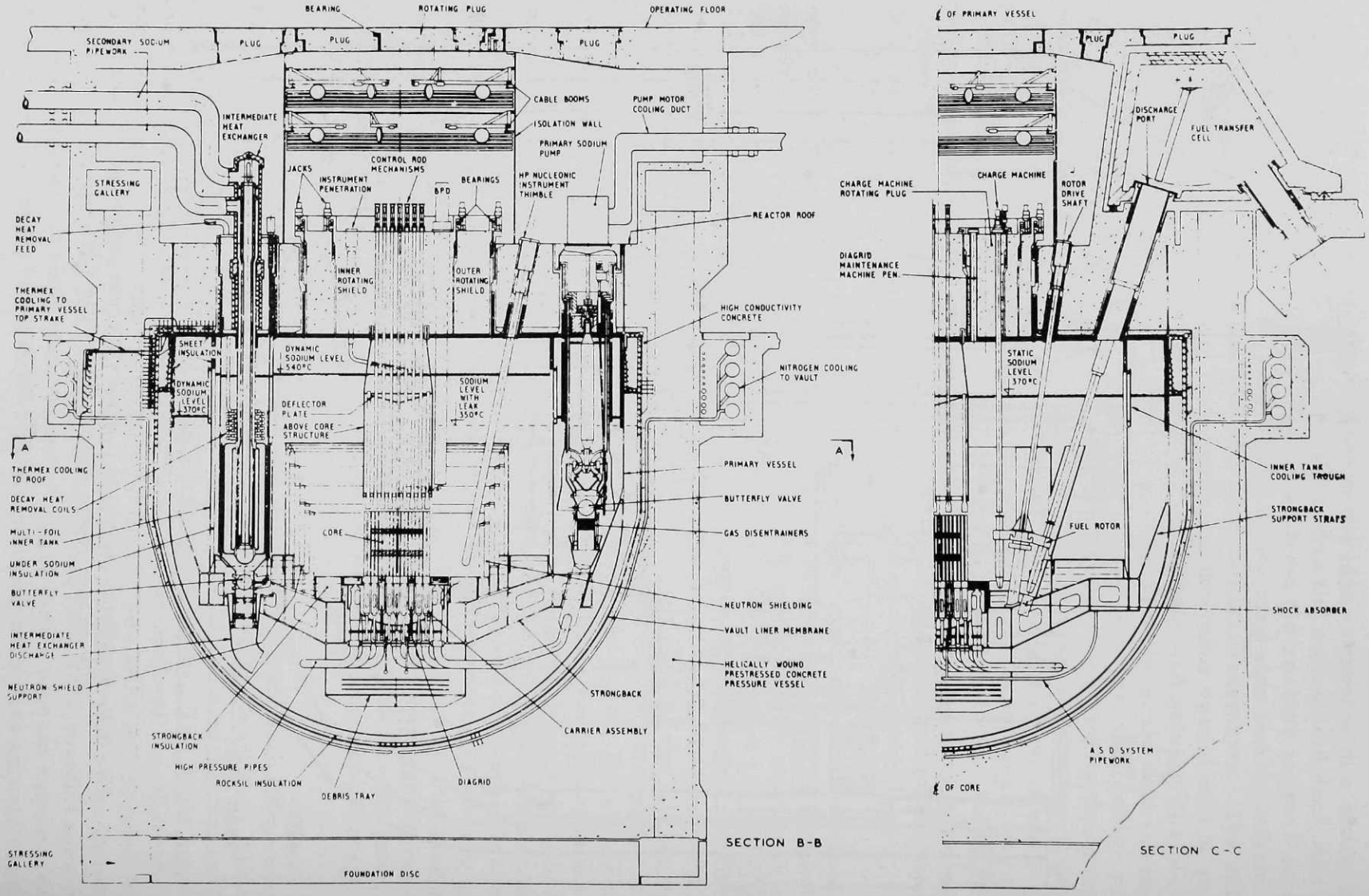


Fig. VI.21. SuperPhenix Concept with Hydraulic Connection Only between IHX and Inner Vessel[18] (courtesy of Centrale des Revues)



(A) CROSS SECTION OF REACTOR

(B) ACTIVE FUEL HANDLING SYSTEM

Fig. VI.22. Elevation View of CFR Concept[19] (courtesy of Nuclear Power Company Limited)

if used, would eliminate the large sleeve valve at the primary inlet. The direct downward coolant exit from the IHX has some merit over the PFR arrangement, in which a large fraction of the flow turned 180° and flowed upward over a baffle. System head losses should be reduced, and natural-convection flow should be enhanced. The IHX/inner-vessel arrangement also reduces radial differential movements at the penetration, since the strongback is insulated on the hot side and, thereby, should operate at close to the reactor inlet temperature.

The CFR pump/reactor-inlet piping arrangement is different than that in PFR. The CFR version (Fig. VI.22) is basically the kind used in Phenix and EBR-II, with the pump and its casing totally removable and connected to the inlet line via a slip joint using piston rings. This joint appears attached to the strongback, and allowances must be provided for differential motion between the pump and the strongback, and between the strongback and the inlet lines.

Shown in Fig. VI.22 is the large drop in sodium level that must be accommodated in these large systems in the event that a leak occurs in the main tank. The drop in liquid level is, of course, a function of the relative volume of the space between the main tank and the guard tank, and the volume of the main tank. Large space allowances must be made for periodic inspection and seismic and HCDA events, so the annulus volume between the tanks is relatively large. Notice that under the adverse conditions of a leak, and with a reduced sodium temperature, the primary inlet to the IHX must remain flooded.

b. Miscellaneous Considerations

The number of secondary loops, IHXs per secondary loop, and primary pumps are design options. In particular, the number of primary pumps is independent of the number of secondary loops if the outlet of an IHX is not ducted or piped directly to a particular pump (only BN-600 has this feature). The arrangements that are most common are (1) three secondary loops, two IHXs per loop, and three primary pumps; and (2) four secondary loops, two IHXs per loop, and four primary pumps. The merits of three versus four loops depend upon value judgments that include not only the primary system but the entire plant; these value judgments are outside the scope of this study. Typical three- and four-loop arrangements are illustrated in Fig. VI.23.

Operation of the plant at part load with one or more secondary loops shut down is an important design consideration. Under these conditions the primary system must be capable of operating satisfactorily. Reasonably uniform temperature distributions must be maintained within the system. In the pump-suction region of the main tank the operator has flexibility to decide whether to operate all pumps at part capacity or to operate fewer pumps at full capacity. No such flexibility exists for the outlet-plenum region and IHX operation. The design must accommodate the maldistribution of flow related to part-load operation.

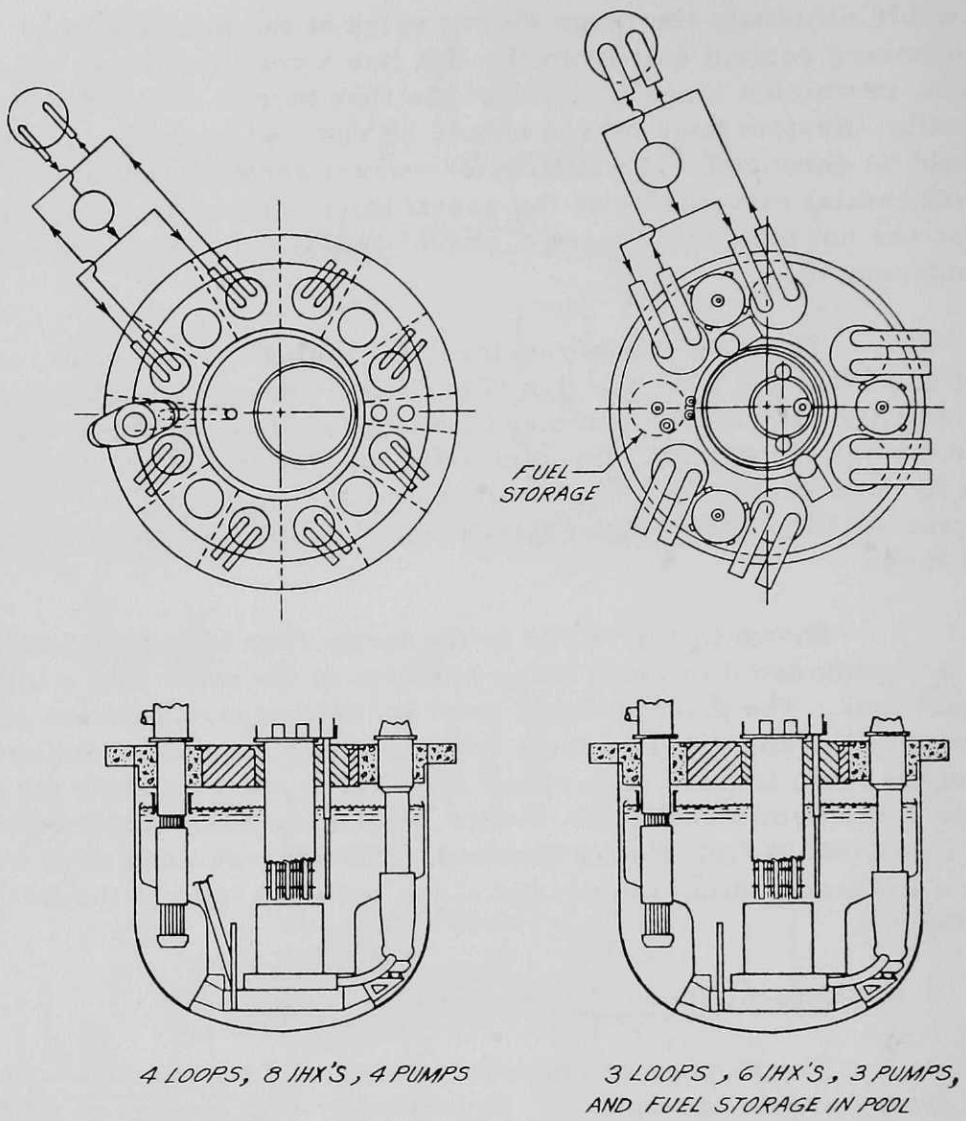


Fig. VI.23. Three- and Four-loop Primary-system Arrangements

Part-load operation requires, of course, an isolation valve for each IHX. A number of approaches are possible, as illustrated in Fig. VI.24. Figure VI.24a shows the method used in Phenix and PFR, and which is applicable to an EBR-II-type IHX. This valve is a large cylindrical gate that can be lowered to isolate the primary inlet to the IHX. The valve has a large seat area and may have fairly large leakage rates. Experience to date in both Phenix and PFR indicates that this approach is satisfactory. Figure VI.24b shows an alternate location for an inlet valve, providing that a (piped) closed-outlet-plenum arrangement is used as in EBR-II and that the outlet piping is removable. A more conventional valve is possible, with lower leakage rates. Figures VI.24c and VI.24d illustrate valves at the exit of the IHX. Figure VI.24c shows the BN-600 approach. Here the outlet of the IHX is ducted directly to a pump. Figure VI.24d is the CFR concept (shown also in Fig. VI.22), which can employ a rather conventional butterfly-type valve in the primary outlet of the IHX. The length of the actuating rod may be a minor problem.

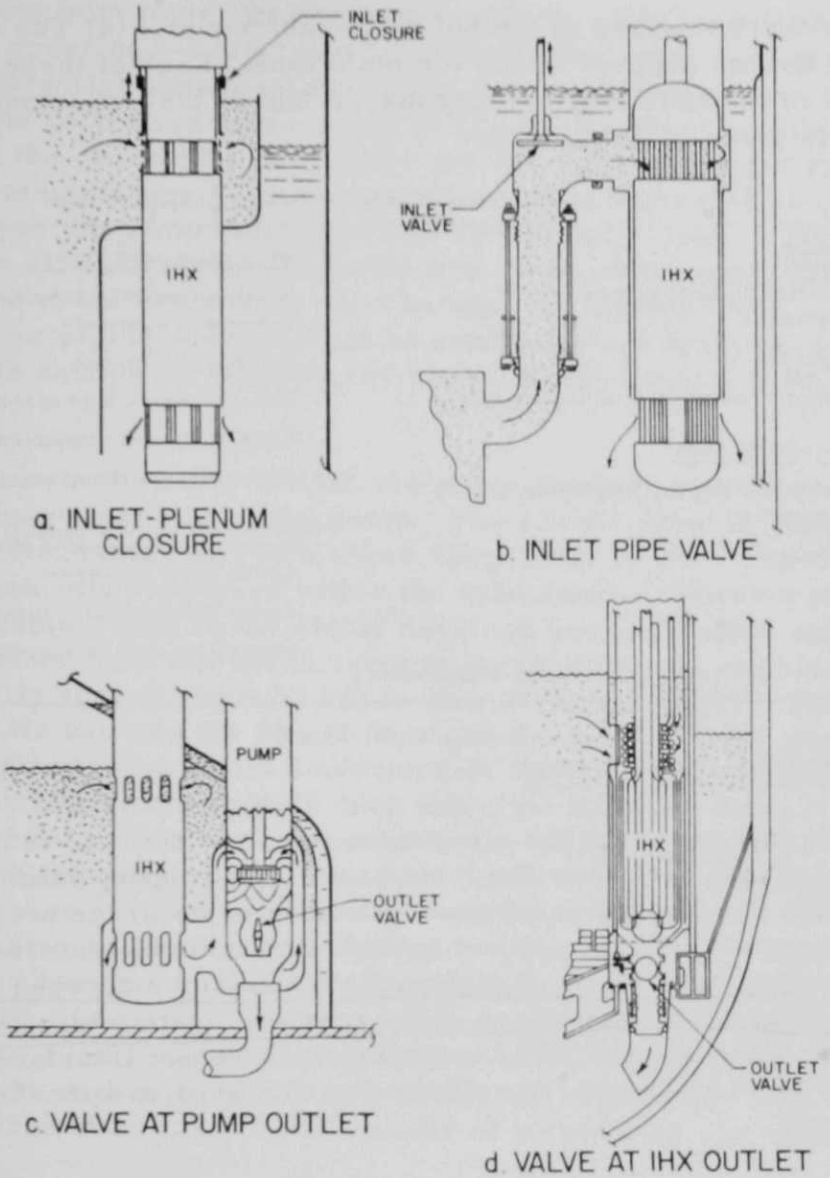


Fig. VI.24. Methods of Isolating IHX for Part-load Operation

D. Support of Primary Pumps and IHXs

The primary pumps and intermediate heat exchangers typically weigh 125 to 200 tons and have a diameter of 7-10 ft with an overall length of 60-80 ft. These units interface with both hot and cold structures. The nature and severity of these interface problems will be influenced greatly by the choice of component support.

Four methods of component support have been used to date: (1) The units are bolted directly to the cold shield deck over the main tank. (2) The weight of the units is supported by the cold shield deck, and there is lateral movement capability relative to the shield deck. (3) The units are fixed to

nozzles that carry the load to the hot main-tank cover. (4) The load is carried to a bottom support within the main tank. Each of these approaches is discussed in the following paragraphs. A list of the important design considerations is given in Table VI.5.

TABLE VI.5. Key Component-support Design Considerations

Fixed to shield deck	Fixed to main-tank cover
Large differential thermal expansions between components	Small differential thermal expansion between components
Seismically secure	Seismically secure
Little movement of secondary piping nozzles	IHX secondary pipe nozzles move with tank
Moves relative to shield deck	Main-tank cover required
Reduced differential thermal expansion between components	No bellows in cover nozzles
Seismically poor	Fixed to tank bottom-support structure
IHX secondary pipe nozzles move with tank	Large axial expansion
Main-tank cover required	Large bellows in main-tank nozzles
Large bellows in tank-cover nozzles for components	Seismicity may be a problem

1. Shield-deck Support

In this approach the component is bolted directly to the cold (room-temperature) shield deck over the main tank. The support point of the component remains fixed in space whereas all other parts of the primary system and the component itself are subject to high system temperatures and expand accordingly. Accommodation of differential expansions between the component and the interfacing structures of the primary system is a key engineering concern. The pump and the IHX have uniquely different interface problems which are, in turn, influenced greatly by the choice of an open- or closed-plenum concept.

a. Open-plenum Concept

First, consider the open-plenum design. Both the pump and the IHX are fixed in place radially; only axial expansion of the units proper is important. This approach is used in PFR and is planned for use in Super-Phenix. The normal differential axial movement between the pump discharge line and the mating connection with the reactor inlet-plenum piping is small, since the main tank and the pump expand downward about the same amount; both are exposed to the reactor inlet-temperature sodium. The design, however, must provide for the possibility of excessive temperatures in the system and mismatches in transient response characteristics of the interfacing structures.

Significant radial differential expansion will occur, since the centerline of the pump is fixed by the cold (room-temperature) shield deck,

and the reactor inlet piping grows in response to a 720°F environment; movements of the order of 2-2.5 in. can be expected for large systems. The magnitude of differential movement that must be accommodated can be reduced by locating the centerline of the pump in-line with a "hot" position of the centerline of the connecting inlet pipe--for example, at a location determined by the mean temperature between operation and refueling. The mating connection between the pump discharge line and the inlet-plenum piping must accommodate all such movements with little load on the pump discharge nozzle. Leakage rates at the connection must be kept reasonably small, and galling of the flex joint or piping support must be avoided, since maintenance of the inlet-plenum line is extremely difficult and should not be counted upon as a design basis.

In the case of PFR, the pump discharge piping is permanently attached to the well casing of the pump. The coolant from the pump flows through four discharge lines to a check valve (Fig. VI.25). The check-valve body is permanently supported within the main tank. The valve internals are attached to a shield plug in the shield deck and are removable. Four lines carry the coolant from the check valve to locations in the reactor inlet plenum that are equally spaced around a 90° section of the periphery. Thus, a fixed structure exists between the shield deck and the reactor inlet plenum. The flexibility of this system must accommodate the differential expansions resulting mainly from a cold shield deck and a hot system. Normally the PFR piping sees 752°F sodium but can possibly see higher temperatures. The piping between the pump casing and check valve must not load the casing laterally, because small clearances exist between the pump and the well casing in the area of the discharge diffuser. The use of four small lines reflects the need for piping flexibility in this part of the system. The use of multiple, small lines from the check valve to the inlet plenum has a number of advantages. Flexibility is provided, good flow distribution is maintained to the inlet plenum with a pump out of service, and a guillotine-type failure of the pipe has less of an impact if a number of small lines are employed.

The PFR approach, as shown in Fig. VI.25, requires close tolerances between the removable impeller assembly and the permanently installed casing and volute. This may remove a design flexibility normally included in a pump--the ability to initially oversize the impeller so adjustment of the impeller diameter can be made later to meet as-built system head requirements.

A concern with the PFR (and Phenix) open-plenum concepts is the possibility that the hot sodium in the outlet plenum might, in some unlikely way, be mixed suddenly with the cold sodium, causing the cold-sodium-region temperature to increase rapidly. A mismatch in response to this situation must be accommodated by differential movement between the pump line and plenum piping at the flex joint.

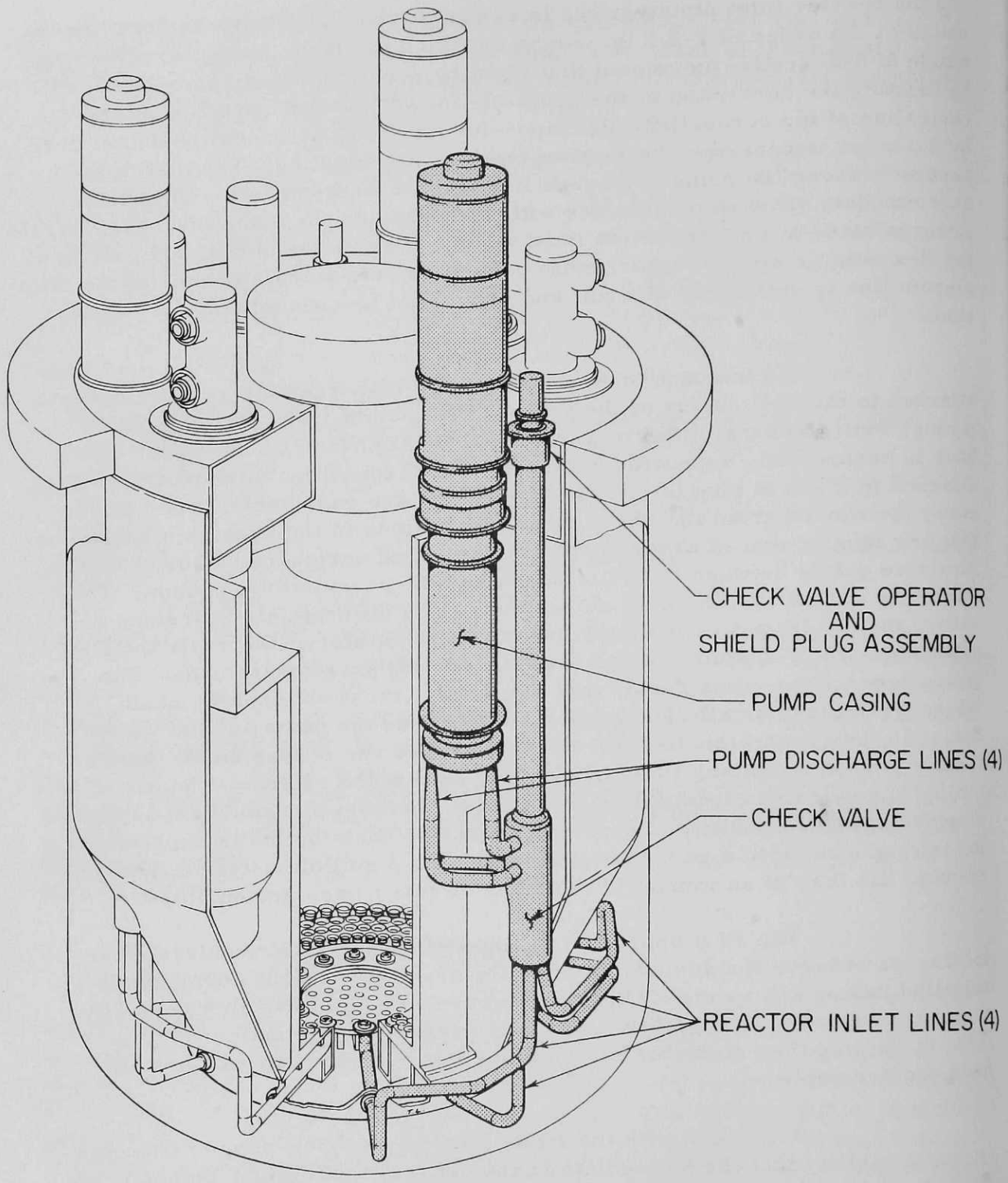


Fig. VI.25. PFR Pump/Check Valve/Inlet Piping Assembly

Continuing with the open-plenum concept, the IHX interfaces with the inner vessel as shown in Fig. VI.8 for SuperPhenix and Fig. VI.12 for PFR. The IHX will expand axially and remain fixed radially by virtue of being bolted to the room-temperature shield deck. The inner vessel will grow axially and radially. A large differential radial movement (3 in. or more for large systems such as SuperPhenix) is likely between the IHX and the inner vessel during normal operating conditions. As with the pump, the design must accommodate system temperatures greater than normal. For example, the outlet-plenum sodium temperature can increase due to abnormal operations such as might occur under failure of all pumps, and this increase would cause higher inner-vessel metal temperatures and greater differential movements.

The secure attachment of the heavy pump and IHX units to the shield deck mitigates seismic-design problems. The support point for these units is subject to modest accelerations, since building accelerations at this elevation experience only small amplification of the ground motion. The components are, or can be, encased in a cylindrical well structure with modest clearances, say 0.5-1 in. Below the sodium surface within the tank these clearances normally are filled with sodium and, as a consequence (as noted in Sec. IV), the well structure may provide significant lateral seismic support.

The secondary-system inlet and outlet nozzles to the IHX will move very little. This is desirable, especially if the secondary piping penetrates a containment barrier close to the IHX.

b. Closed-plenum Concept

The closed-plenum concept has somewhat different design considerations. Support of the pumps and IHXs from the shield deck probably excludes the use of a separate hot cover for the main tank as in EBR-II. The reason, of course, is that large (2 in. or greater) radial motions between the tank cover and the component support would occur. Accommodation of motions this large would undoubtedly require a leaktight flexible device (bellows) having diametral dimensions somewhat larger than the diameter of the IHX shield plug.

The pump design considerations are similar to those of the open-plenum concept except that heat-conduction and radiation effects on the pump casing are less of a problem.

Design of the outlet piping and/or the IHX well casing is extremely difficult. With the centerline of the IHX fixed in space the relative radial growth of the reactor, reactor-outlet plenum, outlet piping, and IHX plenum as compared to the cold shield deck is 3 in. or greater. Accommodating this movement by fixed piping will impose severe design and economic penalties on the system. Even the use of a removable pipe section, as discussed earlier, may not be feasible. One alternative might be to permit radial

movement via a piping concept illustrated in Fig. VI.15d. Another alternative is to accommodate radial motion between the IHX and its well casing, as suggested in Fig. VI.15f.

2. Shield-deck Support with Lateral Movement Capability

This type of support is used in Phenix. (Interestingly, it is not being employed in SuperPhenix.) The component is supported on the cold shield deck on a device that permits lateral (radial) motion relative to the shield deck. This approach uses a main-tank cover with equipment nozzles that extend into the upper area of the shield deck, as shown in Fig. VI.26. Large-diameter bellows are required in the nozzles to accommodate the large (3 in. or greater for 1200-MWe-size systems) relative motion between the main-tank cover and the shield deck. These nozzles do not carry the weight of the component and serve only to provide a cover-gas seal. Expansion of the cover as transmitted to the components is accommodated by lateral (radial) movement of the component on the support devices. Figure VI.27 illustrates this technique as used in Phenix.

Any outward radial motion of the IHXs relative to the cold shield deck eases the differential expansion between the IHXs and the inner vessel.

Outward radial motion of the pumps likewise eases the differential expansion at the flex joint between each pump and its line to the reactor inlet plenum--up to a point. Should the thermal growth of the roof cause the pumps to move radially more than the reactor inlet line at the flex-joint connection, then pump motion would over-compensate, creating a differential expansion of an opposite nature, i.e., the pump has greater radial motion than its connecting pipe.

Some of the axial and radial movements provided for the pump-discharge-line flex joint in Phenix are shown in Fig. VI.28. The 1382°F (750°C) condition is the temperature of the system if all forms of decay heat removal are lost except the capability to radiate heat from the main tank to the surrounding structures (clearly an extremely unlikely situation).

The large bellows in the shield deck area are a problem. The design of a suitable seismic support system is likely to be challenging. The pump and IHX units are unrestrained, space above the shield deck is limited, active seismic devices must be tested periodically, and, if within the tank, must be compatible with a sodium system. Finally, the secondary-system piping must accommodate the radial motion of the IHX without large reaction forces which may cause undesirable stresses on any special support devices.

3. Support by Main-tank Cover

Components can be supported on nozzles that carry the load to a hot main-tank cover. This concept, of course, requires a cover capable of

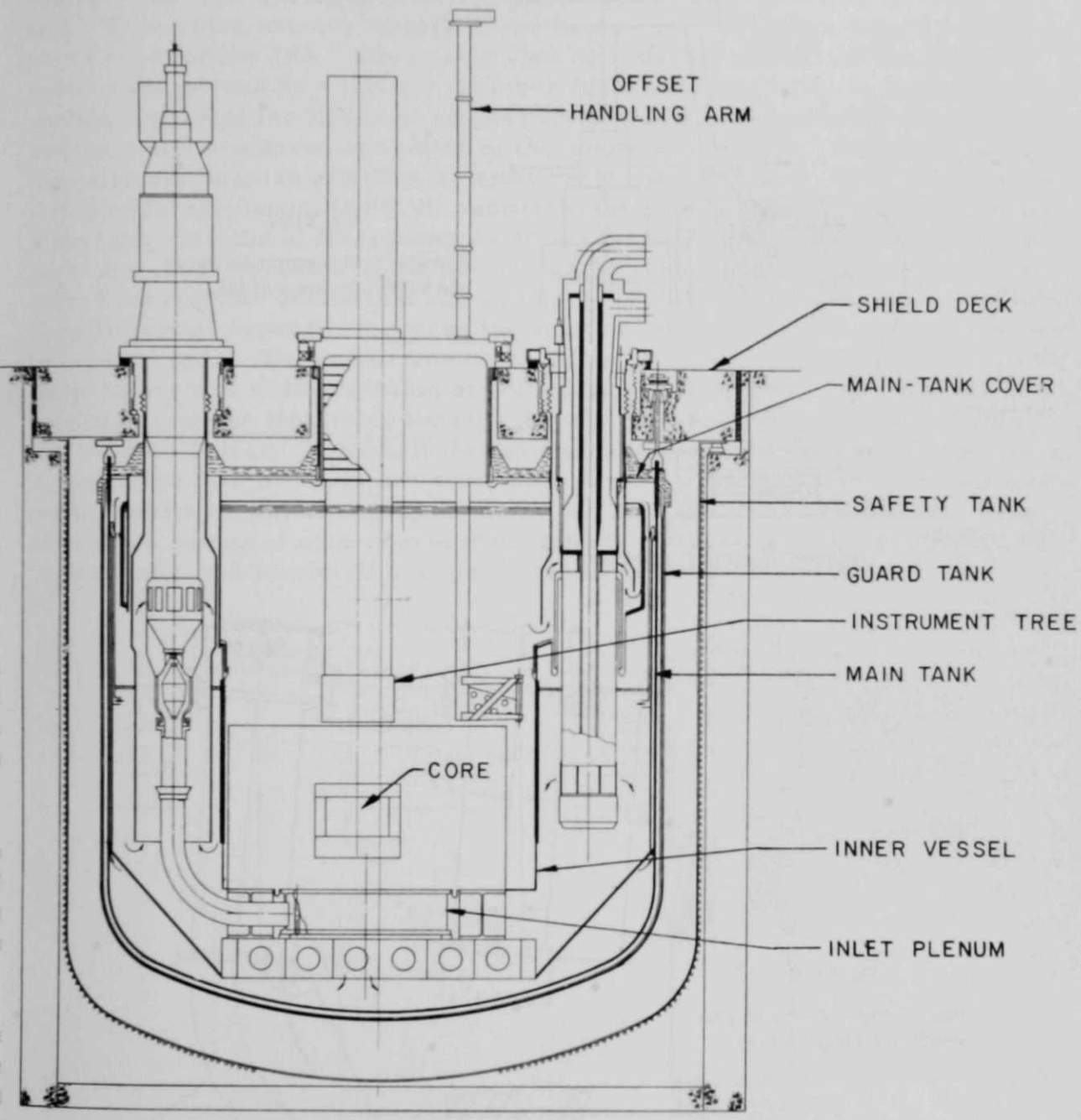


Fig. VI.26. Phenix Primary System

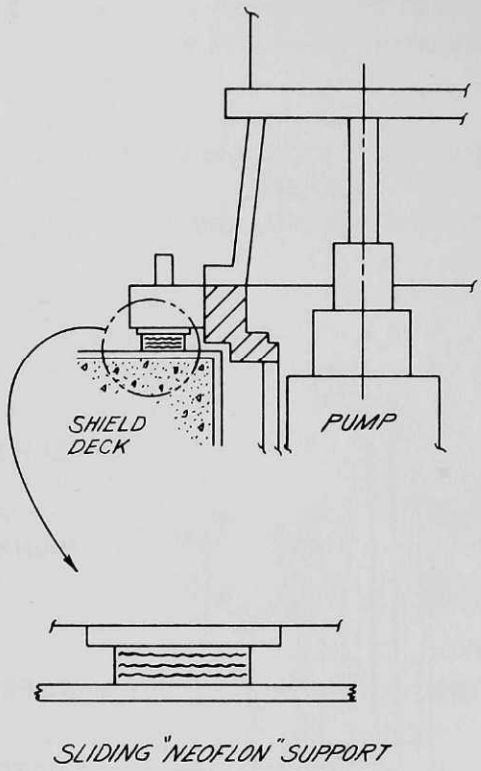


Fig. VI.27

Typical Phenix Support for Pumps and IHXs[20] (courtesy of IAEA)

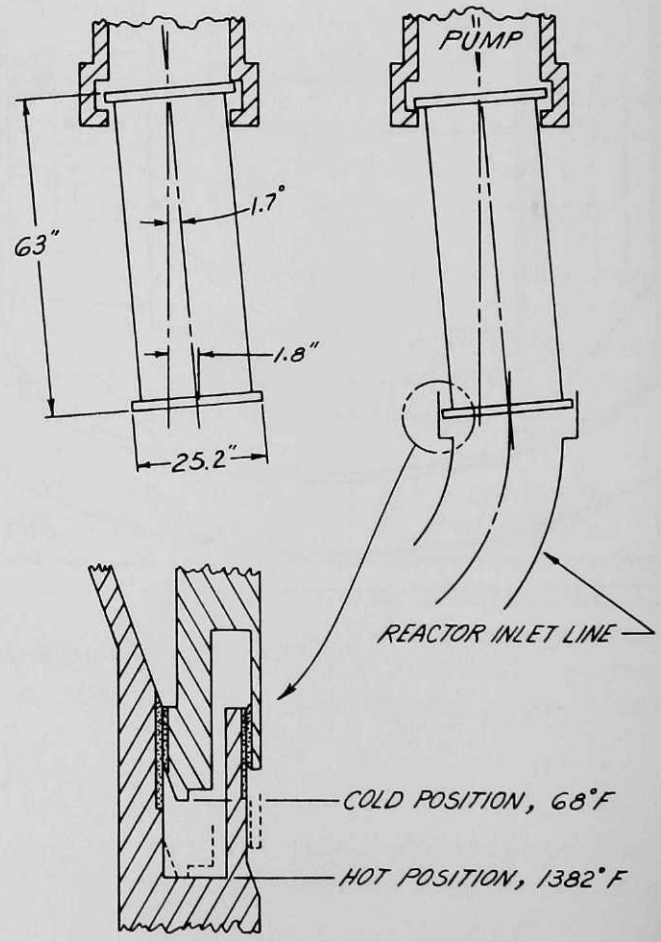


Fig. VI.28

Phenix Pump-outlet-pipe Flex Joint[20] (courtesy of IAEA)

carrying the load of these components. Figure VI.29 illustrates an arrangement of this type as employed for EBR-II, with and without the IHX in the well casing. The well casing is welded to and supported by the cover of the main tank. The casing extends upward from the cover, providing a nozzle and support flange for the IHX. The casing also extends downward into the primary sodium and serves as a plenum chamber for the IHX, a buffer to limit primary-coolant bypass of the IHX, and support for boron-steel shielding material to reduce neutron-induced activation of the secondary sodium. The well casing may also serve as an effective seismic restraint. The main-tank cover is hot. For the closed-plenum (EBR-II) approach, the cover as well as the rest of the main tank operates at the reactor inlet temperature; for the open-plenum concepts the cover operates at the reactor outlet temperature. Clearance is required between the component support nozzle and the shield deck to accommodate the differential expansion between the cold shield deck and the nozzles attached to the hot cover. These movements range up to 3.5 in., depending on the cover temperature and radial location of the component support nozzle. Shielding is needed to prevent streaming through these large annuli between the nozzles and the shield deck. In EBR-II this shielding is located in the main-tank cover. A controlled flow of air is provided through the clearance annulus of each component nozzle to maintain proper shield deck and nozzle temperatures. This requires a device at each nozzle that controls air access to the nozzle/shield deck annulus and can itself accommodate differential movements.

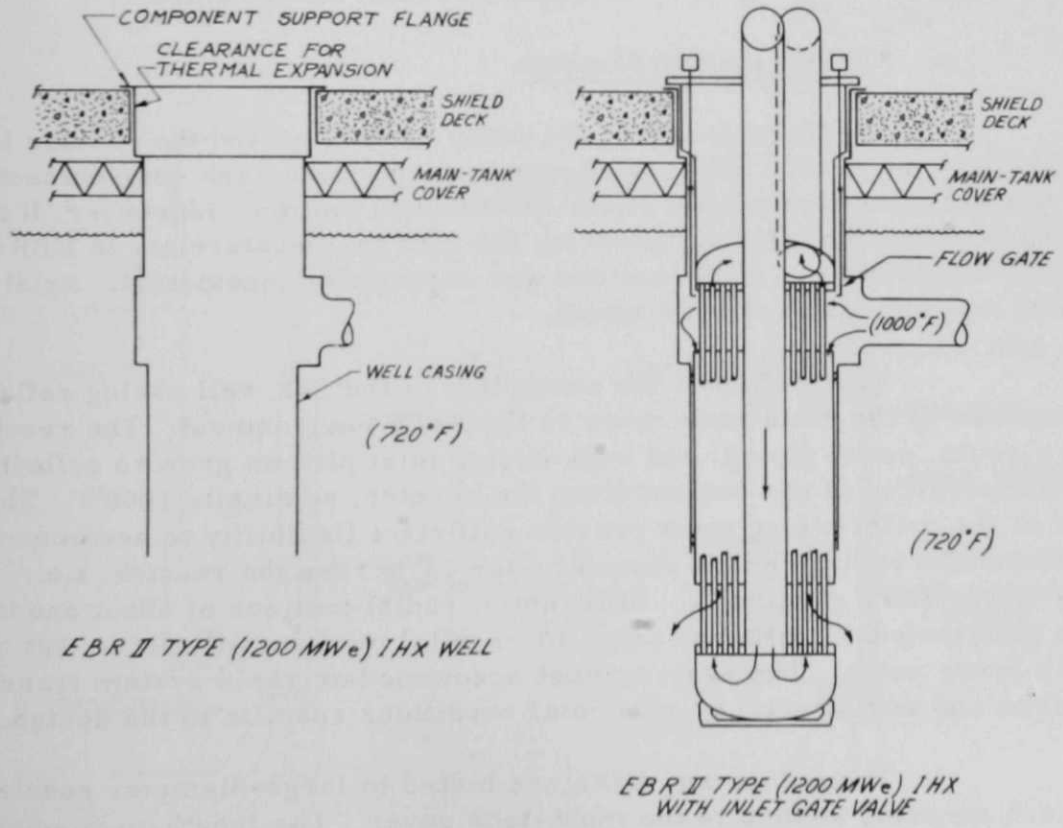


Fig. VI.29. EBR-II IHX Well (left) and EBR-II Well with IHX in Place (right)

The specific design factors will depend upon the concept that is chosen, e.g., an open- or closed-plenum type of primary coolant system.

a. Open-plenum Concept

The centerline of the pumps will move radially outward farther than the pipe to the inlet plenum, since the cover which supports the pump responds to the 1000°F outlet sodium and the inlet piping responds to the 720°F inlet-sodium environment. Differential radial movements in the range of one inch will occur from these operating conditions. Allowance for larger movements may be required, however, to reflect unusual and abnormal operations that are specific to the plant design.

Differential radial movement between the IHX and the penetration through the inner vessel is small, since nominally the inner vessel and the main-tank cover should each reflect the temperature of the sodium in the outlet plenum. The interface problems will be aggravated to the degree to which the inner-vessel and cover temperatures differ. Differences in the response of these two structures to transient sodium temperatures in the outlet plenum will cause relative motions at the IHX inner-vessel penetration. In particular, a sudden rise in plenum temperature, as illustrated in Fig. VI.14, could see a considerable lag in tank cover response; whereas the inner vessel probably responds rapidly to the transient sodium temperature.

b. Closed-plenum Concept

Bert

The movements of the pump centerline and the reactor inlet piping reflect the nearly isothermal nature of the main-tank environment and there would be relatively little radial differential motion. Moreover, if fuel handling is performed without lowering the tank temperature, as in EBR-II, transient-temperature considerations are essentially nonexistent. Axial differential motions similarly are small.

The motion of the centerline of the IHX well casing reflects the response of the main-tank cover to the 720°F environment. The reactor-outlet plenum, outlet piping, and well-casing inlet plenum grow to reflect the mean temperature of the coolant from the reactor, nominally 1000°F. The design of the outlet piping must provide sufficient flexibility to accommodate the differential radial motion caused by the ΔT across the reactor, i.e., 1000 - 720 = 280°F nominally. Differential radial motions of about one inch can be anticipated. Similarly, axial differential motions will occur, but of a slightly lower value. The system must accommodate rapid system transient conditions and any unusual or abnormal conditions specific to the design.

The pumps and IHXs are bolted to large-diameter nozzles which are securely welded to the main-tank cover. The length-to-diameter ratio (L/D) of the nozzle is one or less so the nozzle is structurally stiff.

Therefore, the nozzle support flange responds to a seismic occurrence in essentially the same fashion as the main-tank cover, i.e., the cover motion is not amplified much at the component support point. As noted previously, the well casing may serve to support the lower portion of the components, thereby providing seismic support.

The secondary IHX nozzles will move radially in response to the temperature of the main-tank cover. The secondary piping must not impose large lateral loads on the IHX; such loads could cause unbalanced lateral forces on the main tank and large stresses at the attachment of the IHX well to the main-tank cover. Axial motion of the nozzles probably will be small.

4. Bottom Support

The BN-600 components are bottom-supported. As illustrated in Fig. VI.16, the primary system is supported from the bottom on a ring girder (common support girder) at the outer circumference of the main tank. Within the main tank and supported by the common support girder is a strongback structure that supports the tank internals, pumps, and IHXs. As shown in Fig. VI.30, each component is supported at the shield-deck level on the flange of a cylindrical well that carries the load of the component to the lower support structure, located within the main tank.

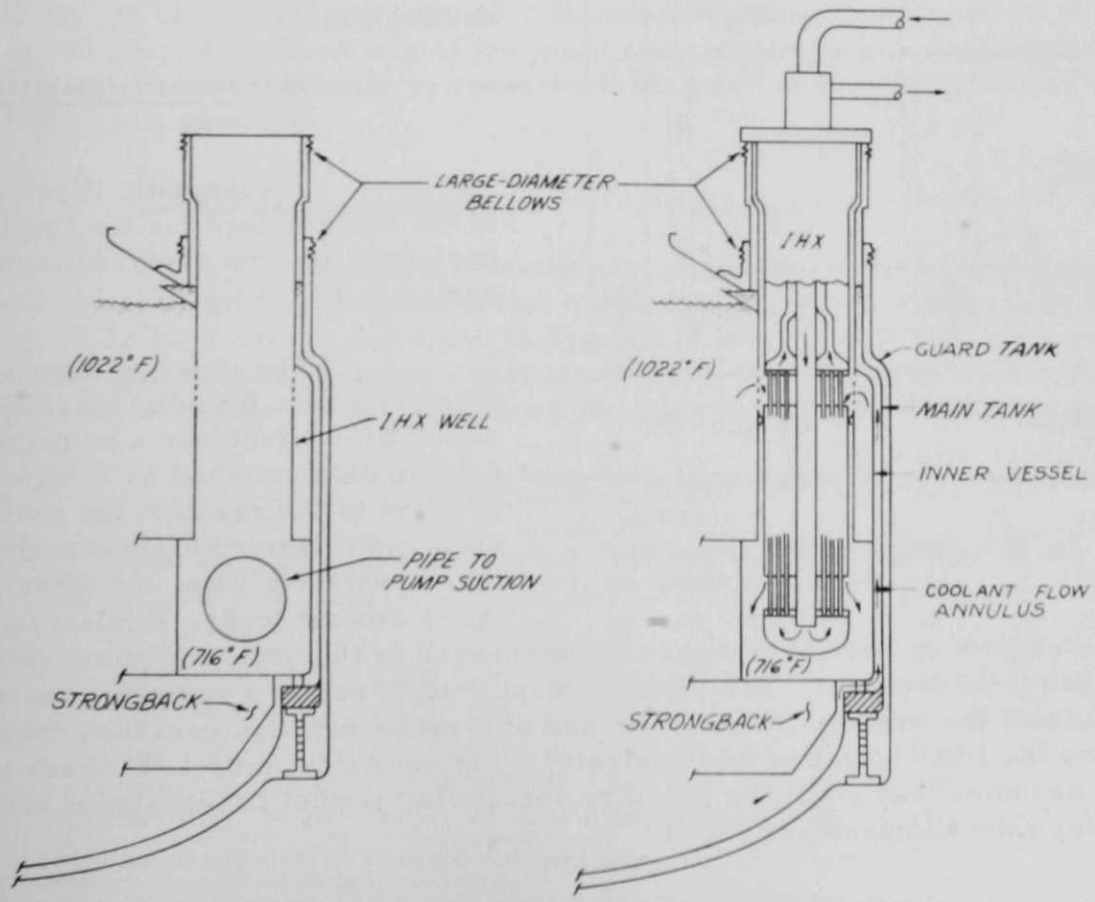


Fig. VI.30. BN-600 IHX Well (left) and BN-600 IHX Well with IHX in Place (right)

This approach requires the use of large-diameter bellows to accommodate differential movements between the well casings and the tank penetration nozzles, and between the main-tank and guard-tank nozzles. Large axial movements of the component support flange can occur, ranging up to 4-6 in. Such movements will cause secondary-piping problems if the secondary piping penetrates a containment barrier in close proximity to the IHX.

Seismicity may be a problem, since the L/D of the component wells is fairly large.

E. Natural-convection Cooling for Pool-type LMFBR Cores

Scoping analyses were performed to examine the emergency and shutdown cooling capabilities of certain pool-type LMFBR designs. The calculations are not intended for design purposes, but only to show trends and qualitative comparisons between the open-plenum system (Phenix, PFR, SuperPhenix, etc.) and the closed-plenum system (1200-MWe-size EBR-II).

The calculations were made with a transient-heat-transfer and fluid-flow computer program written at Argonne National Laboratory.[21] The studies were concentrated on a relatively few limiting cases in an attempt to assess the significance of major system parameters on natural-convection-cooling capability.

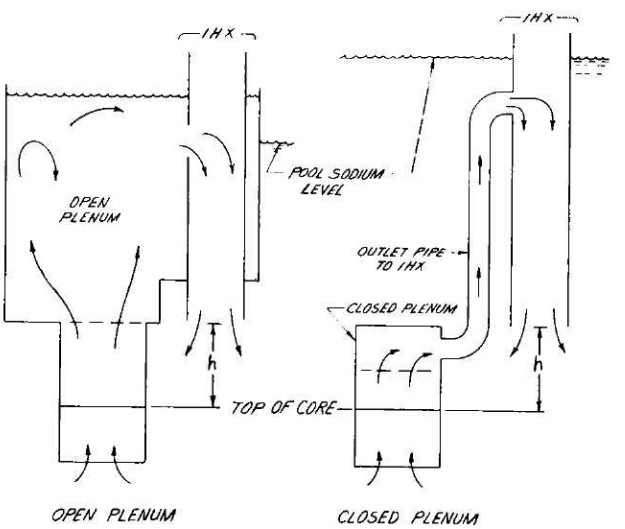


Fig. VI.31. Models of LMFBR Systems for Natural-convection Studies

1. Models and Analysis Methods

A schematic illustration of the models used for the two LMFBR pool-type designs analyzed is presented in Fig. VI.31.

The piped-system (closed-plenum) model, based on the 1200-MWe-size EBR-II, features a primary tank 69 ft in diameter and 53 ft high. In addition to the reactor, the pool contains six intermediate heat exchangers, three coolant pumps, and three primary coolant loops. Coolant from the intermediate heat exchangers is discharged to the pool, and pump suction is taken from the pool. This arrangement, with a positive pumping pressure throughout the system, permits the use of a relatively high pressure drop across the IHX (12 psi in this analysis). The analytical model for these analyses assumes that all of the primary sodium in the pool is initially at normal reactor inlet temperature (711°F).

The hot-center-pool (open-plenum) model was based on a system having a main tank 69 ft in diameter and 40 ft high. As in SuperPhenix, the tank is divided into hot and cold regions such that approximately half the inventory of sodium is at reactor outlet temperature (1003°F) and the other half is at reactor inlet temperature (711°F). In addition to the reactor, the pool contains eight IHXs, four pumps, and associated piping. Coolant flows by gravity from the upper hot pool through the intermediate heat exchangers; consequently, relatively low-pressure-drop IHXs are required (2.5 psi in this analysis).

The calculations focused on the response of these reactor systems to extreme, limiting-case situations. Most calculations assumed complete loss of external cooling capacity (i.e., no heat removal from the IHXs), thereby leaving only the pool sodium as a heat sink for short-term storage of reactor decay heat. The initial condition of the reactor was full-power and full-flow operation with inlet and outlet temperatures of 711 and 1003°F (average), respectively; the outlet temperature of an average core fuel assembly was 1050°F. Transients were initiated by reactor scram followed 0.3 sec later by trip of all coolant pumps.

The principal parameters were the elevation of the IHX exit above the reactor core outlet (labeled "h" in Fig. VI.31) and the primary-coolant-pump coastdown rate. Three IHX exit elevations and two coastdown rates were used: an "h" of 2.8 ft, 5.2 ft, and 12.2 ft, and flow decay to 10% power in 32 sec and in 140 sec. A delayed trip of the main coolant pumps was also studied to investigate system capability to reestablish convective flow from a near isothermal condition.

2. Results of Calculations

Table VI.6 is a tabular summary of results of the calculations. Curves for average core coolant outlet temperature versus time are presented in Fig. VI.32 for Cases 1 and 5 and in Fig. VI.33 for Cases 3 and 7. Reactor power and flow versus time for a fast coastdown--flow decay to 10% in 32 sec--are presented in Fig. VI.34. As shown by Figs. VI.32 and VI.33, sodium

TABLE VI.6 Results of Natural-convection Analyses--Open- and Closed-plenum Pool Systems[a]

Case	Plenum Type	IHX Elev, ft	Flow %	Flow sec	Power at 600 sec, %	Flow at 600 sec, %	Temp at 600 sec, °F: Core Inlet	Upper Plenum Inlet	Upper Plenum Outlet	IHX Inlet	IHX Outlet	Peak Core Outlet Temp,[b] °F	Min Flow, %
1	Closed	5.2	10	32	3.4	3.3	729	1021	1019	822	822	1199	2.3
2	Closed	12.2	10	32	3.38	3.04	733	1012	1003	823	821	1154	2.7
3	Closed	5.2	10	140	3.38	3.2	733	1026	1014	821	818	1123	2.6
4	Closed[c]	5.2	10	130	3.38	3.7	701	962	955	792	606	1067	2.8
5	Open	5.2	10	32	3.38	2.3	806	1215	1029	1026	1022	1370	1.8
6	Open	2.8	10	32	3.38	2.2	804	1239	1032	1028	1024	1370	1.5
7	Open	5.2	10	140	3.38	2.4	863	1263	968	965	959	1337	-

[a]Loss of flow with scram with no heat removal by secondary system, except Case 4.
[b]Peak outlet temperature of an average core fuel assembly.
[c]Scram with no loss of flow, pump trip at 70 sec after scram; some heat removal by IHX, secondary inlet temp of 600°F.

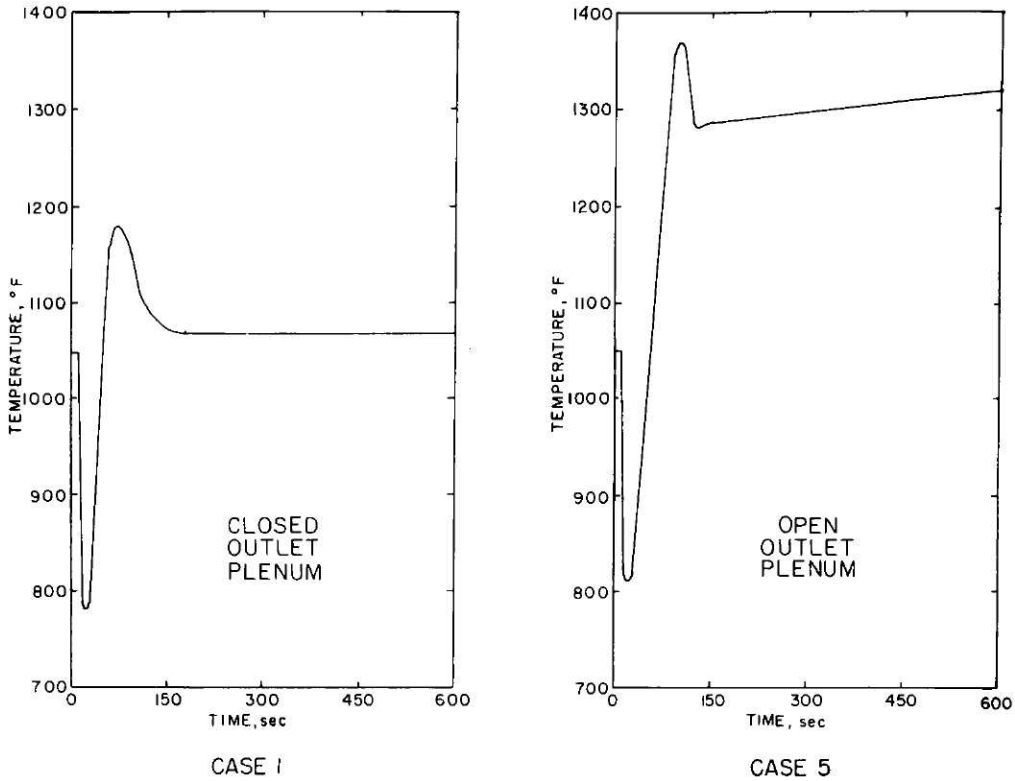


Fig. VI.32. Average Core Outlet Temperature vs Time for Flow Decay to 10% in 32 sec and IHX Outlet 5.2 ft above Top of Core

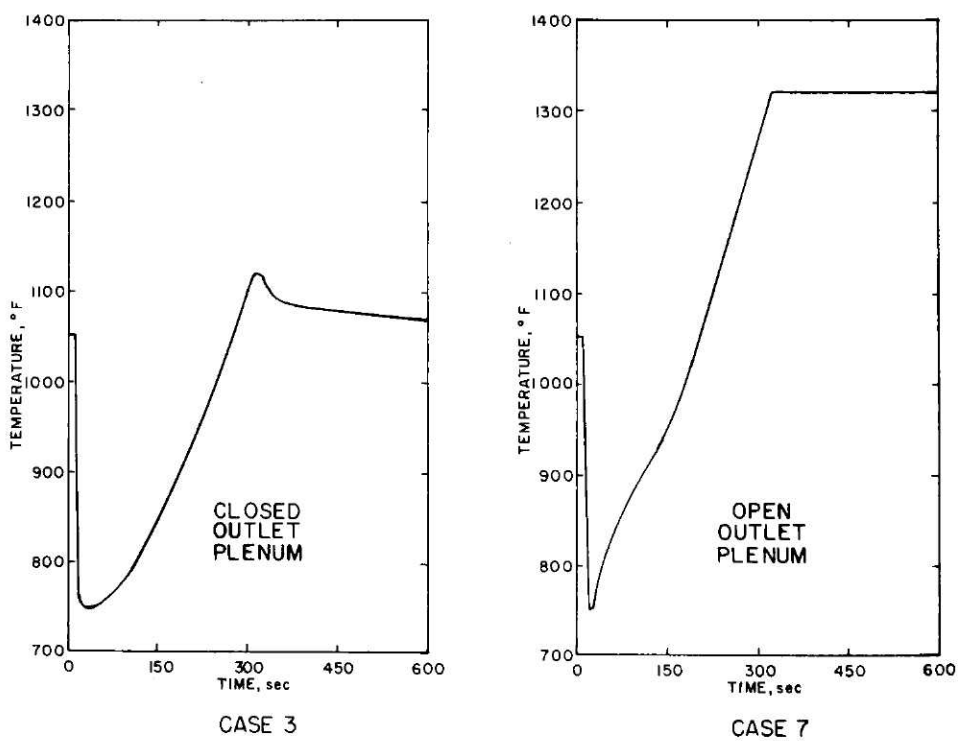


Fig. VI.33. Average Core Outlet Temperature vs Time for Flow Decay to 10% in 140 sec and IHX Outlet 5.2 ft above Top of Core

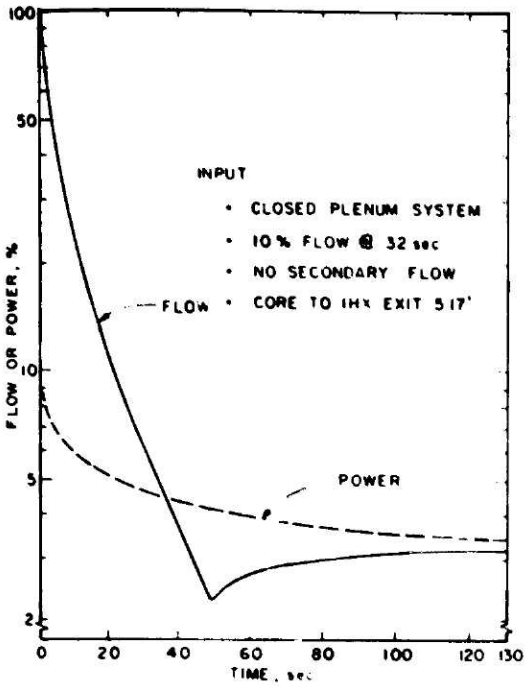


Fig. VI.34. Power and Flow vs Time for Fast Coastdown in Closed-plenum System

temperature from the core undergoes an initial rapid temperature drop (overcooling of the core) due to the power/flow mismatch early in the transients (see Fig. VI.34). This is followed by a subsequent rapid rise and then a drop to a quasi-steady level as the natural-convection flow stabilizes within several minutes. This type of behavior is fairly typical for the two systems at different pump coastdown rates and IHX elevations.

For equivalent IHX elevations, the open-pool cases had higher peak core outlet temperatures (Table VI.6) than the closed-plenum cases. This results from the lower minimum flow (last column in Table VI.6) that occurs during establishment of convection flow. The apparent slightly superior convective-cooling behavior of the closed-plenum system results from the model used for the calculations. The model included a term for the heat loss to the surrounding pool from the plenum outlet pipe. At low convective flow rates the temperature drop in this pipe amounts to about 200°F in most cases (during normal operation the temperature drop is small). The average temperature of the fluid in the outlet pipe is, therefore, higher than the temperature of the fluid in the IHX (which is isothermal, since there is no heat removal). Thus there is a relatively constant convective driving head independent of IHX elevation above the reactor core. This effect, which is not present in the open-plenum system, accounts for the somewhat larger equilibrium convective flow at equivalent IHX elevations that is shown in Table VI.6.

Another difference in behavior of the two systems is shown by the core inlet temperature data in Table VI.6. Core inlet temperatures after ten minutes are higher in the open-plenum system. This results from the rapid equalization of sodium levels in the open-plenum system upon loss of pump flow and discharge of a large volume of hot sodium adiabatically through the heat exchangers into the cold inlet region. The resulting elevated inlet temperature, plus the lower equilibrium convective flow rate (approximately 2.2% versus approximately 3.5%) for the open-plenum system, gives an approximately 200°F higher outlet temperature (compare upper-plenum inlet temperatures at 600 sec in Table VI.6).

The effect that the large sodium inventory in the open plenum has on the temperature entering the IHX is also shown by the data in Table VI.6. Core exit temperature (plenum inlet temperature) is typically about 200°F hotter

than the plenum exit temperature. This results from the analytic model which provides for uniform mixing of cooler sodium in the plenum with the hot sodium exiting from the reactor. This, however, is a transient phenomenon, and these two temperatures would eventually equilibrate.

Calculations for a delayed pump trip (Case 4, Table VI.6) indicate no particular problem in establishing natural-convection flow from a near-isothermal reactor condition, which results when the primary-cooling-pump trip significantly lags reactor scram.

In both systems, the sodium in the main tank acts as a large heat sink. The time to bulk-sodium boiling at the exit of the core is somewhat concept-dependent. However, for both concepts the time available to take corrective action ranges from a few to many hours, depending on thermal head available to provide natural-convection cooling.

These analyses indicate that flow rates and temperatures are sensitive to some subtle features in system configuration. Typically, thermal driving heads are of the order of 0.1 to 0.2 psi, and laminar flow regimes exist in some regions. For purposes of reference, this magnitude of driving head can be produced in a simple, 25-ft-high convective loop by a temperature difference of about 100°F. Examinations of the temperature profiles at various points of the system show that the natural-convection flow rate is surprisingly insensitive to IHX elevation above the core for both systems. Variation in flow rate was even less sensitive than the theoretical cube root value for IHX elevation for a simple system. As discussed above, in the closed-plenum system there is heat loss to the sodium pool from the outlet piping, resulting in approximately 200°F temperature drop between core exit and IHX inlet. This produces a relatively constant convective driving head, independent of IHX elevation, which causes IHX elevation changes to have a smaller effect on convective flow rate. This is reflected by the slight flow change of 3.2 to 3.4% in going from an elevation (h) of 5.2 to 12.2 ft.

The open-plenum system showed a similar insensitivity of flow rate to IHX elevation. This is attributed to the mixing effect in the upper plenum, which produces a cooler liquid leg in the IHX. The result is a reduced sensitivity of convective flow rate to IHX elevation, as shown by the flow-rate change of 2.2 to 2.3% in going from an elevation of 2.8 to 5.2 ft. These results were obtained for a perfect mixing model for the upper plenum, wherein the plenum has no vertical temperature gradients, and none of the flow bypasses directly to the IHX.

3. Conclusions

The results of the natural-convection-cooling calculations indicate that both the closed- and open-plenum LMFBR systems may have good natural-convection-cooling capability even under severe hypothetical cooling-system

failure conditions. Furthermore, the large thermal inertia of the sodium pool typically gives a time of an hour or more before boiling would occur in the complete absence of external cooling.

Some of the differences between the open- and closed-plenum systems which influence convection cooling are summarized in Table VI.7.

TABLE VI.7. Summary of Features of Open- and Closed-plenum Systems Which Influence Natural-convection-cooling Behavior

Closed Plenum	Open Plenum
Small thermal inertia in outlet plenum and piping	Large thermal inertia in outlet plenum.
Reactor inlet temperature immune from large or rapid changes	Reactor inlet temperature changes rapidly with simultaneous failure to remove heat from all IHXs coincident with pump trip.
Total thermal inertia of bulk sodium greater than for open plenum, since all sodium is at reactor inlet temperature. This allows more time for auxiliary systems to become operative.	Hot-sodium outlet-plenum volume about the same as cold-sodium volume in main tank.
Outlet-pipe heat losses to pool are substantial under low-flow conditions and are beneficial.	Temperature of sodium into IHX not sensitive to changes in coolant temperature from the core.
Flow pattern in outlet of plenum and piping well defined.	IHX flow determined by level difference between hot and cold regions in main tank--this level drops rapidly on pump trip.
	Flow pattern in outlet plenum uncertain.

F. Auxiliary Decay-heat-removal Systems

There are two elements to decay-heat removal: (1) cooling of the reactor and (2) transport and dissipation of the reactor heat to the environment. The former has been discussed in Sec. VI.E. The latter is the subject of this section.

The designs of all LMFBRs include features to assure that normal heat-transport paths remain operable for dissipation of decay heat. Examples of such features are redundant secondary and steam systems, redundancy of equipment and flexibility of operation of the steam and water systems, assured supplies of water, redundancy and diversity of electric power sources for all vital equipment, etc. These assurance measures are assumed as normal. In the context of this discussion we deal with systems, frequently called auxiliary or emergency decay-heat-removal systems, which serve when all other normally available systems no longer are serviceable, for whatever reason.

The pool-type system can readily accommodate an auxiliary decay-heat-removal system. In this section the approaches currently in use are described and then discussed from a generic point of view.

1. PFR and EBR-II

These two facilities use somewhat similar approaches. Each auxiliary decay-heat-removal system is a fully natural-convection coolant system and includes sodium-to-NaK heat exchangers in the main tank and NaK-to-air heat exchangers to dissipate the heat to the environment.

The PFR system employs a forced-convection air system which will also operate by natural convection at slightly higher temperatures. In PFR the in-tank exchangers consist of a coil of pipe positioned within the primary-coolant inlet-plenum region of each IHX (Fig. VI.12). The NaK lines to each heat exchanger coil pass through the shield plug of the IHX. The coil is removable with the IHX for maintenance. The coils in the two IHXs associated with one secondary loop serve a single NaK-to-air heat exchanger. Thus, three completely independent natural-circulation loops are provided for dissipation of decay heat from the main tank. Each loop has sufficient heat-transport capability (5 MWt) to protect the tank and its contents.[22]

The EBR-II employs two in-tank, immersion-type bayonet heat exchangers, each capable of protecting the system. Figure VI.35 illustrates the design.

Both the PFR and EBR-II employ various shield-cooling, sodium-purification, and other heat-removal systems which remove heat from the primary system. None, however, is counted upon for decay-heat removal.

2. Phenix

Decay-heat removal is provided by natural circulation of the secondary sodium through the steam generator and heat dissipation to the atmosphere by natural circulation of air through the steam-generator enclosure. The steam generator, Fig. VI.36, is comprised of hairpin-shaped pipes containing secondary sodium. The pipes are not insulated and are located within a large, tall, insulated enclosure (case). The case has openings (normally closed) that can be opened for natural circulation of air through the enclosure. Each loop can dissipate about 5 MWt.

In event that all three secondary loops fail completely as a heat-transport system, the primary system will slowly increase in temperature, bringing into effect an ultimate, water-cooled, emergency heat-removal circuit. The safety tank that surrounds the main-tank and guard-tank complex is covered by a network of tubes of 1.34 in. (34 mm) dia, 0.160 in. (4 mm) wall (Fig. VI.37). The tubes are spaced 5.9 in. (150 mm) apart, and neighboring tubes belong to separate circuits (two circuits are provided). The system capability is approximately 2 MWt for a primary-system temperature of about 1382°F (750°C). To permit more ready heat transfer from the main tank under high-temperature conditions, the guard tank for the main tank is equipped with

special insulation. As shown in Fig. VI.38, the insulation is composed of four sheets of 0.004-in. (0.1-mm) stainless steel which is held apart by a metal grating. The total thickness of the insulation is 0.79 in. (20 mm). At 750°C the thermal resistance of this insulation is one-fourth its value at normal operating temperature, 525°C.

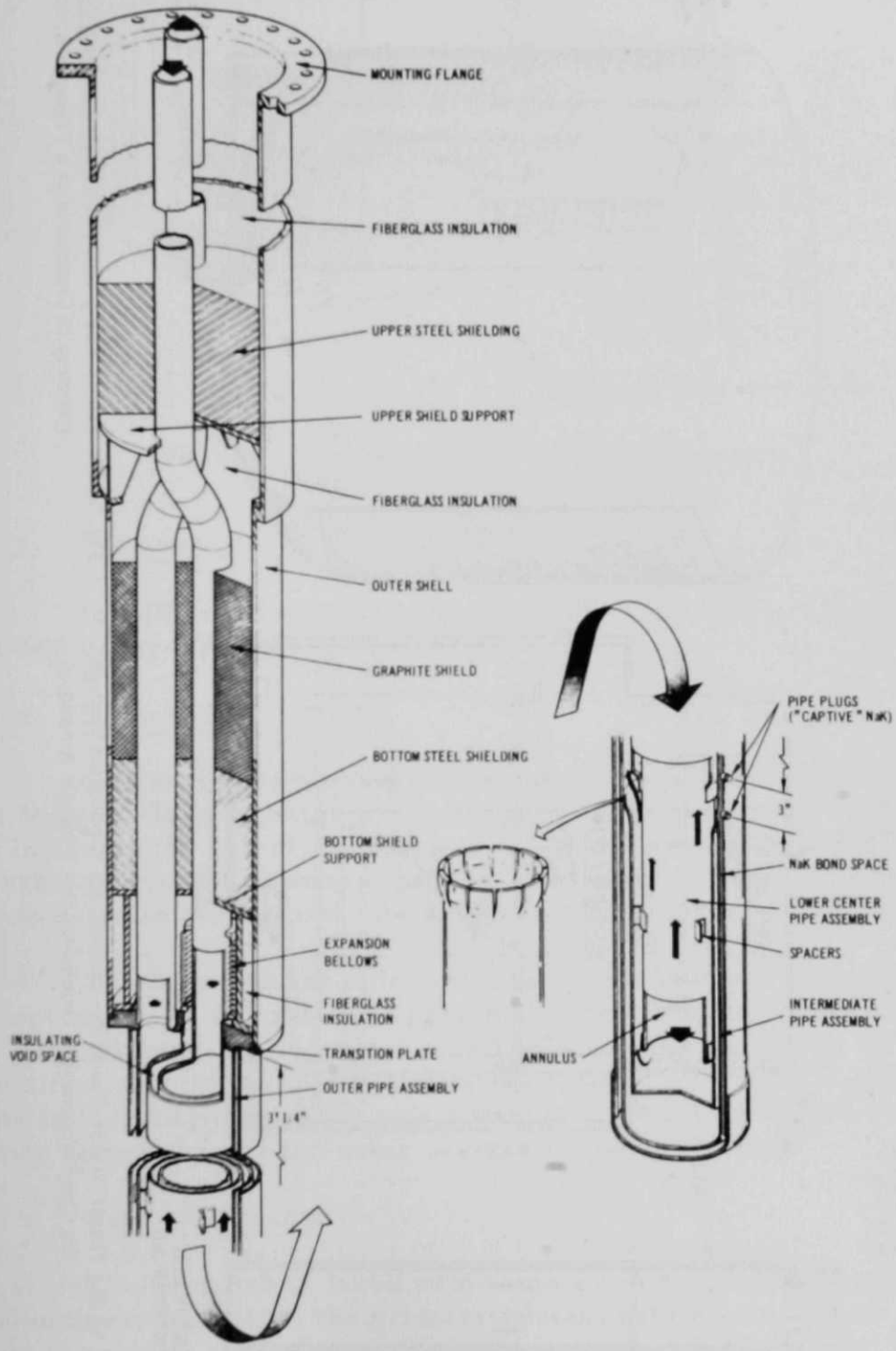


Fig. VI.35. EBR-II Auxiliary Decay-heat-removal Heat Exchanger

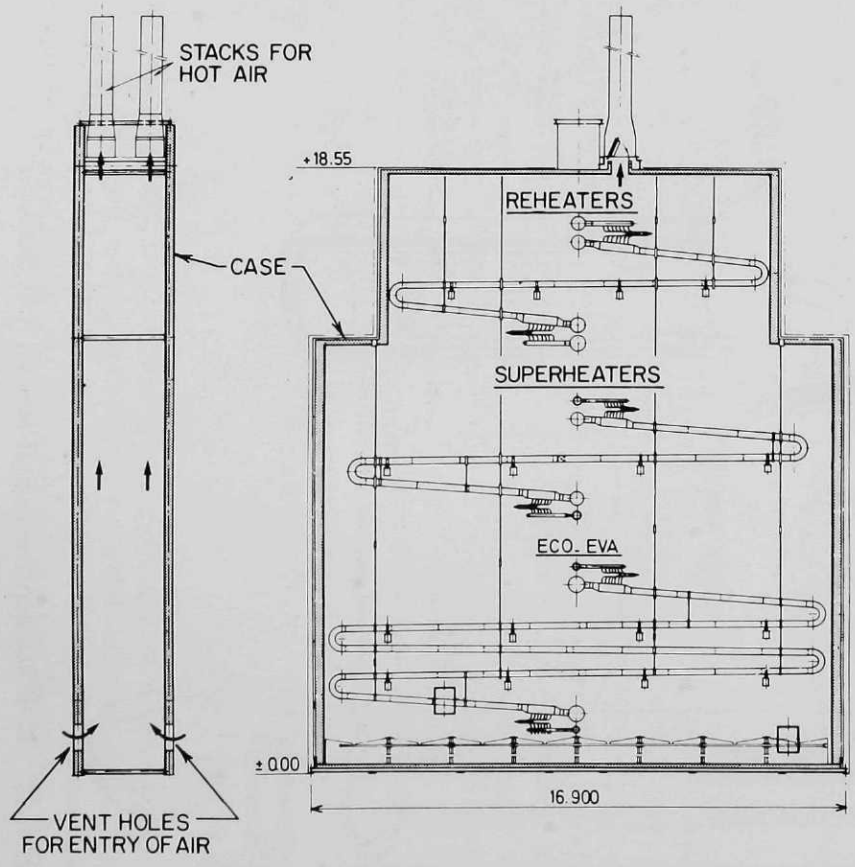


Fig. VI.36. Phenix Steam Generator--Natural Air Circulation through the Case[23] (courtesy of Commissariat a l'Energie Atomique)

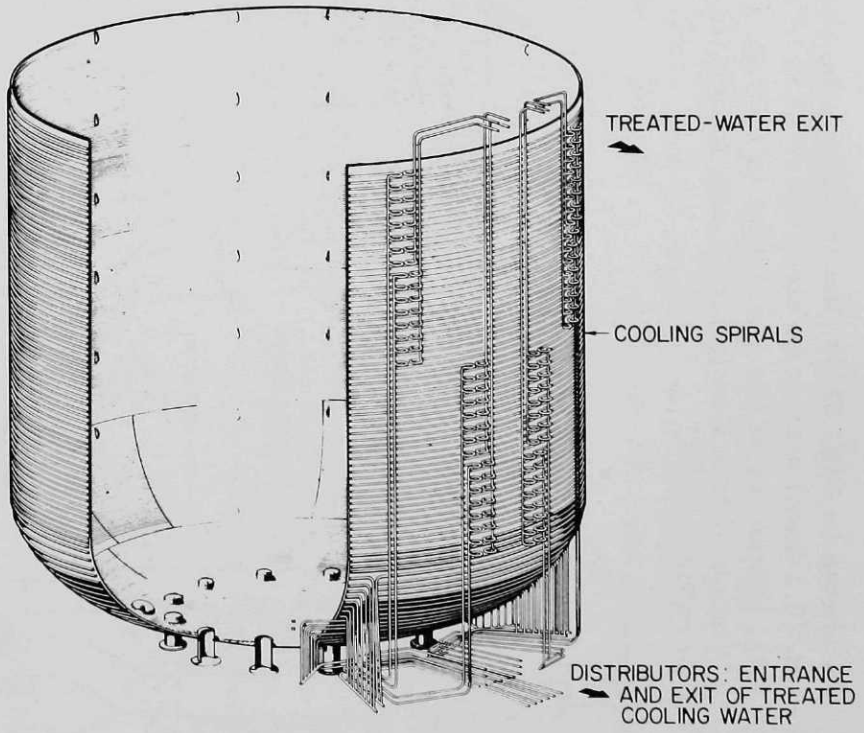


Fig. VI.37. Phenix Safety Tank--Ultimate Emergency Cooling System[23] (courtesy of Commissariat a l'Energie Atomique)

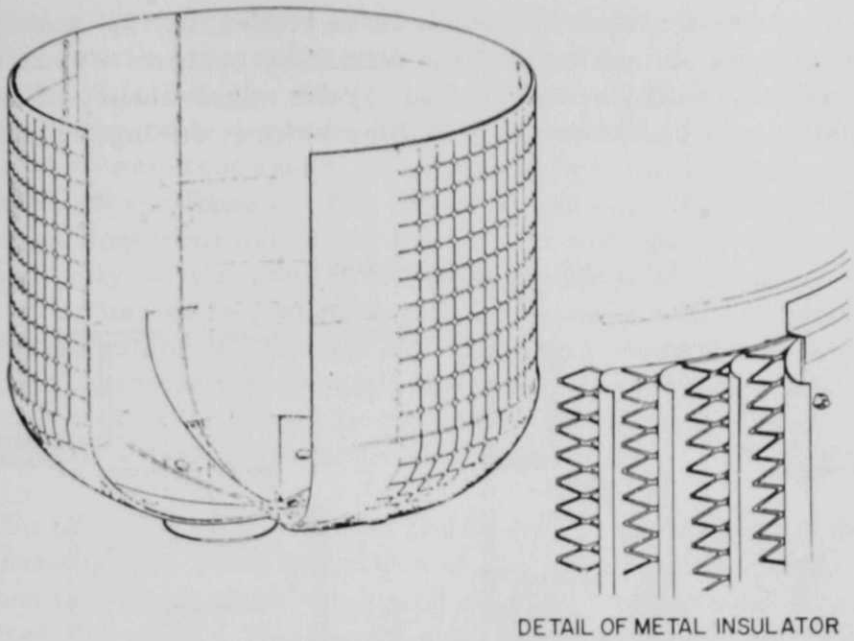


Fig. VI.38. Phenix Guard Tank and Its Insulation[23] (courtesy of Commissariat a l'Energie Atomique)

3. BN-600

Insufficient information is available in the literature to describe the BN-600 decay-heat-removal system.

4. SuperPhenix

The design of the steam generators for SuperPhenix does not permit decay-heat dissipation by air circulation as in Phenix. The current design indicates the use of a natural-convection, secondary-sodium-to-air heat-exchange system as shown schematically in Fig. VI.39. A valve must be opened to activate the system. A capability of 7.5 MWt per loop is planned.[24]

The SuperPhenix ultimate decay-heat-removal system is similar in concept to that of Phenix. The principal differences are that (1) the main tank is not insulated, (2) the guard tank has been eliminated and the safety tank retained, and (3) the piping for the ultimate, emergency, cooling-water systems is located between the safety tank and the insulated concrete wall of the cavity rather than on the outer surface of the safety tank.

5. Generic Aspects

The designer is faced with some immediate choices. He may opt to remove decay heat from the primary system with the secondary sodium systems, or to provide separate, independent systems for decay-heat removal from the main tank, or to do both. Key considerations in these choices are

(1) the nature of the accident that must be survived, (2) the assessment of the capability of in-tank components of the auxiliary system versus the capability of the IHXs to survive the accident, and (3) the vulnerability of normal and auxiliary systems to common-mode failure before, during, or after the accident.

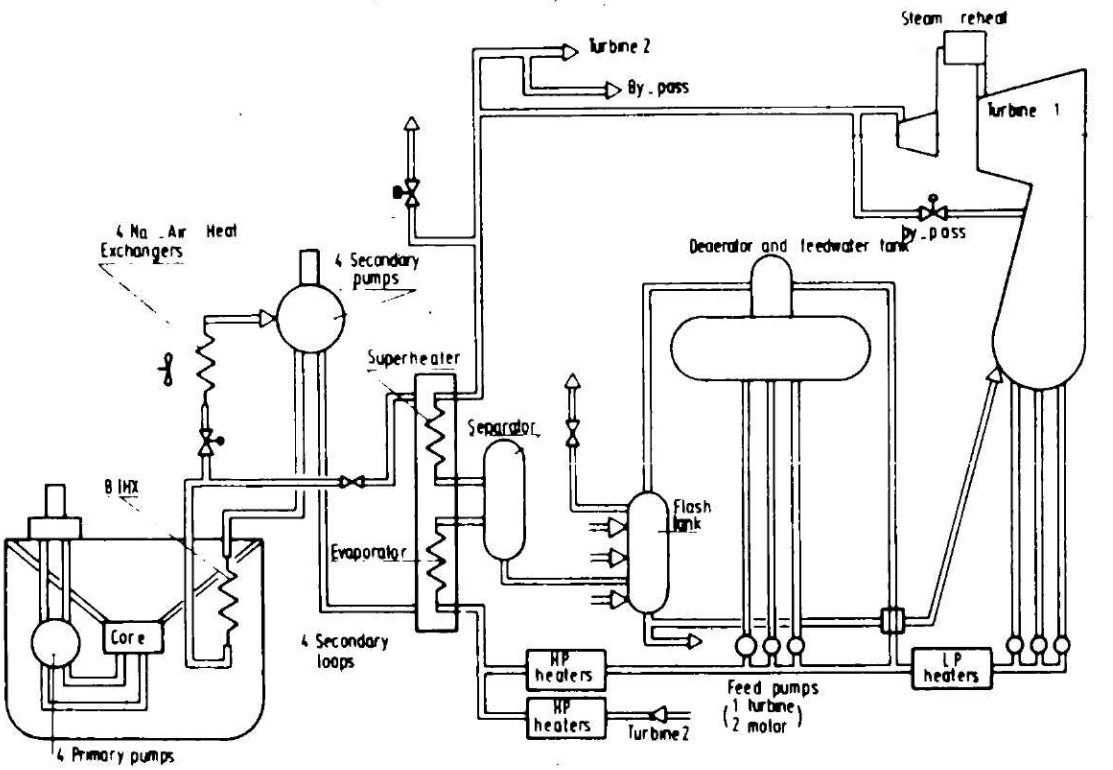


Fig. VI.39. SuperPhenix--Plant Flow Diagram[24] (courtesy of American Nuclear Society)

For this discussion, the accident will be assumed of the nature of an HCDA. The PFR, Phenix, and EBR-II pool-system designs provide for the possibility that all secondary systems may become inoperable for decay-heat transport. Each provides an independent system for decay-heat dissipation from the primary system.

The vulnerability of components within the main tank is unsettled. Much depends upon the magnitude of the accident and resultant loads upon in-tank structures. The kind, design, and location of the component also will influence survivability. The coil that is used in the PFR IHX would seem to have a high degree of survivability. On the other hand, the PFR IHX, which is supported by a central pipe, might be more susceptible to damage than an IHX which is supported at its outer circumference and protected by a strong well casing.

The use of a pipe coil in the inlet plenum of the IHX has a number of advantages and disadvantages. The coil is located so as to promote thermal

head and natural-convection flow rates. This reduces maximum core temperatures. The coil is removable with the IHX, thereby reducing the number of openings in the shield deck. The trade-off here is that the IHX must be removed to get at the coil--a more difficult operation. Coils are flexible and thereby resilient to accident loads, but may exhibit tendencies to vibrate under normal coolant-flow conditions. The effect of the coil on hydraulic resistance of the IHX and on flow distribution to the tubes must not be adverse. The coil approach is probably best accommodated by an IHX with primary coolant on the tube side. The use of two coils in parallel (one in each of two IHXs) will require careful balancing of the hydraulic resistances of these two parallel auxiliary loops. Inherently, these natural-convection systems will have low resistance, and small differences in hydraulic characteristics of the parallel loops could unbalance flow and starve one of the coils.

The depth of submergence of the coil is an important design consideration. If the coil is to be operative in the event that the main tank fails (and the sodium is contained by the guard or safety tank), the coil must remain submerged after the event. The depth of initial submergence will depend upon the volume of the annulus between the main tank and guard tank relative to the volume of sodium in the main tank and the tank L/D ratio. In general, a tank about 70 ft in diameter and with an L/D ratio of about one will experience a drop of about a half foot of sodium for each 1 in. of annulus uniformly surrounding the tank. The annulus clearance between the main and guard tanks is likely to depend upon clearance requirements for design seismic or HCDA events. Clearances of 9-18 in. are likely, although current illustrations for SuperPhenix indicate a clearance of about 30 in. Such clearances clearly would cause large changes in the sodium level in the event of a leak in the main tank. Deep submergence of the IHX inlet plenum limits the flexibility of heat exchanger design, since the elevation of the IHX primary exit relative to the top of the core may be fixed for natural-convection purposes.

Other auxiliary, in-tank, heat exchanger designs are possible, such as the bayonet type used in EBR-II. These units occupy independent locations in the tank. Each is connected to its own natural-convection auxiliary heat-dissipation system. This approach has no significant interaction with the main primary coolant system. It neither adds to nor detracts from the capability of the primary system to convect thermally. Locations in the shield deck are required. Each heat exchanger is individually removable for maintenance. Providing adequate heat-transfer area while maintaining a simple, reliable geometry with a small cross-sectional area (shield-deck plug area) may be a problem. If used with an open-plenum system, penetrations through the inner vessel will be required, with adequate provisions in the bayonet heat exchanger to account for the sudden change in axial temperature profile across the inner-vessel penetration area.

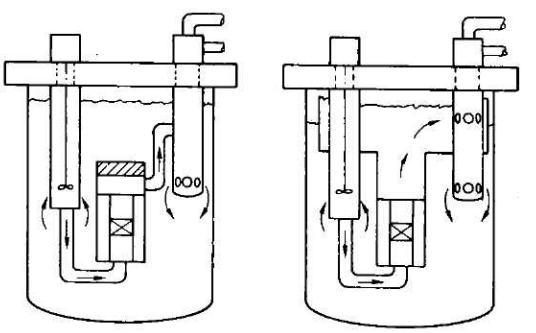
VII. SAFETY-RELATED ASPECTS OF POOL-TYPE LMFBRs

This section describes the results of a preliminary review and evaluation of some of the safety-related characteristics of pool-type LMFBR systems. Accordingly, the work described herein was an attempt to delineate in a qualitative and, where possible, quantitative fashion the capabilities of pool-type LMFBR designs to accommodate certain postulated accidents. This safety review was divided into four principal steps: (1) review of pool concepts, and definition of general safety-related characteristics, (2) definition of safety considerations, features, and requirements, (3) consideration of unique safety features of pool-type systems, and (4) where possible, quantitative analysis and assessment of particular safety-related features. Throughout, the main focus was on specific design features unique to pool-type LMFBR systems; accordingly, those design features which have similar implications for all LMFBR systems were not addressed.

The safety reviews were based on reactor concepts which are discussed in detail in other sections of this report (e.g., Sec. VI); however, for purposes of completeness and continuity, a summary of the main design approaches and their salient features are presented in the following section.

A. Common Characteristics of Pool-type Reactors

In the pool-type LMFBR, all components of the primary system are immersed in sodium coolant contained within one main tank. These components include the reactor, pumps, intermediate heat exchangers (IHXs), piping system, and fuel handling mechanisms. The various design approaches for the pool-type systems discussed can generally be divided into hot-pool and cold-pool categories. A cold-pool approach (Fig. VII.1a) is represented by the EBR-II design, wherein the entire boundary and essentially all the coolant of the main tank are isothermal at the reactor coolant inlet temperature. In the hot-pool approach (Fig. VII.1b), which is predominant among the various European reactor designs (e.g., PFR, Phenix, SuperPhenix, BN-600, CFR), much of the pool top surface and a significant part of the main-tank contents are at the reactor coolant outlet temperature. Differences between the various design approaches, as they relate to safety, are described in the following sections.



a) COLD POOL b) HOT POOL

Fig. VII.1

Typical Pool-reactor Configurations

B. Safety Approach for LMFBRs

The generally accepted approach to reactor safety in the United States as applied to LMFBRs consists of the following three levels.[25]

1. Level 1

Provide a simple, reliable, and functional design free of defects, with inherent safe performance, and fabricated and operated to the highest proven standards. The situations which will normally be handled by this level of protection are full-power operation, start-up and shutdown, random fuel-element failures, and refueling.

2. Level 2

Provide protection systems with adequate speed of response to cope with all identified transients. This level should accommodate events that, based on experience, are expected to occur at least once in the life of the power plant; they include loss of off-site power, loss of power to one pump, operator errors, and spurious scrams. In addition, this level should accommodate unlikely faults which are not expected to occur individually, but any one of which might occur during the life of the plant. Examples of such faults include pump seizure or failure of heat-dump system.

3. Level 3

Provide extra capability to cope with extremely unlikely events that are never expected to occur, and incorporate additional design features to provide prudent margins for unforeseen events. This level should accommodate events which are not expected to occur during the plant lifetime, including the maximum design flood, earthquake, or tornado, a very large sodium fire, or a very large sodium-water reaction. In addition, it should provide design margins to accommodate postulated events, such as the hypothetical core disruptive accident (HCDA), including possible significant core meltdown.

The safety approach for LMFBRs closely parallels that which is currently used for light water reactors (LWRs) and is based on the very conservative approach taken by the licensing and regulatory authorities in the United States. It is a balanced approach which considers high-probability, low-consequence events as well as low-probability, high-consequence events. The following discussion of the generic safety characteristics of pool-type LMFBR systems is organized and presented under the three levels as delineated above.

C. Safety Characteristics of Pool-type LMFBRs

1. Level 1

The first level of safety assures reliable plant operation and prevention of accidents during normal operating conditions through the intrinsic

features of the reactor design, such as quality assurance, redundancy, testability, inspectability, and fail-safe characteristics. In this context, a significant feature of pool-type systems is that they can be designed with a simple, low-stress, redundant containment envelope (main tank and guard tank). It is believed that the main and guard tanks can be shop- and field-fabricated with a high degree of assurance that they are reliable. They are fabricated of relatively simple plates by straightforward welding techniques; hence, this integrity is essentially assured by reliable fabrication and inspection techniques. Experience with the fabrication and operation of somewhat similar tanks in the United States has been quite good. Useful experience has been gained with the EBR-II reactor main tank, as well as with more recent work in the fabrication and erection of large, non-nuclear, steel tanks (see Sec. VIII).

In pool-type LMFBRs, problems with design and layout of primary piping systems to accommodate thermal expansions and questions about reliability have been greatly reduced. Reliability requirements for the very small amount of primary-system piping and other components immersed in the main-tank sodium are lessened, since the consequences of leaks are quite tolerable. Thus, the inventory of radioactive primary sodium coolant is contained within a single enclosure, providing an extremely high degree of protection against design or fabrication errors which could lead to loss of primary coolant during operation.

As discussed in Sec. IX, provisions can also be made for in-service inspection of the main and guard tanks. Moreover, degradation of the tank during its operating lifetime is expected to be little for the following reasons: (1) The main tank, in most cases, operates isothermally at or near the reactor inlet temperature and is not subjected to significant operating thermal transients; (2) there are no penetrations in the main-tank walls or bottom, and, except for places where the tank thickness changes or where supports for internal structures are located, there are no points of stress concentration;* and (3) the main tank is far removed from the reactor, so neutron radiation damage will be minimal because the neutron flux at the tank wall is low.

All these features help assure a continued reliability of the primary tank over its operating lifetime and help assure its integrity under HCDA loads.

2. Level 2

The second level of safety provides protection against anticipated faults and unlikely events (e.g., partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel handling errors) that might occur in spite of the care taken in design, construction, and operation of the plant. This additional level of protection is provided by engineered safeguards, including

*This advantage would be negated to a certain extent if internal structures, such as the reactor support skirt, are welded to the tank wall. This practice would give rise to locally high stresses.

redundant critical components and protection devices or systems designed to assure that such events will be prevented or arrested so as to mitigate the consequences.

a. Decay-heat-removal Capability

As stated earlier, a common characteristic of all pool-type systems is the large sodium inventory contained within the one main tank. This sodium is always available for decay-heat removal and provides a large thermal capacity for energy storage in the event of complete or partial failure of the plant's secondary heat-removal system. The average temperature and volume of the primary-sodium pool, however, differ with each reactor design. The cold-pool design, as represented by the EBR-II approach, has almost the entire sodium inventory at reactor inlet temperature (about 700°F). On the other hand, the hot-pool designs represented by PFR, Phenix, SuperPhenix, and CFR, have a higher average sodium temperature, since a large fraction of the sodium inventory is in the hot pool, at reactor outlet temperature. In the BN-600 reactor, the average temperature is quite high, since most of the sodium inventory is in the hot pool, at reactor outlet temperature. With appropriate care taken in the design, pool-type systems have excellent capability for removing core decay heat by natural convection, even with the secondary coolant loops (IHXs) inoperative. (See Sec. VI.E for some calculations made in connection with this study.)

Since the total quantity and average temperature of the sodium are different for each of the systems, the available energy storage in the pool below sodium boiling temperature (1620°F) is different for each system. Nevertheless, the pool typically provides a time of hours rather than minutes from reactor shutdown to boiling at the exit of the reactor (with no removal of heat from the pool by auxiliary cooling systems). (See Sec. VI.E.) Also, as a general rule, it is quite easy to incorporate a completely redundant decay-heat-removal system for the pool. This can be accomplished by: (1) a separate bayonet-type heat exchanger extending into the pool (EBR-II approach), (2) removing heat through the main-tank walls by a cooling coil outside the main and guard tanks (Phenix and SuperPhenix approach), or (3) providing a redundant cooling coil within the IHX (PFR and CFR approach). Although the large primary-sodium inventory does not remove the requirement for an auxiliary type of cooling system, it does greatly ease the requirements for speed and reliability of such a system, since there is reasonable time to activate it manually or, possibly, to locate and repair malfunctions.

b. Thermal Inertia of Pool

In addition to its heat-storage capability, the sodium pool provides significant thermal inertia and plays a significant role in damping of temperature transients occurring as a result of power and/or flow transients. In both hot- and cold-pool systems, the thermal inertia of the cold pool (with the exception of BN-600, which has essentially no cold pool) effectively isolates the

pump and reactor inlet from temperature increases or decreases resulting from either loss of secondary heat-removal capability (through the IHXs) or reactor trip. Thus, in the pool-type system the reactor coolant pumps and inlet plenum see essentially constant, or very slowly changing, sodium temperatures during both of these types of transients.

In all designs of pool systems, the pump is located in the cold leg, and there appears to be little or no incentive for any other approach. A resulting advantage is elimination of the need to prevent possible damage due to thermal shock by rapidly shutting down the pumps in coincidence with reactor trips. There is actually an incentive to provide coolant pumps with relatively long coastdown times, thereby enhancing the likelihood of continued cooling while auxiliary systems are brought into play and/or natural-convection cooling is stabilizing. This approach is used in the SuperPhenix design, wherein the electrical generator supplying the pump motor is equipped with a flywheel, thereby providing a coastdown time of about 50 seconds to 50% flow. The cold-leg pump makes it possible to continue operating the pumps after reactor scram and thus reduce the probability of the situation where there are spurious pump trips with failure to scram the reactor.

There has been some thought given to the possibility of designing the reactor core with a strong negative expansion coefficient. This has particularly been a design goal of the SuperPhenix design team. In the event of loss of flow with failure of reactor scram, the reactor could then be brought to some steady-state low power below some nondamaging core temperature on the negative feedback from the increased coolant temperature and resultant core thermal expansion.

c. Integrity of Primary Loop

Pool-type primary systems appear to be less vulnerable to the sudden rupture of reactor inlet piping. Moreover, the effects of a pipe break are believed to be tolerable.

Several design features add inherent safety advantages. Because the piping system is submerged, the outlet piping can fail without seriously affecting the cooling of the core. The inlet piping is the area of chief focus. This piping is usually of relatively short length, has few bends, operates isothermally at reactor inlet temperature, and is subject to only minor thermal transients. The use of flexible pipe-to-pump connections ensures low reaction loads and related stresses. The lack of insulation, electric pipe heaters, or extensive pipe support systems eliminates a number of potential failure mechanisms. Immersion of the piping in the bulk sodium may serve to dampen seismic accelerations. Corrosion is not a significant factor.

Conversely, pool-type piping is extremely difficult to inspect, flexible joints at the pump discharge may be vulnerable to seismic loads, and vibration always must be carefully considered.

The effects of a guillotine failure of an inlet pipe are believed to be tolerable. This belief is based upon the following considerations rather than a thorough and exhaustive analysis, which was outside the limited scope of this study. First, a large reduction in flow rate through the core can be tolerated before endangering the safety of the facility. For example, based on nominal core temperatures, a 50% reduction in flow rate would leave a margin to boiling with failure to scram the reactor. Second, the flow rate through the core during a pipe break depends upon the relative hydraulic resistances for flow through the system and out the break versus flow through the core and IHXs. The relative resistances of these two flow paths dictate the division of flow. If the hydraulic resistance through the core and IHX is normally low, as is the case for most pool concepts, a favorable flow division should obtain with commensurate adequate cooling of the core. Third, the submergence of the inlet piping in the bulk sodium provides a constant back pressure which reduces the available ΔP across the pipe break, thus further reducing the flow rate through the pipe break and enhancing the fraction of flow through the core. Finally, no loss of coolant occurs, and natural-convection cooling of the core is not inhibited.

The above considerations coupled with the design flexibility to provide a multiplicity of inlet lines if needed (SuperPhenix has eight, CFR has twenty-four) give strong indication that the pipe-break accident in pool systems can be accommodated without undue design complexities or serious impact upon safety of the facility.

3. Level 3

The third level of protection supplements the first two levels by providing acceptable plant response to extremely unlikely faults such as pipe leaks, large sodium fires, or large sodium-water reactions.[25] In general, this level also takes into account plant response to maximum flood, earthquake, or tornado. Design features to accommodate hypothetical core disruptive accidents (HCDAs) are also included.

a. Maximum Floods and Tornadoes

We have not identified any differences in the expected behavior of pool-type LMFBR systems from that of other LMFBRs in their response to these natural phenomena. In general, the same type of design approaches and precautions would be expected to be used for pool type and other LMFBR systems.

b. Earthquakes

There are probably some advantages to the pool-type systems in resisting damage from earthquakes. This is discussed in detail in Sec. IV.

c. Hypothetical Core Disruptive Accidents (HCDAs)

Several features of the pool-type LMFBR could be helpful in absorbing the energy release and after-effects of an HCDA. These include the large cross-sectional area of the main tank as well as the large sodium inventory.

The large sodium pool could be helpful as an energy dissipater or absorber in the event of an HCDA. The pressure waves will be attenuated, and furthermore, the bubble of fuel, fission gas, and sodium vapor will have a large sodium pool in which to dissipate its thermal energy by radiation or rapid heat-conduction and -convection processes.

Several safety-related advantages can be attributed to the large cross-sectional area of the main tank, especially in terms of an HCDA and its after-effects. The first of these is that the sodium-slug energy, as a result of such an event, will be dissipated over a large area. The slug velocity will be reduced, thus giving a lower impact pressure. This results in a lower total loading on the rotating plug, and the reduced force per unit area may facilitate design of a shield deck able to sustain HCDA loads. This point will be discussed in further detail in Sec. VII.D.6, following. The volume of the cover-gas gap is also large, and this results in a lower, quasi-steady static pressure.

During and after a postulated HCDA, fuel can be expected to be dispersed both upward and downward, although the proportioning between the two directions is not known at this time and may be very dependent on the details of the accident sequence. Some of the fuel dispersed upward could conceivably become lodged in the IHXs or pipes in sufficient quantities to cause melt-through of these components, and most of such fuel would fall on surfaces throughout the tank, including the bottom. Meltthrough of the pipes and IHXs would apparently not cause a problem, since these components are within the main tank, and the fuel would eventually migrate to the bottom or remain coolable at some other position. Breach of these primary-system components will not cause primary-sodium loss nor jeopardize coolability. The fuel dispersed downward from the core would also come to rest on the core support structure, reactor support skirts, and main-tank bottom.

The fuel thus dispersed on the main-tank bottom must then be analyzed relative to the degree to which it can be shown to be in a coolable, subcritical configuration that would not thermally attack and cause failure of the main tank and resultant loss of sodium. If the main-tank bottom were to have a large radius of curvature and the fuel were well dispersed, it would be possible to cool a large fraction of the debris with the large sodium inventory present. If adequate cooling cannot be demonstrated because, say, "natural" dispersal cannot be shown, or for any other reason, then an engineered core-debris retention system may be considered. If this system were believed necessary, the pool concept looks amenable to an in-vessel (i.e., in-tank) design to ensure debris coolability, using natural convection, without failure of the

main tank. It should be emphasized that considerations relative to the need for a postaccident heat-removal system are in no way special to pool systems, but pool systems may be particularly amenable to natural-convection debris cooling with no engineered device or with an in-vessel engineered device.

Results of a calculation of the structural response of a pool-type reactor having either a 70-ft or 35-ft diameter are presented in the following Sec. VII.D.

D. Evaluation of Containment Capabilities

1. Selection of Parameters and Scope

One of the concerns in the evaluation of the safety of LMFBRs is the demonstration of the ability of the system to sustain an HCDA core disruptive accident (HCDA). This survival is assured by showing that the main tank, or coolant boundary, will not be breached and that the core can be maintained in a coolable, subcritical configuration. As has been mentioned, the pool concept employing a large-diameter tank seems to offer some added safety margins over a system of similar thermal-power rating using a smaller vessel diameter.

This preliminary evaluation of the containment capability of the pool concept was accomplished as follows. First, a reference reactor concept was selected. The reactor configuration chosen was based essentially on the 1200-MWe scale-up of the EBR-II reactor, but with an integral shield-deck design (no separate main-tank cover). The reactor region, including the core, blankets, fission-gas plenum, and shielding, was based on the 3000-MWt reference reactor design described in Sec. III. One exception should be noted here, however. Unlike the EBR-II concept, the model used in this evaluation did not include the heavy reactor-vessel cover employed in the EBR-II design, since it was difficult to include this component in the mathematical model. The resulting model, then, approximates, to a degree, a hot-center-pool design.

After selection of the overall reactor configuration, an attempt was made to include the main features of the pool system. Because of the complexity inherent in the design, and the ability of the containment computer code (REXCO-HEP)[26] to accommodate only certain design details, the selection was limited to the main tank, the reactor vessel, the details of the reactor region noted above, the core support structure excluding the support skirt, and a rigid representation of the shield deck and plug structures. Details of the design selection follow in Sec. VII.D.2. As a first approximation, the pumps and IHXs were ignored.

To obtain a representative evaluation based on safety experience with other U. S. fast reactors, including the Fast Test Reactor and the Clinch River Breeder Reactor, a representative pressure-volume curve was selected. This curve was determined by an independent calculation and has been used as an arbitrary representation of a possible upper bound for the reference reactor

core. It should be emphasized at this point that no detailed, mechanistic evaluation of the reference reactor under accident conditions was made, but there is no reason to believe at this time that the results of such an analysis would be substantially different from those reported here.

Finally, calculation of the structural response of the system was performed, and several variables such as energy partitioning, vessel strains, pressure loadings, and component forces were determined. Separate scoping-type analyses were made to evaluate the response of the plugs and shield-deck structures. The effect of the variation of several parameters, including main-tank wall thickness and diameter, on the system structural response was also determined with a combination of computer and scoping calculations.

Separate evaluations were made of the effect on the core flow due to a single, double-ended, pipe rupture and natural-convection-cooling capabilities. These two items are discussed in more detail in Sec. VI.B.

2. Mathematical Model

The mathematical representation of the reactor described in the preceding section is shown in Fig. VII.2. This figure shows the Lagrangian

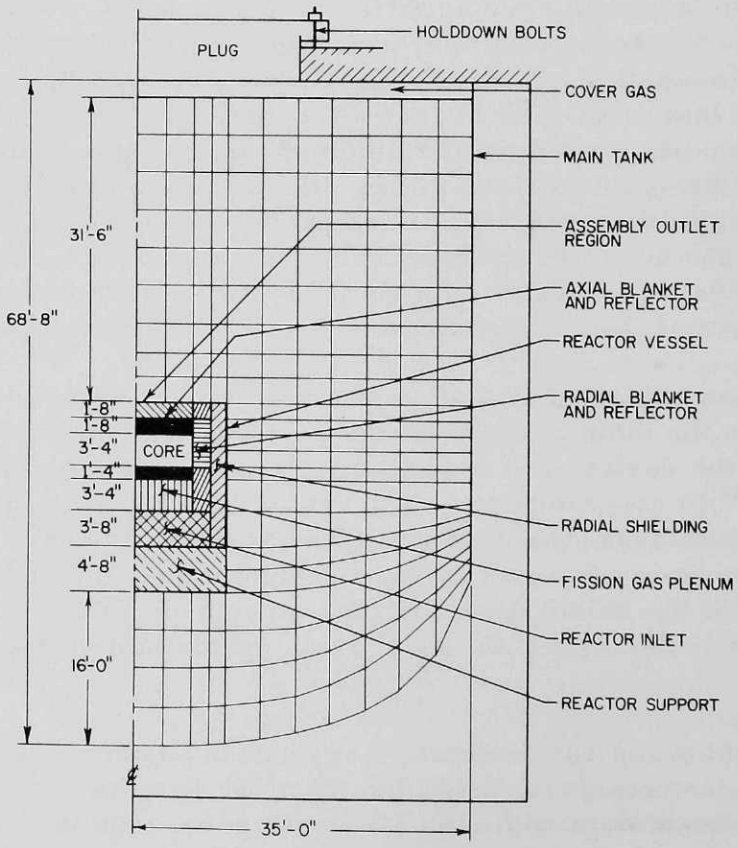


Fig. VII.2. Mathematical Model of Reference Reactor with 70-ft-dia Main Tank for Containment Calculations

mesh used in the REXCO containment code. The vessel chosen had a diameter of 70 ft and a total height of 68 ft 8 in. The reactor region includes representations of the core, axial reflectors and blankets, radial reflectors and blankets, fission-gas plenum, core support structure, and reactor vessel. Between the top of the sodium pool and the bottom of the shield deck and plug structure there is a cover-gas volume 20 in. deep. This gap is variable from design to design.

Although the plug and shield decks are included in the figure, they were assumed to be rigid in the actual calculational model. The plug is shown attached to the shield deck by a set of holddown bolts. Although the code would be capable of calculating the effect the plug motion has on the shield deck and wall structures, it was felt to be conservative to ignore this effect. The main tank is assumed to be rigidly attached to the shield deck so that neither axial nor radial movement is permitted at this point.

The pressure-volume curve used to describe the core region is shown in Fig. VII.3. This curve represents an adiabatic expansion of the fuel vapor; it has been normalized to the final volume and pressure expected in the core expansion and is an input requirement for the REXCO code. The initial core volume (V), at the start of the calculation, was 10.34 m^3. Based on this volume, the V/V_F ratio is 0.0022, where V_F is the final volume of the expanded core. This particular p-v curve is representative of a core with an average core temperature of 4800°K at the time of disruption. The total energy available in expanding the core to the final quasi-static pressure is about 1400 MW-sec; if the expansion were continued down to one bar, the total energy available would be about 2720 MW-sec. It should be noted that in fact the core is not allowed to expand to this low pressure, and the total work done by the core is substantially less.

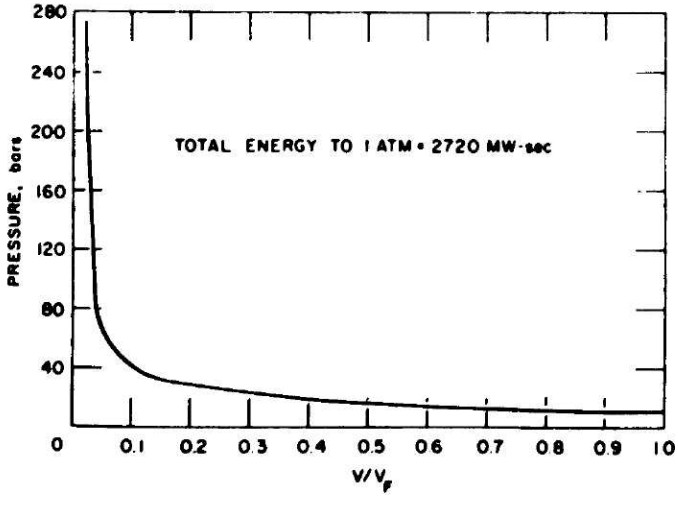


Fig. VII.3

Normalized Core Pressure–Volume Curve for 4800°K Average Core Temperature

3. Model Conservatisms and Implications

There were several conservatisms inherent in this analysis that should be addressed in order to put the following results into the proper

perspective. The conservatisms were dictated both by time and analytical limitations, but were felt not to affect the general conclusions which could be drawn for the pool system. Among the conservatisms were (a) the rigid shield deck-and-plug assembly, (b) neglect of internal shrouds, skirts, baffles, and liners, and (c) neglect of pumps, IHXs, instrument trees, and other energy-absorbing structures.

Neglect of the energy-absorbing capability of the deck-and-plug assembly has essentially two effects on the calculational results. The first is that, since the assembly is considered rigid, the loadings during impact are somewhat overestimated. Thus, calculated upper-tank-wall deformations are probably greater than would be expected if shield deck motion were included, although the amount is not precisely known. Second, when performing independent, decoupled, calculations on the plug and shield deck, there is again a tendency to overestimate stresses and deflections because of the higher force loadings. Further, the energy absorbed by the plug and shield deck, which could be substantial, is neglected.

The internal shrouds, skirts, baffles, and liners were neglected, since these were strongly dependent upon the design. For example, the normalized 1200-MWe EBR-II design included a liner and reactor support skirt that covered almost the entire height of the main tank. The presence of this shell gives a greater effective wall thickness despite a narrow annulus of coolant between the shell and the main tank. Past experience with scale models[27,28] has shown that two shells separated by a small annular gap deform as a shell with a thickness equal to the sum of the individual thicknesses. More will be said about this when we discuss the effect of wall thickness on the calculations.

Lastly, significant internal structures such as the pumps, IHXs, and instrument trees have not been included. Although similar internal components have been neglected in loop-type-reactor calculations without affecting the results, it is not clear that this would hold true for the pool system. Further evaluation of the energy-absorbing capabilities of these components would be useful. Pool-system designs include up to four coolant loops which would need as many as four pumps and eight heat exchangers arranged peripherally near the main-tank wall. These components could conceivably protect the main tank somewhat from long-duration (tens of milliseconds) blast loadings, and would thereby reduce the strains in the tank. The energy difference could be either absorbed in these components or directed to the plug and shield deck.

4. General Results

The following is a description of the general results of a case using the model described previously with a 2-in.-thick Type 304 stainless steel main-tank wall. The purpose of this presentation is to give a general feeling for the results in terms of slug arrival times, energy partitioning, and vessel strains and component loading.

As an indication of the time for the sodium slug to contact the shield deck-and-plug assembly, the average sodium-slug surface displacement as a function of time is shown in Fig. VII.4. The plug impact commences at about 90 msec, with the slug first touching the center of the plug. Full slug contact occurs at about 110 msec, with further axial motion stopped by the rigid shield deck.

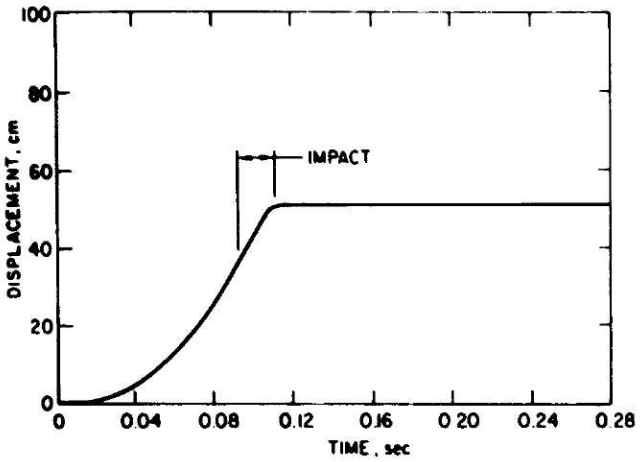


Fig. VII.4

Average Displacement of Sodium-slug Surface as Function of Time after an Energy Release

The core expansion energy as a function of time is shown in Fig. VII.5. This is the work done by the expanding core on its surroundings. The important item to note here is that the expansion energy reaches a peak of about 1400 MW-sec at a time of 280 msec. The leveling off of the core energy release is an indication that the system has reached equilibrium. The average pressure in the system at this time is about 5-7 bar and is considered equal to the final quasi-static equilibrium pressure present within the system at the termination of the event.

Fig. VII.5

Core Expansion Energy as Function of Time after Energy Release

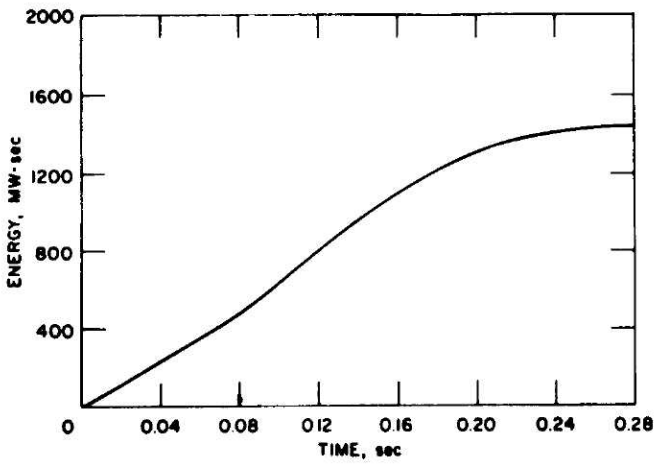


Figure VII.6 depicts the energy partitioning in the various system components during the core expansion. Up to slug impact, at approximately 90 msec, the major portion of the energy is partitioned into the axial and radial kinetic energy (K.E.) of the sodium and other internals. Some energy is absorbed in the reactor vessel. At slug impact, the axial kinetic energy is

transformed into radial kinetic energy of the fluid. This radial kinetic energy finally is absorbed by the vessel wall and appears as the main-tank strain energy (S.E.). At about 280 msec, the main-tank strain energy reaches a maximum while the axial and radial kinetic energy asymptotically approaches zero.

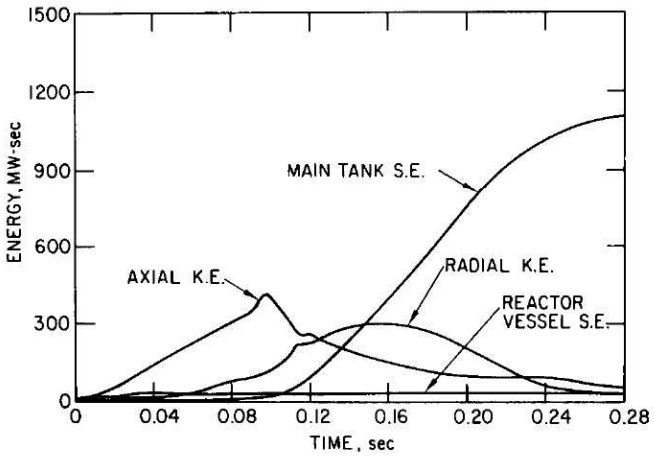


Fig. VII.6

Energy Partitioning during Core Expansion to Equilibrium in 70-ft-dia Main Tank

Some general observations can be made for this particular system. It appears that approximately three-fourths of the actual total core energy release appears as main-tank strain energy at equilibrium. The rest of the energy is partitioned among strain energy in elastic-plastic components, some axial and radial kinetic energy, compressive energy in the sodium, and some energy absorbed in the reactor vessel. This result is borne out by similar parametric studies in loop-type systems and is helpful in extrapolating results to show the effects of parametric variations. These will be discussed further in the following Sec. VII.D.5. It should be noted that design variations in the reactor region may allow significant energy absorption at this point while providing protection for the main tank. The EBR-II concept, with its enclosed reactor vessel, would prove effective in this connection, although it is not clear at this time if other problems may thus be created. A further study is needed.

5. Parametric Variations

From a designer's viewpoint, one of the more useful exercises which can be performed when studying the generic characteristics of the pool system is the effect of parametric changes on the containment response. It should be pointed out that although an exhaustive parametric study was not undertaken, general trends were deduced from some relatively detailed calculations and supportive scoping analysis. Aside from the sensitivity of the HCDA source to input parameters, the other parameters which were looked at briefly, include (a) equivalent wall thickness, and (b) main-tank diameter. This is by no means a complete list of interesting parameters, but some knowledge of the effect of varying these two should prove helpful in determining the overall response of a given, specific system and in denoting possible effects of design changes on containment response.

a. Equivalent Wall Thickness

It was obvious in the initial review of the current reactor designs that a wall thickness representing only the main tank would not suffice. Although the reference reactor system described earlier had a nominal main-tank wall thickness of 2 in., several important shells and skirts were neglected. For example, in the scale-up of the EBR-II reactor a $1\frac{1}{2}$-in.-thick steel liner is part of the reactor-support structural system. The steel liner is separated from the main-tank wall by a relatively narrow annulus filled with sodium. Further, in the SuperPhenix design, the thermal baffling necessary to keep the main-tank wall cool during operation is composed of several cylindrical shells in addition to the full liner which separates the hot-center-pool sodium from the cold-side coolant.

What is the significance of these liners and baffles in terms of energy absorption? It was found in scale-model containment tests for the Fast Test Reactor, performed by the Stanford Research Institute for the Hanford Engineering Development Laboratory,[27] that an inner shell, such as the thermal liner which is separated from the main tank by a relatively thin, sodium-filled annulus, would deform just like the reactor-vessel wall under HCDA loads. Further, in performing REXCO calculations,[28] it was found that a good comparison could be made between the experiments and the calculations if the thickness of the thermal liner and reactor vessel wall were added together to form an equivalent vessel wall thickness. Physically this approximation is valid due to the good structural coupling between the shells by the relatively incompressible sodium in the annulus. Thus, when calculating the energy-absorption capability and final vessel strain of the main-tank wall, one should include all baffles and liners and thus define an equivalent wall thickness.

The method used to arrive at this variation in this study was as follows. Two calculations were performed. The first assumed a tank of Type 304 stainless steel at about 800°F and with a wall thickness of 2 in. The second used a similar tank which turned out to have an equivalent wall thickness of about 8 in. Thus, the equivalent wall-thickness range of possible interest was essentially bracketed. The energy absorbed in the 2-in.-wall tank was about 1100 MW-sec for an actual total core energy release of 1400 MW-sec. The other tank had a wall energy absorption of about 700 MW-sec for an actual total core energy release of 1000 MW-sec. The same initial core volume ($10.3\ m^3$) and average core temperature (4800°K) were used in both cases (as well as in all other cases discussed). The maximum tank-wall hoop strain for the 2-in. wall was 10.6%; the equivalent 8-in. wall had a maximum hoop strain of 2.6%. Longitudinal strains in each case were negligible for these preliminary calculations. One interesting fact should be noted from the above. For thicker equivalent wall thicknesses, the actual total core energy release is reduced. The reason for this is that the equilibrium pressure increases as the wall thickness increases, thus limiting the total work done by the core during expansion. Although this may not be too much of an issue with wall deformations, the higher equilibrium pressure may be important in shield deck-and-plug design.

Figure VII.7 shows the variation of maximum tank hoop strain as a function of the equivalent wall thickness for the 70-ft-dia tank. This curve is dependent, of course, upon the assumed stress-strain properties of the steel. As the equivalent wall thickness is increased, the maximum vessel strain decreases, approaching zero asymptotically. Similarly, the total energy release decreases as well, but this is not only a function of the vessel-wall deflection, but of the cover-gas volume between the top of the sodium pool and the bottom of the shield deck. In this particular set of calculations, in which a 70-ft vessel diameter was assumed with a 20-in. inert-cover-gas gap, the actual total core energy release would be about 600 MW-sec assuming zero wall deformation.

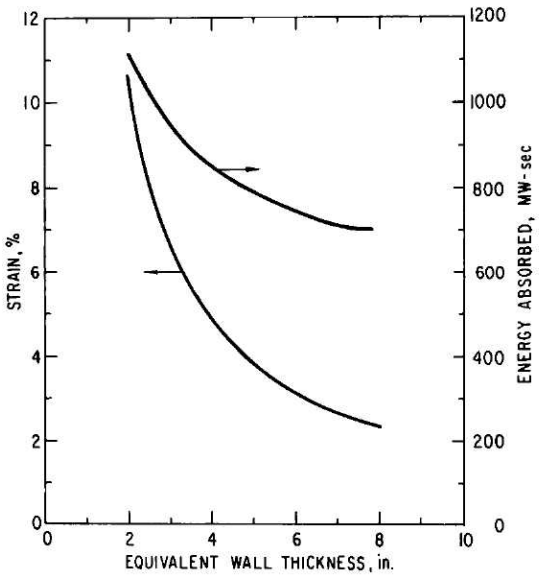


Fig. VII.7. Maximum Vessel Hoop Strain and Absorbed Energy vs Equivalent Wall Thickness for 70-ft-dia Main Tank

What does this curve mean for the several designs that are currently being considered? As noted earlier, both the EBR-II scale-up and the SuperPhenix reactor have two or more tanks. The walls of these tanks together would give an equivalent wall thickness of about $3\frac{1}{2}$-4 in. Referring to Fig. VII.7, this would correspond, roughly, to about a 5% maximum main-tank strain. Another inch of equivalent wall thickness would decrease the maximum tank strain to about 4%. These numbers are not dissimilar to results of calculations of this type that have been performed for much smaller systems with correspondingly smaller energy releases.[29]

b. Primary-tank Diameter

An interesting calculation was performed to ascertain what effect a large variation in tank diameter would have on vessel deformation and the loadings expected on the shield deck and rotating-plug region. The calculation was made using exactly the same reactor as for the 70-ft-dia tank, but the diameter was halved to 35 ft. The mathematical configuration used in the calculation is shown in Fig. VII.8. The reactor region, including the core, blankets, fission-gas plenum, reflectors, and assumed core support region is the same as that shown in Fig. VII.2. Here, however, the main primary tank has been reduced in diameter and a suitable approximately ellipsoidal shell was designed to represent the bottom of the tank. The HCDA source was the same as that used in the previous case, and the main-tank wall thickness was again 2 in.

The energy partitioning is shown in Fig. VII.9. The slug impact occurs at approximately 60 msec, as compared to 90 msec for the

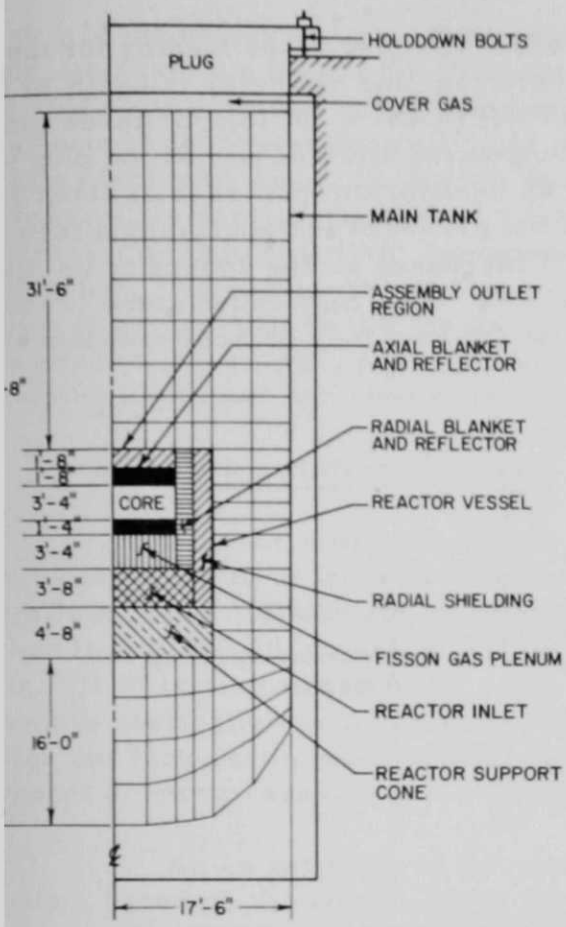


Fig. VII.8. Mathematical Model of Reference Reactor with 35-ft-dia Main Tank

70-ft-dia tank. This difference is due to a "gun barreling" effect acting in the smaller vessel, which translates into a higher sodium-slug velocity and hence an earlier impact time. Equilibrium is reached at about 160 msec, as compared to 280 msec for the larger tank. The actual total energy release from the core during expansion is about 775 MW-sec. The main tank absorbs about 640 MW-sec maximum, a somewhat higher percentage than usual, but in the generally expected range.

The maximum main-tank hoop strain was found to be 13.9% for the 35-ft-dia tank as opposed to 10.6% for the 70-ft-dia tank. This represents about a 31% larger strain for the smaller tank. Significant differences were also noted in the plug and shield deck loadings. The force loading on the 30-ft-dia plug for the 70-ft-dia tank is shown in Fig. VII.10. The peak force is about 690 MN (1.551×10^8 lb_f); the force then diminishes in a triangular shape to a quasi-equilibrium force of about 90 MN (2.023×10^7 lb_f), which

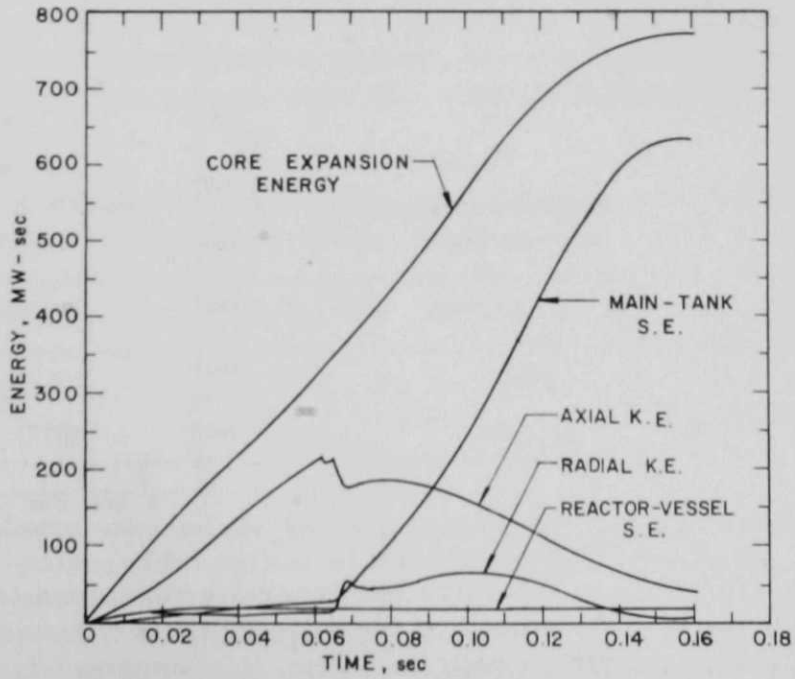


Fig. VII.9

nergy Partitioning during Core Ex-
ansion to Equilibrium in 35-ft-dia
lain Tank

corresponds to a pressure of about 5-7 bar. The plug force loading for the 35-ft-dia tank is shown in Fig. VII.11, where the plug diameter is again 30 ft. Here, the peak impact force is about 2000 MN (4.451×10^8 lb_f), or three times that for the larger tank. Similarly, the long-term force acting on the plug is three times as large and corresponds to an equilibrium pressure of about 20-25 bars. It should be noted that although the pressure is reduced by a factor of three, the total force on the shield deck increases as the square of its diameter, thus giving a total force increase of 33%. This may cause some problems in shield deck response, but it appears that the load may be accommodated by current designs.

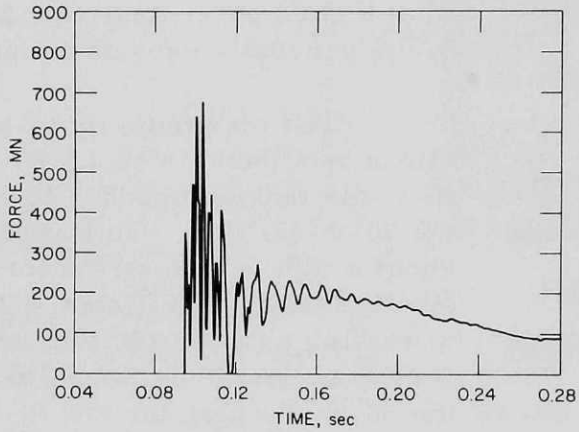


Fig. VII.10

Plug Force-loading History: 70-ft-dia Tank

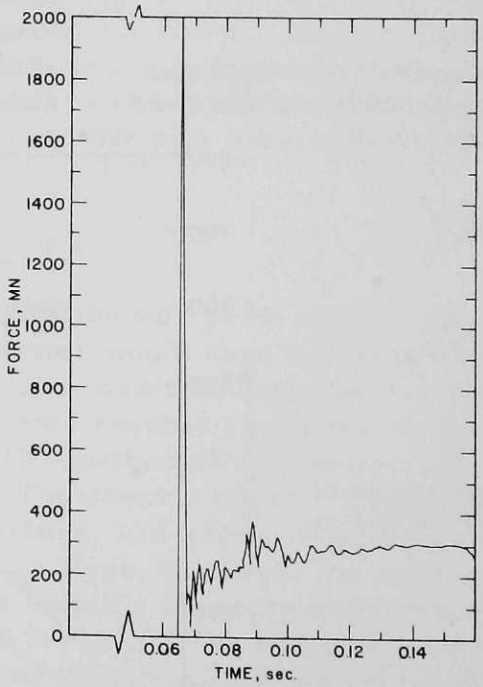


Fig. VII.11

Plug Force-loading History: 35-ft-dia Tank

What are the general conclusions to be drawn from these calculations? Assuming that the reactor region, source term, wall thickness, and cover-gas depth remain the same, it is apparent for a smaller diameter tank

that: (1) the actual energy released is reduced, since the smaller diameter tank can sustain a higher equilibrium pressure for the same wall thickness; (2) slug velocities and impact pressures increase while the time at which impact occurs decreases due to a "gun barreling" effect; and (3) maximum vessel strains increase due to the higher impact pressure and higher equilibrium pressure. In varying the vessel diameter, the designer must accommodate the necessary primary-system components, such as pumps, IHXs, and fuel handling equipment. He should be aware, as demonstrated by these calculations, that there are trade-offs in the design. Although time did not permit investigating the effect of cover-gas depth on the structural response, it is realized that this could be important and should be considered.

6. Deck and Plug Response

Several scoping calculations were made to determine, if possible, the safety margins of the shield deck and rotating-plug structures in response to HCDA loads. The basis for this analysis was the time-dependent loading given in Fig. VII.10. It should be noted that the force-time loading depicted in Fig. VII.10 is an overestimate, since the plug and shield deck were not allowed to move vertically during impact. Since the sodium is relatively incompressible, the long-term force loading could be reduced, and a potentially large amount of energy absorbed, if shield deck deformation were allowed.

As an estimate of the plug response to the force loading, a bolting system based on the Clinch River Breeder Reactor (CRBR) approach was assumed. The CRBR design includes 92 bolts of Type SA-453 steel arranged on a 273.8-in. diameter. The bolts were 3 in. in diameter and 60 in. long. The cross-sectional area per bolt was 7.068 in.^{2} With an assumed bolt-circle diameter of 30 ft for the pool plug, which nominally was used in some of the normalized 1200-MWe designs, the corresponding number of bolts is 120, giving a total bolt area of 848 sq in. Next, the force-versus-deflection curve for the holddown bolts was determined. Some values from the curve are listed in Table VII.1.

TABLE VII.1. Calculated Plug Holddown-bolt Force and Deflection Data: 70-ft-dia Tank

Force, MN	Deflection, cm	Force, MN	Deflection, cm
383	0.52	528	2.44
487	0.97	558	6.096

The maximum force exerted on the plug is 690 MN, but this force occurs in a string of small-pulse-width spikes which can be neglected as a first approximation. If one assumes that at worst a force of 300 MN is applied over a time span of 20-30 msec, then the maximum elongation is about 0.16 in., and the bolts are still in the elastic region. Thus, it would appear that for a

bolting system with strength characteristics similar to that for CRBR, there would possibly be reasonable assurance that the plug would remain in place during and after an HCDA. It is not clear at this time what the requirements would be on total sodium leakage during and after an HCDA.

The shield deck posed a somewhat more formidable analytical problem, since it was difficult to determine a reference design which provided sufficient detail. To gain some insight as to what might be expected, a design nominally taken from the EBR-II scale-up was used. The main features of the structure include 15 radial beams with webs 10 ft high and 3 in. thick. The flanges were assumed to be 12 in. wide and 4 in. thick. The beams were nominally about 16.5 ft long. The effect of concrete was neglected while the effect of a 4-in. (10.2-m)-thick base plate (a nominal thickness which could be reduced) was included. The analysis assumed, conservatively, that the beams were simply cantilevered with built-in ends. This neglects the stiffness and support provided by the central ring girder, which ties all the beams together. The maximum stress calculated for the structure was about 31.5 ksi for a uniform shield deck pressure load of 100 psi. From the calculation on the 70-ft-dia tank, the quasistatic pressure was about 100 psi. If a high-strength steel were used, with a yield approaching 80-90 ksi, a considerable margin would exist. The margin probably could be enhanced by including reinforced concrete in the structure. The deflections calculated were also small; however, the dynamic structural response due to the slug loading was not calculated.

Using the results of some scale-model tests and finite-element calculations, the French have shown that the SuperPhenix plugs would deflect over 20 in. under HCDA loads.[30,31] It is not clear at this time what magnitude of load was used in the analysis, but it should be related to the 800-MNsec energy release which has been assumed for the SuperPhenix containment evaluation. This energy release was based on a coherent fuel coolant interaction with a relatively high conversion efficiency. Given the large deformations calculated, it is clear that the loading of the shield deck and plug structure was quite high, but it is not clear what role the explosive tests played in this evaluation.

E. Summary and Conclusions

Based on the foregoing qualitative and quantitative evaluation of the safety aspects of the pool-reactor concept, the results and conclusions can be summarized as follows:

1. There are several generic characteristics of pool reactors which may be suitable to help eliminate, mitigate, and contain both likely and very unlikely accident events. Among these potentially inherent safety characteristics are: (a) a large sodium inventory to act as a heat sink and protect system components from thermal shocks, (b) a large main tank at relatively low operating temperatures and with no piping penetrations beneath the sodium

surface, (c) "cold-leg" pumps with short piping runs to the reactor inlet plenum submersed in the cold inlet sodium pool, and (d) a large cross-sectional area of the main tank to help dissipate and absorb HCDA above-core bubbles and reduce both transient and long-term loads and to provide the potential for debris retention and postaccident heat-removal capability.

2. Preliminary containment calculations using somewhat conservative assumptions indicate that the pool concept, because of the above generic characteristics, can accommodate significant energy releases from an HCDA. The study gives a reasonable indication that the main-tank, deck, and plug structures can be designed to survive the given HCDA loads without failure of the main tank or significant breach of the shield deck.

3. Both the cold-pool and hot-center-pool systems can accommodate loss-of-piping-integrity (LOPI) accidents without losing primary-system coolant. The probability exists, based on steady-state scoping calculations which should be the subject of further study, that both systems can provide sufficient core flow to avoid boiling. Further, with simultaneous reactor and pump trip at pipe rupture, longer pump coastdown times could be incorporated to aid initial decay-heat removal and institute natural-convection cooling.

4. Designing the system to provide adequate natural-convection cooling upon loss of all pumping power with reactor scram appears feasible for intact geometries. Preliminary calculations (see Sec. VI.E) have shown that both the cold-pool and hot-center-pool designs can provide adequate cooling margins.

5. Preliminary assessment of the current pool designs (work done outside this study[32]) indicate that in-tank cooling should occur for a significant quantity of core debris following an HCDA. It is not clear at this time that the entire core inventory is coolable without the presence of an engineered core-retention device. The fraction of the core debris which is coolable by inherent processes is likely to be, to a significant degree, a function of the radius of curvature of the vessel bottom.

VIII. CONSTRUCTIBILITY

This section contains considerations that need to be addressed in the construction of a pool LMFBR. Recommended design approaches are offered to enhance cost-effective construction features. Descriptions of fabrication and erection methods for EBR-II, Phenix, and PFR are included to assist the reader to visualize the constructional needs and some solutions already accomplished.

A. Construction Considerations

The design of any nuclear reactor primary system requires careful planning to assure the requisite quality of installation and achievement of design objectives. The main features concerning construction that distinguish the pool system are the compactness of the primary system and the relatively large size of the main structural components such as the main tank and the shield deck. These features make it even more important that very careful planning of the construction of the system begin at the outset of design and remain integrated with the design effort. Recognition of the iterative nature of the construction and design of a pool system is essential to achieve improved economy and quality of construction.

The following listing is believed to represent the major considerations involved in construction of the pool LMFBR. Each consideration is discussed following the listing.

- Incorporate specific construction needs into design.
- Incorporate specific inspection and testing needs into design.
- Perform detailed planning of construction methods and sequence.
- Coordinate design and construction of primary system with structures and equipment immediately surrounding the pool.
- Maximize the use of shop prefabrication.
- Avoid specification of unrealistic, restrictive tolerances, especially on the large, massive, main structural components.
- Plan to minimize the amount of field welding and locate welds in areas which facilitate field welding.

1. Incorporate Construction Needs into Design

To implement this effectively, the designer must have a good understanding of the exact construction methods and sequences available for LMFBR pool systems. It may be necessary for the design specifications to spell out what methods and sequences are to be employed. Such an approach has been used--with good results--but only when detailed input from qualified

vessel manufacturers and constructors was obtained and followed. For EBR-II, very detailed erection sequences were specified both in narrative and graphic form to assure not only proper vessel erection but also effective coordination with adjacent (and sometimes concurrent) construction. We understand that the construction and design forces for Phenix employed a similar approach. It must be recognized that this whole approach is an iterative one; freezing of certain design aspects too early could result in unnecessary restraints both on other design features and on options for construction methods.

Design of certain components will not only affect design of other areas, but will probably set the basic construction sequence. For example, the use of a main-tank cover (such as in EBR-II and Phenix) permits different choices as to types of tank support, and thus subsequent field erection procedures may be different. With the elimination of the main tank cover (such as in PFR and SuperPhenix), the vessels must be supported by gastight support skirts welded to the shield deck, and construction procedures will be correspondingly affected.

2. Incorporate Inspection and Testing Needs into Design

Many of the comments made above on construction needs also apply to this area. The main emphasis in this area is perhaps the need to provide in the design adequate means for in-service inspection and testing. This area is a subject of current interest in LWR construction, and the basic principles generally may apply to the LMFBR. Two factors seem unique to the pool system in this area:

i. A considerable amount of the primary coolant boundary (main tank) and guard tank will be field welded, and the ability to inspect, both initially and subsequently in-service, must be provided in the design.

ii. Most, if not all, of the present ASME Code inspection and testing requirements on LMFBRs apply to loop or piped systems. Anticipating the nature of future code (and NRC) requirements concerning inspection and testing of LMFBRs will be necessary. At this time, this area must be considered an open question; for example, if no main tank cover is used, even the boundary of ASME code coverage is not clear.

3. Perform Detailed Construction Planning

The importance of this area is perhaps obvious. What requires emphasis, however, is that the amount of detail such planning should achieve is quite significant if maximum benefit is to be obtained. The designer, vessel manufacturers, and general constructors should collaborate in this planning, to assure the best of input and to improve understanding of the plan's purposes. The plans should be formalized; in some cases, perhaps they should be shown in drawings.

One aspect of the detailed planning referred to here is that it should become a contractual requirement on all contractors involved. Experience proves the value of not only gaining input from experienced vendors, but also of ensuring that all parties understand the nature and purpose of the methods and sequences selected.

Sections VIII.B-VIII.D below contain brief descriptions of construction of EBR-II, Phenix, and PFR. For all these reactors, considerable effort was devoted to construction planning, and to the best of our knowledge, such plans were by and large followed as prepared.

4. Coordinate Primary System Design with Surrounding Structure

The pool system is a compact design, and every effort should be made to allow free access to the main pool area during construction. To a large extent, this may dictate (and has dictated) the design of surrounding structures, shields, and equipment--not only to facilitate access, but also to minimize construction time.

5. Maximize Shop Prefabrication

Since the pool system utilizes numerous large structural components, efforts to reduce field erection time and cost should prove of great value. Shop prefabrication will contribute to such savings. Several examples of specific prefabricated parts are discussed in Secs. VIII.B-VIII.D below.

It is interesting that the structural concepts of the shield deck for EBR-II, Phenix, and PFR are quite similar. Although the exact arrangement of such parts as the circular support rings and ring girders varies, the ability to prefabricate is preserved and was used in all these reactors. Figure VIII.1 shows the basic prefabrication shape (or segment) possible for the shield deck and for the main-tank cover if used. The segments are made as large as shipping limitations will permit.

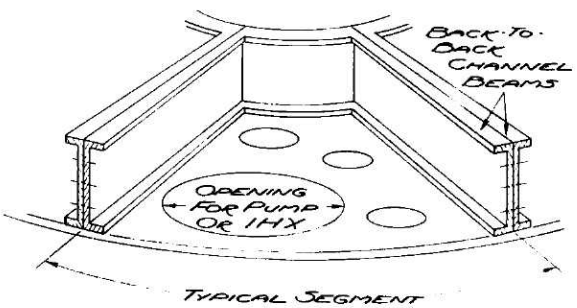


Fig. VIII.1

Prefabricated Segment for Shield Deck

6. Avoid Unrealistic and Restrictive Tolerances

Because of the massiveness of the main structural elements, this consideration can have very significant impact on project cost. The main tank and the guard tank are primarily liquid containers. Tolerances for all

but the reactor support structure may be only slightly more restrictive than that used in large field-erected storage vessels.

7. Facilitate Field Welding

The main tank and the guard tank for a large LMFBR pool-type plant will be large: 70 ft in diameter and 500-600 tons each. The average metal thickness of the tank walls will probably be of the order of $1\frac{1}{2}$ to 2 in. Welding of austenitic stainless steels of thicknesses greater than $1\frac{1}{2}$ in. does not require stress-relieving under current rules, although local field-stress-relieving in the field can be, and has been, successfully accomplished if it is desired or required. It should be noted that welded free-standing carbon steel reactor containment vessels of thicknesses near those above have been successfully erected. Also, stainless steel welded vessels of 6 in. thickness and about 18 ft in diameter have been successfully constructed; in addition, stainless steel vessels of thickness slightly greater than 1 in. and about 62 ft in diameter have been field erected.[36]

The amount of field welding for the major structural components will be significant. Efforts that can pay off in cost savings and improved weld quality include locating weld joints for easy access for welding, minimizing the shell-plate thicknesses at the welds (by using transition plates or forgings to develop locally thicker plates), and assuring correct and uniform quality of welding and plate material. One of the more important aspects in this area is that, with the annulus between the main tank and guard tank generally 12 in. or less, it will be necessary to erect the vessels separately. (Some designs, such as SuperPhenix, have a gap of about 2 ft.) Therefore, the capability must be provided to handle vessels weighing up to 600 tons.

B. Fabrication and Erection of EBR-II

For EBR-II, as well as for other pool LMFBR systems, the design is greatly influenced by considerations of fabrication and erection. This section describes briefly the salient features of the fabrication and erection of EBR-II. Figures VIII.2-VIII.4 are overall plans and cross sections that identify the major areas discussed. The material is essentially that which appeared in Reference 33.

1. Fabrication

a. Main Tank

Because of the size and complexity of the main-tank cover, it was fabricated separately from the remainder of the tank to assure adherence to schedule. The Rock Island Arsenal in Illinois, and, later, the Watertown Arsenal in Massachusetts, fabricated the two cover halves complete and shipped them to the site in Idaho. The remaining portions of the main tank were fabricated by the Chicago Bridge and Iron Company, which also field erected the main-tank assembly.

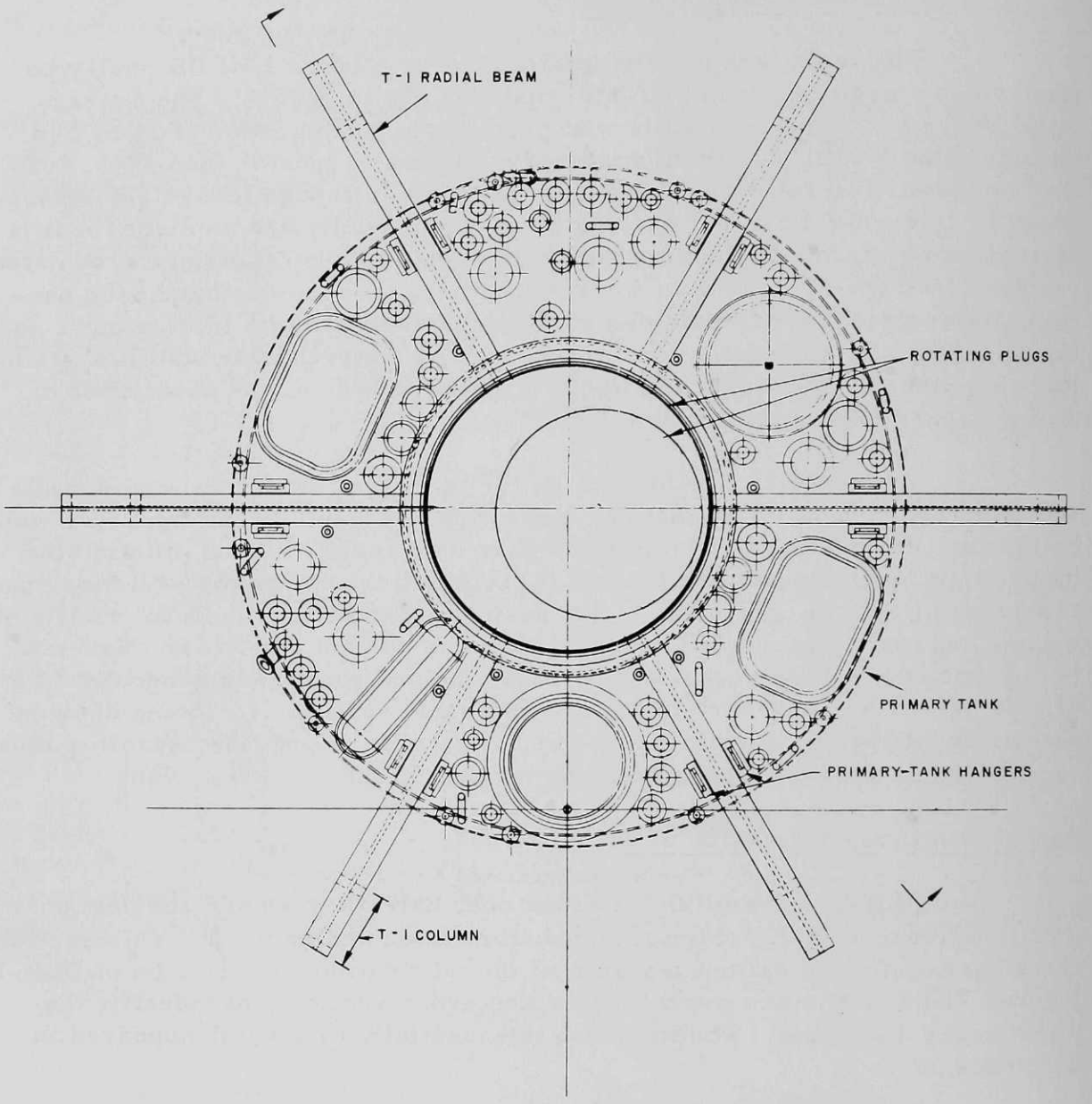


Fig. VIII.2. Plan View of EBR-II Primary Tank (Main and guard tanks, main-tank cover) Support Structure

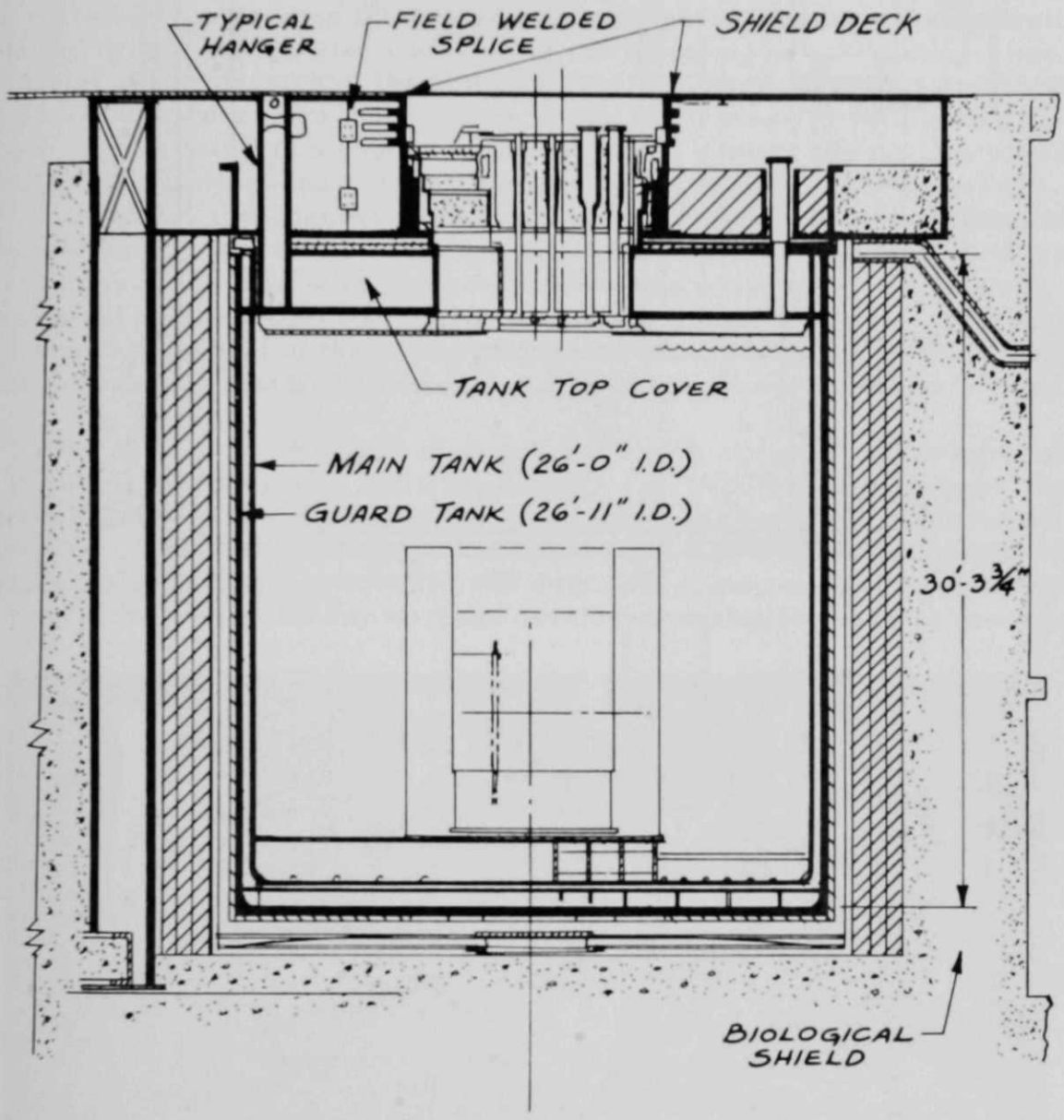


Fig. VIII.3. Cross Section of EBR-II Main Tank and Its Support Structure

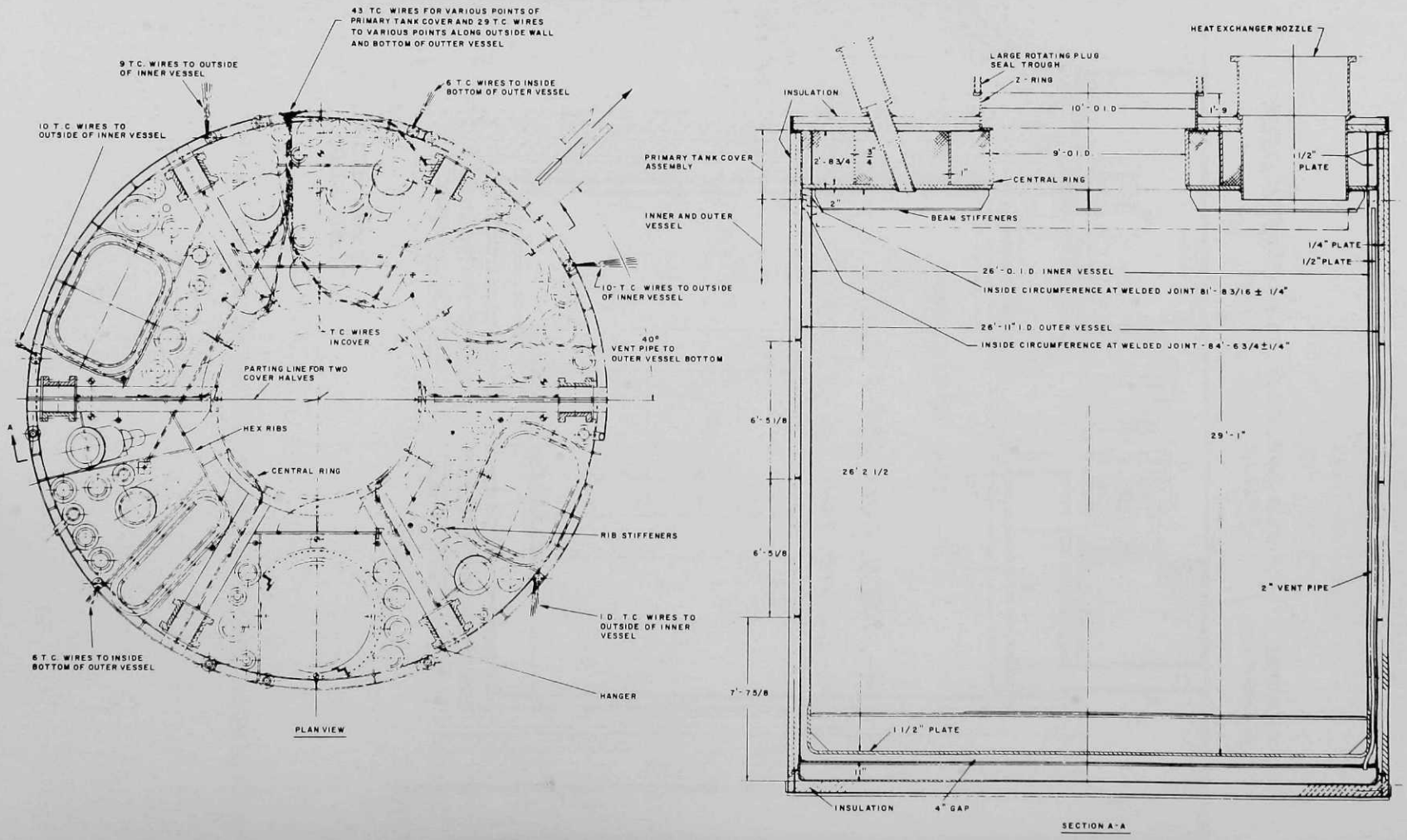


Fig. VIII.4. EBR-II Main Tank Assembly

(1) Cover

Location and alignment are critical for several of the machined nozzles in the main-tank cover and required accurate positioning and machining in the shop fabrication. In addition, the numerous reinforcing ribs and nozzle guide plates complicated the actual sequence of fabrication. To assist in the planning of the fabrication of the cover, Argonne constructed an accurate quarter-size model of the entire cover made of aluminum and wood. Figure VIII.5 is a photograph of the model without any nozzles present, and Fig. VIII.6 shows the completed model. This model proved to be very useful, especially in uncovering difficult and sometimes virtually impossible areas for proper assembly of the many individual pieces. The yoke and pin assemblies at the tops of the hangers are models of the ones used during transfer of the main tank load from temporary support to the shield deck. A full-size wood model of the actual hanger and roller assembly was also constructed to study design refinements and fabrication and erection problems.

The basic structures for each cover half were fabricated at the Rock Island Arsenal and are shown in Fig. VIII.7. The numerous "Swiss-cheese" plates shown were used for gross positioning of the nozzles during assembly. The 2-in.-thick bottom plates were purchased with the nozzle holes rough cut. A precision theodolite was employed during all phases of assembly to keep close control on any warpage developing during the welding operations.

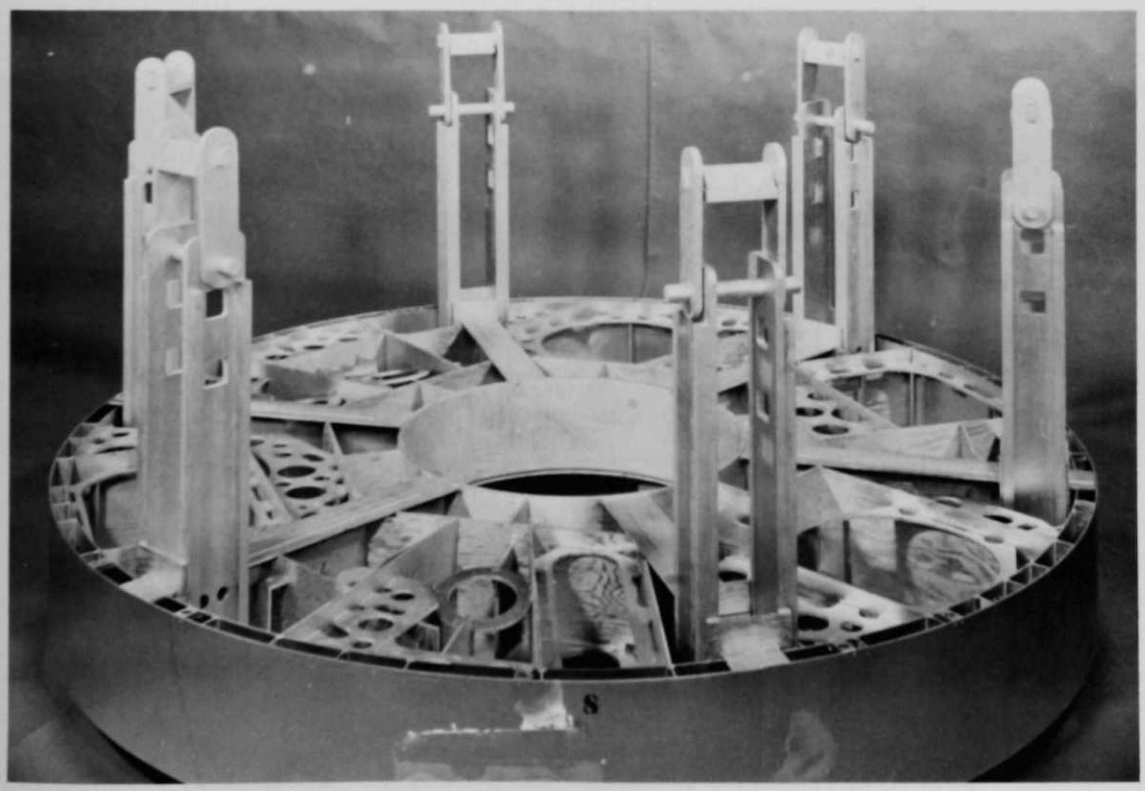


Fig. VIII.5. Model of EBR-II Main-tank Cover without Nozzles. ANL Neg. No. 111-7320.

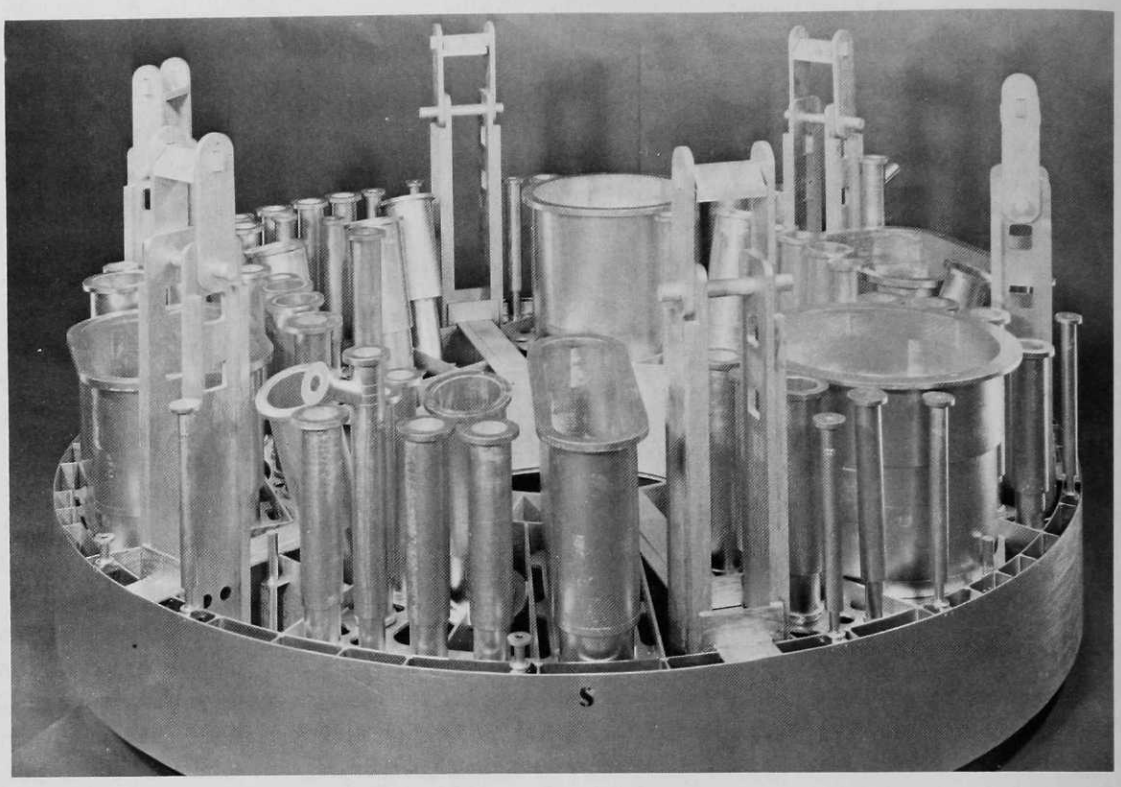


Fig. VIII.6. Model of EBR-II Main-tank Cover with Nozzles. ANL Neg. No. 111-7321.

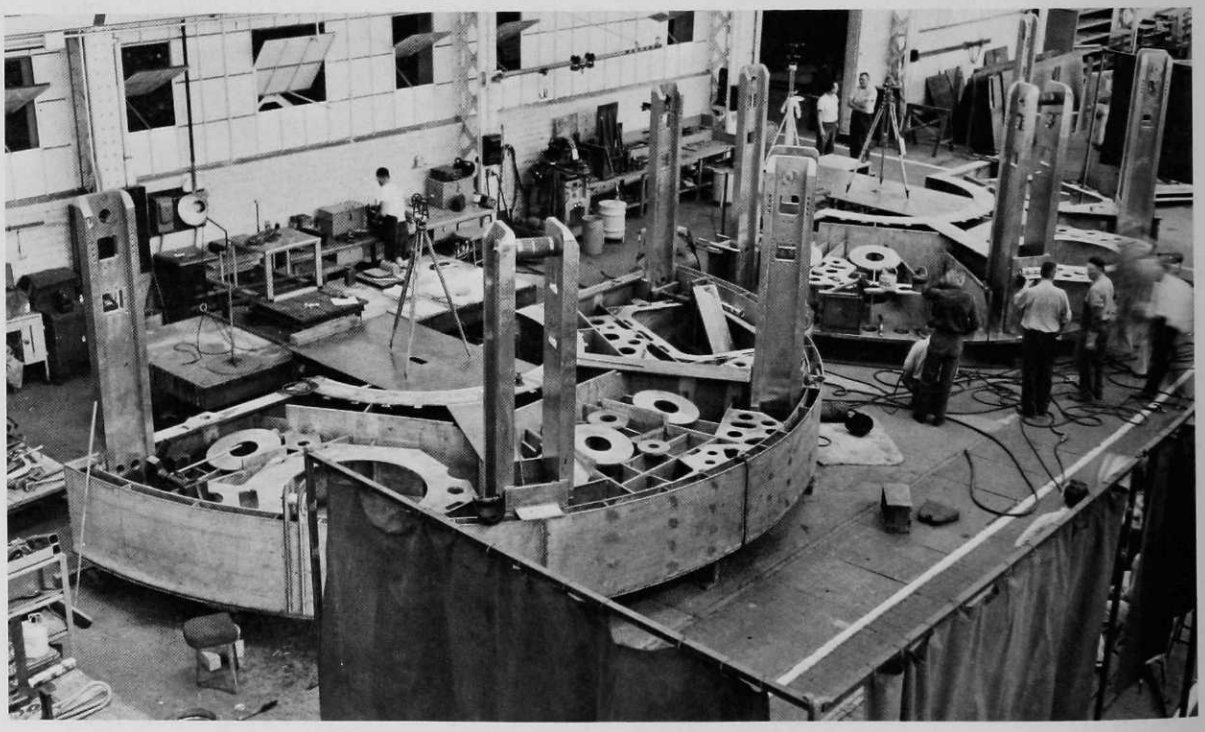


Fig. VIII.7. Halves of EBR-II Main-tank Cover during Fabrication. ANL Neg. No. 201-3339.

Upon completion of the basic cover halves, each half was placed in a furnace, and a partial stress relief was performed at 850°F.* Again to accelerate schedules, one cover half was sent to the Watertown Arsenal (in Massachusetts) for installation of the nozzles and final machining, and the other remained at Rock Island to be completed.

To facilitate field assembly and ensure a tight fit, the mating surface on each cover half was prepared to give metal-to-metal contact for a depth of about 6 in. from the top and 6 in. from the bottom.

Although the same template was used by both arsenals in drilling the bolt holes in the mating surfaces (the matching back-to-back channel beams), the risk was too great to completely rely on accurate machining for this crucial field fitup. To solve this problem, the matching bolt holes were drilled about 1/2 in. oversize, and stainless steel bushings with eccentric holes drilled through were provided for the fitted bolts. Thus, inserting one bushing in each half of the mating surfaces made it possible to line up the bolt holes in the bushings by merely rotating the bushings within the oversized hole. This feature greatly facilitated field assembly, which was accomplished quite easily. Dowel-pin holes and scribe marks were used to accurately align the two halves in the field.

The larger nozzles and those not of circular cross section were fabricated of rolled plates. All the remaining nozzles were made by centrifugal casting. The flange of each was machined before it was inserted into the cover assembly. Each nozzle was secured to the cover at the bottom 2-in. plate by extending the nozzle through the plate and welding with a partial-penetration groove weld with fillet reinforcement. Each weld was individually helium leak tested.

Figure VIII.8 is a photograph of a completed cover half. The large circular nozzle at the right is for the single IHX used on EBR-II; the large rectangular nozzle to the left is for one of the two primary pumps. Figure VIII.9 is a closeup view of this cover half with the pump nozzle at the extreme left of the photograph. The central circular ring and hexagonal ribs that are later connected by a top cover plate in the field assembly are apparent in the figure. Figure VIII.10 is a photograph of this cover half on a special railroad car ready for shipment. The blind flanges shown, used for protection of gasket faces, also provided the seal for a final helium leak test made in the field upon completion of the entire main tank.

(2) Walls and Bottoms of Main Tank and Guard Tank

The bottom of the guard tank was fabricated in two symmetrical halves. A 2-ft width of the top plate was left off to provide access in the field to the butt-weld joining the guard-tank bottom plate. The main

*This temperature was chosen as a value greater than the structure would later experience, but low enough to avoid sensitization of the stainless steel.

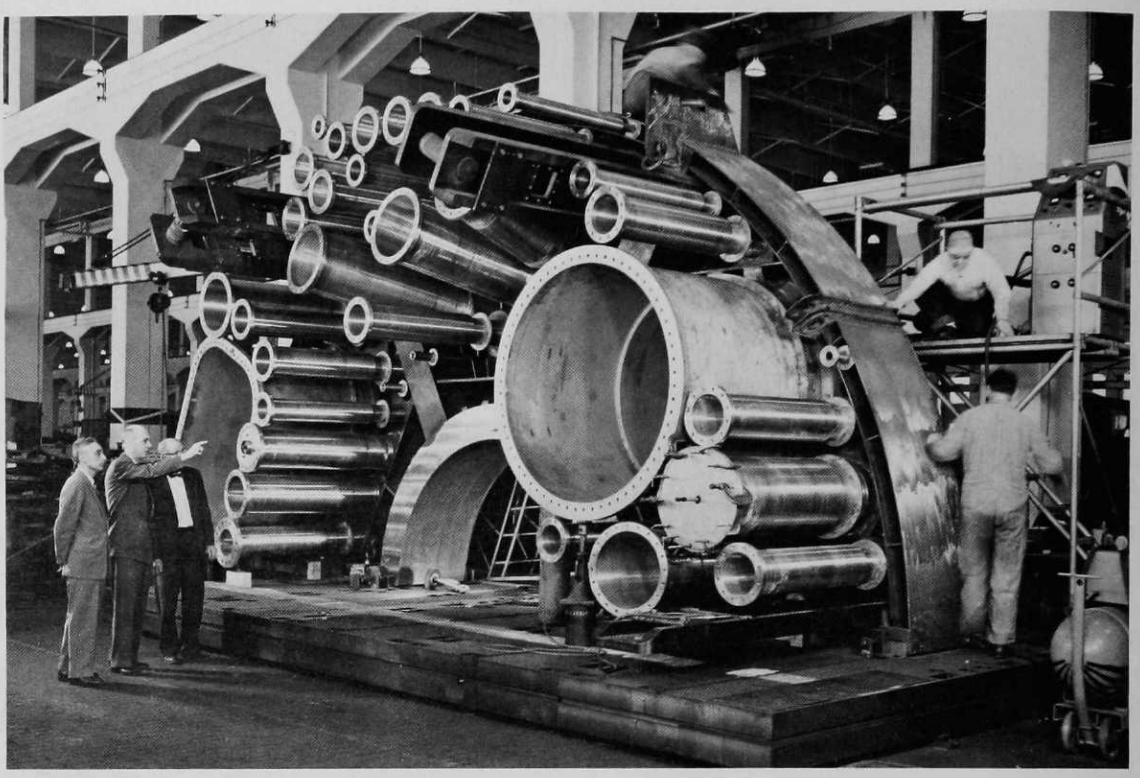


Fig. VIII.8. View of Completed Half of EBR-II Main-tank Cover. ANL Neg. No. 201-3760.

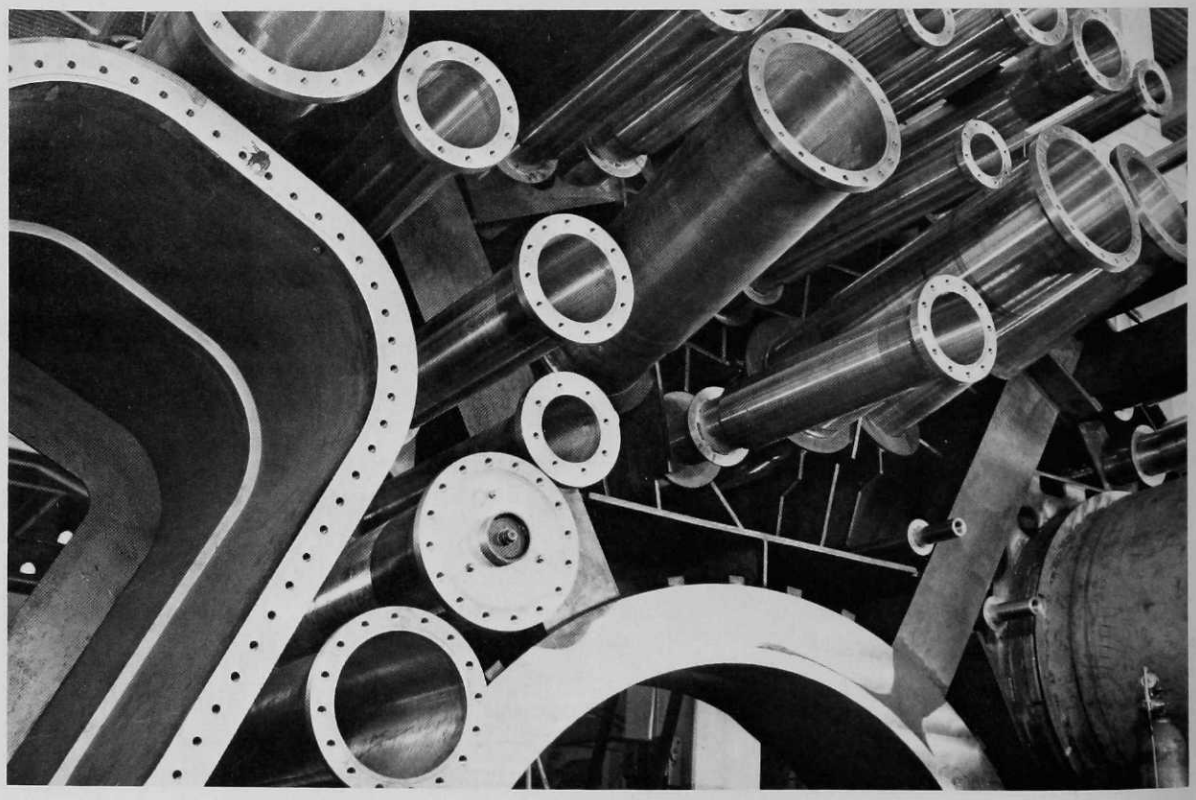


Fig. VIII.9. Closeup View of Completed Cover Half. ANL Neg. No. 201-3765.

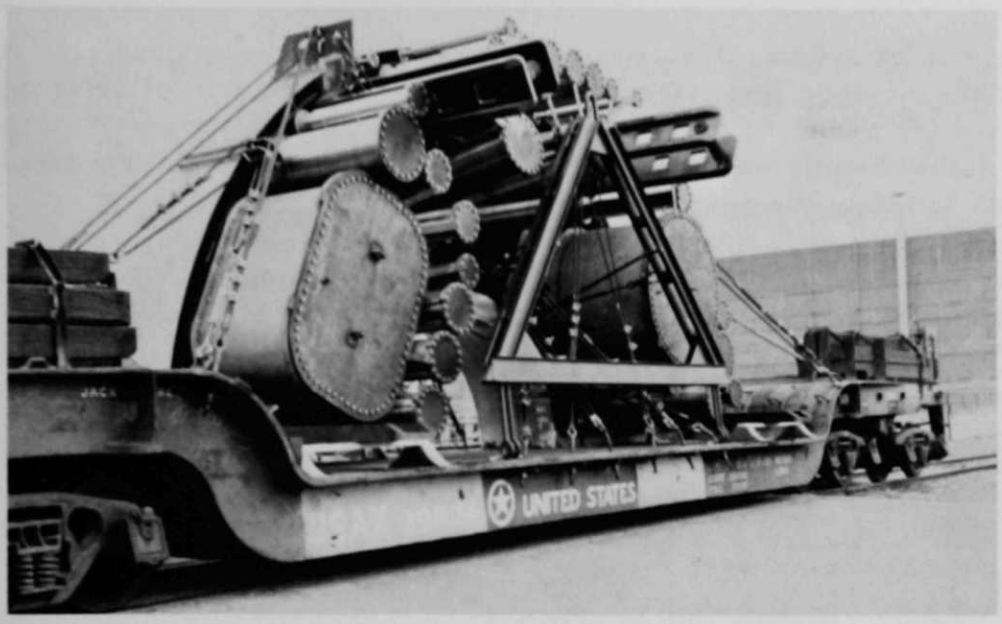


Fig. VIII.10. View of Cover Half on Special Rail Car. ANL Neg. No. 201-3805.

tank bottom was fabricated in three pieces. One piece consisted of the central sections of the bottom including the eight main stiffener beams, a central portion of a 3/4-in. top plate, and a $1\frac{1}{2}$ in. circular plate. The outer portions of the main-tank bottom comprised the other two pieces.

All the bottom-structure weldments were partially stress-relieved at 850°F.

b. Shield Deck

The entire shield deck was fabricated of T-1 Constructional Alloy by the Nordberg Manufacturing Company in Milwaukee, Wisconsin. The columns and bottom beams were required in the field several months before the top structure and were fabricated first. Field erection of the top T-1 structure was by the Graver Tank Manufacturing Company.

All base T-1 material was preheated to about 300°F, with interpass temperatures of 250°F. Both manual and automatic submerged-arc welding were employed.

(1) Top Structure

The machined surfaces in the T-1 central ring were very critical. This machined ring supports the two heavy rotating plugs, which have a combined weight of over 200 tons. The high accuracy required in the support of these rotating plugs is very important, as the plugs house and locate precision-built mechanisms that are essential to the safe and efficient operation of the reactor. As a result of these requirements on the T-1 ring,

considerable effort was devoted to the machining of the central ring in order to maintain the specified tolerances. An elaborate system of reference surfaces used for accurate machining of the ring and shop assembly of the ring and six radial beams was established. The following procedural steps were employed to attain the desired weldment:

(a) The central ring assembly was completely fabricated except for final machining of the ring insert supporting the rotating plugs.

(b) The six radial beams were fabricated with sufficient overrun radially to allow for errors in erecting the T-1 columns.

(c) The six radial beams and the central ring were assembled, aligned, and welded together. Reference points were placed on the bottom of the ring, and from field measurements of the erected T-1 columns the ends of the beams were marked for proper lengths.

(d) Splice plates were tacked to the radial beams, and the beam web and splice plates were match-drilled to ensure reproducibility of the shop assembly later in the field.

(e) The radial beams were cut off at the splice joints, and their ends were trimmed to the field-measured lengths.

(f) The central ring was machined using reference marks for centering and leveling located on the bottom of the ring.

(2) Columns and Bottom Beams

The columns and the bottom beams were fabricated by welding together precut and edge-prepared plates. Automatic welding (submerged-arc) was employed successfully for the bottom built-up channels. However, the use of very heavy (4-in.) plates in the columns proved to be a detriment to the successful submerged-arc welding of the columns. Upon cooling of the welds on the first two columns, several random transverse cracks in the welds appeared. After many attempts to correct this by modifying the welding process, the fabricator finally repaired the cracked welds manually and used manual welds to fabricate the remaining columns. No further difficulty was experienced in cracking of the welds.

2. Erection

The actual assembly and installation of the primary tank,* support structure, and shield deck involved extensive preplanning, detailed engineering design of interdependent components and of integration within massive structures, and finally prefabrication of those units amenable to commercial transport and field erection within the reactor building. Moreover, space limitations within the reactor building (78 ft in diameter) necessitated a sequence of field

*In this section, the EBR-II nomenclature "primary tank" is used to denote the main and guard tanks and their top cover.

erection involving preliminary assembly and alignment of components on temporary staging, and final assembly within the cavity of the biological shield with the aid of the overhead polar crane in the building.

The sequence in which these operations were performed is pictured in Figs. VIII.11 through VIII.25. Shortly after the concreting operations began in the reactor building, the six T-1 columns and bottom channel beams were erected and braced (as shown in Fig. VIII.11). Concrete operations continued with the construction of the radial biological shield. Figure VIII.12 shows the hoop reinforcing for the biological shield and some of the radial offset ducts of the shield-cooling system. Figure VIII.13 is a view of the completed cavity formed by the air-baffle vessel, blast shield, and biological shield. The tops of the T-1 columns may be seen.

The first step in the erection of the primary tank was the assembly of the main- and guard-tank bottoms on the operating floor of the reactor building. Figure VIII.14 shows the bottom of the main tank during welding of the three prefabricated sections. Upon completion of the tank bottoms, both the main- and guard-tank walls were erected and welded concurrently. All butt welds were radiographed simultaneously with erection. Upon completion

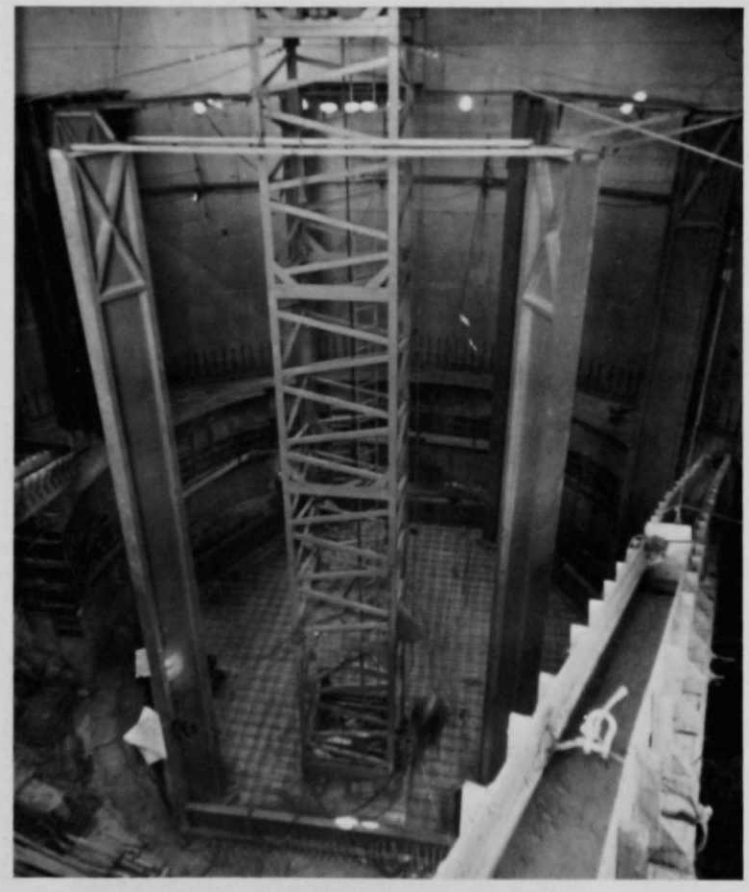


Fig. VIII.11. View of EBR-II T-1 Columns in Place. ANL Neg. No. 103-5206.

Fig. VIII.12. Construction of EBR-II Biological Shield. ANL Neg. No. 103-5270.

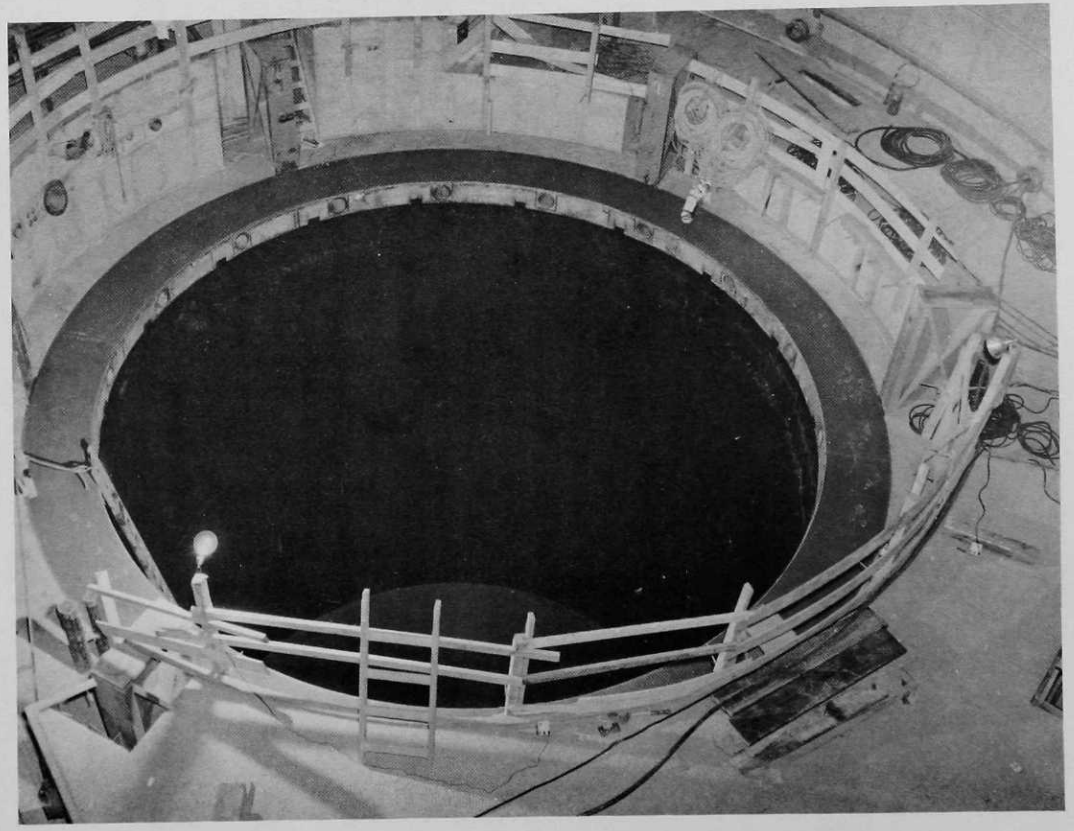


Fig. VIII.13. EBR-II Blast-shield Cavity. ANL Neg. No. 103-5676.

Fig. VIII.14. EBR-II Main Tank Bottom under Construction. ANL Neg. No. 103-5700.

of the guard tank, the bottom insulation and stainless steel pan were installed. Although a helium leak test was required for the entire primary tank, the guard-tank bottom, because of its sandwich construction, was able to be tested before completion of the tank. The guard tank was then lowered into the cavity, and its weight was transferred from the crane to three temporary support beams cantilevered out from the operating floor of the reactor building.

While work progressed on the main tank, the primary-tank cover halves were brought into the building for assembly into one piece (as shown in Fig. VIII.15). Space limitations reached their peak once the two cover halves were placed on the operating floor and work was progressing on all three components (see Fig. VIII.16). At the time this photograph was taken, the cover halves were being welded together, work on the main tank (to the left) was being completed, and thermocouples and wiring were being installed on the guard tank (right). In addition to the obvious problems of coordination and scheduling, the primary-tank contractor was required to complete the entire job, including hanging the tank from the T-1 floor support structure, in three months, which was accomplished.

With the guard tank suspended in the cavity, the completed main tank was lowered into the guard tank and cribbed up on the bottom of the guard tank about 5 ft to put the main tank about shoulder height when standing on top of the blast shield (as shown in Fig. VIII.17).

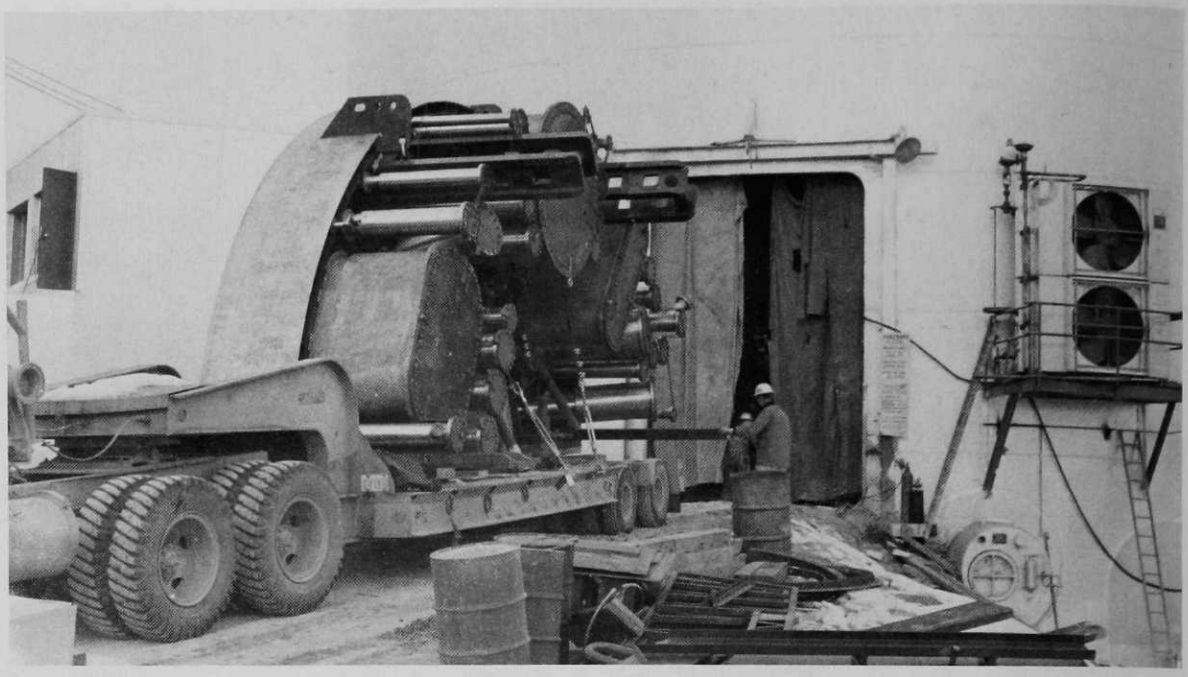


Fig. VIII.15. Moving Cover Half into EBR-II Reactor Building. ANL Neg. No. 103-5709.

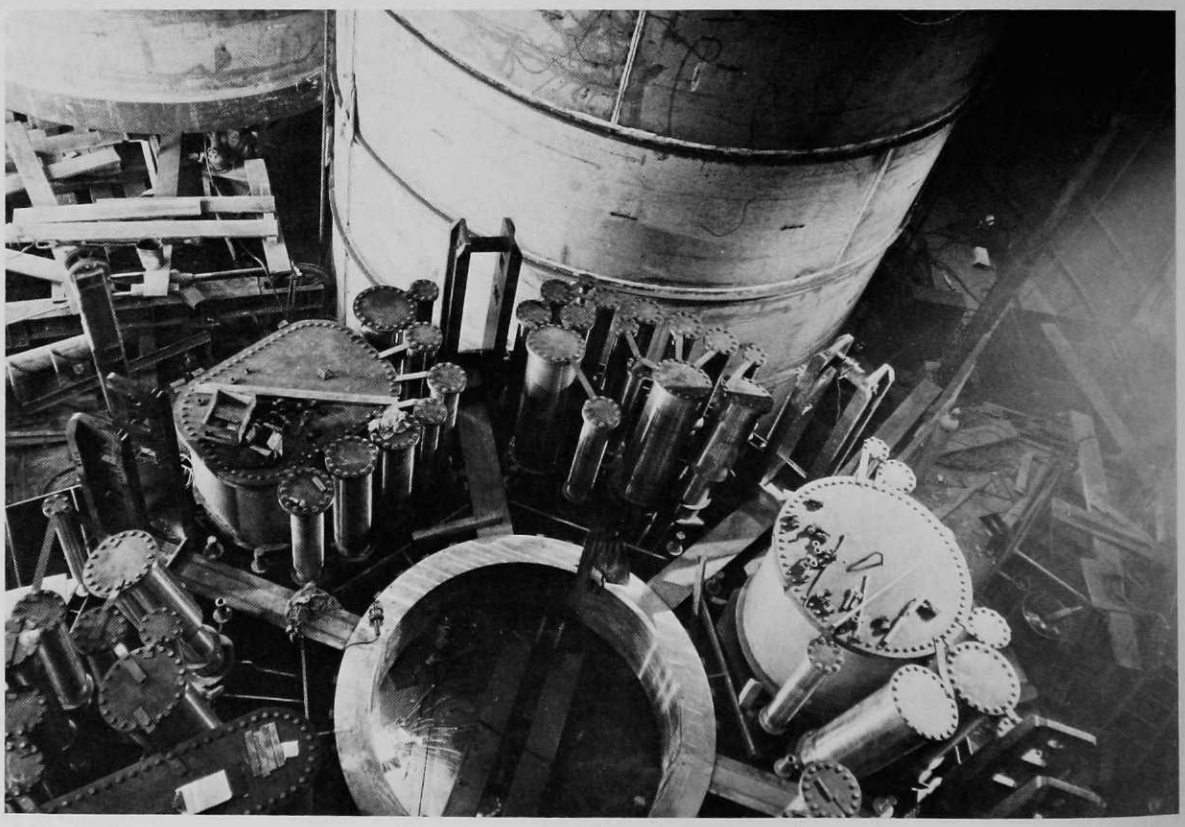


Fig. VIII.16. Overall View of Erection Work in EBR-II Reactor Building. ANL Neg. No. 103-5725.

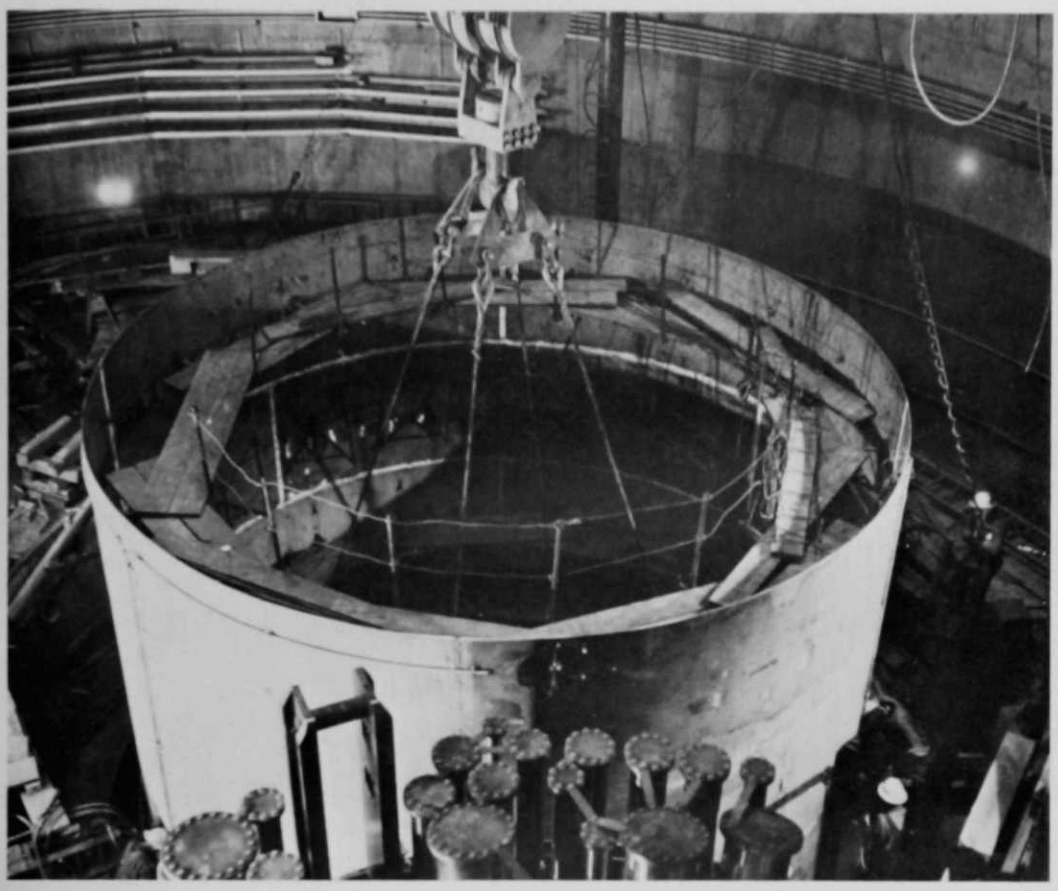


Fig. VIII.17. Lowering EBR-II Main Tank into Guard Tank. ANL Neg. No. 103-5728.

The assembled primary tank cover was then brought to the center of the cavity and raised to the underside of the crane bridge (as shown in Fig. VIII.18). At this point, the primary-tank cover was connected to each of three cables passing over columns with the other end of the cables fastened to large steel boxes filled with about 45 tons of 3/8-in.-dia steel balls (used later for shielding). This counterweight system was used because the weight of the completed primary tank exceeded the design rating of the crane, and it was imperative that no unnecessary risks be taken during this phase of the construction. The six beam stiffeners on the underside of the cover are evident from Fig. VIII.18.

The primary tank cover was then lowered, aligned, and welded to the main tank. The weld was then radiographed. The assembly of the cover and the main tank was then pressurized to 5 psig with helium, and the welds were probed for leaks. The main tank and primary-tank cover assembly was then lowered, aligned, and welded to the guard tank. This last closure weld was the only double-butt weld that was not radiographed.* The annulus between the main and guard tanks was then pressurized with helium to 5 psig, while a

*Subsequent R&D has developed techniques such that double-walled construction can, under certain circumstances, be successfully radiographed.

similar pressure was maintained inside the main tank to prevent its collapse. All welds on the guard tank were then probed for helium leakage. No leaks were found in either tank.

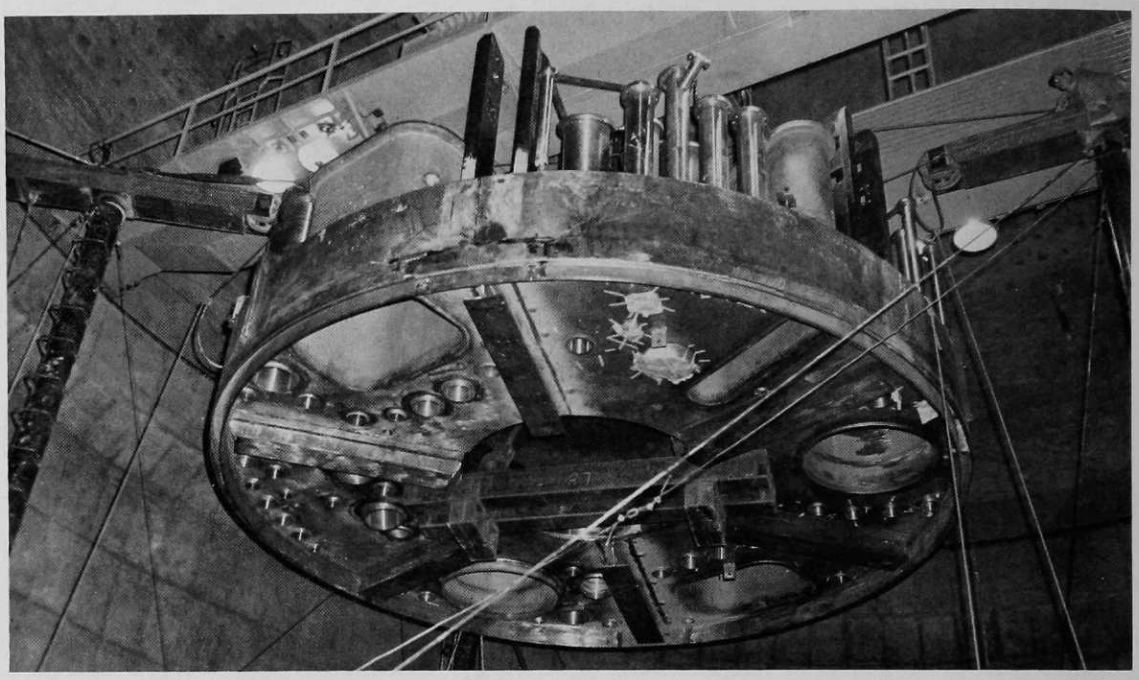


Fig. VIII.18. Raising of EBR-II Main-tank Cover. ANL Neg. No. 103-5742.

Figure VIII.19 shows the completed primary tank raised for application of the wall insulation. Figure VIII.20 shows the raised assembly with the top part of the side insulation covered with the stainless steel jacket and the lower portion wrapped with the aluminum foil. The primary-tank assembly was then raised, the cavity was cleaned, and the tanks lowered and positioned for the installation of the T-1 structure and its load transferred to the three temporary beams as shown in Fig. VIII.21. Each of the three temporary support beams contained a hydraulic jacking system used to level the main-tank assembly fairly close to its final level elevation. Since the erection tolerances for the primary tank and for the shield deck are referred to each other, it was necessary to establish a reasonably fixed datum for the shield deck installation. At this point, the use of the operating floor was turned back to the general contractor for installation of the T-1 structure. This next phase of the work was completed in approximately two weeks.

The first step in the erection of the shield deck was to move in the central ring and six radial beams and to bolt up the splice joints. Figure VIII.22 shows the ring being set on cribbing. Figure VIII.23 shows the top spider partially assembled. After the splice joints were bolted, the assembly was positioned over the primary-tank assembly and lowered to make certain there would be no interferences. The T-1 structure assembly was then

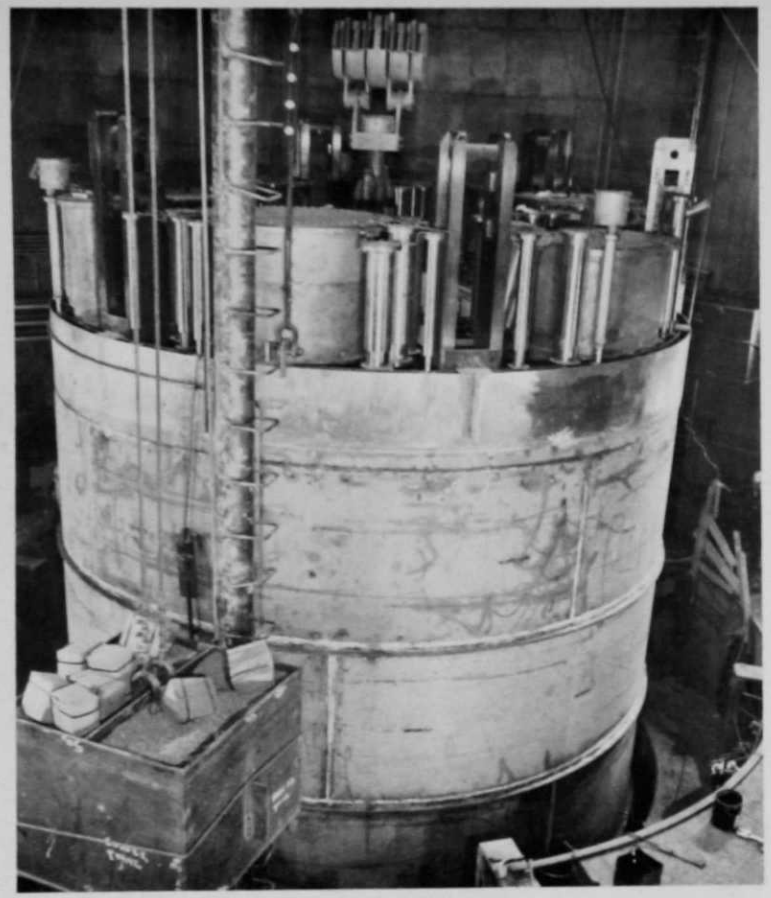


Fig. VIII.19. View of Fully Welded EBR-II Primary Tank. ANL Neg. No. 103-5758.

Fig. VIII.20. Completing EBR-II Insulation on Outside of Guard Tank. ANL Neg. No. 103-5771.

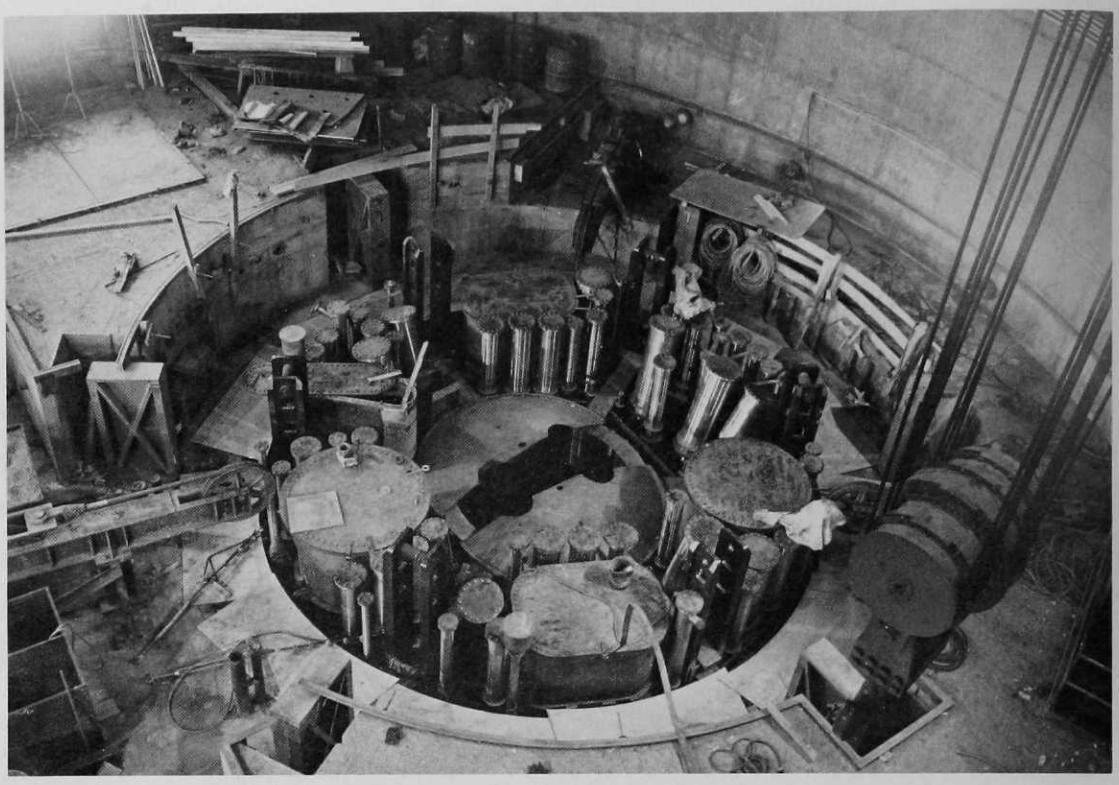


Fig. VIII.21. EBR-II Primary Tank Suspended from Temporary Supports. ANL Neg. No. 103-5773.

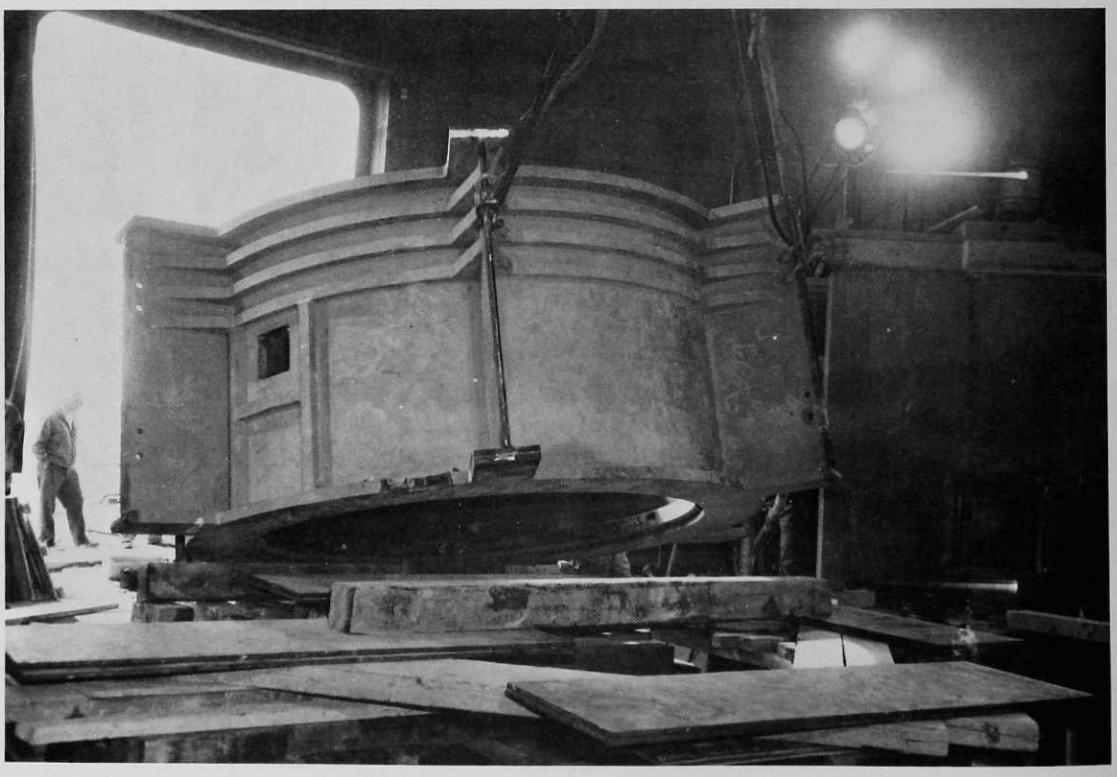


Fig. VIII.22. Moving T-1 Central Ring into EBR-II Reactor Building. ANL Neg. No. 103-5777.

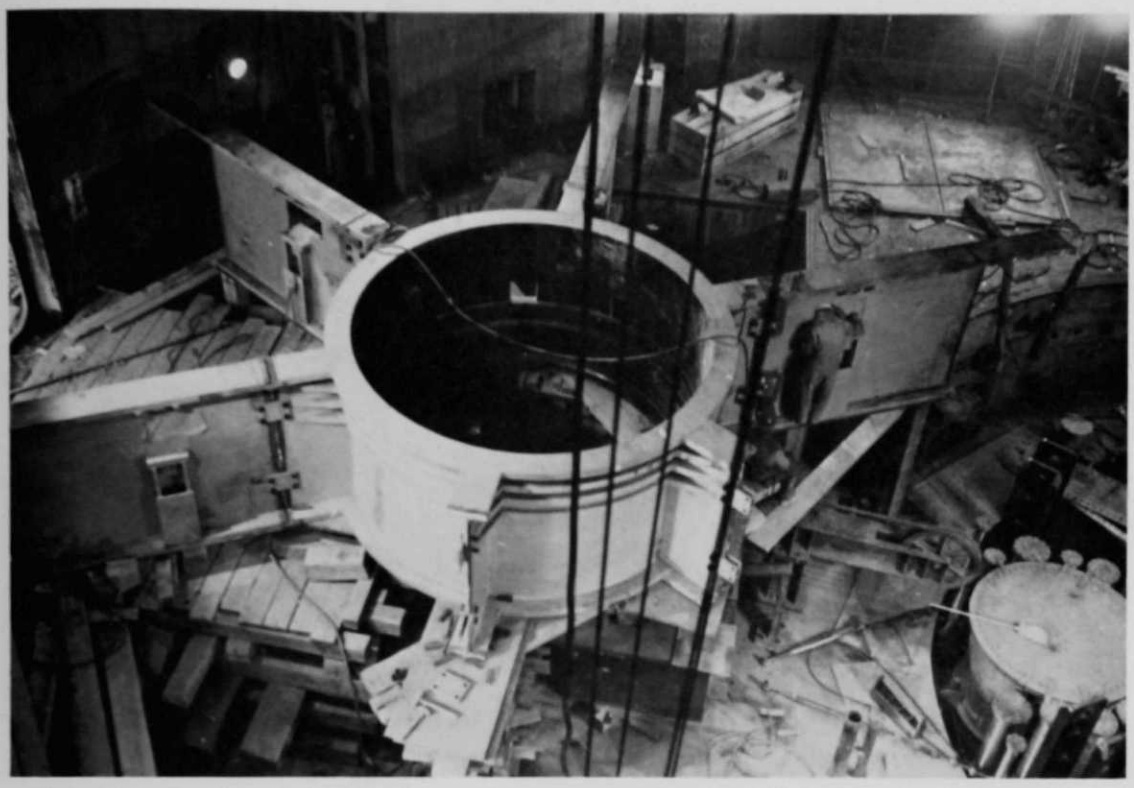


Fig. VIII.23. Temporary Assembly of EBR-II T-1 Top Structure. ANL Neg. No. 103-5797.

propped up on the columns, and the splice joints were fully welded (see Fig. VIII.24). This procedure permitted a close watch on any distortion of the radial beams which might have caused a misalignment at the columns. The spider was then lowered onto the columns and leveled (see Fig. VIII.25). The T-1 structure was leveled with a single bolt at each column passing through a threaded hole in the beam top flange. A water-level micrometer was used to assure the levelness of the horizontal portion of the machined step in the central-ring assembly.

C. Fabrication and Erection of Phenix

The design features of the Phenix are described in the open literature. The important structural aspects of the Phenix are described briefly in Sec. IV. Basically, the design consists of a main tank and a guard tank. The design also uses a third safety tank. The two inner tanks are connected to a top plate closure about $2\frac{1}{2}$ in. thick. (See Fig. VIII.26.) The main tank is suspended from 22 hangers which transfer the load to "bridge-type" bearing pads consisting of laminated steel and neoprene plates.

This part will describe some of the published information available on fabrication and construction. The material was selected from Reference 34.

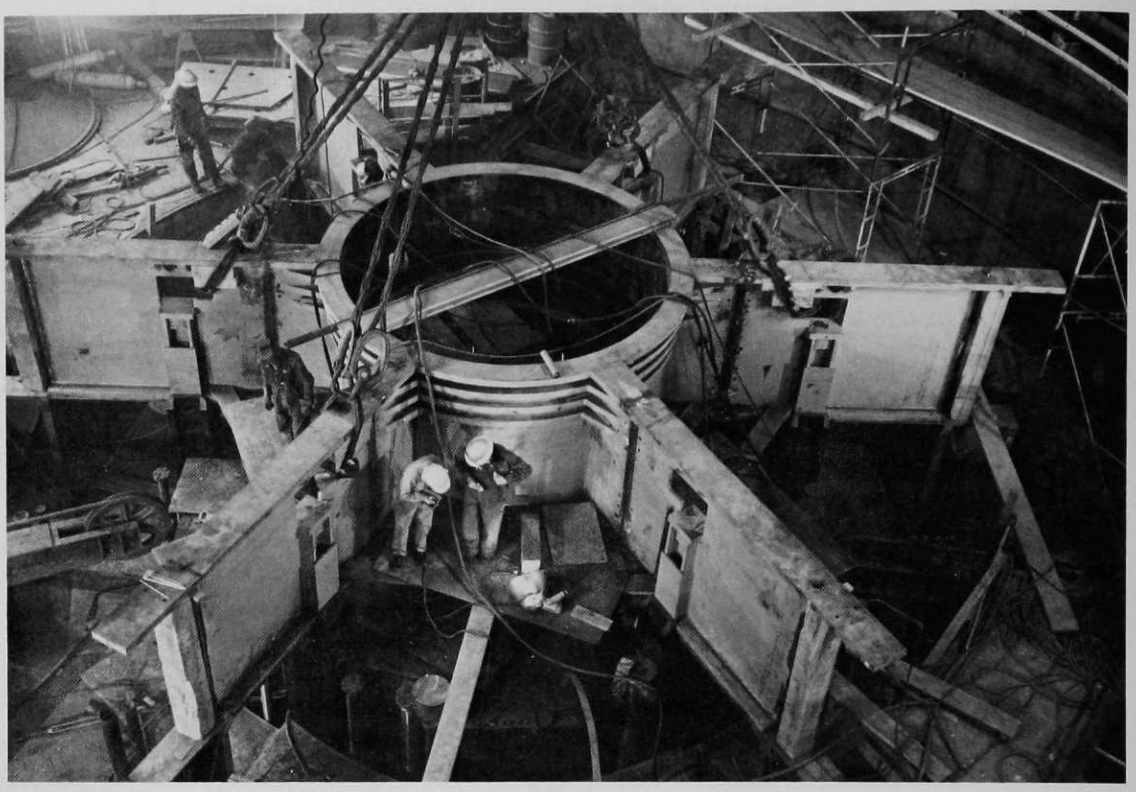


Fig. VIII.24. Completed EBR-II T-1 Top Structure, Ready for Lowering. ANL Neg. No. 103-5803.

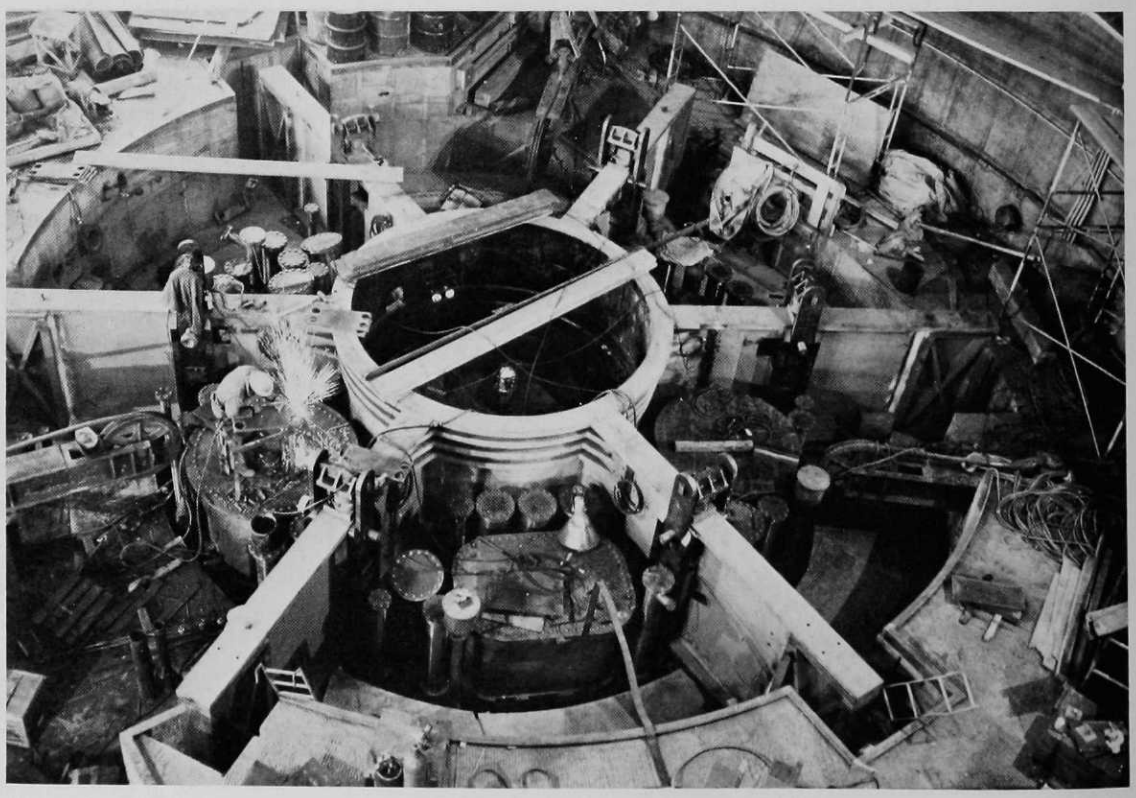


Fig. VIII.25. Transfer of EBR-II Primary Tank to T-1 Top Structure. ANL Neg. No. 103-5809.

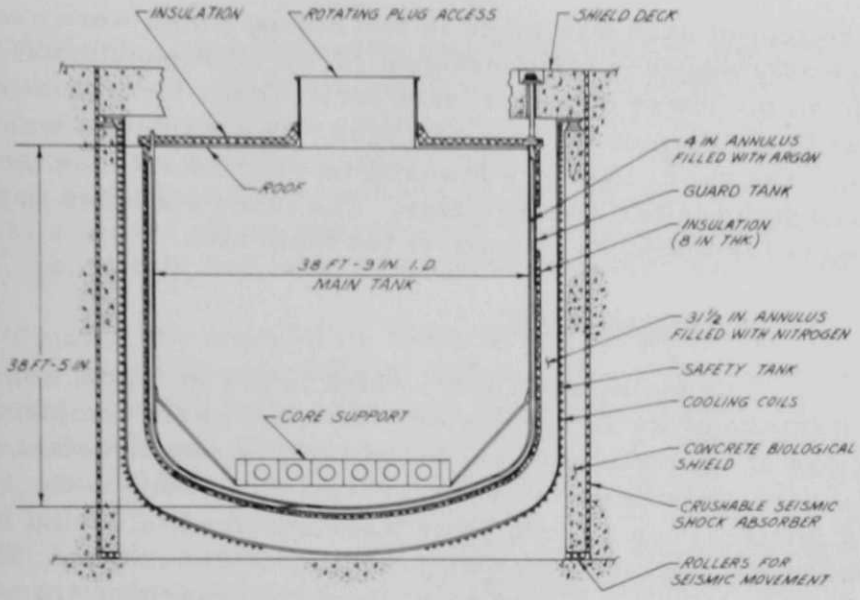


Fig. VIII.26. Concept of Phenix Tanks and Core Support Structure

1. Shop Prefabrication

a. Safety Tank

It is virtually impossible to build large components, such as vessels 39 to 46 ft in diameter, at the works, owing to the very difficult transport problems to which this would give rise. The carbon steel plates for this large vessel were rolled, cut, and chamfered for welding. The various parts were then trial assembled and joined in the shop. The trial assembly was then dismantled, and the parts were sent to the erection shop at the site. There the vessel was put together again and welded. This method was perfectly worked out and supervised by the manufacturer, and went completely according to plan. The tank is, of course, suspended under the shield deck, but during this construction period it rests on the ground through twelve legs welded to the bottom. The safety tank was then fitted with its double circuit made of cooling tubes which were joined to the outer surface by a continuous weld seam applied automatically by the MIG process. It was then sandblasted and sprayed with a zinc coat. The cooling circuits (water) were then given an internal surface treatment by continuous bath circulation. After they were dried out, nitrogen was introduced into the tubing. The tank was then moved by trolley to the reactor unit, lifted by double-trolley crane, and lowered into the concrete cavity where it rested temporarily on its legs until it was fitted into place under the upper shield.

b. Main Tank and Guard Tank

Full use of shop prefabrication was chosen for these stainless steel tanks. The components were shop-fabricated to the greatest transportable

size. The bottom of each was made in two halves which were assembled in the on-site workshop; then six preshaped 19.7 x 23 ft panels were fixed to the bottom to form the lower course of each tank. Studs for insulation support were welded to the outside of the guard tank. The insulation was then fitted into place and the guard tank was lowered into the safety tank on six jacks passing through the latter's twelve legs. The other six holes in the safety tank were used for the jacks to support the main tank.

c. Main-tank Cover

The main-tank cover, which is 2.4 in. thick, required particular care. Because of its size, it was made in two parts complete with the upper sections of the main tank and guard tank. From the start of fabrication, the plate sectors were heat treated after welding to relieve the residual stress in the thick joints. Then the two parts were surfaced, allowing for the flatness tolerance and the requirements for penetration location. The upper section of the tank walls were planed to produce the transition from the 2.4-in. section at the main-tank cover to 0.6 in. at the top course of the tank. Openings were then made in the main-tank cover for components passing through it, and were fitted with cylindrical stiffeners (nozzle extensions).

The two parts of the main-tank cover were then placed in front of a boring machine to calibrate the diameter of their openings and their relative positions of these openings. They were then transported to the site, where they were welded together. Only then were the central openings for the plug cut out and its corresponding strengthener ring fitted.

The upper part of the reactor outlet duct was delivered on site in two parts. The greatest construction difficulties concern the penetration tolerances of the step (between hot and cold sodium), since the insertion of the components requires that the successive passages through the three separately made parts--shield deck, main-tank cover, and reactor outlet duct step--should be accurately aligned. This is why the roof components were machined so as to eliminate any uncertainty as to size, and to assure a clear vertical passageway for equipment.

The shield deck, 48.3 ft in diameter and 4.4 ft thick, was made in six sections. To preserve good dimensional stability, each section was heat treated according to the grade of steel. The components were assembled in the on-site shop, and the complete shield was then moved into the reactor unit and set in place after shield concreting took place.

2. Field Assembly

Construction and assembly activities at the Phenix site were separated into two basic areas: (1) overall civil engineering activities; and (2) basic mechanical engineering work. This was done to minimize constraints that each of these activities placed on the other.

While the concrete cavity was being poured, the tanks themselves were being constructed. Once the concrete cavity was finished, it was possible to lower successively into position the safety tank, the guard tank, and the main tank (with its reactor-support structure). The reactor inlet-plenum foundation was built; then the reactor inlet-plenum itself, with its feed pipes, was set on its stellited plates and keyed to the reactor support structure. Then the neutron shield and reactor support structure were completed.

Finally, the completely assembled reactor outlet duct was lowered into the reactor unit and welded to the neutron shield diagrid support. The roof was placed over the main and guard tanks and the welding rim was welded to the main-tank course. Then the guard tank was raised to its joining level and welded to its welding rim.* The shield deck was placed in position and adjusted for level. The 22 bearing points take the 1000 ton load of the concrete-filled shield. Finally, the main-tank hangers were welded to the tank welding rims and the mountings between the tank and the shield deck were placed in position.

This mounting system is made with STUP cushions composed of a sandwich of alternating metal and neoprene layers. Such an arrangement reduces the need for great accuracy between the bearing surface of the hangers and that of the upper shield, owing to the vertical flexibility of the cushions. Furthermore, the lateral flexibility of the cushion allows the roof to expand. Tests made on a model enabled distortion-stress curves for the cushion assemblies to be plotted.

The safety tank then was raised up to the level of the welded rim integral with the shield deck and welded. All the tooling was withdrawn and removed through a manhole. Then the apertures in the 12 legs were closed by welding semi-spherical cups over them, and the manhole was sealed.

3. Testing and Inspection

Continuous testing and inspection was carried out throughout construction. The fabrication history of each component was carefully recorded. Each batch of welding electrodes was given an acceptance test, and the results were documented. The welding processes were agreed to by the fabricator and designer. The position of each plate and corresponding weld joints were recorded, so the complete history of each assembly is known and can be referred to after completion. The main leak-tight tanks were examined by ultrasonics, and all welds were x-rayed. Geometrical dimensions and shapes were inspected as assembling progressed. The position of the assemblies was checked by theodolite relative to fixed points placed in the concrete as the work progressed.

The French believe that the availability of an on-site assembly shop, specially designed to handle the large separate pieces, had many

*This procedure was quite similar to that used on EBR-II.

advantages. It reduced the need to find suitable nearby shops qualified to perform such work (the number of such shops was quite low in 1969). Further, the shop could be designed and built to do the exact job at hand. The French credit having an on-site shop with their maintaining good schedule progress.

For both the carbon steel safety tank and the stainless steel main and guard tanks, the welding processes and electrodes were jointly agreed upon by the project management and fabricator. The welding work all progressed smoothly, without delay, with the number of defects always below specified limits.*

D. Fabrication and Construction of PFR**

Figure VIII.27 shows a cross section of PFR. Because of their large size, the PFR main tank, guard tank, and shield deck were delivered to the site in prefabricated sections, and assembled in the reactor building and another large temporary building that was erected nearby. The reactor building was also used as an assembly workshop for the intermediate heat exchangers and the rotating shield plug.

The guard tank, the main tank, and reactor support structure are all welded to the roof, and novel methods of supporting these in the reactor vault during construction were adopted. The guard tank was supported by floating it in water in the vault, the main tank being hung from its rim by temporary hooks. The reactor support structure was carried by temporary cantilevers from the top of the vault, and the shield deck was supported on a jacking tower about 51.2 ft high.

The construction procedure was to lower the shield deck on its jacks to a level where it could be welded to the reactor support structure. The next stage was to lift the guard tank and main tank by raising the water level to a position where the latter could be welded, and finally to raise the guard tank to its welding position.

As indicated above, the main structural components (e.g., the reactor support structure and main tank) are supported from the shield deck. Since these large units and the reactor outlet duct (whose function is to separate the hot from the cool sodium and which is carried by the reactor support structure) all sit one inside the other, they had to be in position in the reactor vault before the shield deck could be brought over into position.

Access for large loads to the construction site at Dounreay is difficult. The nearest seaport which could handle them is at Wick, 36 miles from the site and connected to it by narrow roads that limited to 18 ft the maximum

*Except for the need to repair part of one horizontal course on the guard tank, this same statement applies also to EBR-II.

**This material is a condensation of Reference 35. The reader is referred to that publication for construction photographs that could not be reproduced here, and additional detail about PFR construction.

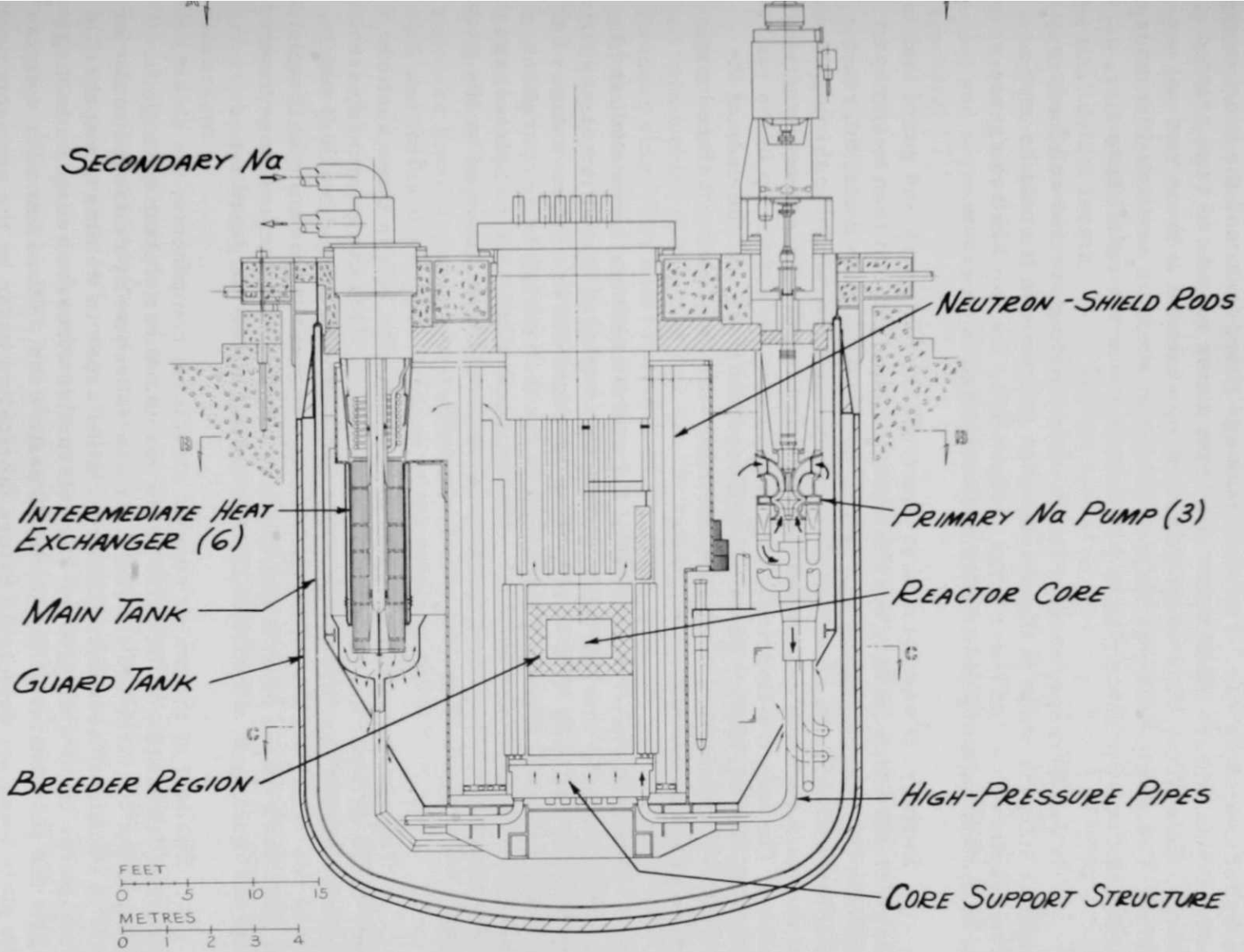


Fig. VIII.27. Cross Section of PFR

width of load that could be transported. Bridges limited the maximum weight to 90 tons. The large structural components mentioned above had therefore to be fabricated on site. The main tank and guard tank, and the reactor support structure were made up on site from plates rolled and trial erected at works; the reactor outlet-duct was made up at works in three vertical sections. The shield deck was shipped to site in six main sections (the inner ring complete, and the outer ring in five sectors) plus ten radial beams.

It was arranged to have the reactor building erected and made weather-tight at an early stage in the construction program so it could be used as a construction shop for these large components. Its two 125-ton cranes were of sufficient capacity to lift all these completed components.

Initially, it was planned to construct the main tank and guard tank in a large construction facility near the reactor building, and then to transport them to the reactor building on trolley rails. As it turned out, the reactor building proved to be an excellent construction area, and only the bottom domes of the main tank and guard tank were erected in the adjacent construction facility. The upper cylindrical parts of the tanks were made up in the reactor building, and all mating parts brought together there. In the case of the ferritic-steel guard tank, the bottom dome was fully welded before being transferred to the reactor building.

Welding and stress relief of the guard tank were completed, and the tank was lowered into the vault, where, as mentioned above, it was supported by flooding the vault with water. This arrangement has the advantage of giving a ready means of adjusting the height of the tank during later operations. By this time the dome and cylinder of the main tank had been completed, and they were then brought together in the area which had been occupied by the guard tank.

The reactor outlet duct had been sent to the site in three sections, and only three vertical welds had to be made there. This component is essentially cylindrical, but has three large lobes protruding at the top, each of which holds two of the six intermediate heat exchangers. At the time of completion of field work on the duct, it and the reactor support structure were lowered into the main tank, which was by then sitting inside the guard tank.

The last of the major reactor structural components, the shield deck, incurred fabrication difficulties at the manufacturer's plant and significantly delayed the construction schedule. It was subsequently delivered to the job site in six major sections. The difficulties arose in welding the spokes* through the outer ring girder: the contraction stresses set up in welding pulled apart the plate material in the radial beam webs. These lamellar defects in the plate were not detectable before fabrication began by the ultrasonic scanning specified. There was considerable variation in the amount of defects from sector to sector depending on the detailed characteristics of the plates used for

*Radial beam webs in the shield deck.

each. They were found by ultrasonic inspection techniques specially developed to check these cruciform welds. The trouble was overcome by repeatedly cutting out the defects and refilling with weld metal. This process took considerably longer to complete than expected, and with the advantage of hindsight, one can see that it could well have been advantageous to begin fabrication again with better quality plate material and improved fabrication procedures. Each cycle of the repair process, however, seemed to result in a reduction of the total defect length, and it always appeared possible that the variations in procedure that were being tried would soon produce satisfactory results. The normal precaution against running into this type of trouble was taken care of by carrying out the welds on a full scale mock-up but this gave no sign of the extent and seriousness of the problem that was to come in the shield deck structure.

There is every confidence that the above troubles would not recur on another fabrication of this type, so it is not considered necessary to change the support concept for the large reactors to follow PFR. (Indeed, other shield decks have been built without such difficulties.)

By the time the completed shield deck assembly was moved over the reactor vault, the five outer sections (each of which included two radial beams) had been welded into a ring and the ten radial beams welded to connect this to the inner ring. The outer member of the outer ring girder carries the guard tank, and the inner member carries both the main tank and the reactor support structure through a forged ring with a section like an inverted tuning fork. The reactor support structure is supported by a series of straps from the inner leg of the "tuning fork" (to minimize thermal stresses), and the main tank is welded completely to the other leg. By the time the shield deck was moved over the vault, the "tuning fork" ring had been welded to it, as had the stainless steel diaphragm plate that covers the bottom surface of the shield deck and forms the upper boundary of the primary circuit.

A temporary construction frame had been erected around the vault to support the roof higher than the other components in the vault so that the penetration liners passing through the shield deck (which provide access to the primary pumps, heat exchangers, etc.) could be welded to the diaphragm plate before the packs of laminar insulation were raised into position below the diaphragm.

During lowering of the shield deck (with penetrations and insulation complete), jacks were used because the weight of the shield deck was now too high for the cranes to handle. Welding of the main tank and then the guard tank to the shield deck underside completed the main structural work on the reactor. The shield deck was then lowered into its final position and filled with shielding.

During the above welding operations, assembling of the internal components proceeded with access through the large central hole in the shield deck occupied ultimately by the rotating shield plug. This hole is large enough for the core support structure and neutron shield rods to be loaded through it. Those components whose position in the reactor did not permit them to be loaded in this way were placed in the main tank in temporary storage positions before the shield deck was lowered.

IX. NOTES ON IN-SERVICE INSPECTION AND MAINTENANCE

The material in this section is presented in two parts. The first part deals with in-service inspection of pool-type LMFBRs, and the second with maintenance of such units. The purpose of the section is to assist in evaluation of the pool concept and in comparison of the different design approaches of the pool concept in regard to possible needs and capabilities for primary system in-service inspection and maintenance.

A. In-Service Inspection

Time limitations did not permit a definitive study of in-service inspection. The treatment provided is neither detailed nor comprehensive. Nevertheless, it is hoped that the brief observations made might be of interest and useful. For this report, in-service inspection is assumed limited to the examination of components or parts of components which are "important to safety" as generally indicated in 10 CFR-50 and Regulatory Guide 1.26. The scope of in-service inspection considerations is taken to be the same as that reflected in the ASME Code, Sec. XI (rules for in-service inspection of nuclear power plants). First, the state of the art in respect to rules, standards, and procedures for in-service inspection is briefly reviewed. Then, the major aspects of inspection to be treated are described. Finally, a number of major components potentially subject to in-service inspection are discussed.

1. Status of Codes and Standards

Section XI of the ASME Boiler and Pressure Vessel Code states the rules for in-service inspection of nuclear power plants. The rules for LWRs have been published in Sec. XI (Division 1) and are in effect. The rules for gas-cooled reactors (Division 2) will shortly be published for trial use and comment. An ASME subgroup presently is working on a draft of rules for LMFBRs (Division 3). Division 3 rules are expected to apply to all types of LMFBRs; however, they presently address only a single concept of the LMFBR--the loop concept.

The Division 1 requirements for in-service inspection of LWRs are useful here as a general background and as an aid in patterning the discussion. Obviously, their direct applicability is limited by the major differences between water-cooled, high-pressure, low-temperature LWR systems and the sodium-cooled, low-pressure, high-temperature LMFBR systems. The requirements under consideration in Division 3 for LMFBRs are also of help in the discussion, but their usefulness is limited by the fact that they may have to be clarified or expanded (to account for differences between pool and loop type systems) at such time as the pool concept is addressed.

It should be noted that EBR-II experience has provided some information relative to in-service inspection, as well as to maintenance. Buildup

of radiation level on component surfaces, handling of components immersed in primary sodium, amassing of information on troublesome components, etc., are areas of EBR-II experience which can assist in assessing the impact of the pool concept in these areas. The reader may find this information in the extensive open literature on EBR-II; a brief review is given under B. Maintenance.

2. Major Aspects of Inspection

The impact of the use of the pool-type concept in respect to inspection activities is discussed for most of the major components in the primary system, and is discussed in terms of the same key aspects for each component. Division 1 (and Division 3 draft) of Sec. XI are helpful in ascertaining the key aspects of in-service inspection and identifying the components potentially to be subjected to inspection. Based on that material, the following three aspects have been selected as the basis for evaluation: (a) accessibility, (b) importance to safety (requirement to inspect), and (c) type of examination. Other aspects covered in Sec. XI, such as examination evaluation, repair procedures, and systems tests, are not addressed. Before discussing individual components, a few general statements are made about the selected aspects.

Accessibility: Each major primary system component treated is briefly evaluated in terms of its accessibility in a pool-type system. Accessibility considerations include spatial aspects, nature of the surrounding environment, radiation level, special conditions such as the presence of insulation or shielding, etc. In these evaluations it is assumed that the reactor is in a shutdown state so that the primary system sodium is at a temperature in the neighborhood of 300-400°F; from an in-service inspection viewpoint, the lower temperature is preferred because it allows more ready accessibility and is more compatible with certain of the examination equipment required.

Importance to safety: As noted earlier, in this discussion only components which are "important to safety" are assumed possibly to require in-service inspection. Much of the terminology in 10 CFR-50 and Regulatory Guide 1.26 related to evaluation of the importance to safety is not directly applicable or clearly transferable to pool-type LMFBR systems, so an estimation of the most likely determination of importance to safety is attempted in the discussion of each component. In assessing the importance to safety, a qualitative evaluation of the consequence of component failure is made.

Types of examination: The three general methods or types of examinations specifically mentioned in Sec. XI are volumetric, surface, and visual. In addition, the required continuous monitoring for leakage of each system or component containing liquid metal is specified.

The two volumetric examinations which are commonly used are radiographic and ultrasonic (UT). Radiographic testing has a limited application for LMFBR primary systems, primarily because only the outside surface

of a component is usually accessible; also, the radiation level in some locations may be high. Ultrasonic testing of austenitic stainless steel sections of thicknesses of interest may be feasible, but at this time such testing must be considered not developed to the level of reliability desired. Any need for accomplishing such testing at elevated temperature and with restricted access adds further to the difficulty. In view of this, UT development programs are presently being conducted under government sponsorship at HEDL, ORNL, and ANL; private firms are also performing research and development to improve UT techniques for commercial application on LMFBRs. The apparent success of the French and German UT programs is noted. It is the intention in the SuperPhenix plant to try to inspect the main tank both visually and by UT, and a French UT development program is being carried out for this specific purpose.

Surface examination is of somewhat limited application for LMFBR in-service inspection: in particular, the use of liquid penetrants is restricted by the relatively high temperature of primary-system sodium boundaries and restricted access. The use of magnetic-particle examination is limited generally by the employment of austenitic stainless steel in essentially all areas.

To aid in performance of visual examination, remotely operated TV cameras have been developed. The specifications for the visual inspections of the CRBR reactor vessel include a capability of the TV camera to see cracks in the surface at 400°F. The camera is attached to a cable which is supported from a ring structure located above the annular space between the reactor vessel and guard vessel. The camera is mounted on a device whose wheels contact the outside surface of the reactor vessel and the inside surface of the guard vessel. The device moves the camera to the desired locations where the camera is able to examine the vessel sides, bottom, and nozzles on the accessible surfaces.

Development efforts are also being expended on under-sodium "viewing" devices. However, the degree of resolution that seems to be attainable from these devices is much less than that generally desired for in-service inspection.

To the above types of examination should be added direct mechanical distortion measurement as a means for monitoring structural deformation.

3. Discussion of Inspection of Key Components

Although the general approach of the present comments on in-service inspection is based on Sec. XI, the material presented is much less detailed. The scope here is in general limited to the discussion of components and areas unique to pool-type systems or at least potentially significantly different from those of other types of LMFBR systems. Only five component groupings are considered.

The five component groupings are (a) main tank and guard tank, (b) tank support structure, (c) main tank cover and shield deck, (d) reactor support structure, and (e) primary system piping (or ducting). The impact of the pool design on the in-service inspection of the components in these five groups is discussed in the remainder of Sec. A.3 in terms of the three aspects discussed above--accessibility, importance to safety, and type of examination.

a. Main and Guard Tanks

Accessibility: The inside of the main tank is not readily accessible because of its submergence in primary sodium (and, in some concepts, because of the presence of a protecting thermal baffle). The outside of the guard tank in most concepts is not readily accessible because of the presence of thermal insulation and, possibly, heating or cooling equipment. However, the space incorporated between the two tanks, nominally at least 9 in. and possibly as much as 28 in. in radial thickness, provides ample spatial access to the entire outer surface of the main tank walls and bottom closure and to the entire inner surface of the guard tank walls and bottom closure. To introduce inspection equipment into the annulus, inspection penetrations are provided vertically through the shield deck (and through the main tank cover where one exists) at appropriate intervals around the tank periphery. The spatial access is unencumbered by presence of nozzles or other protuberances on the tank walls--the tank walls and tank bottom are continuous and smooth. The atmosphere is an inert gas (most likely nitrogen) at the same temperature as the main tank (300-400°F during inspection activities). It is likely that a low level of radiation will exist after elapse of a short time for decay of ^{24}Na activity, because of the large amount of shielding (including sodium) within the main tank. Thus, accessibility for inspection of the tank wall and bottom of both tanks is considered good.

Importance to safety: The main tank is the normal primary system coolant boundary. The guard tank is provided as a backup; it becomes the coolant boundary in case of main tank leakage or failure. If main tank leakage should occur, a reduction in sodium level (of several feet) could result within the tank. However, there would result no effect on continuity of reactor coolant flow, and capability for removal of reactor decay heat, even by natural convection alone, would remain unimpaired. This is because the IHX and pump inlets are physically located by design so as to remain adequately below the sodium free surface in such an event. This also assumes, of course, that the guard tank does not leak. If the guard tank were also to leak, a significant loss of coolant situation would result; that is, the sodium level could fall to the point where flow of coolant through the reactor by either forced or natural convection would be unattainable. In addition, in some concepts the guard tank is an integral part of the primary containment system, and any leakage or failure of the guard tank would constitute violation of the containment system.

For these reasons, both tanks are viewed as important to safety, but not necessarily in the same degree. It is considered that periodic in-service inspection of the main tank will be required. The need for inspection of the guard tank is less certain, but it possibly may be required. It is assumed that careful monitoring of the inter-tank gap throughout service life to detect any leakage of sodium from the main tank would be required.

Type of examination: Remote visual inspection of all welds in the walls and bottom of both tanks can be made; a track system for positioning of the camera probably would be utilized. Remote ultrasonic examination of such components appears at this time to be difficult and of doubtful reliability, as discussed earlier, but may in the future be shown (as a result of ongoing research and development programs) to be an acceptable method. Continuous monitoring of the inter-tank gap to detect and alarm any sodium leakage through the main tank wall is clearly feasible (and is assumed required as an important element of the in-service inspection program).

b. Tank Support Structure

Support structures only for top-supported main tanks and guard tanks are considered here, as only one plant has been built or proposed which utilizes bottom-supported tanks. There are two general types of such structures, and the inspection assessment is different for the two. One type is the skirt support, in which the cylindrical skirt is simply an extension of the wall of the tank at its upper end, with the top of the skirt attaching to the shield deck. The other type employs the hanger concept, in which a number of discrete hangers spaced at intervals around the tank periphery are attached at their lower end to the tank and at their upper end to the shield deck.

Accessibility: With the skirt type of support, accessibility is generally identical to that for main tank and guard tank walls as described in a, above, and is therefore considered good. A possible exception is that some sort of insulation or convection inhibitor might be required near the top of the annulus between the two tanks which potentially could interfere with access--this would have to be designed deliberately to minimize interference.

With hanger type support, the top end of each hanger (together with its load pad, rollers, bridge pad, pin connection, or other top fixture) is located near the upper surface of the shield deck and should be readily accessible spatially; also, it is in an air atmosphere, at room temperature, with negligible radiation. Accordingly, accessibility here must be considered excellent. The body of the hanger at lower elevation and its area of attachment to the tank (underneath the shield deck), however, see substantially different conditions. In particular, spatial access probably is significantly restricted even after the removable shield blocks assumed provided around each hanger have been temporarily removed. In the area of attachment to the tank, detachable sections of tank thermal insulation assumed provided also would have

to be removed. Although an air environment would exist, temperatures would not be high (they would be equal to or lower than that of the tank), and radiation level would not be high, inspection would nevertheless tend to be difficult as it would have to be carried out remotely (from the top of the shield deck) on a weldment of irregular configuration and in very cramped quarters. In addition, portions of the welds between the hangers and the tank might be below the seal plate of the tank cover, and thus located in the cover gas region. Therefore, accessibility for inspection of these portions of the hanger type support structure is judged generally poor--although a detailed design which enables reasonably good access is not necessarily ruled out, especially if some compromise in other design objectives is accepted.

Importance to safety: The main-tank support structure supports and retains in position the main tank, which contains all of the primary system coolant and in some concepts supports the reactor as well. Its failure, if sufficiently complete, could result in dropping of the main tank and loading up the guard tank, with potential for subsequent leakage and loss of coolant. Conceivably, in some concepts, insertion of excess reactivity into the reactor (by dropping the reactor away from the control rods) also could result. The guard tank support structure similarly supports and maintains in position the guard tank, which is normally empty but possibly could become filled with primary sodium or otherwise loaded, as above, as a result of main tank leakage or failure; in the latter circumstances, the additional failure of the guard tank support structure potentially could lead to loss of coolant or, in some concepts, to loss of primary containment. Thus, both structures are viewed as important to safety, although in differing degree. Accordingly, it is considered that periodic in-service inspection will be required of the main tank support structure (skirt or hangers); it may possibly be required for the guard tank support structure.

Type of examination: With the skirt type of support, the types of examination that can be conducted and the examination equipment usable are essentially the same as described in a. Main and Guard Tanks, above, for the tank walls. With hanger type support, remote visual examination of all welds probably is practical. Volumetric examination, however, would be more difficult. Near the bottom end of the hanger, and especially in the region of attachment of the hanger to the tank cover, ultrasonic examination would appear to be particularly difficult, or perhaps impossible, partly because of the restricted access and, even more, because of the irregular configuration of the weldment.

c. Main Tank Cover and Shield Deck

In existing or proposed pool-type LMFBRs, two different general design approaches have been taken in respect to the main-tank (and guard-tank) top closure. One is to provide a separate, gastight tank cover and, in addition, an independent shield deck positioned above it. The top

surface of the tank cover in this case must be well insulated to prevent excessive heat loss and to prevent overheating of the shield deck. The other approach is to combine the two functions and provide only a shield deck which serves also as the tank cover. In this arrangement, insulation must be provided on the undersurface of the deck (inside the main tank) to prevent overheating of the deck.

Accessibility: The top surface of the main tank cover, if such a cover is employed, is considered inaccessible for in-service inspection. This is because of the extremely limited spatial access (down through the shield deck) to the air gap between cover and deck, and because removal of the cover insulation to enable examination would in such conditions be infeasible. Also, any meaningful remote inspection (other than visual) of the underside of the cover by insertion of equipment downward through penetrations extending through the deck and cover would be extremely difficult because of the limited and indirect access to most areas of the cover surface, the presence of sodium vapor, risk of dropping equipment into the primary coolant, etc.

In respect to the shield deck, with or without existence of a separate tank cover, accessibility for inspection is somewhat limited. The top and upper portions of the structure would be near the operating floor, in an air atmosphere at room temperature, and almost entirely exposed and accessible. These portions would include the top flanges and upper web sections of the structure main beams, the top plate (if any), and, probably, many of the stiffening ribs, depending upon the details of the design. However, the lower portions of the same elements and the base plate of the deck would tend to be virtually inaccessible because of the concrete (or other) shielding fill in the deck. It might be feasible, however, to provide tunnel-like access voids in certain locations such as along the bottom of main radial beams to permit visual inspection of the flange-to-web welds, if this were necessary. Also, by incorporating removable shielding sections, access from the top to specific areas of the baseplate (or beam or web bottoms) probably could be provided. When a separate tank cover is employed, access to the underside of the deck baseplate is possible, although quite difficult, by insertion of equipment downward through penetrations in the deck and into the gap between cover and deck. When the integral deck approach is employed, access to the underside of the deck baseplate is not possible because of the presence of nonremovable insulation covering the plate (the insulation itself, however, could be visually inspected by insertion of equipment through deck penetrations). General access to the shield deck strength members (and to components mounted on the deck) for purposes of monitoring for gross deformation should be good, however.

Importance to safety: The main tank cover is discussed first. In at least one design (BN-600) employing a separate main tank cover, the cover is used to support the rotating plugs. Because the control rod drives, as well as other mechanisms, are mounted on these plugs, the tank cover in

this case clearly is important to safety, and in-service inspection of the cover (or of critical portions of it) might well be required. In some other designs employing a main tank cover, the cover supports no major components and none of direct significance to safety; in this case the cover serves only to effect a gastight seal for the primary system cover gas space and probably need not be considered sufficiently important to safety as to require in-service inspection. In still other designs, the situation lies between these two; the cover is used to support major components such as primary pumps, IHXs, shutdown coolers, etc. For the latter systems, the probable effect on safety of possible modes of cover failure would have to be analyzed in order to determine whether, or to what extent, in-service inspection of the cover might be required.

The shield deck is discussed next. In almost all designs (at least one exception is BN-600), the shield deck is the structure to which the main (and guard) tank support structure attaches; gross failure of the deck conceivably could result in loss of support of the tank. The deck also is used in almost all cases to support the rotating plugs on which the control rod drives are mounted. For at least these reasons, shield deck integrity is related to safety. Also, in some designs, possible importance to safety may derive from use of the deck to support the primary pumps, IHXs, shutdown coolers, etc. Again, analysis of the probable effect on safety of possible modes of deck failure would have to be made to determine the nature and extent of deck in-service inspection, if any, that might be required. In this connection it is noted that, from a structural viewpoint, minor weld cracks or weld imperfections in the deck structure can probably be tolerated, since they are quite unlikely to propagate. More important, development of a significant safety effect would require rather significant cracking and distortion, so that monitoring of deck deformation may be more important than examining for cracks.

Type of examination: It appears clear that close-up visual examination can be accomplished on the upper portions of the deck structure. Even volumetric or surface examination in these areas probably could be accomplished if it were deemed necessary. Areas of the lower portions of the structure which are of particular interest and for which access from above is arranged in the design also could be remotely examined visually. Possibly, UT examination could be accommodated in certain of the latter areas, but only with difficulty. Mechanical distortion measurements on the deck could be made quite readily to monitor deck deformation.

d. Reactor Support Structure

The term "reactor support structure" is used here to mean the structure that supports the core support structure and reactor assemblies, core barrel and reactor vessel (if any), reactor inlet plenum, radial shielding surrounding the reactor, etc. In some cases the structure consists essentially

of an upward-facing conical shell (reactor support cone) joined at its inner end to the core support structure and at its outer end either to the wall of the main tank or to a separate vertical cylindrical shell (which, in turn, is hung from the shield deck or the main tank cover). In other cases, it consists of a downward-facing conical shell joined at its inner end to the core support structure and at its outer end to the bottom of either the main tank or a separate tank provided for the purpose.

Accessibility: No portion of the conical shell, whether of the upward-facing or downward-facing type, is normally accessible for inspection because of the total submergence of the shell (at 30 to 50 ft depth) in primary sodium. By first unloading the reactor and then draining all the sodium from the main tank, access could be achieved through penetrations in the deck (and tank cover, if any). However, this requires provision of sufficient on-site tankage (which, for SuperPhenix, is being provided) and, obviously, would require a large amount of downtime. In addition, the presence of residual sodium on the shell surfaces, even with draining at highest practicable temperature, would make meaningful inspection other than visual inspection exceedingly difficult and, at the present state of the art, quite impractical.

Where a separate cylindrical shell or tank is employed as part of the reactor support structure, accessibility of the shell or tank is essentially the same as that for the conical shell because these parts also are totally submerged in the primary sodium. Where the main tank is used as part of the reactor support structure, accessibility of this part for inspection is good, as described under a. Main and Guard Tanks.

Importance to safety: Sufficient degradation of the reactor support structure during service potentially could lead to failure. Conceivably, failure could occur under normal loading, but possibly more significant is possible structure failure under postulated earthquake or HCDA conditions due to undetected degradation. Since gross failure conceivably could result in loss of reactor support, the structure is clearly related to safety. However, the probable effect on safety of possible modes of reactor support structure failure would have to be analyzed in order to determine whether, or to what extent, in-service inspection of the structure would be required. In this connection, it might be noted that the downward-facing conical shell (primarily in a state of compressive stress) may be considered somewhat less susceptible to progressive degradation than an upward-facing conical shell (primary in a state of tensile stress). It should also be noted that development of a significant safety effect would require rather significant cracking and distortion, so that monitoring of reactor support structure deformation may be more important than examining for cracks.

Type of examination: Neither regular remote visual inspection nor volumetric examination normally can be accomplished on the support structure conical shell or on the separate cylindrical skirt or tank (if used), because

of lack of access due to the total submergence of these structures. Use of under-sodium viewing devices during service might possibly enable detection of cracks of exceptionally large width, but this certainly could not produce results equivalent to those of regular remote visual inspection. However, physical measurements to various selected points on the structures probably could be made periodically during service which would provide indication of any gross deformation. Infrequent, periodic visual examination of these structures could be conducted by first draining the tank as described above under Accessibility, but only with the penalties and difficulties described; any volumetric examination by this means appears impractical at this time. If the main tank is used as part of the reactor support structure, in-service inspection of this part can be accomplished as described under a. Main and Guard Tanks.

e. Primary System Piping or Ducting

All pool-type concepts employ piping from primary pump outlet to reactor inlet and either piping or ducting* from reactor outlet to IHX inlet. No piping or ducting is used from IHX outlet to pump inlet, this function being served by the main tank (an exception occurring in the BN-600 concept, in which ducting is used).

Accessibility: All primary system piping and ducting is totally submerged in the bulk sodium of the main tank. Accordingly, its accessibility is generally the same as for the reactor support cone, as described under d. Reactor Support Structure. However, in some concepts, portions of the piping system are deliberately made removable from the tank so that they can be inspected and maintained or replaced during service if this should appear desirable. Removal is either in conjunction with removal of a primary pump to which the part (such as a flexible bellows assembly) is attached, or through a special shield plug provided for the purpose. To enable removal, use of a quick-disconnect or (more difficult) a bolted joint is required.

Importance to safety: Failure of the reactor outlet piping or ducting does not represent a significant safety problem. This is because: the piping or ducting is not a part of the primary coolant boundary, and its failure does not result in any loss of coolant from the primary system; the failure does not reduce the coolant flow rate through the reactor; and, the resulting reduced rate of heat removal from the main tank through the IHXs produces relatively slow heatup of the reactor inlet coolant (and primary system) because of the very large thermal capacity of the bulk sodium and tank components, so that time would be available for manual shutdown even if reactor scram were to fail.

*In various concepts, ducting from the reactor outlet to IHX inlet is sometimes referred to as the inner tank, inner vessel, redan, outlet flow barrier, etc.

Failure of the reactor inlet piping, for the same reason as above, also does not result in any loss of coolant from the primary system. Similarly, the resulting change in reactor inlet (and bulk sodium) temperature would be slow. The failure could, however, rapidly reduce coolant flow rate through the reactor and increase correspondingly the core and reactor outlet temperatures. Calculations nevertheless indicate that for a pool-type LMFBR with appropriate design of the inlet piping, a double-ended instantaneous rupture of any inlet pipe (even at the inlet plenum nozzle and either with or without reactor scram) would not raise coolant temperature within the reactor (peak local, including all hot channel factors) to the boiling temperature. This result is achieved basically because the hydraulic resistance from the reactor inlet plenum through the reactor and IHXs out to the bulk sodium is adequately low compared to the hydraulic resistance (for reversed flow) from the inlet plenum out through the break to the bulk sodium, so that an adequately large fraction of the total coolant flow continues through the reactor with only an acceptably small fraction going out the break. The resistance in the direction through the reactor in most pool-type LMFBRs tends to be low because of the minimal outlet piping and the very low pressure drop of the IHX. The resistance in the direction out the break can be made sufficiently high without undue penalty by such measures as using two relatively small diameter pipes, instead of one large pipe, from each pump to the inlet plenum. This resistance is significantly augmented in the pool-type LMFBR by the static pressure of the tank sodium external to the break.

In view of the above, it appears likely that in-service inspection would not be required either of reactor outlet piping (or ducting) or inlet piping.

Type of examination: The type of in-service examination that could be conducted on the reactor outlet piping or ducting and the reactor inlet piping is essentially the same as that described for the reactor support cone (conical shell) described under d. Reactor Support Structure.

B. Maintenance

Capability for performing appropriate maintenance is recognized as a requirement of vital importance in LMFBR plants, as in all nuclear plants. For this reason, even though the Pool Study Group work scope did not permit a concerted study of the subject, some brief comments on maintenance are presented here which may be useful. As in the preceding discussion on in-service inspection, the discussion is limited to components within the main tank, with emphasis on aspects for which maintenance considerations potentially may be significantly different from those for LMFBR systems other than pool type.

Some general remarks on maintenance of pool-type systems are made first, and then some of the pertinent experience achieved to date in this country is described.

1. General

The general arrangement of the primary system of a pool-type LMFBR is characterized by very close grouping of the major components. Included among such components are the reactor, primary pumps, IHXs, primary piping, fuel handling mechanisms, etc. All of these are located totally or mostly within the main tank. Accordingly, the main tank cover (or shield deck) penetrations for all of these components that must be removable for maintenance, plus the rotating plugs and numerous auxiliary mechanisms, are positioned within a circular area of diameter equal to the main tank diameter--about 62-70 ft for a 1200 MWe unit. As rough estimation, the large rotating plug required might be 22-32 ft in major diameter, while 9 to 16 large penetrations (nozzles) of 8-11 ft diameter for pumps and IHXs and on the order of 50 small nozzles of 1-3 ft diameter for shutdown coolers, in-tank sodium purification systems, instrumentation, etc., might be needed. Thus, the shield deck area tends to be crowded, and great care must be taken in design to assure adequate room for access to each component. The most restrictive access requirement relates to component removal. For essentially all removable components, removal is accomplished by disconnecting the component from its support nozzle and vertically lifting it into a cask, a rigid caisson (a cylinder-piston arrangement), or a flexible bag while maintaining a continuous seal between tank atmosphere and building atmosphere. Thus, adequate space must be provided to enable access to the entire periphery of each affected nozzle flange, permit attachment of cask, caisson, or bag to the nozzle or adjacent deck, and accommodate any required sealing provisions including a slide valve or "trap door" to close off the nozzle opening after the component has been removed. Also, where significant shielding is required around the removed component (basically, only for the relatively small components removed from the reactor itself), space must be available for accommodation of such shielding without interference. Potential interferences stem primarily from adjacent nozzle flanges (and the upward-projecting components or mechanisms) and from the shield-deck beam flanges; ribbing in the deck is probably at too low an elevation (below the nozzle flanges) to produce interference. Provision of minimum clearances between adjacent nozzle flanges and/or beam flanges of approximately 6-8 in. probably is adequate to preclude restriction of access from this source. Obviously, care also is required to locate the many cable runs, instrument leads, cooling pipes, and other such required equipment above the shield deck in such manner as not to unduly interfere with component removal or servicing. That this can be done in a practical manner is probably best attested to by the EBR-II experience. Because of its small size (about 27-ft main tank diameter) but essentially same number of support nozzles (total of 67) as expected in a commercial unit, the deck above the EBR-II main tank tends to be even more crowded than for a commercial-size system. The maintenance experience with EBR-II, nevertheless, has been good.

It should be noted that some primary system components are not removable from the main tank for maintenance and, because they are submerged in sodium within the main tank, could be serviced only with extreme difficulty or not at all. Principal examples of such components are the fixed portions of the reactor inlet piping, the reactor outlet ducting (or fixed portions of outlet piping, if any), and the reactor support structure. (See Sec. IX.A.3.d and IX.A.3.e for further comments relating to accessibility and inspection of these components.) Accordingly, it is necessary to design and construct all such components in such a manner as to make negligible the probability of their failure during the service life of the plant.

Maintenance of reactor core instrumentation is essentially the same as for other types of LMFBRs; in all types, sensors and leads for such instrumentation must be designed to be remotely and easily replaced. Maintenance of other instrumentation in the primary system, however, is more difficult because much of it is submerged in sodium and some of it is most conveniently mounted on components or structures which are not removable. Because of this potentially restricted access, it is of paramount importance to locate as large a fraction of the instrumentation as possible on the removable components (pumps, IHXs, etc.) and provide for remote replaceability of the remaining critical instrumentation (or arrange for sufficient redundancy or appropriate quality and integrity) to assure retention of an adequate instrumentation base throughout plant life.

The reactor, fuel handling mechanisms, rotating plugs, and the control rod drives and other mechanisms mounted on the rotating plugs are either identical to or very similar to those employed in LMFBR plants of types other than the pool type; hence, their maintenance complexities and considerations are also similar and are not addressed here. Similarly, the maintenance of most auxiliary system components is not basically different.

In general, experience with pool-type systems (EBR-II, PFR, Phenix) to date has not revealed excessive difficulty with maintenance. On the contrary, based primarily on EBR-II experience (because PFR and Phenix have been in operation for only one or two years and little pertinent maintenance information from them has become available), maintenance experience appears to have been quite satisfactory. In the following section a brief review of EBR-II experience over the last dozen years is presented.

2. EBR-II Maintenance Experience

a. Operating History

After initial sodium filling of the EBR-II primary and secondary systems in January-February 1963, wet critical experiments were performed (November 1963). The approach to power started in July 1964 and was concluded in March 1965, when the reactor power was raised to 45 MWt. Until

May 1965, the plant operated in fulfilling the initial design goal: demonstration of a sodium-cooled fast breeder reactor as a central station power plant. In May 1965, EBR-II began its role as an irradiation test facility with the insertion of the first experimental assembly. Reactor power was raised to 50 MWt in August 1968 and to 62.5 MWt in September 1969.

The burnup limit for the core driver fuel has been raised significantly above its initial conservative level, and the length of reactor runs has been increased to 2700 MWt-days to obtain maximum utilization of the fuel. At the end of 1974, the cumulative thermal and electrical power production of EBR-II were 2,070,383 MW-hours and 576,207 MW-hours, respectively.

Because the primary function of EBR-II is that of an irradiation facility, the irradiation program determines to a large extent the programming of operation of the reactor. At the present time, EBR-II performs five to six reactor runs each year. The length of each reactor run is 2700 MWt-hours (43.2 days at full-power operation). About seven days are allowed between runs to permit insertion and removal of experiments, reactor refueling and maintenance. Each year the plant is shut down for 45 to 60 days for scheduled annual maintenance. During the annual shutdown, the sodium systems are lowered to about 350°F, the secondary sodium is drained to its storage tank, and the power plant is cooled to ambient temperature. Major maintenance and inspections and any desired major modifications are carried out during this extended outage.

b. General

A preventive maintenance program for in-sodium components is inherently quite limited. The philosophy at EBR-II has been to provide close surveillance of all sodium components. This is particularly true of fuel handling components and primary system components in the main tank. Whenever imminent trouble with an in-sodium component is indicated, a plan of action is immediately formulated, and the component is repaired or replaced as soon as plant conditions permit. This "in-sodium philosophy" contrasts with that used in the steam system, in which the conventional components are disassembled for inspection or replaced on a preestablished frequency.

Some difficulties due to various causes have been experienced with major in-sodium components. Normally, these were readily resolved by a minor design modification. In general, excellent service has been obtained from the in-sodium components, particularly considering the experimental nature of some of the equipment. To date, no failures or incidents have occurred in which serious in-core or out-of-core consequences resulted. The primary effect of failures of in-sodium components has been additional reactor downtime.

In-sodium maintenance activities can be divided arbitrarily into three general classes:

(1) Work involving components that are highly radioactive because of neutron activation and sodium contamination (i.e., components that reside in or penetrate the reactor vessel).

(2) Work involving components that are somewhat contaminated by sodium and fission products, but are not highly radioactive (i.e., components external to the reactor vessel, but in the main tank).

(3) Work involving components that are fairly accessible, but penetrate into sodium systems (primary and secondary) and may or may not be radioactive.

Examples of Class (1) components involved are control rod drives, main fuel gripper, and assembly holddown. These components all penetrate the reactor vessel cover and, in addition to being sodium-contaminated, become highly radioactive because of neutron activation. A control rod drive that is sodium covered and fission product contaminated and has a radiation level of 300 to 500 R/hr requires considerable care in removal, even without any attempt to repair it. For Class (1) items, identical interchangeable assemblies are kept in a spare parts inventory. For example, if a control rod drive fails, the plant is shut down and the primary system is cooled. Detailed procedures for removal operations are available, as is all handling equipment that is required. In two days the drive can be replaced with a spare unit, and the plant can be returned to power. The failed drive is then disassembled at leisure, and many of the nonactivated parts are reused in building a new spare.

Class (2) work involves components that are outside the reactor vessel, but reside in the main tank. The problem of high radiation level is not present, but problems with surface contamination by sodium and fission products are present. Examples of Class (2) components are the fuel transfer arm and the primary pumps. These components can be removed, cleaned, repaired and reinstalled. The size and expense involved with Class (2) components have precluded the stocking of complete spare assemblies. Instead, only long-lead parts, easily damaged components, or parts that are expected to wear out are kept in the spare parts inventory. Examples are bellows, bearings, and especially complicated shafts. If a Class (2) component must be removed, it can be repaired and returned to service in a short time.

Class (3) work involves components that are generally more readily accessible, but penetrate into sodium systems. Examples include the components in the primary and secondary auxiliary systems (such as the sodium purification systems). For this class of components, most work is performed on a routine scheduled maintenance program. The frequency of inspection and maintenance is based on past operating history, manufacturer's

recommendations, and the importance of the equipment to plant operation. Most Class (3) items are plant-controlling items only in an indirect sense, as they may not require an immediate reactor shutdown. Careful scheduling and planning for Class (3) maintenance limit plant downtime for these items.

Removal of components from the main tank is a job that has been done routinely and without significant problems. For example, since the main tank was filled with sodium, a control rod drive has been removed and reinstalled sixteen times, the fuel gripper has been removed and reinstalled four times and the fuel holddown mechanism once, the transfer arm has been removed and reinstalled once, and a primary pump has been removed and reinstalled three times.

General precautions taken when major components are being removed from or inserted into the primary tank are the lowering of the temperature of the main tank to as low as 260-350°F (depending on the configuration of the component to be removed), the maintaining of an inert-gas atmosphere on the component, and the maintaining of a positive pressure of the main tank cover gas of about 1/8 in. water gauge to minimize air inleakage or cover gas outleakage. Also, to permit radioactive sodium and cover gas to decay, the actual removal operations are not usually attempted for a couple of days after shutdown. Personnel wear full protective clothing during the removal.

Only one incident of any kind has occurred during the many interchanges of components within the main tank. A chemical reaction took place when the auxiliary gripper plug was being reinserted. A small amount of sodium was expelled around the plug shaft and up through the tank penetration. Investigation indicated that the auxiliary gripper plug had not been adequately dried after cleaning, and the residual alcohol-water cleaning solution reacted with the sodium. Damage to equipment due to the expelled sodium was insignificant, and contamination was minor. The auxiliary gripper plug was not damaged.

c. Control Rod Drives

A control rod drive is a typical example of components that are removed or installed in the main tank. As mentioned, this particular component has been removed and replaced sixteen times. Mechanical problems necessitating component removal have occurred ten times in control rod drives--three times from gripper jaw actuation difficulty, and seven times from bellows failures in the drive shafts. On six occasions, accommodation of in-reactor experiments necessitated drive removal. It should be noted that five of the original twelve drives have never been disturbed or removed and have operated perfectly for over eleven years. The following describes the basic equipment and procedure used for removal.

A special small-diameter cask and several attachments were originally designed for removing the first failed control rod drive. This cask and its handling procedures have been found to be applicable to almost any component that has been installed in an EBR-II control rod position.

The cask consists of a 31-ft-long piece of 3-in. Schedule 40 pipe. A 20-in.-dia flange is attached at the upper end, and a sliding gate valve is provided at the lower end. A piston (which also acts as the lifting attachment) using double O-ring seals fits inside the long tube. The surface of the piston and tube are both coated with grease to reduce friction and maintain an argon seal over the full piston travel. The cask is supported by the use of the reactor building crane and a spreader bar. A dynamometer and chainfall are attached to the piston in the cask. To operate the chainfall 30 ft above floor level, a special basket is provided for two operators and is supported by a second crane hook.

When highly radioactive components are removed, this arrangement allows the operation to be controlled remotely. The operators in the crane basket have control of the crane, control of the removal cask, and control of the lifting piston in the cask from a distance of at least 25 ft. The only operations required at the floor area are the opening and closing of the cask and the main tank isolation valves. If required, these operations are done manually by the use of extension rods and local operator shielding.

The proximity of the adjacent control rod drives (which are less than 6 in. apart, center to center) and the central support column to a drive being removed does not allow much room for radiation shielding on the cask. The bottom 3 ft of the cask has approximately 3/8 in. of lead shielding, which is all that can be installed. The remote-operating capabilities of this system have compensated for the minimal shielding, and control rod drives which are usually at a radiation level of 300 to 500 R/hr can easily be removed or installed with individual personnel exposures of less than 25 mR. It is expected that in a commercial LMFBR, because components and clearances would be significantly larger, it would be considerably easier to provide the desired shielding.

d. Fuel Handling System

The fuel handling system is one of the most complex systems at EBR-II. The components making up the fuel handling system are probably the most difficult to maintain and repair, and require the talents of personnel specifically trained for this work.

Most of the fuel handling components are original equipment and have operated many thousands of times with only minor difficulties. Some design changes were required after initial checkout, and a few changes were made after operating experience was gained; but, in general, the system has

performed very satisfactorily. As of January 1, 1975, operating experience with the fuel handling system consisted of over 15,000 satisfactory operations. There appears to be no reason why the fuel handling components will not continue to function properly for an indefinite period in the future. Replacement or repair of any component, if one fails, has been demonstrated in a number of instances.

The rotating-plug seals and their control system have been the principal sources of the minor fuel handling delays that have been experienced since the fuel handling system became operational. The fuel handling system has extensive control-circuit protective interlocking; this interlocking is used to prevent damage to reactor, reactor assemblies, and the fuel handling equipment, and also to assure precise positioning of the gripper mechanism over any desired point in the reactor. Although effective in preventing equipment damage, this interlocking occasionally causes delays. Time required to troubleshoot and repair control circuitry and equipment is extended by the limited accessibility of relays and position limit switches. It should be noted that total delays in fuel handling, from all sources, has been small.

(1) Gripper Mechanism

The gripper mechanism is an electromechanical device that is used to insert assemblies into, or extract assemblies from, the reactor vessel support grid. The gripper mechanism grasps the upper adapter of the assembly and moves the assembly vertically out of or into the reactor vessel. In January 1964, the gripper was removed from the main tank to investigate the cause of sticking. A large quantity of sodium and sodium oxide was found on the shaft and guide tube at a point just above the sodium level. To minimize the buildup of sodium oxide between the gripper shaft and its guide tube, three gas-vent holes were drilled in the guide tube. These holes allowed the argon cover gas to enter and pressurize the previously stagnant space in the guide tube annulus and greatly reduced the tendency for air inleakage due to a pumping action when the gripper shaft was raised and lowered.

Later the same year, the gripper was again removed. Damage had occurred to the gripper guide funnel when the funnel interfered with the holddown while the gripper was being lowered. The guide funnel was redesigned to have less tendency to catch on the holddown. The gripper operated routinely until 1967, when difficulty was encountered in releasing an assembly to the transfer arm. The gripper was removed, and a visual inspection was made remotely of the gripper jaws and sensing blade. This inspection revealed that the gripper was in proper working order; the difficulty was found to be due to the assembly top adapter, not the gripper.

No problems were encountered for the next seven years. In 1974, the gripper and guide tube were both removed following sticky operation of the gripper shaft and an infrequent malfunction of the gripper jaws. The

gripper jaws were inspected with mirrors and a camera, and were found to be in satisfactory condition. Once again, a buildup of sodium oxide was found on the gripper shaft and guide tube just above the sodium level. After removal of the sodium oxide buildup, the gripper and guide tube were reinstalled and have since operated satisfactorily.

(2) Holddown Mechanism

The holddown mechanism operates in conjunction with the gripper mechanism. During insertion or removal of an assembly from the reactor vessel, the holddown mechanism functions as a guide funnel and holds down and prevents inadvertent extraction or movement of adjacent assemblies. The mechanism has operated virtually trouble-free for over 25,000 cycles, with one exception. Early in 1970, the holddown began sticking near the upper end of its vertical travel. The problem appeared to be similar to the problem with the gripper mechanism in 1964. A sodium oxide buildup was apparently forming in the cover-gas space between the holddown shaft and the guide tube. The design of the holddown provided for removal of the guide tube without removal of the holddown shaft. To provide better circulation of cover gas in the guide tube, the guide tube was removed, cleaned, modified, and reinstalled.

(3) Transfer Arm Mechanism

The transfer arm mechanism is a manually operated and electrically interlocked device that transports assemblies and other core components within the main tank. One difficulty occurred which required the transfer arm mechanism to be removed from the main tank. In February 1966, the locking/sensing pin drive mechanism failed because of a defective weld and severe oxidation of the shaft bushings; these bushings were exposed to a 700°F air environment. The transfer arm and its shielding plug were removed by raising them into a specially fabricated rubberized nylon bag that maintained the inert gas atmosphere. The lower end of the bag was attached to an adapter section. This adapter contained several viewing windows and a cover plate. After removal of the transfer arm and its plug, the tank opening was sealed with the cover plate. The transfer arm was removed from its enclosing bag after having cooled to room temperature. Use of the rubberized nylon bag proved very successful. The bag required a minimum of time to design and fabricate, and was not expensive. The transfer arm has operated reliably since reinstallation, which was by the same method.

(4) Large and Small Rotating-Plug Seals

The rotating-plug seals, which are a trough-and-blade assembly filled with a low-melting alloy of 42 wt % tin and 58 wt % bismuth, are designed to contain the primary cover gas and permit rotation of the plugs for fuel handling. Past experience has shown that the tin-bismuth alloy, which is heated to above its melting point (280°F) during the fuel handling operations,

oxidizes at its surface on the air side of the trough. The accumulation of this oxide, plus uneven heat distributions in the seal trough, have caused chronic sticking problems with the large and small rotating plugs. To initially achieve free rotation, the seal alloy has sometimes required heating to elevated temperatures (400 to 500°F), and manual force has been applied to the plugs.

Starting in 1966, access holes were drilled from the top of the plugs and through the side of the seal troughs (above the alloy) so the outer annuli of the seals could be inspected and cleaned. A program of periodic cleaning of the seals was initiated in conjunction with careful temperature control. Several cleaning methods have been used; these methods include vacuuming, brushing, scraping, and skimming of the seal alloy. A routine procedure of cleaning and brushing, now scheduled between every other reactor run, has improved the reliability of plug rotation. The program of cleaning the seals, and the use of elevated seal alloy temperatures for initial rotation, have significantly reduced the frequency of plug sticking.

An additional inspection and maintenance hole was drilled through the top of the large plug over the inner annulus of the seal in 1975. Tools and techniques are now being prepared to enable removal of sodium and sodium oxide deposits from the gap between plug and structure. It appears that with frequent removal of the oxide, relatively trouble-free rotation of the plugs can be achieved, although the cleaning is a time-consuming maintenance item.

e. Primary Pumps

Initial operation of the primary sodium system began in April 1963. Binding occurred in both pumps during their first six months of operation. Failure was caused by excessive rubbing between the pump shaft and the lower labyrinth. Both pumps were removed and modified to eliminate the rubbing. Modification consisted of replacing the pump shafts and increasing the clearances between the shaft and the lower labyrinth and baffle plates.

Since initial modification, primary pump No. 2 has performed satisfactorily for over 53,000 hours. Pump No. 1 experienced two minor incidents of binding while shut down, attributed to the buildup of sodium and/or sodium oxide in the running clearance between the pump shaft and lower labyrinth. After each binding, the shaft was manually rotated by the application of about 200 ft-lb of torque to begin rotation. The shaft then rotated freely. The pump subsequently performed satisfactorily until mid-1970, when periodic increases in power consumption were noted with no corresponding change in primary sodium flow. Further investigation indicated that excessive torques were required to rotate the pump shaft. Although some success was achieved in providing freer rotation of the shaft, the pump was scheduled for removal and examination as a precautionary measure. Inspection after removal and disassembly revealed a considerable buildup of sodium and sodium

oxide on the pump shaft, inside the lower labyrinth, and on the shield-plug liner above the labyrinth. Shaft rubbing had occurred in these areas. Sodium oxide had corroded or eroded the internal serrations of the lower labyrinth. The lower labyrinth was replaced with a unit of modified design, and minor rework was done on the baffle-assembly drain holes to improve the argon purge in a manner expected to reduce the buildup of sodium oxide.

Since the repair of this pump, no other problems, except minor malfunctions of the electrical control components, have been encountered.

The same basic techniques were used in removing the primary pumps as used for the control rod drives, but special equipment had to be designed and fabricated because of the size and shape of these components. No augmentation of shielding was required because of the low radiation level.

To permit removal of the primary pumps, a cylindrical caisson and special lower adapter were built. The cylindrical caisson was installed over the pump, and the pump was lifted by the building crane into the caisson in an inert gas atmosphere. A sliding piston (with rubber piston rings) in the caisson maintained the inert gas atmosphere as the pump was raised out of the sodium tank. The special lower adapter served as the connector between the pump mounting flange and the caisson. This also contained an isolating device in the form of a trap door which closed to seal the sodium tank upon removal of the pump and allowed the isolation of the pump in the caisson. After the pump was lifted into the caisson and isolated, it was cooled to room temperature. Air was slowly introduced to oxidize the sodium adhering to the pump. After oxidation of the sodium, the pump was removed from the caisson, cleaned and repaired. The pump was inserted by essentially the reverse of the removal procedure.

f. Instrumentation

Instrumentation failures have occurred within the main tank; these primarily have been failures of thermocouples, resistance thermometers, and pressure transducers or flowmeters (sensors or their leads). Some were within the reactor, such as assembly outlet thermocouples, and others outside the reactor. The assembly outlet thermocouples were designed to be replaceable, and the failed elements were easily replaced. Many of the other sensors (and leads) were not so designed, however, and the failed units could not be replaced or repaired. Because of the redundancy of instrumentation provided originally, sufficient usable instrumentation remains to enable operation. In a commercial LMFBR it would be prudent to design a larger fraction of the primary-system instrumentation to be readily replaceable.

g. Intermediate Heat Exchanger (IHX)

The IHX is located within the main tank. The primary sodium from the reactor passes through the shell side of the tube bundle and returns to the primary tank. The nonradioactive secondary sodium flows through the tube side to remove heat from the primary circulation system. Except for a single event involving a drain tube, operation of the IHX has been trouble-free. On November 14, 1970, a banging noise was heard in the vicinity of the IHX. Indications were that the noise source was within the IHX inlet pipe. A port for access to the IHX internals was installed on the inlet-pipe elbow over a rectangular hole that was cut in the elbow. Visual examinations, by the use of both a periscope and a remote TV system, revealed that two support clips holding a 1-in.-dia drain tube in place were not in their originally installed condition. The top clip was loose, and the bottom clip was missing. This condition allowed the tube to move and vibrate against the wall of the 12-in.-dia inlet pipe. The upper clip was removed; the drain tube was cut at the top and bottom and removed. The section cut out of the inlet elbow was rewelded in place, and the secondary system was restored to operational status. Quiet operation of the IHX verified that the repair was successful.

h. Conclusions

A review of the operating experience of EBR-II reveals that all the original design objectives have been successfully demonstrated. In addition, the usefulness of the facility has been greatly extended by redirection of its mission to conduct an ambitious irradiation and testing program. The unique design of the fuel handling system, which provides for very rapid refueling of the reactor, results in highly efficient use of the reactor.

The facility has operated very well and continuously except for short shutdowns required by modifications, maintenance, refueling, and minor repair. EBR-II has demonstrated reliability and maintainability.

Maintenance experience at EBR-II over the past 12 years has shown that maintenance is not a difficult task for a sodium-cooled pool-type system of this size. In fact, in the heat-transfer piping system there is very little maintenance--less than in the conventional steam systems. Repair of components such as fuel handling components is practical, and if replacement units are available, exotic repair facilities are not necessary. The maintenance personnel for the reactor system must be specially trained, but the normal mechanical work force for repair and maintenance requires only three men. The radiation exposure of personnel, both maintenance and operational, is low.

APPENDIX A

Notes on Shielding for a Commercial Pool-type LMFBR

1. General Considerations

The neutron shielding problem in an LMFBR is basically a problem of slowing down fast neutrons. Once they have been slowed down, they are absorbed in most materials in a relatively short distance. Heavy materials such as steel, which have high inelastic scattering cross sections, are more effective in stopping the high-energy neutrons ($E > 0.1$ MeV) than the common moderator materials. Some pertinent basic nuclear properties of iron and carbon are given in Table A.1. The macroscopic fast-neutron removal cross sections (Σ_{rem}) shown are from measurements performed with the Bulk Shielding Facility at the Oak Ridge National Laboratory. The fast-neutron removal cross section of iron is seen to be almost three times that of carbon.

TABLE A.1 Basic Properties of Shield Materials

Material	Fast-neutron Removal Cross Section (Σ_{rem}), cm^{-1}	Thermal-neutron Slowing-down Power ($\xi\Sigma_s$), cm^{-1}	Thermal-neutron Capture Cross Section (Σ_c), cm^{-1}	3-MeV Gamma Removal Cross Section (μ_t), cm^{-1}
Iron	0.16	0.032	0.214	0.284
Carbon	0.06	0.060	0.00026	0.080
Sodium	0.03	0.008	0.013	0.034
Boron	0.13	0.084	103.	0.070

The thermal-neutron slowing-down power ($\xi\Sigma_s$) of carbon is twice that of iron; however, this is of little consequence in fast reactor shielding, where thermalization is less important than stopping of the high-energy neutrons. The high-energy neutrons tend to scatter in the forward direction and travel longer distances between collisions than thermal or epithermal neutrons. Required shield thicknesses, therefore, depend primarily on the fast-neutron removal cross section. If the shield is sufficiently thick to stop fast neutrons, it will generally be more than sufficiently thick to remove the less energetic neutrons.

The low capture cross section of carbon is a distinct advantage when carbon is used as a moderator, but a disadvantage in shielding applications. The macroscopic gamma removal cross section for 3-MeV gammas is much higher for iron (0.284 cm^{-1}) than for carbon (0.080 cm^{-1}). Iron, however, has the serious disadvantage of producing energetic (7 and 9 MeV) gamma rays as a result of neutron capture. Whenever these secondary gammas become limiting in the shield design, it is important to include in the design a material, such as boron, that has a high neutron capture cross section and produces no secondary gamma radiation.

The relative shielding characteristics of steel (iron) as discussed above suggest that use of shielding composed entirely of steel without any graphite (carbon) may well be feasible, with consequent elimination of requirements for in-service replaceability. This is the type of shielding system examined in this Appendix.

The shielding provided must reduce the neutron flux to acceptable levels primarily with respect to: (1) radiation damage in structural materials; (2) activation of repairable components; (3) activation of the secondary (intermediate) system sodium; and, (4) biological dose to operating and maintenance personnel. Some general indications of the amounts of shielding found likely to be required for each of these purposes in a pool-type LMFBR are given below.

In respect to radiation damage in structural materials, the principal components requiring protection are the core barrel, instrument tree, and core support structure, all other major components being subjected to a significantly lower fluence. Adequate protection can be achieved by providing approximately 13 in. of depleted uranium blanket and 12 in. of steel shielding (the reactor reflector) between the reactor core and these components.

In a pool-type system in which components requiring removal for maintenance, such as pumps and intermediate heat exchangers (IHXs), are located in the sodium pool at about, say 16 ft from the reactor vessel, the activation of these components by neutron capture can be kept to levels that allow efficient maintenance by providing the equivalent of about 16 in. of steel shielding additional to the 12 in. required for protection of the core barrel. (Ease of maintenance may be significantly affected, however, by fission products and corrosion products that may be deposited on the component surfaces.)

Activation of the secondary system sodium can be kept to acceptable levels by providing adequate shielding between the core and the IHXs. In a pool-type system, approximately 18 in. (minimal) of steel shielding additional to the 12 in. required for the protection of the core barrel are needed for this purpose. This again assumes the presence of about 14 ft of sodium between the reactor and the IHXs. The neutrons contributing significantly to the activation of the secondary sodium have energies in the range of 2 to 300 keV. The sodium in the main tank is an effective shield for the secondary sodium in the IHXs. As shown in Table A.1, the fast-neutron and gamma removal cross sections of sodium are about half those of carbon. Sodium, therefore, is not as good a shield material as carbon, but it is a very significant one. If the sodium were not present in the tank, approximately 3 ft of steel would be required to replace the 14 ft of sodium between the reactor and the IHXs.

Shielding requirements for biological protection outside the main tank stem from the gamma flux from the activated sodium, the neutron flux at the tank boundary (including its effect in producing secondary gammas by neutron capture in the external biological shield), and to a lesser extent the gamma flux

from the reactor itself. In the radial direction, the 18 in. of steel shielding provided outside the reactor reflector--required to suitably limit the secondary system sodium activation--is adequate to limit these sources so that biological dose limits can be achieved with a reasonable thickness of biological shield, for example, 6 ft of 200-lb/ft^3 concrete. In the vertical direction, the introduction of about 21 in. of additional shielding (in excess of the 12 in. of steel above the upper axial blanket mentioned earlier), either by extension of the top portion of the reactor assemblies or by incorporation of additional steel within the reactor vessel cover, is adequate to limit the same sources if a biological shield thickness of 5 ft of 240-lb/ft^3 concrete or 30 in. of steel is used. With these shielding thicknesses (equivalent to 21 in. of steel vertically and 18 in. radially) locally surrounding the reactor, the neutron flux level and the level of gamma flux from the core are reduced to practically negligible values at the tank boundary. The dominant source of the potential biological dose outside the main tank is, then, the gamma flux from the activated primary system sodium. With the amounts of shielding indicated locally surrounding the reactor, almost all of the sodium activation occurs within the core and blanket of the reactor, and additional shielding would be of only minor value in reducing further the level of activity. A specific activity of about 3 Ci/ℓ is reached. For adequate biological protection outside the tank, the approximate biological shield thicknesses stated above are needed.

2. General Criteria

The principal shielding criteria assumed are listed below. These are general and essentially qualitative. More specific criteria are described in Part 3 of this Appendix.

The shielding should be designed so that the following conditions are met:

(1) Design lifetimes of the various structural components are not reduced by excessive radiation damage or excessive heating rates.

Accordingly, the neutron fluence incurred by each component must be suitably limited, depending upon the material, operating temperature, and required ductility. Approximate total fluence limits for some typical conditions are shown in Table A.2. These are for base metal; for welds, somewhat lower values are appropriate, dependent upon the type of weld. The specific limits used for the study are given later, and although stated in terms of fast ($E > 0.1$ MeV) fluence, may reasonably be compared with the total fluence limits indicated.

(2) Maintenance of repairable components is not excessively inhibited by component radiation due to neutron activation.

For major components, such as primary pumps and IHX tube bundles, it is assumed that dose rates at the component surfaces at the time of completion of component removal from the system should not exceed about 20 mR/hr from this source.

(3) Routine access to secondary (intermediate) system piping and components is not precluded by radiation due to neutron activation of secondary system sodium.

(4) Personnel access to the various operating stations both during normal operation and during shutdown can be achieved while maintaining occupational radiation exposures as low as practicable.

TABLE A.2. Approximate Total Fluence Limits for Types 316 and 304 Stainless Steel

Material	Temperature, °F	Fluence Limit, 10^{22} n/cm^2 5% Total Elongation	10% Total Elongation
SS 316	700	2.0	1.6
SS 316	800	2.5	1.9
SS 316	900	2.5	1.9
SS 316	1000	2.4	1.8
SS 304	700	1.7	0.9
SS 304	800	2.1	1.1
SS 304	900	2.1	1.1
SS 304	1000	2.3	1.2

A breakdown of design dose rates for various plant areas generally similar to that proposed for CRBR is employed. Five general types of areas, or zones, are recognized. "Routinely occupied" areas (Zone I) are those in which personnel are present for a period of two hours or more per day, but for not more than forty hours per week. "Nonroutinely, but frequently occupied" areas (Zone II) are those occupied for less than two hours per day. "Periodically occupied" areas (Zone III) are those occupied for not more than one hour per week, entered essentially only for routine inspection or maintenance. "Nonroutinely, occasionally occupied" areas (Zone IV) are those entered only on very selective occasions. The last class consists of those areas which are "normally inaccessible" (Zone V). Table A.3 shows the design dose rates assumed for the various areas; all are considered conservative and fall well within the guidelines of 10 CFR Part 20.

Of principal importance to this study are Zone I and Zone II areas. Assumed to be Zone I are: areas immediately external to the radial biological shield surrounding the guard tank; the area immediately above the rotating plugs; and the areas above the shield deck over the main tank. Assumed to be Zone II are areas immediately surrounding the intermediate system piping, pumps, steam generator, etc.

TABLE A.3. Biological Criteria

Area or Zone	Maximum Design Dose Rate, mrem/hr	Zone Description	Design Occupancy Time
Zone I	0.2	Routinely occupied	Full time
Zone II	2.0	Nonroutinely, frequently occupied	500 hr per yr
Zone III	20	Periodically occupied	50 hr per yr
Zone IV	200	Nonroutinely, occasionally occupied	5 hr per yr
Zone V	200	Normally inaccessible	None

The above general criteria are not meant, of course, to be comprehensive. Numerous other shielding requirements exist, but these are not treated in this report. Examples are: Spent fuel stored in the main tank, if any, must be shielded adequately to preclude excessive fission heat generation or significant addition to dose rate outside the biological shielding; and shielding must assure that background radiation in the vicinity of neutron detectors is kept to a tolerable level considering the sensitivity and required lifetime of the detectors.

3. Shielding Scheme for 1200-MWe EBR-II-type LMFBR

A shield system design was developed for the normalized (1200-MWe) EBR-II pool concept, and preliminary calculations made to determine the system effectiveness. While the specific quantitative results obtained apply only to this particular system, the qualitative results and conclusions are considered to apply generally to the other pool-type reactors reviewed in the study.

A schematic vertical section of the primary system area of a conceptual 1200-MWe pool-type LMFBR based on the EBR-II design approach is shown in Fig. A.1. Represented are the reactor, an IHX, the main tank, guard tank, main-tank cover, rotating shield plug, top biological shield (shield deck), and radial biological shield. The reactor employed is the reference reactor (with minor deviations) described in Sec. III. Proceeding radially outward from the reactor core, the principal locations requiring consideration of shielding are noted in the figure: (1) the core barrel; (2) the IHX tube bundle; (3) the intermediate system sodium within the IHX; and (4) the area immediately external to the radial biological shield. In Fig. A.2, proceeding vertically from the reactor core, similar locations are noted: (5) the reactor assembly support grid; (6) the instrument tree positioned in the reactor outlet plenum; (7) the area on top of the rotating plugs; (8) the general area above the shield deck; and (9) the peripheral area above the shield deck.

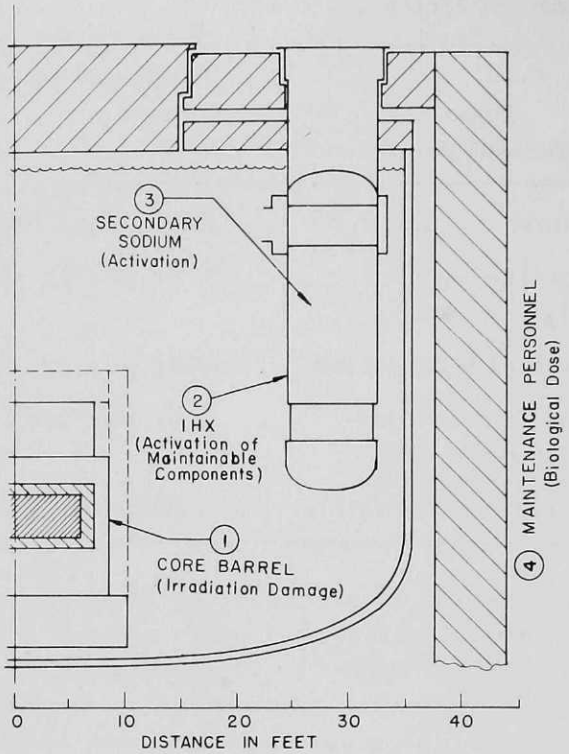


Fig. A.1

Schematic Vertical Section of Conceptual Reference Reactor, Showing Shielding Areas: Proceeding Radially from Core

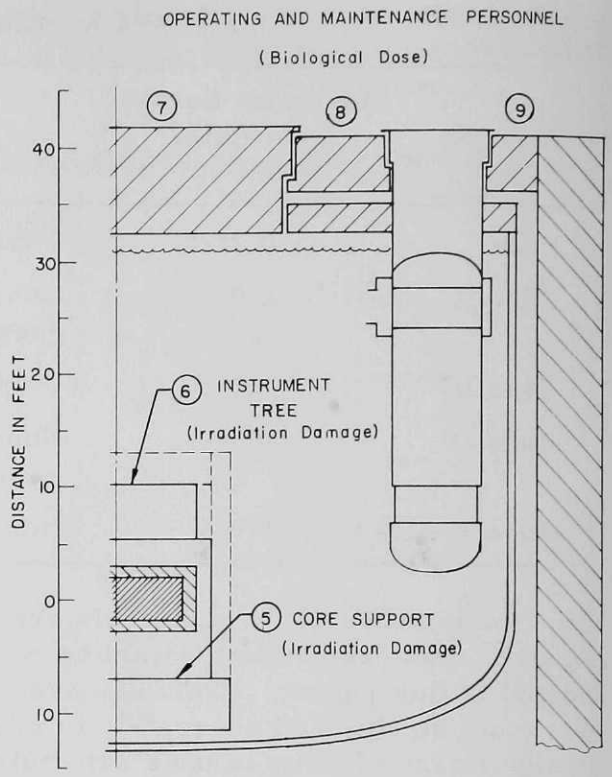


Fig. A.2

Schematic Vertical Section of Conceptual Reference Reactor, Showing Shielding Areas: Proceeding Vertically from Core

The principal shielding criteria which it is assumed must be met at each of the above locations are described next.

Core barrel. The reactor radial blanket and reflector (shield) must reduce the fast flux from the core sufficiently to prevent excessive radiation damage to the core barrel. The critical areas of the core barrel are those in the high-fluence region near the elevation of the core midplane. At this location, the temperature will be near the inlet sodium temperature (720°F). In order to assure adequate residual ductility in the structure at this temperature, it is assumed that the neutron fluence, over the life of the plant, should not exceed about 1.1×10^{22} neutrons/cm^2 for neutrons with energies greater than 0.1 MeV. Assuming a 30-year life and a plant capacity factor of 0.82, this corresponds to an allowable flux level of 1.4×10^{13}/cm^2-sec ($E > 0.1$ MeV).

IHX tube bundles and primary pumps. Those components which will be or may be removed from the main tank at some time during the life of the reactor for repair or maintenance, such as IHX tube bundles and primary pumps, must be protected from excessive neutron activation. For maintenance of these components without additional burden from avoidable radiation, it is assumed that the dose rate at their surfaces from this source should be less

than about 20 mR/hr 48 hr after shutdown. At the locations of these components, an equilibrium flux spectrum characterized by the nuclear properties of the interposed primary sodium is established. The thermal component of this spectrum is primarily responsible for the activation of the steel components. The required dose rate limit can be achieved if the thermal flux on the surface of the repairable components is kept to less than about $10^8/cm^2$-sec. In the design life of the plant, as above, this results in a specific activity of the component steel of about 10^{-5} Ci/cm^3 or less. (It should be noted however, that the induced activity could be significantly less than the activity from surface contamination by fission products and corrosion products. Consideration should be given in system design and operation to minimization of the surface contamination.)

Intermediate system sodium within IHX. In order to permit routine, frequent access to the intermediate system sodium piping, pumps, steam generator, and other components, the dose rate at the surfaces of these components (Zone II areas) should be kept to less than 2 mR/hr. Calculations indicate that for a system volume to IHX volume ratio of 20 or higher, an IHX diameter on the order of 9 ft, and using a 2-meter-dia sphere as representative of the largest system source, this dose rate can be achieved if the total flux incident on the nearest edge of the IHX does not exceed about 5×10^7 n/cm^2-sec. This results in a specific activity (^{24}Na) of the intermediate sodium of about 10^{-5} Ci/ℓ or less.

Area external to radial biological shield. Outside the radial biological shield, a Zone I area, the dose rate should be small enough to allow continuous access for operating and maintenance personnel. Accordingly, it is assumed that shielding provisions should be designed to maintain the dose rate in such areas at 0.2 mR/hr or less. The corresponding allowable neutron fluxes are noted later, under "Areas above rotating plug and shield deck."

Support grid and instrument tree. Unless adequately protected, the reactor assembly support grid and the instrument tree could undergo excessive radiation damage. Because replacement of these components during the lifetime of the plant would be difficult or impossible, such protection is required. The support grid operates at a temperature of 720°F. To preserve adequate residual ductility, it is assumed that the neutron fluence on this component should be kept to less than 1.1×10^{22} neutrons/cm^2 for neutrons with energies greater than 0.1 MeV. For the assumed 30-year life at 0.82 plant capacity factor, this corresponds to an allowable flux level of 1.4×10^{13} n/cm^2-sec. The instrument tree is at the temperature of the reactor outlet sodium (1000°F), and to assure sufficient residual ductility of this weldment, it is estimated that the fluence should not exceed approximately 3×10^{21} neutrons/cm^2. For the service indicated, this corresponds to an allowable flux of 4×10^{12} n/cm^2-sec.

Areas above rotating plug and shield deck. The areas above the rotating plug and shield deck should be available for routine operation and maintenance.

Accordingly, the dose rate for these (Zone I) areas should be less than 0.2 mR/hr total. This dose rate may result from gamma rays or from neutrons. The corresponding allowable neutron flux depends on the energy of the neutrons. As representative examples, a flux of 200 thermal neutrons/cm^2-sec or 3 fast ($E > 1.0$ MeV) neutrons/cm^2-sec is allowable.

The shielding criteria for all of the critical areas discussed are summarized in Table A.4.

TABLE A.4. Summary of Specific Shielding Criteria

Location	Criterion
1	Fluence >0.1 MeV = 1.1×10^{22} n/cm^2; corresponds to flux >0.1 MeV = 1.4×10^{13} n/cm^2-sec (0.82 capacity factor, 30-year life).
2	<20 mR/hr (from induced activity) at component surface after 48 hr; corresponds to thermal flux ≅ 10^8 n/cm^2-sec.
3	<2 mR/hr at surface of secondary system piping and components; corresponds to total flux ≅ 5×10^7 n/cm^2-sec.
4	<0.2 mR/hr dose rate; corresponds either to (1) thermal flux = 200 n/cm^2-sec or fast flux = 3 n/cm^2-sec, or (2) gamma dose rate = <0.2 mR/hr.
5	Same as for Location 1.
6	Fluence >0.1 MeV = 3×10^{21} n/cm^2; corresponds to flux >0.1 MeV = 4×10^{12} n/cm^2-sec (0.82 capacity factor, 30-year life).
7-9	Same as for Location 4.

The principal locations for placement of shielding to meet the above criteria are listed in Table A.5.

TABLE A.5. Principal Locations of Shielding Materials

A. Reactor shield (reflector) assemblies	G. Above upper axial blanket, within reactor assemblies
B. Radial shield surrounding reactor	H. Within reactor vessel cover, if any
C. Bulk sodium surrounding reactor	I. Bulk sodium above reactor vessel
D. Local, at IHXs or pumps	J. Within rotating plug(s)
E. Radial biological shield surrounding tank cavity	K. Within main-tank cover
F. Below lower axial blanket, within reactor assemblies	L. Within operating floor structure (or shield deck)

In the radial direction, the neutron flux beyond the radial blanket is initially reduced by a radial zone of stainless steel (the shield, or reflector zone) inside the core barrel, identified as location (A) in Fig. A.3. Beyond this, a radial shield surrounding the reactor (B) and the primary sodium in the tank (C) are effective in stopping the high-energy flux from the reactor and absorbing the neutrons. At location (D), additional neutron removal can be accomplished locally at the heat exchangers, if necessary, by placing a thin layer of shielding material such as borated steel around the exchangers. Finally, the biological shield (E) reduces both neutron and the gamma flux so that personnel may operate in the regions beyond the biological shield.

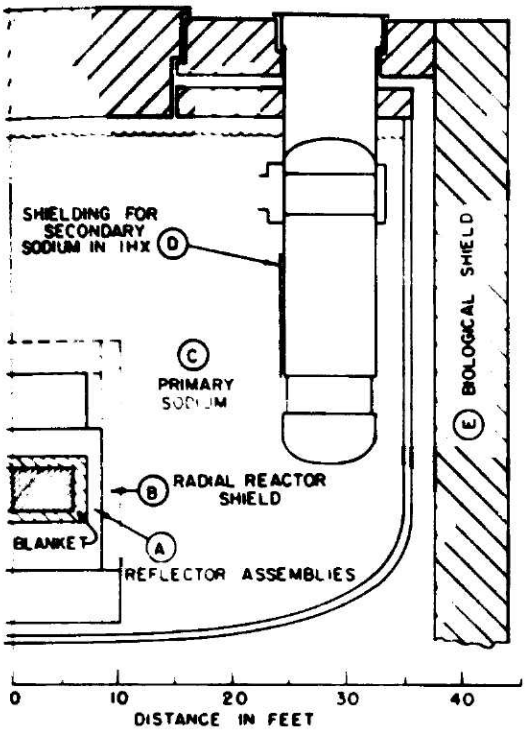


Fig. A.3. Options for Shielding: Proceeding Radially from Core

The radial shielding materials and thicknesses assumed for the normalized EBR-II pool-type concept were, in order, from the reactor core radially outward: reactor radial blanket--a 13-in.-thick section of 60 v/o depleted uranium oxide, 16 v/o SS, and 24 v/o Na; reactor shield zone (or reflector)--a 13-in.-thick section of 86 v/o SS and 14 v/o Na; radial shielding surrounding the reactor--a $6\frac{3}{4}$-in.-thick region of sodium (between reflector and core barrel), followed by a $20\frac{1}{4}$-in.-thick section of about 88 v/o SS and 12 v/o Na, the outermost 6 in. of which is 1/2 w/o borated SS;* bulk sodium--thickness of about $13\frac{1}{2}$ ft to leading edge of IHX shells, or about 24 ft to main tank wall; local shielding surrounding IHX--none assumed; tank walls (reactor support skirt, main tank, and guard tank)--$4\frac{1}{2}$-in. total thickness of SS; and, biological shield--a 6-ft thickness of 200-lb/ft^3 concrete.

Shielding calculation results are reported in terms of the above configuration, although some modifications which should not significantly alter the results were actually employed in the analysis. In particular, the reactor shield zone thickness assumed was one inch less than that stated above, while the radial shielding surrounding the reactor was three inches more in SS thickness and about nine inches less in Na thickness.

The locations of vertical shielding are shown in Fig. A.4. The neutron flux beyond the axial blankets is initially reduced by a reflector and shielding region incorporated within the fuel assemblies below and above the core, locations (F) and (G) in the figure, respectively. The amount of shielding required in the top end of the assembly is affected by the decision to include or omit a reactor vessel cover. If a cover is provided, advantage is taken of it to incorporate additional shielding there (location H); if not, the equivalent amount of shielding would be added to that within the top ends of the fuel assemblies, making them longer. The sodium pool (I) above the reactor vessel cover aids further in stopping the high-energy flux from the reactor and absorbing neutrons. Finally, the rotating shield plug (J) directly above the reactor serves to reduce the neutron and gamma flux to biologically acceptable levels. Similarly, the main-tank cover (K and K') and the shield deck (L and L') reduce the dose rate to acceptable levels on the operating floor. The

*More exactly, as described in Sec. III, this section consists of a 2-in. core barrel, two 4-in. SS cylinders, a $1\frac{1}{2}$-in. SS reactor vessel, a 1/4-in. SS reactor shroud, and a borated 6-in. SS cylinder, all separated by 1/2-in. Na gaps--thus, steel thickness alone is slightly less than 18 in.

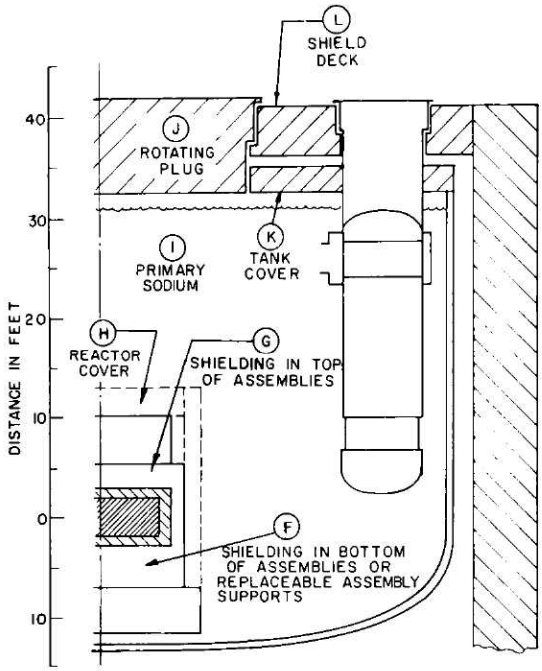


Fig. A.4. Options for Shielding: Proceeding Vertically from Core

required shielding thicknesses near the edge of the tank (K' and L') may be somewhat less than at the center of the tank (K and L) due to the larger distance from the reactor and the edge effect in the radioactive primary sodium pool.

The vertical shielding materials and thicknesses assumed for the normalized EBR-II pool-type concept were, in order, from the reactor core upward: reactor axial blanket--a 13-in.-high section of 36 v/o depleted uranium oxide, 22 v/o SS, and 41 v/o Na; reflector and shielding region* (also including fuel pellet holddown, element and closures, plenum, handling socket, etc.)--a 27-in.-high section of 44 v/o SS and 56 v/o Na; reactor outlet plenum (also including instrument and holddown tree, control-rod drive lines, etc.)--a 5-ft-high section of about 5 v/o SS and 95 v/o Na; reactor vessel cover--3 ft in height, of 67 v/o SS and 33 v/o void; sodium pool above the cover--height of about 17 ft; rotating shield plug (directly above reactor)--$7\frac{3}{4}$ ft in height, of 32 v/o SS and 68 v/o void (equivalent to 30 in. of SS); and, outboard of the rotating plug, the main-tank cover and shield deck--a 3-in.-thick cover of SS and a 5-ft-thick deck primarily of heavy (220 lb/ft^3) concrete (equivalent to 27 in. of SS).

Again, the shielding calculation results are reported in terms of the above configuration, although some modifications which should not significantly alter the results were actually employed in the analysis. In particular, the 27-in. reflector and shielding region assumed was about 3.7 in. less in SS thickness than stated above and the same amount more in Na thickness, the 5-ft-high outlet plenum was 3 in. less in SS thickness and the same amount more in Na thickness, and the vessel cover was 100% SS but only 2 ft in height (i.e., same thickness of SS).

The calculated neutron flux as a function of radius is shown in Fig. A.5. The highest energy flux is seen to fall off extremely rapidly from the edge of the core outward. The flux of energy greater than 0.11 MeV is reduced by approximately five orders of magnitude as it traverses the radial blanket, the reflector, the core barrel, the in-vessel shielding, the reactor vessel, and the

*It should be noted here that the design of the conceptual reference reactor fuel assembly provides essentially the same effective amount of shielding below the bottom axial blanket, which is effective in protecting the reactor assembly support structure.

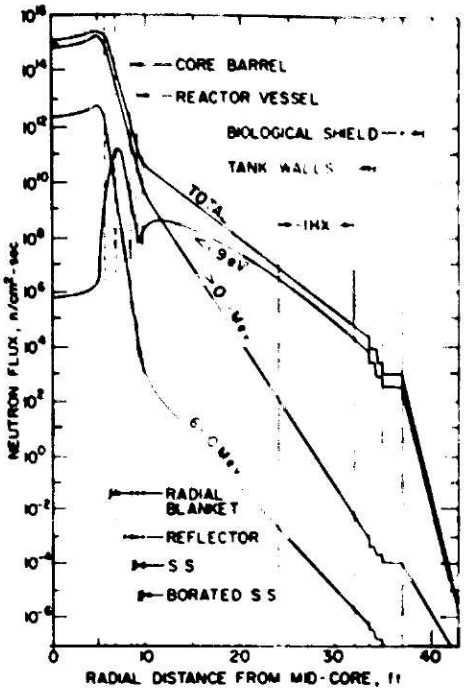


Fig. A.5. Radial Distribution of Neutron Flux in Reference Reactor

stainless steel shield outside the reactor vessel. It is reduced more than another twelve orders of magnitude as it traverses the approximate 24 ft of sodium in the main tank. Thermal flux decreases substantially more slowly, but nevertheless is reduced to the order of 10^3 n/cm^2-sec at the outer surface of the main tank. After traversing only a fraction of the radial biological shield, the total neutron flux is reduced to a negligible value.

The thermal-neutron flux at each point results primarily from neutrons slowing down from the higher energy neutron groups. Neutron diffusion affects the value of the thermal-neutron flux at a point in a secondary manner. The thermal-neutron flux is established as a fraction of the flux in the higher energy groups by the neutron scattering and absorbing characteristics of each material. For example, in the core region, the thermal-neutron flux is several orders of magnitude less than the flux greater than 0.11 MeV, while in the primary sodium tank, the thermal flux is several orders of magnitude higher than the fast flux. The equilibrium value of the thermal flux is established in the sodium tank even though borated stainless steel is used to prevent thermal neutrons from diffusing from the core to the bulk sodium. The borated stainless steel serves mainly to capture neutrons diffusing outward from the core so that they are not captured in the sodium in the tank, thereby increasing the activity of the primary sodium.

The shielding provided in the reference design appears to be adequate in respect to the radiation damage to the core barrel and is clearly more than adequate in respect to the biological dose (due to neutron flux) effected outside the biological shield. It also is more than adequate from the standpoint of activation of the major repairable components (IHXs, pumps, etc.). It is just adequate in respect to limiting activation of the secondary sodium; however, significantly greater protection could be provided here simply by adding a plate of borated stainless steel (as is done in EBR-II) to the exterior of the IHX on the side facing the reactor.

The neutron flux distribution in the vertical direction is shown in Fig. A.6. The vertical flux distribution is qualitatively similar to the distribution in the radial direction. The shielding provided appears to be adequate from the standpoint of radiation damage to the instrument tree (located in the reactor outlet plenum, immediately above the tops of assemblies). This also is true of the reactor assembly support structure, not shown in the figure, the top surface of which sees virtually the same neutron flux as the lower portion

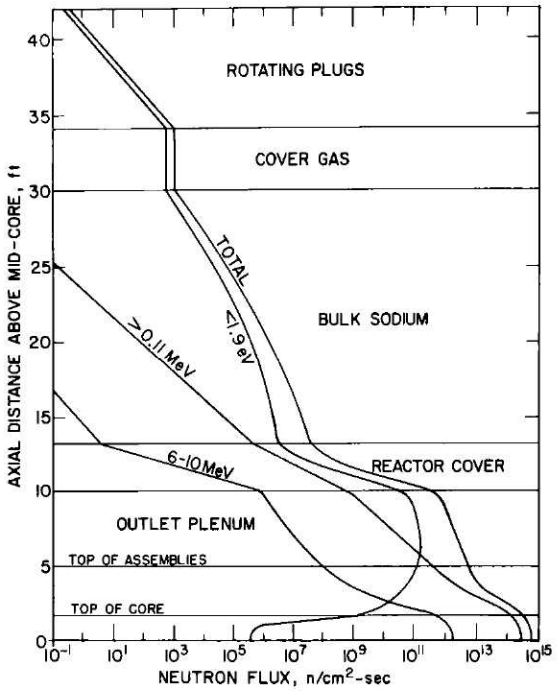


Fig. A.6. Axial Distribution of Neutron Flux in Reference Reactor

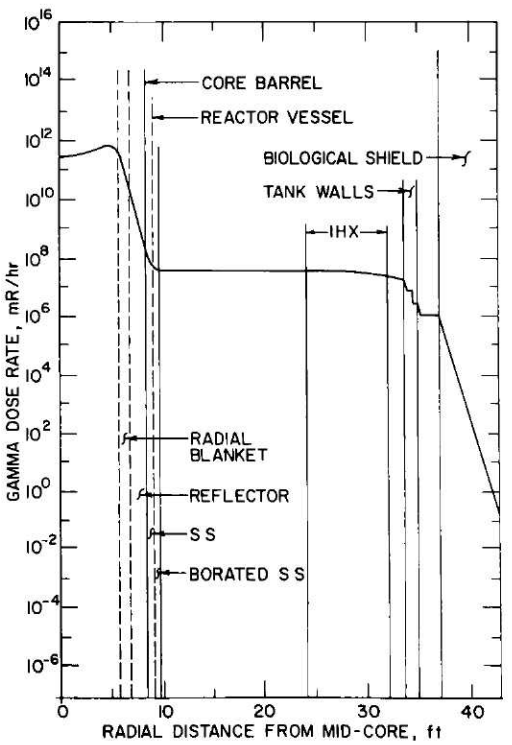
of the instrument tree. Again, only a fraction of the shielding provided in the rotating plug is required to reduce the neutron flux to an insignificant level.

The flux calculations made were based on core and blanket compositions at the start of the first cycle. Plutonium buildup in the blankets during operation would increase neutron flux levels at the outer edges of both the radial and axial blankets (and outward therefrom) by up to approximately 30% above the levels indicated. This effect, however, does not alter the general conclusions on shielding adequacy noted above.

The amounts of shielding provided in the radial biological shielding, the rotating plugs, and the shield deck are required primarily to attenuate gamma dose rates to an acceptable level. Figure A.7 shows the gamma dose rate distribution in the radial direction. The gamma dose from primary gammas originating in the core and from secondary gammas originating in the radial shield surrounding the reactor is attenuated in the radial shield surrounding the reactor.

Fig. A.7

Radial Distribution of Gamma Dose Rate in Reference Reactor



The gamma dose rate is further attenuated in the bulk sodium in the tank surrounding the radial shield. About a foot from the exterior of the radial shield the gamma dose rate due to the primary and secondary gamma rays is reduced sufficiently that it approximately equals the dose rate from the gammas originating in the sodium itself. From this radius outward, the gamma dose is dominated by the gammas from the activated sodium. The tank walls and the biological shield attenuate the sodium gamma rays to a level which will permit continuous access for operation and maintenance.

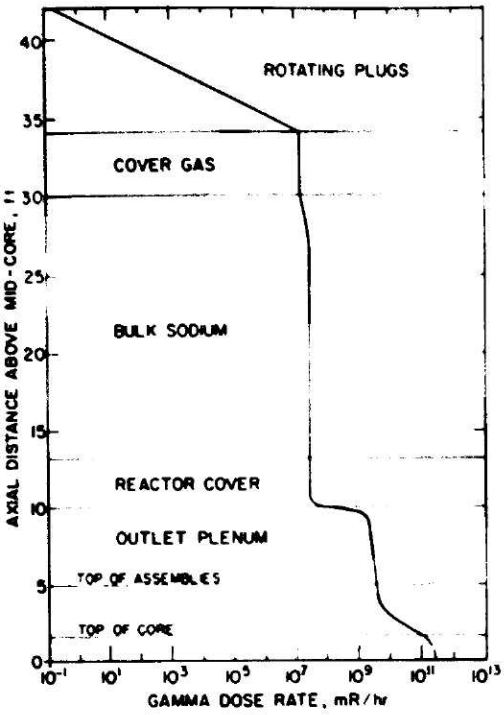


Fig. A.8. Axial Distribution of Gamma Dose Rate in Reference Reactor

The variation of the gamma dose rate in the vertical direction is shown in Fig. A.8. The vertical profile is similar to the radial profile in that the dose rate drops monotonically with increasing distance until it falls to the level of the gamma dose rate in the bulk sodium. This gamma dose rate is attenuated in the rotating plugs at the center of the tank and in the main-tank cover and shield deck over the remainder of the tank. The vertical distributions of the gamma dose rate through the main-tank cover and shield deck are shown in Fig. A.9.

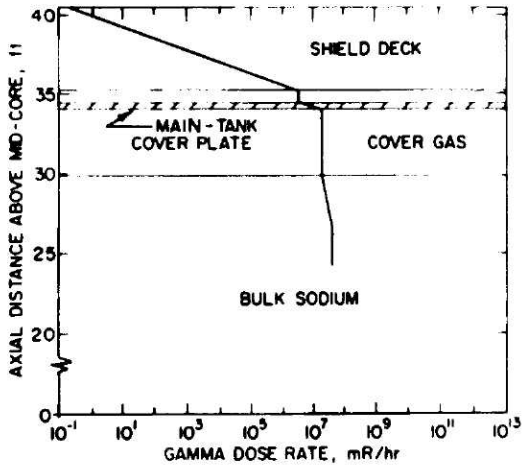


Fig. A.9

Axial Distribution of Gamma Dose Rate through Main-tank Cover Plate and Shield Deck

4. Method of Calculation

The specific model used for the calculations was a simplification of the 1200-MWe normalized EBR-II concept. The geometry was simplified somewhat so that it could be described entirely by cylinders of revolution. This model, shown in Fig. A.10, was developed so that neutron transport theory could be applied to determine the energy-dependent neutron and gamma flux

distributions throughout the primary system. The neutron and gamma energy structures used for these calculations are shown in Table A.6. The calculations were normalized so that the neutron and gamma fluxes calculated correspond to the steady-state operation of the reactor at 3019 MWt.

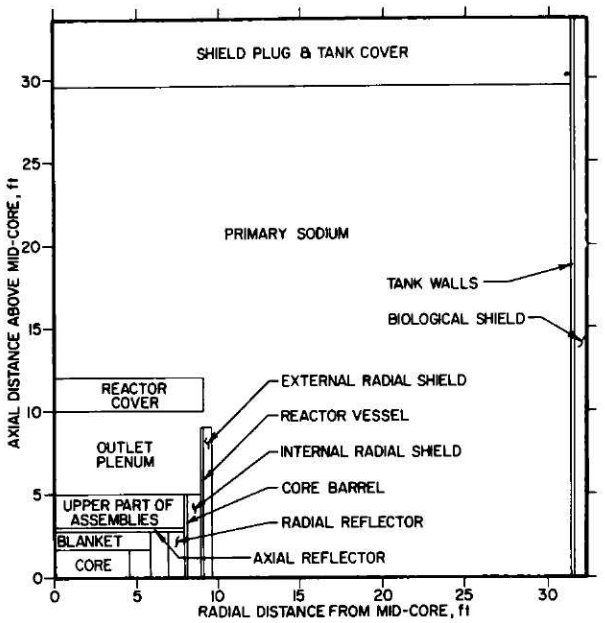


Fig. A.10

Model Used for Calculating Energy-dependent Neutron and Gamma Flux Distribution

TABLE A.6. Neutron and Gamma Energy Structures Used in Calculations

Energy Group	Upper Energy Limit, eV		Energy Group	Upper Energy Limit, eV	
	Neutron	Gamma		Neutron	Gamma
1	1.00000×10^7	1.0×10^7	16	5.53084×10^3	7.0×10^5
2	6.06531×10^6	8.0×10^6	17	3.35463×10^3	6.0×10^5
3	3.67879×10^6	7.0×10^6	18	2.03468×10^3	4.0×10^5
4	2.2313×10^6	6.5×10^6	19	1.2341×10^3	2.0×10^5
5	1.35335×10^6	6.0×10^6	20	7.48518×10^2	1.0×10^5
6	8.2085×10^5	5.5×10^6	21	4.53999×10^2	
7	4.97871×10^5	5.0×10^6	22	2.75364×10^2	
8	3.01974×10^5	4.5×10^6	23	1.67017×10^2	
9	1.83156×10^5	4.0×10^6	24	1.01301×10^2	
10	1.11090×10^5	3.5×10^6	25	6.14421×10^1	
11	6.73795×10^4	3.0×10^6	26	3.72665×10^1	
12	4.08677×10^4	2.6×10^6	27	1.37096×10^1	
13	2.47875×10^4	2.2×10^6	28	5.04348×10^0	
14	1.50344×10^4	1.8×10^6	29	1.85539×10^0	
15	9.11882×10^3	1.35×10^6			

The neutron cross sections used in these calculations were derived from the ENDF/B Version 3 library and collapsed to the multigroup structure with the use of the MC^2 code. The gamma transport cross sections were obtained from the MUG code, which computes Compton scattering cross sections using the Klien-Nishina formula and adds photoelectric and pair-production cross sections from its associated library. The gamma emission spectra for neutron interactions were obtained in the appropriate group structure from the code, and combined with neutron interaction cross sections to obtain gamma

production cross sections. The neutron transport theory code DOT was then used with neutron cross sections to calculate multigroup neutron flux levels throughout the system. These fluxes were used with the gamma production cross sections to calculate the gamma source as a function of space. This gamma source was then used as the starting source for a gamma transport calculation performed with the DOT code.

More specifically, the gamma-heating calculations were done in the following steps:

1. The neutron flux distribution was computed with the transport code.

2. The photon source distribution was obtained at each point by taking the inner product of the gamma production cross section matrix Σ_{ij} (where the first subscript denotes the gamma energy group and the second subscript denotes the neutron energy group) and the neutron flux vector Φ_i (where the subscript denotes the neutron energy group). The gamma production cross section Σ_{ij} is defined as the number of gamma rays produced in energy group i per second per cubic centimeter per unit of neutron flux in energy group j. The macroscopic gamma production cross section is computed taking into consideration the gamma rays produced by fission (including gamma rays emitted by fission products, assuming steady-state operation), neutron capture, and neutron inelastic scattering.

3. The photon flux that results from the computed photon source was determined by using the DOT code with gamma-ray scattering and total cross sections.

4. The rate of heat production was determined by multiplying the photon flux vector ψ_i (where the subscript denotes the gamma energy group) by the gamma energy-absorption cross section vector Σ_i^a. The energy-absorption cross section for group i is defined as the rate at which gamma-ray energy is converted into heat per unit of photon flux in group i.

Contour maps of the neutron flux throughout the primary system are given as Figs. A.11-A.17. The high-energy neutron flux distribution is shown in Fig. A.11. The neutron flux distribution for neutrons of lower energy are shown in Figs. A.12-A.17. These figures show graphically the escape of neutrons through the outlet piping, represented as an annulus of sodium in these calculations. Although neutron leakage through the outlet piping is significant, the neutron flux level in the tank around the outlet piping does not exceed the neutron flux level outside the radial shield at the core midplane.

The calculated gamma source densities are indicated on the contour maps given as Figs. A.18-A.21. These source densities in the sodium regions do not take into account the mixing of the sodium. Actually, the sodium is so well mixed that the gamma source densities throughout the sodium pool are constant. The gamma fluxes and gamma dose arising from the mixed sodium sources are obtained from a separate calculation. These results were combined with the results of the transport theory calculation and used to plot the curves in Figs. A.7-A.9.

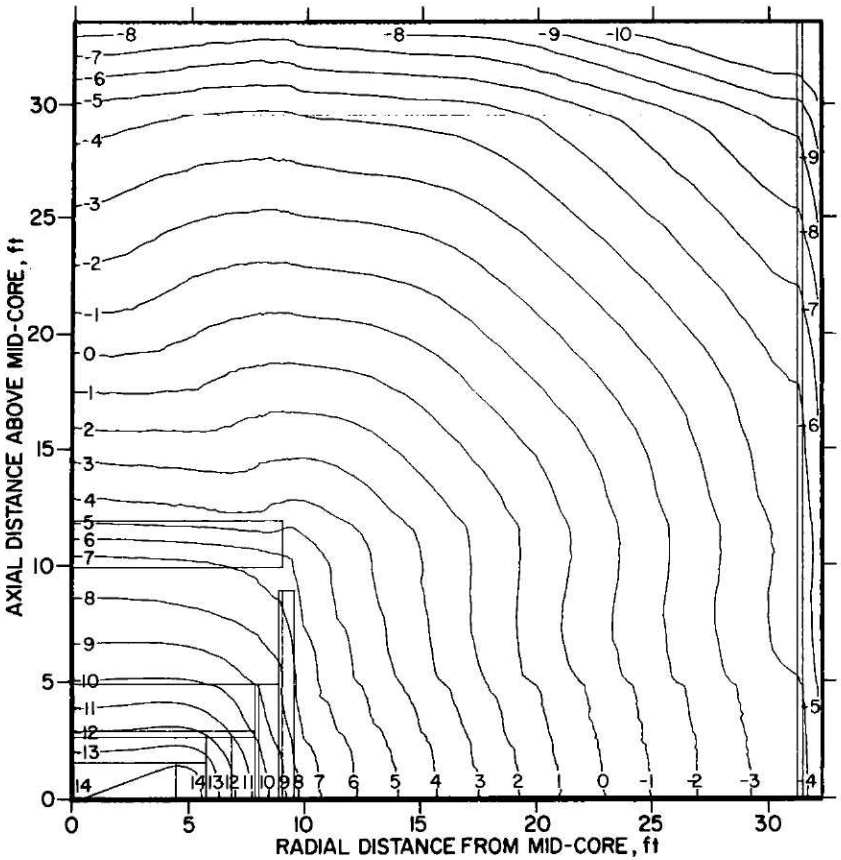


Fig. A.11. Contour Map of Neutron Flux throughout Primary System of Reference Reactor: Energy Group 6

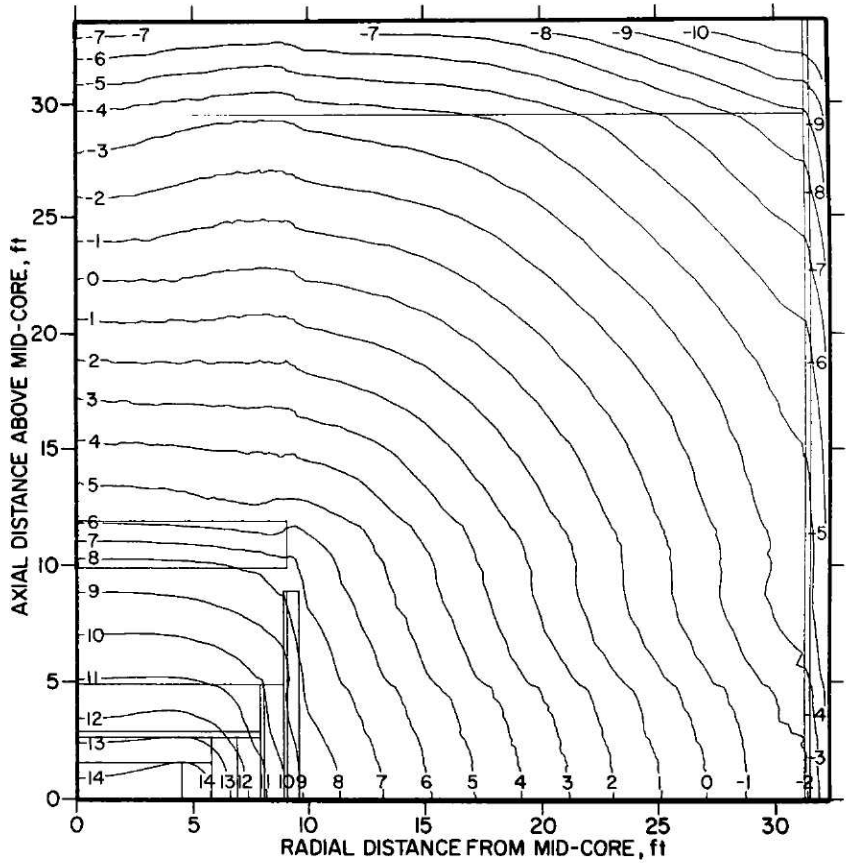


Fig. A.12. Contour Map of Neutron Flux throughout Primary System of Reference Reactor: Energy Group 9

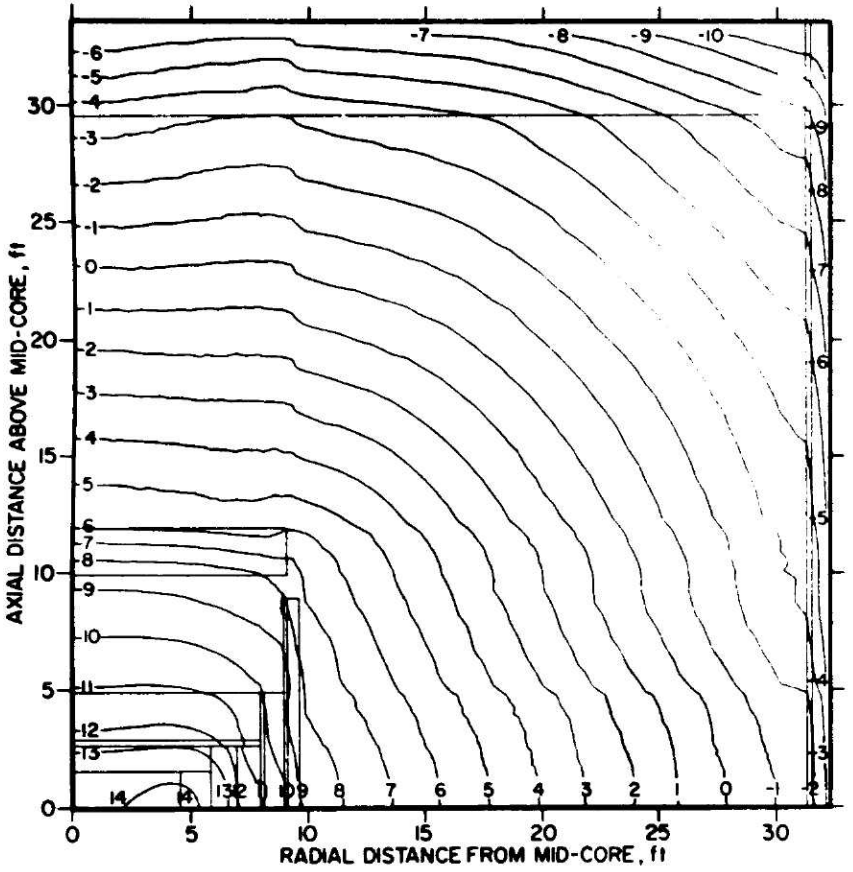


Fig. A.13. Contour Map of Neutron Flux throughout Primary System of Reference Reactor: Energy Group 12

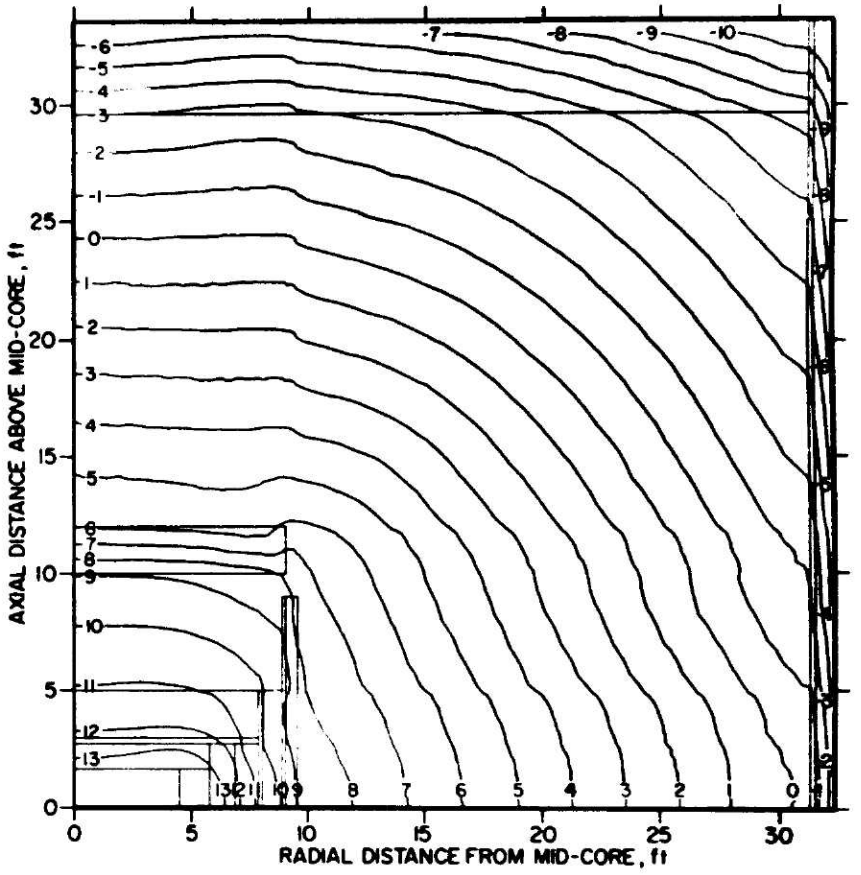


Fig. A.14. Contour Map of Neutron Flux throughout Primary System of Reference Reactor Energy Group 15

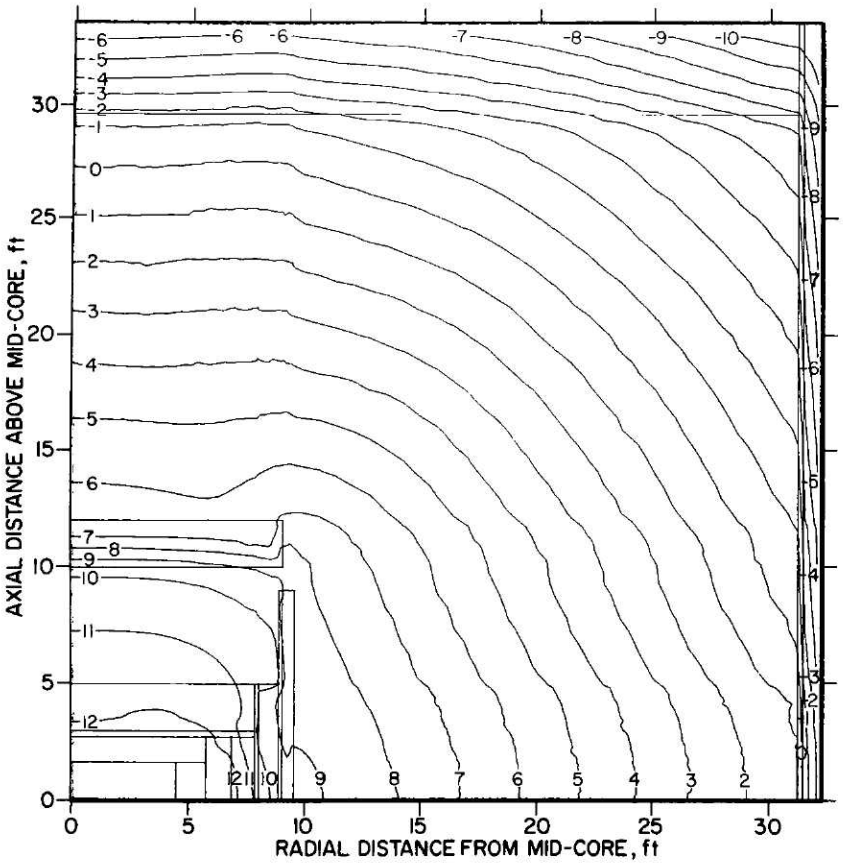


Fig. A.15. Contour Map of Neutron Flux throughout Primary System of Reference Reactor: Energy Group 20

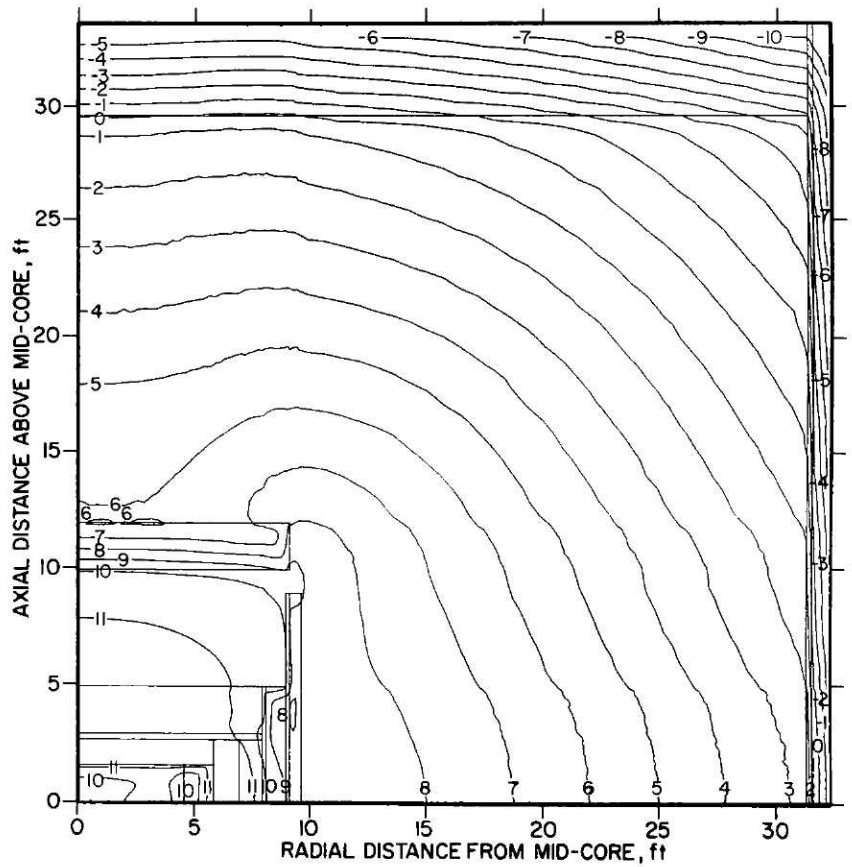


Fig. A.16. Contour Map of Neutron Flux throughout Primary System of Reference Reactor: Energy Group 25

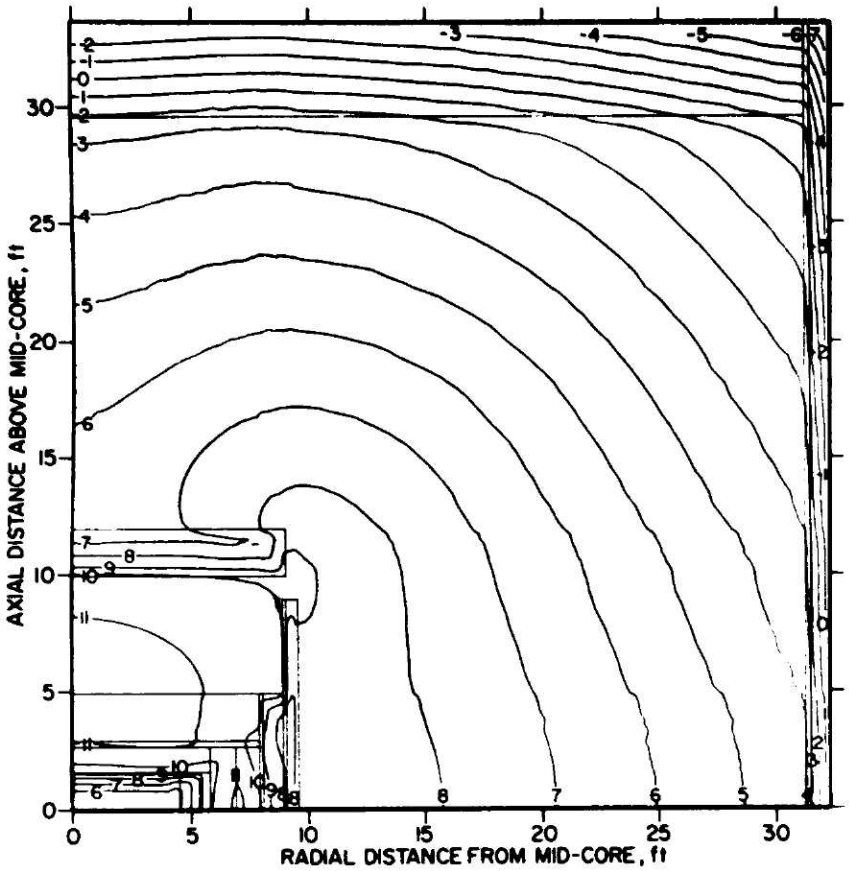


Fig. A.17. Contour Map of Neutron Flux throughout Primary System of Reference Reactor: Energy Group 29

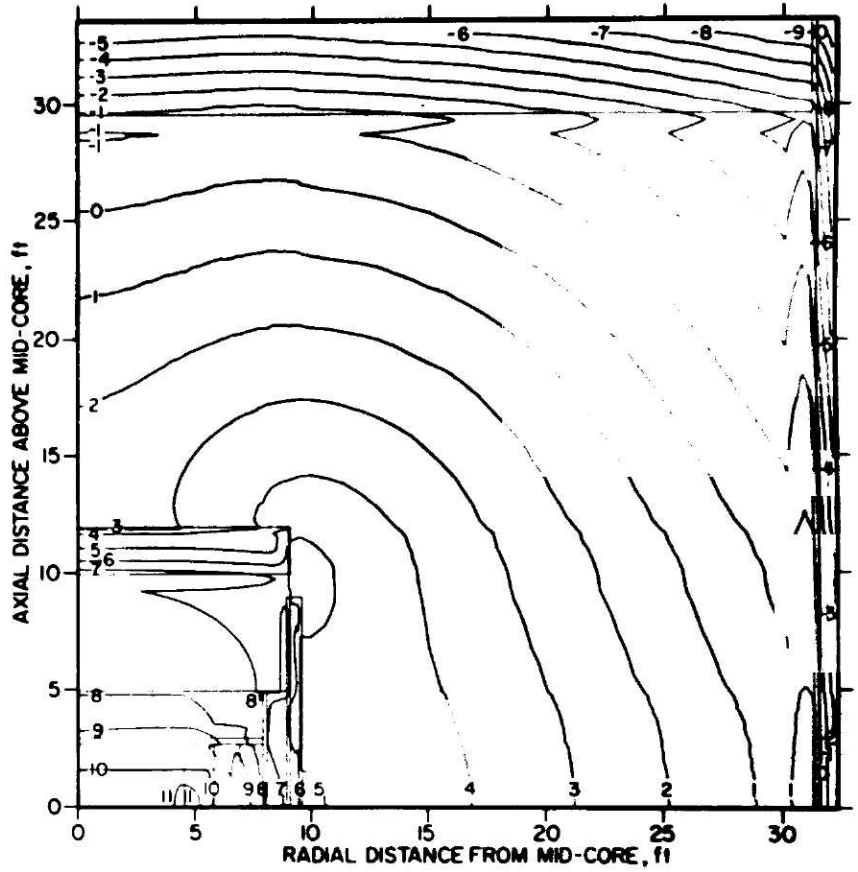


Fig. A.18. Contour Map of Gamma Source Densities throughout Primary System of Reference Reactor: Energy Group 6

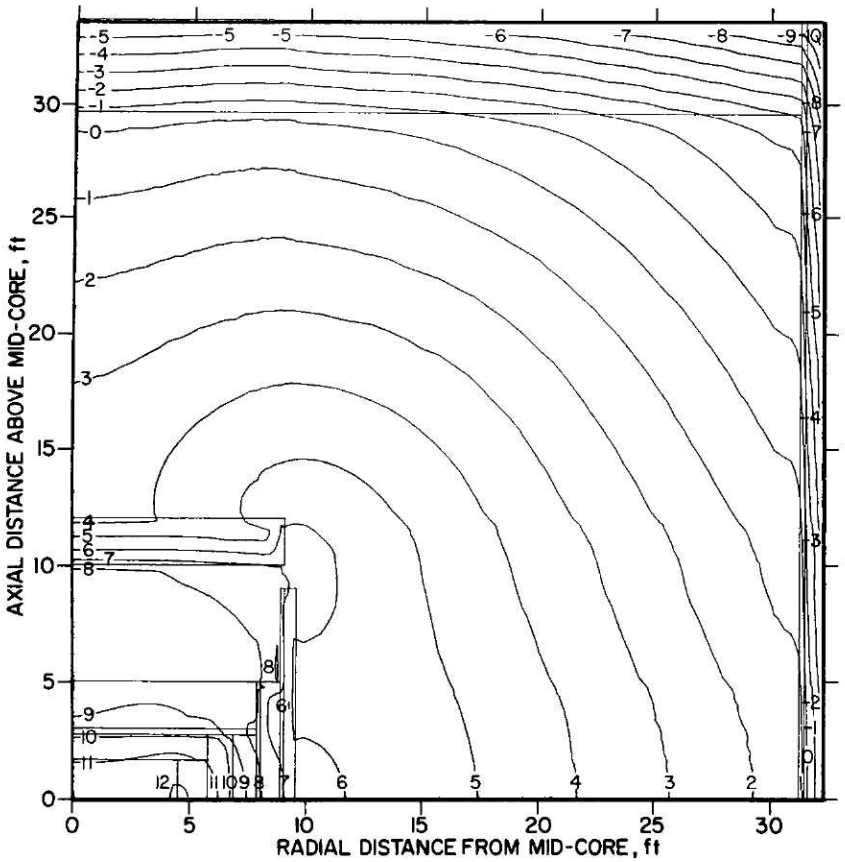


Fig. A.19. Contour Map of Gamma Source Densities throughout Primary System of Reference Reactor: Energy Group 9

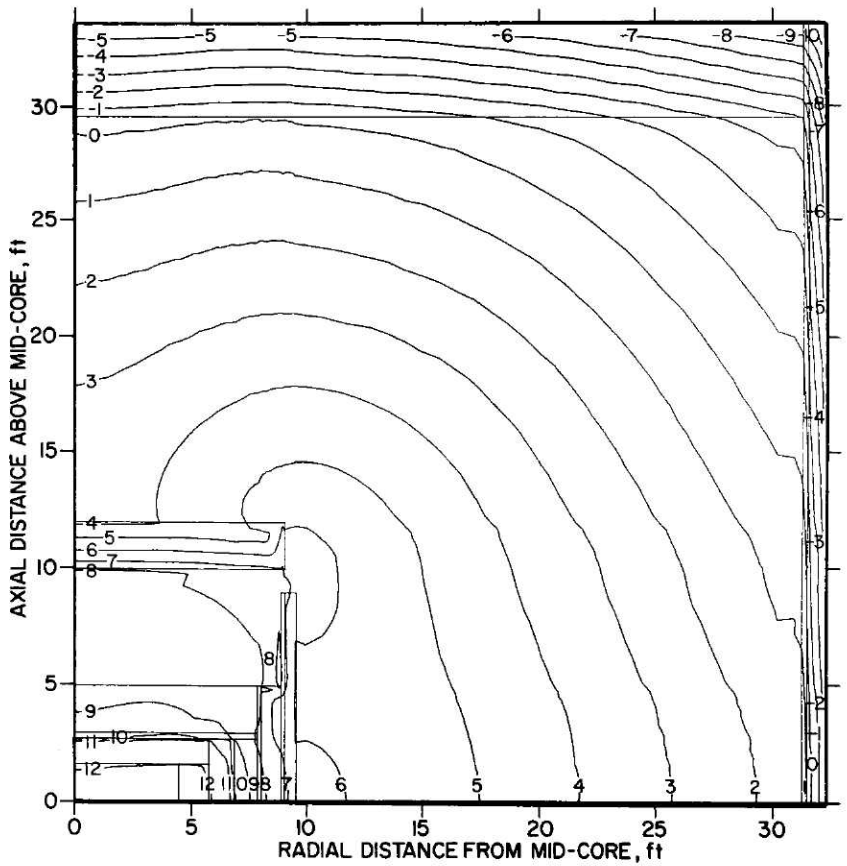


Fig. A.20. Contour Map of Gamma Source Densities throughout Primary System of Reference Reactor: Energy Group 12

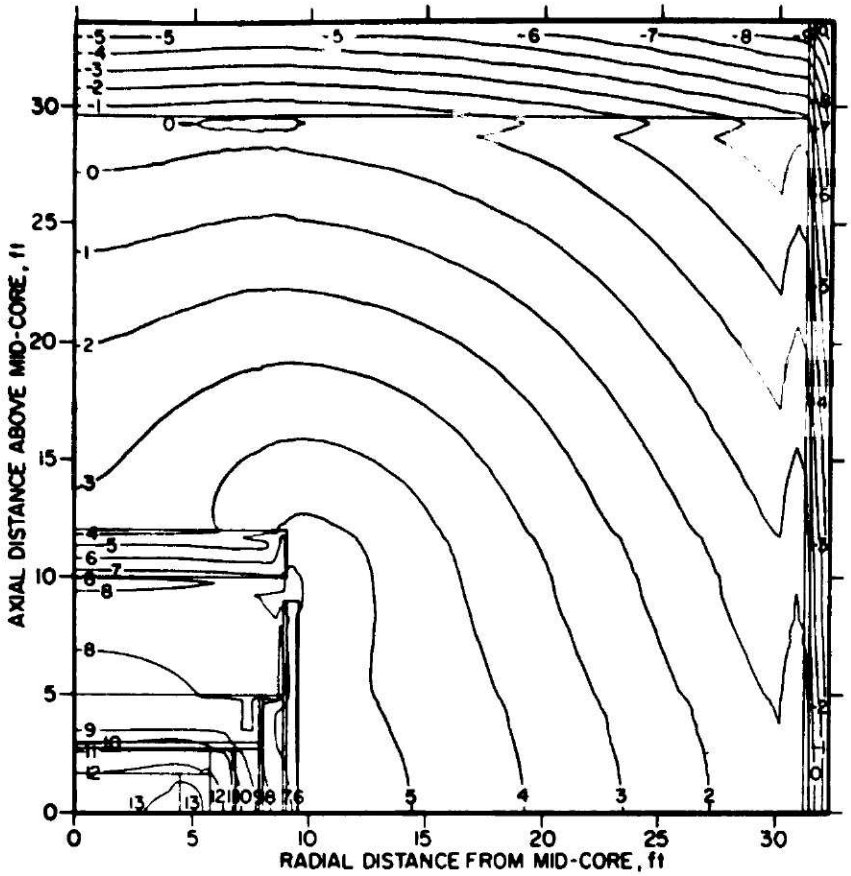


Fig. A.21. Contour Map of Gamma Source Densities throughout Primary System of Reference Reactor Energy Group 15

The attenuation of the neutron flux through the steel in the reactor vessel cover and the radial reflector is evident in the contour maps. The contour maps of the gamma source densities show graphically the discontinuities in the gamma source densities which result from discontinuities in the macroscopic gamma production cross section at the material boundaries. The gamma production rate in steel is several orders of magnitude higher than the gamma production rate in sodium. This leads to the discontinuities in gamma production rates shown in the gamma source density contour maps.

Activation of the primary sodium occurs primarily in the core region. Sodium activation rates in each region of the system are shown in Table A.7. The activation rate in each region with a sodium atom density of 1, as obtained from the calculated neutron fluxes and activation cross sections, is shown in column 2 of the table. The sodium atom density in each region is shown in column 3, and the total activation per second in each region is given in column 4. Approximately half of the sodium activations occur in the inner and outer core regions, and an additional 22% of the sodium activations occur in the axial and radial blankets. The inlet and outlet plenums contribute about 10% to the sodium activation rate, and the axial and radial reflectors contribute about 6%. Less than 1% of the sodium activations occur in the sodium pool. The total sodium activation rate in the system is approximately 3×10^{17} atoms/sec.

TABLE A.7. Sodium Activation in Primary System of Reference Reactor

Region	Activations per sec/(atom/cm^3)	Sodium Atoms per cm^3	Activations per sec	Fraction of Total Activations per sec
Inner core	1.17×10^{19}	0.00838	9.80×10^{16}	0.33
Outer core	8.07×10^{18}	0.00838	6.76×10^{16}	0.22
Radial blanket	4.01×10^{18}	0.00538	2.16×10^{16}	0.07
Radial reflector	9.31×10^{17}	0.00311	2.90×10^{15}	0.01
Axial blanket	5.40×10^{18}	0.00840	4.54×10^{16}	0.15
Axial reflector	1.81×10^{18}	0.00907	1.64×10^{16}	0.05
Upper assembly	4.82×10^{17}	0.02042	9.84×10^{15}	0.03
Core barrel	1.98×10^{16}	0.0	0	-
Reactor radial shield (internal)	2.71×10^{16}	0.0	0	-
Reactor vessel	1.01×10^{16}	0.0	0	-
Reactor radial shield (external)	4.56×10^{15}	0.0	0	-
Sodium pool	1.73×10^{16}	0.02269	3.92×10^{14}	0.001
Support skirt and tank walls	3.31×10^{10}	0.0	0	-
Radial biological shield	3.44×10^{9}	0.0	0	-
Inlet and outlet plenums	1.72×10^{18}	0.02269	3.90×10^{16}	0.10
Reactor cover and lower grid	1.65×10^{16}	0.00756	1.25×10^{14}	-
Shield plug and tank cover	6.88×10^{8}	0.0	0	-
Total			3.01×10^{17}	

The total amount of primary sodium in the system is approximately 2.3×10^9 grams. The sodium activity is therefore approximately 1.3×10^8 disintegrations per second per gram. (Since one curie is equal to 3.7×10^{10} disintegrations per second, this implies an activity of approximately 3.5×10^3 microcuries per gram or approximately 3.0 curies per liter. This compares to the measured activity of 2 curies per liter in the primary sodium of EBR-II.) Within the main body of the bulk sodium, the gamma energy absorption rate per unit mass is equal to the gamma energy production rate per unit mass. Since each sodium disintegration produces 4.1 MeV, the gamma energy production (and absorption) rate is 5.3×10^8 MeV per second per gram. Considering sodium absorption of 5.8×10^7 MeV/g to represent a dose of one roentgen (essentially the same value as for biological dose), the gamma dose rate in the bulk sodium is 9.2 roentgens per second. The pool boundary dose rate is one half of this, or 4.6 roentgens per second. This is equivalent to a dose rate of 1.7×10^7 mR/hr. Since the dose on the operating floor should be less than or equal to 0.2 mR/hr, the tank and shield deck must be capable of reducing the gamma flux by a factor of approximately 1.2×10^{-8}. Approximately thirty inches of steel or an equivalent weight of concrete are required to reduce the gamma dose by this factor.

The fact that the activation rate in the sodium pool is only a small percentage of the total activation rate is a result of the effective removal of neutrons in the shielding within reactor assemblies and that immediately surrounding the reactor, including the reactor vessel cover. Neutron capture is enhanced at the outer edge of the radial shield surrounding the reactor by the addition of boron to this steel shield. The boron will, of course, be depleted by neutron capture during the life of the plant; however, calculations indicate that this depletion will be less than 1% in 30 years.

The intermediate system sodium may be protected to some extent from excessive activation by placing borated steel shielding around the IHXs, and the boron may be expected to last for the life of the plant. However, the calculations indicate that such shielding will not generally be necessary. The neutron flux is sufficiently attenuated in the sodium pool before reaching the IHXs that the activity of the intermediate sodium is acceptably low. The shielding indicated for the normalized EBR-II pool-type concept includes about fourteen feet of sodium between the reactor and heat exchanger which reduces the total neutron flux at the nearest edge of the IHX to about 7×10^6 n/cm^2. The resulting calculated dose rate at the surface of a large concentration of intermediate system sodium, such as in a steam generator (represented in the calculation as a sphere of 2-meter diameter), is approximately 0.3 mR/hr, well below the criterion of 2 mR/hr.

The shielding described for the normalized EBR-II pool-type concept, on which the shielding calculations are based, has not been optimized. However, the amounts of shielding indicated for the various locations within the system appear to be fully commensurate with the amounts required to meet the assumed shielding criteria. The general shielding arrangement indicated is believed to be a good point of departure for shield design calculations for a pool-type LMFBR of commercial size.

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APPENDIX B

Preliminary Seismic Studies of Cold and Hot Center Pool-type LMFBR Vessels

The material contained in this Appendix is excerpted from a report prepared for ANL by URS/J. A. Blume and Associates (URS/Blume) with the same title of this Appendix. That report contained results of a limited study performed by URS/Blume (under Argonne's direction) which was intended to provide a preliminary engineering assessment of the resistance of large-diameter vessels for LMFBR pool-type systems to seismic loads. Two systems were analyzed: an expanded (or up-sized) EBR-II system (Cold-Pool Type), and a simulated version of the French SuperPhenix-type (Hot-Pool Type).

The expanded EBR-II concept is based upon, and is similar to, the actual EBR-II operating since 1963 at the Idaho National Engineering Laboratory (INEL). The simulated version of the French SuperPhenix type of system was based primarily upon review of published engineering data pertaining to that system. In both cases, the actual dimensions and material thicknesses assumed must be considered as preliminary, although generally representative of an actual design.

It should be noted that the dimensions used for the URS/Blume study vary somewhat from those shown in the main body of this report. These differences are relatively minor and were the result primarily of the timing of the URS/Blume work.

1. Scope of the Studies

Seismic stress levels in the primary structural elements of the systems--the various tanks and their supports and interconnections--were investigated for a safe shutdown earthquake (SSE) of 0.3 g peak ground acceleration. Because no information was available at the time of the study on the structural configuration or the seismic characteristics of the reactor buildings that house pool-type LMFBRs, certain assumptions had to be made. Since many reactor systems are typically supported at the mid-height of the reactor buildings, the ground SSE values were modified to reflect the expected structural amplification of the motion. The vessels were analyzed for peak support accelerations of 0.6 g in the horizontal direction and 0.4 g in the vertical direction. The results should be viewed in light of these important and assumed values.

To evaluate the structures for the above loading conditions, finite-element models were constructed and analyzed. All load-resisting structural

elements, including the reactor core and the various primary pumps and primary heat exchangers, were taken into consideration, in varying degrees. For the horizontal earthquake accelerations, the bulk liquid sodium was modeled with the aid of fluid elements that account for the hydrodynamic effects due to coupling of the various interacting elements submerged in the liquid sodium.

The (expanded) EBR-II system is a cold-pool type primary system, while the SuperPhenix in a hot-center-pool system in which sodium is contained separately in different compartments of the tanks at two temperatures: approximately 986 and 716°F. The compartment holding the hotter sodium--the hot-center-pool--was included in the computer models for the SuperPhenix, and its structural response due to both horizontal and vertical seismic excitation was obtained and examined.

A separate analysis was carried out to determine the extent of liquid sloshing and the maximum fluid pressures on the supporting shells of the two LMFBR systems. In addition, an approximate 1:70 scale rigid model of the Expanded EBR-II was built from clear acrylic material and was used for preliminary experimental studies. The reactor and its supporting shell were modeled fairly accurately, while the gross effects of the contained equipment were modeled using equivalent circular cylinders. The experimental studies were used primarily to determine the influence of the internal equipment on the sloshing modes of the liquid sodium. Due to the limitations of the study, obviously, such tests were modest, and further work is probably advisable to assure these preliminary findings.

2. Descriptions of the Structural Systems

a. Expanded (or up-sized) EBR-II Primary System

The cold-pool-type primary system of the Expanded EBR-II concept consists of three steel tanks containing liquid sodium coolant, the reactor vessel, three sodium pumps, six intermediate heat exchangers (IHXs), interconnecting piping, and other equipment. The structural configurations are as illustrated in Figs. B.1 and B.2. The various tanks that were analyzed have the following assumed dimensions:

Tank	Height	Inside Diameter	Plate Thickness
Guard Tank	52 ft	71 ft-10 in.	1.5 in.
Primary Tank	61 ft	70 ft-0 in.	2.0 in.
Reactor Support Structure	35 ft	69 ft-6 in.	1.0 in.--cylindrical section 2.0 in.--conical section
Pumps	30 ft	7 ft-6 in.	1.5 in.
Heat Exchangers	30 ft	8 ft-1 in.	1.5 in.
Reactor w/Shielding	24 ft	22 ft-0 in.	1.5 in. and 3.0 in.

The plate thicknesses are greater at various connections and plate discontinuities, as indicated later in Fig. B.7.

The guard tank serves as a secondary barrier for the leakage of sodium into the plant. The main tank is part of the primary system boundary and carries the weight of the sodium. Through interconnections near the shield deck, the main tank also supports the reactor core, the primary tank cover, three pumps, six IHXs, and other equipment. The shield deck and the rotating plugs are supported externally and are not included in the present analysis. The reactor is supported by the reactor core support structure, a perforated cylindrical tank with a conical bottom.

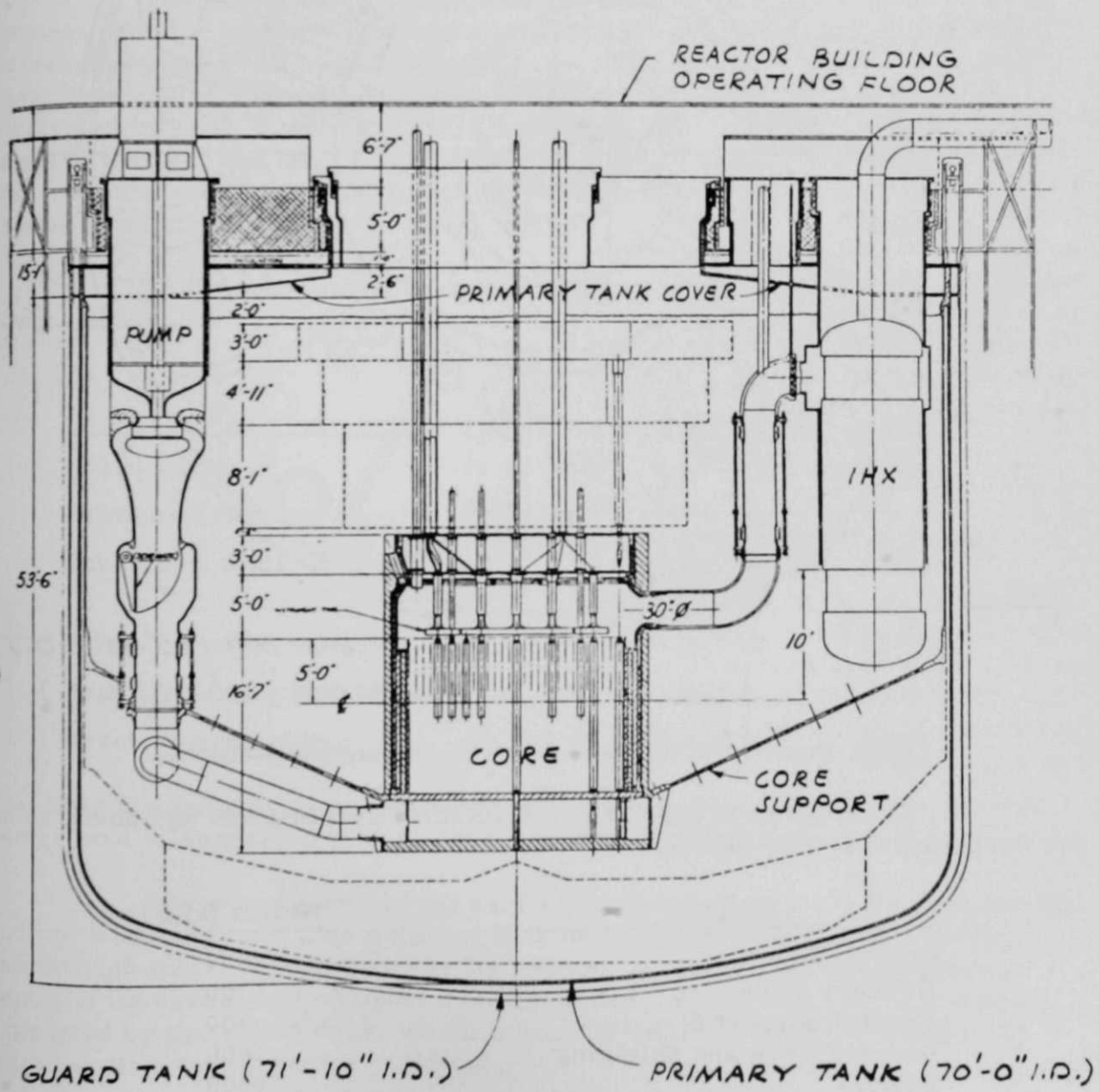

Fig. B.1. Expanded 1200 MWe EBR-II Type (3 loop) Primary System--Elevation View

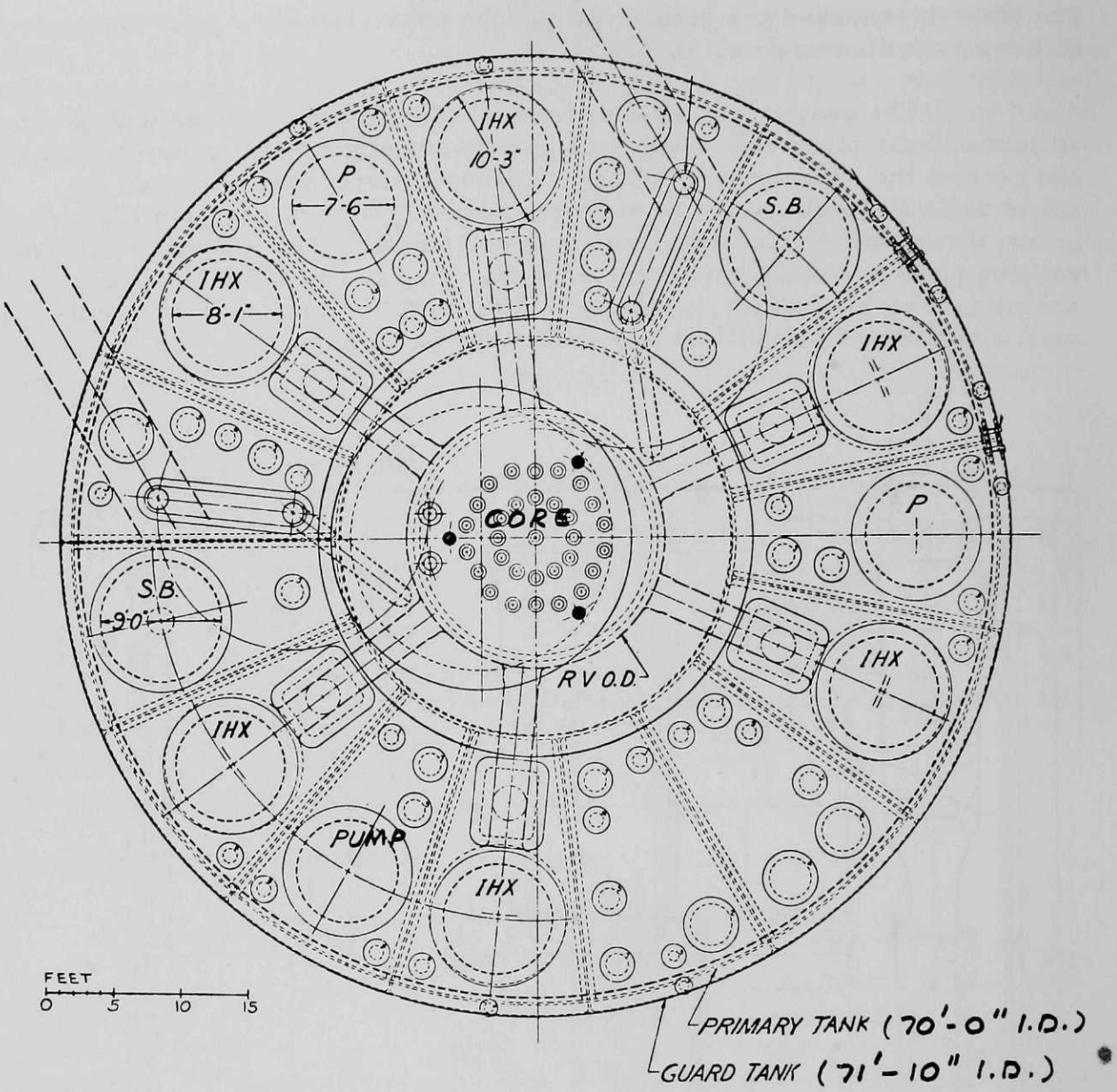

Fig. B.2. Expanded 1200 MWe EBR-II Type (3 loop) Primary System--Plan View

The weights of the various structural systems and equipment items for the analysis are as follows:

Item	Weight (kips)
Guard Tank	900
Primary Tank	1200
Reactor Support Structure	400
Reactor Core and Shielding	1600
Sodium	9000
Pumps (3 at 400 kips)	1200
Heat Exchangers (6 at 400 kips)	2400
Primary Tank Cover	600

The analysis was based on the assumption that the primary system is supported by a continuous skirt, embedded in the massive concrete of the shield deck. The primary tank cover is additionally supported by a skirt support embedded in the shield deck near the rotating plugs.

The operating temperature of the system and of the sodium pool is approximately 720°F. The liquid sodium fills the primary tank to within 4 ft 6 in. of the primary tank cover.

b. Simulated SuperPhenix Primary System

The hot-center-pool primary system of the SuperPhenix concept consists of three inverted and interconnected tanks containing liquid sodium, the reactor vessel and assorted appendages, four primary pumps, eight IHXs, interconnecting piping, and other equipment. The structural configuration is illustrated in Figs. B.3 and B.4. A series of baffles is located between the primary tank and the hot center pool walls. These baffles are essentially nonstructural elements and accordingly are generally neglected in this conceptual seismic analysis.

The various tanks that were analyzed have the following assumed dimensions:

Tank	Height	Inside Diameter	Plate Thickness
Guard Tank	61 ft-0 in.	74 ft-10 in.	1.5 in.
Primary Tank	58 ft-6 in.	70 ft-0 in.	2.0 in.
Hot Center Pool Walls	46 ft-0 in.	35 ft-0 in. to 68 ft-0 in.	1.0 in.
Pump Shroud	22 ft-0 in.	9 ft-0 in.	0.75 in.
Heat Exchanger Shroud	22 ft-0 in.	9 ft-0 in.	0.75 in.
Reactor w/Shielding	24 ft-0 in.	30 ft-0 in.	2.0 in.

The plate thicknesses vary, as shown later in Fig. B.9. In addition, the hot center pool is supported by a divider with a plate thickness of 0.75 in.

The guard tank serves as a secondary barrier for the leakage of sodium into the plant. The primary tank carries the weight of the sodium, and through interconnections near its bottom, it also supports the reactor core and its associated equipment. The pumps and the heat exchangers are supported by the shield deck, which in turn is attached to the surrounding external structure (reactor building).

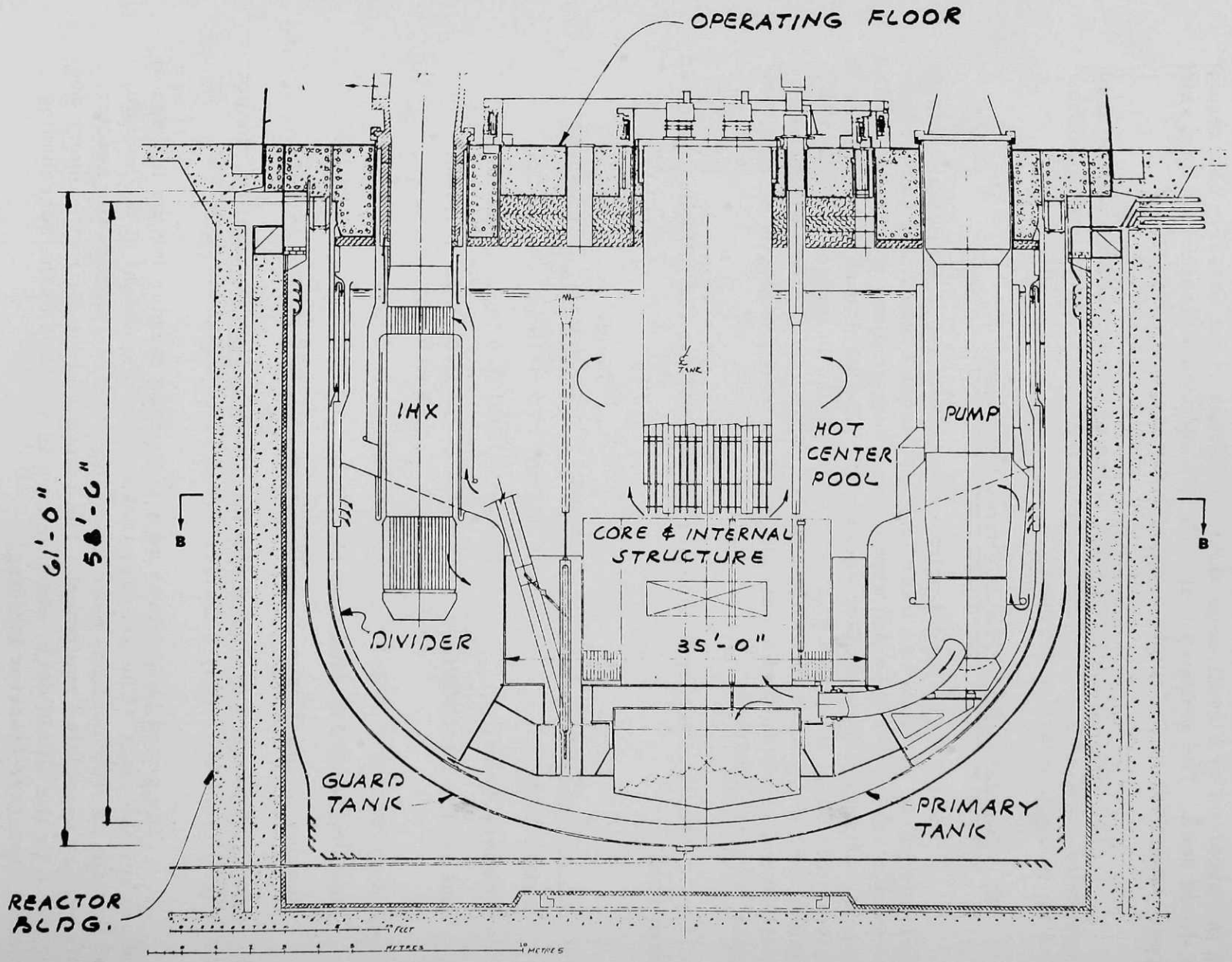

Fig. B.3. SuperPhenix (1200 MWe) Primary System--Elevation View

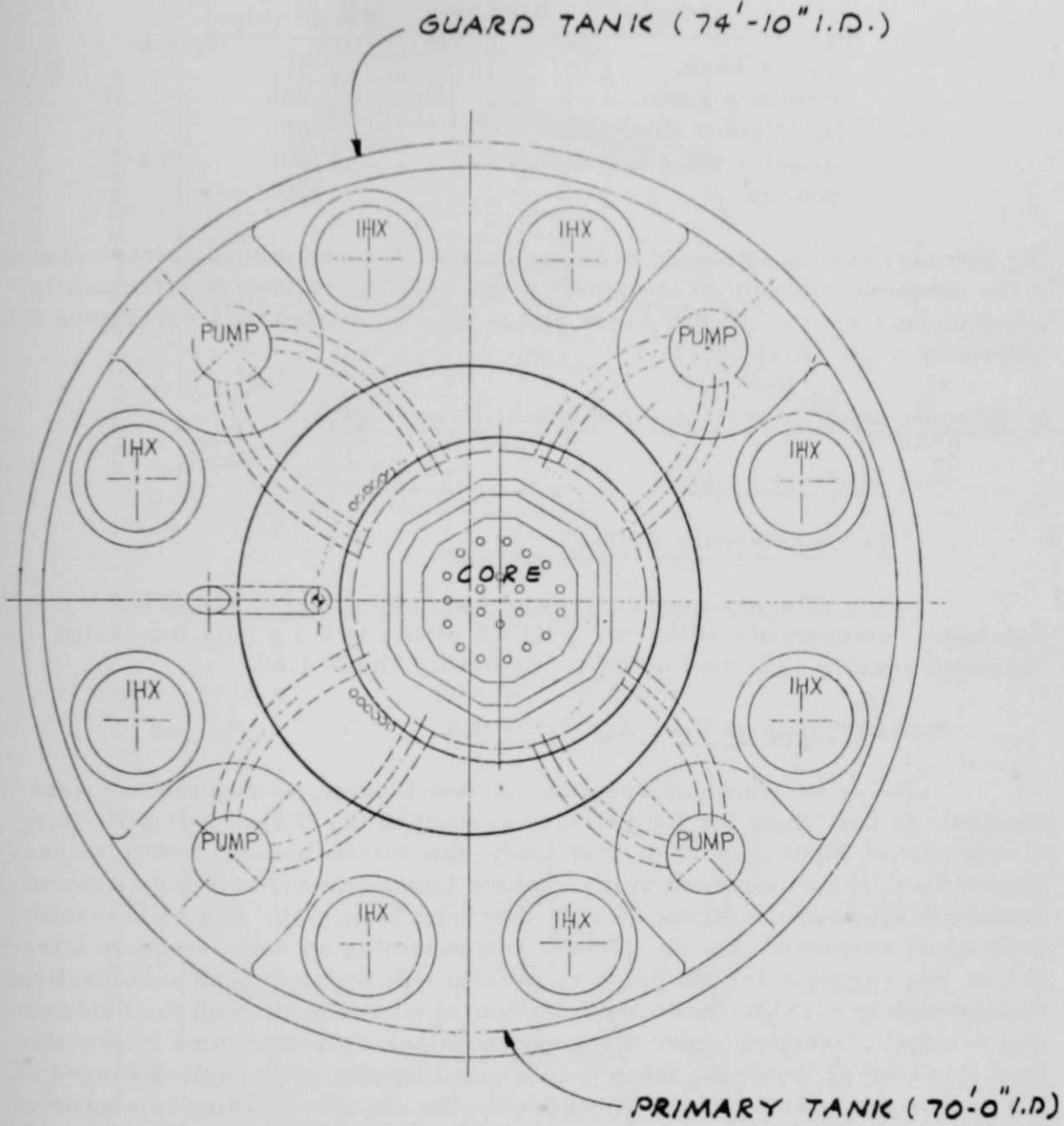

Fig. B.4. SuperPhenix (1200 MWe) Primary System--Plan View

The divider (which partially supports the hot center pool) is perforated, allowing flow of sodium. However, the hot center pool contains the outlet sodium (at approximately 986°F) and separates it from the remaining cold sodium (which is at 716°F). The liquid sodium fills the primary tank and the hot-center-pool to within 4 feet of the bottom of the shield deck (or about 9 feet from the assumed tank supports).

The weights of the various structural systems and equipment items used in the analysis are as follows:

Item	Weight (kips)
Guard Tank	950
Primary Tank	500
Hot Center Pool (Steel Only)	600
Reactor Core and Shielding	1600
Sodium	9000

The primary tank is assumed to be supported by a continuous skirt, embedded in the massive concrete of the shield deck. The guard tank is structurally independent from the primary tank and is also supported by a continuous skirt embedded in the shield deck.

3. Seismic Input Criteria and Analytical Procedures

a. Analysis Criteria

(1) Earthquake Criteria

The selected peak ground acceleration for the SSE for the Category 1 components of the two LMFBR plants is 0.3 g with the design response spectra specified by NRC Regulatory Guide 1.60.

(2) Support Peak Accelerations

At least two possible reactor building configurations were available in this study for the primary system of the (Expanded) EBR-II or the simulated SuperPhenix. In this study, the system is assumed to be suspended from the shield deck of the reactor (containment) building at approximately mid-elevation, about 70 feet above the base slab. If a stiff reactor building is embedded into the ground, and assuming no soil-structure interaction, the supports for the primary system will undergo peak accelerations approximately equal to those of the ground, i.e., 0.3 g for both the horizontal and vertical directions under the given criteria. Soil-structure interaction may increase or decrease these values significantly over limited ranges of the design response spectra. If, however, the reactor building is supported at grade by a base slab (i.e., is not embedded), the response spectra will be altered and amplified at the elevation of the shield deck. Typical reinforced concrete reactor buildings or containment buildings experience horizontal acceleration amplifications between 1.5 and 3.0 at 7% damping, with accompanying modifications of the entire spectra. It was felt to be sufficient for the purposes of this study to amplify the peak SSE acceleration by a factor of 2.0, to 0.6 g at the elevation of the support of the large vessels. In the vertical direction, structural amplifications are significantly lower because of the high stiffnesses of the structures. An amplification of 1.33 was selected, for a vertical SSE acceleration of 0.4 g.

For simplicity, it has also been assumed that the NRC Regulatory Guide 1.60 design response spectra (scaled to the respective peak floor accelerations) would represent the input motion to the vessel supports. The spectra are shown in Figs. B.5 and B.6.

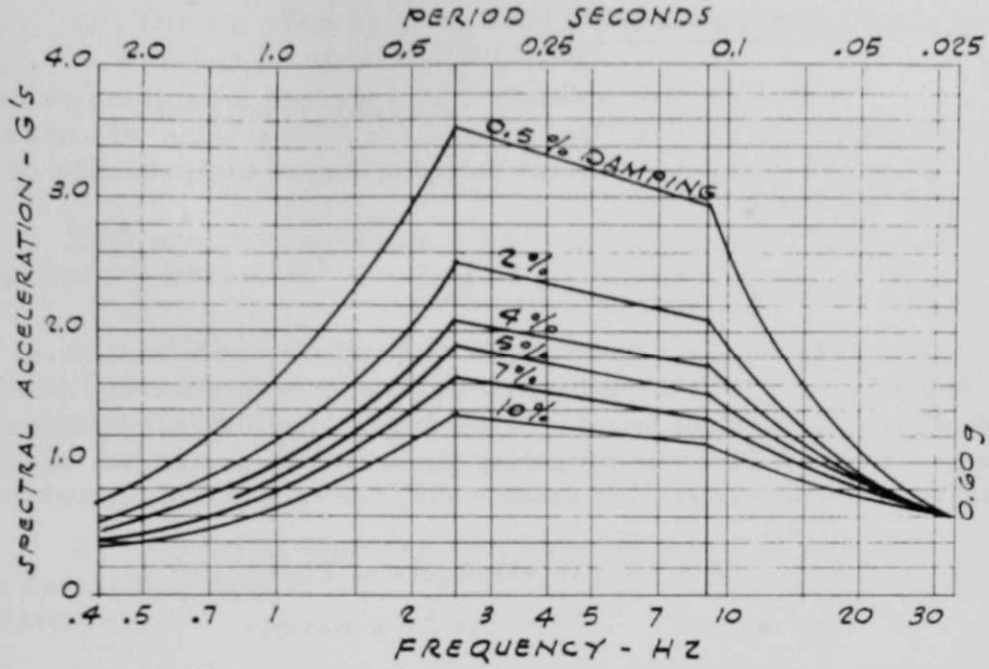

Fig. B.5. LMFBR Primary System SSE Response Spectra--Horizontal Direction

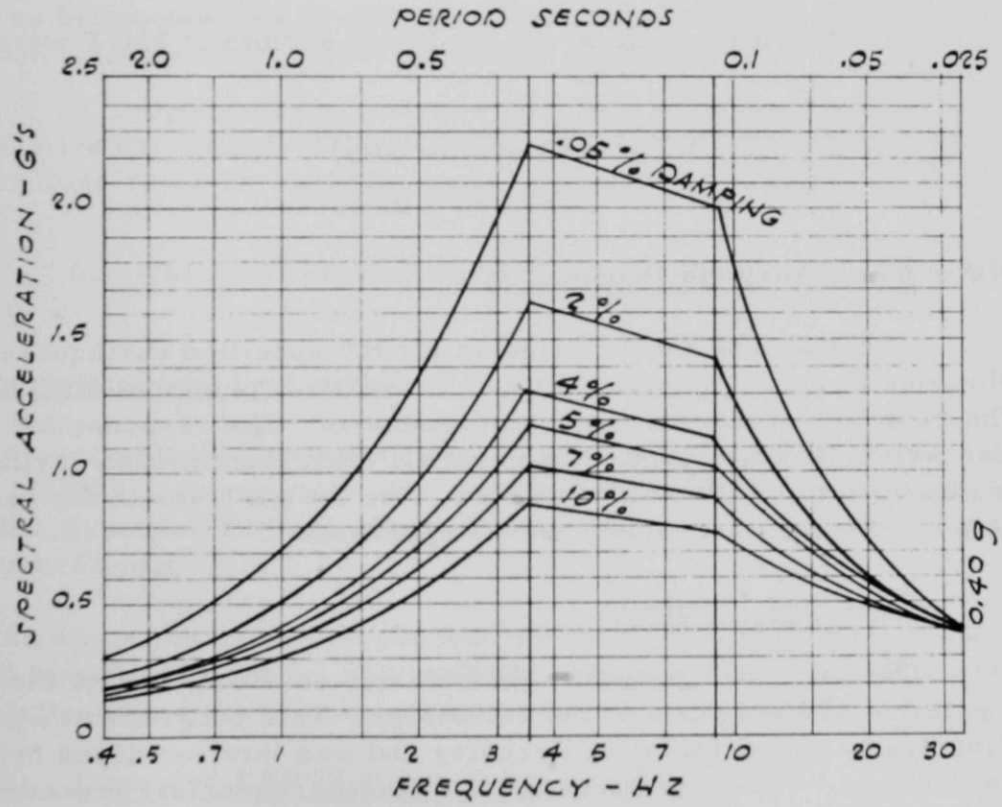

Fig. B.6. LMFBR Primary System SSE Response Spectra--Vertical Direction

(3) Damping Values

NRC Regulatory Guide 1.61 specifies a damping value of 4% of critical damping for welded steel structures. This value was used for the analyses. Possible changes in modal damping due to the presence of the liquid sodium were neglected.

(4) Material Properties--Steel

The hot liquid sodium is at temperatures from about 720°F to about 986°F. The 720°F temperature was selected for the analyses since that is representative of the safety-related parts evaluated. At that temperature, stainless steels have the following approximate properties according to the ASME Boiler and Pressure Vessel Code, Sec. III, Table I-6.0:

$E = 2.45 \times 10^7$ psi (Modulus of Elasticity)

$\nu = 0.30$ (Poisson's Ratio)

$\rho = 490$ lb/ft^3 (Weight Density)

(5) Material Properties--Sodium

The properties of the hot liquid sodium at 720°F were specified as follows:

$\rho = 53.5$ lb/ft^3 (Weight Density)

$\nu = 0.47$ (Poisson's Ratio)

b. Basic Analysis Procedures

The analyses were conducted for the specified earthquake spectral velocities in both the horizontal and the vertical principal directions, using the response spectrum numerical technique. The response for each principal direction was obtained such that all significant modes having frequencies up to about 30 Hz were included. For the analyses in the vertical direction, all the torsional modes were suppressed.

c. Computer Program

The computer program ASHSD4 was used to generate the dynamic characteristics and response of the structures. This program was developed at the University of California at Berkeley and was later modified by Dr. Edward Tsui.[37,38]* It provides three different numerical procedures for the dynamic analyses of lumped-mass, finite-element structural systems:

*Superscripts refer to specific references listed at the end of this report.

1. Direct, step-by-step time integration, using the improved mid-point acceleration scheme
2. Modal superposition
3. Response spectrum

This program provides a wide spectrum of analytical capability, including, but not limited to:

1. Fluid elements have been incorporated. This type of element provides the hydrodynamic effects due to both apparent and virtual masses of the fluid entrapped between concentric flexible shells, including internals and other tubular members such as pumps and intermediate heat exchangers. The resulting mass matrices for these elements have nonzero, off-diagonal terms.

2. A special algorithm for the extraction of eigenvalues for systems having nonnegative definite banded mass matrices has been incorporated.

3. Shell stresses, including the exact axisymmetric solutions for cylindrical shells, are provided in the response printout.

4. Tubular internals, pumps, heat exchangers, etc., can be handled as beam-columns or effective shell elements with arbitrary boundary conditions, i.e., elastic, hinged, or simple supports.

5. Options have been provided for recording absolute accelerations, relative displacements, mode-shape matrices, etc., on special tapes such that floor response spectra, mode shapes, etc., can be automatically plotted.

6. Thicknesses of shell elements are input as functions of elements.

4. Mathematical and Test Models

A series of mathematical models were developed for the analyses of the two LMFBR primary systems. Two basic models were used for the analysis of the Expanded EBR-II. Two similar models were used for the analysis of the SuperPhenix.

An approximate 1:70 scale test model of the Expanded EBR-II was built and tested. The model was used for simple testing of sloshing behavior and the influence of the internal components on the sloshing.

a. Expanded EBR-II Mathematical Models

Two axisymmetric mathematical models were used for the analysis of the Expanded EBR-II. The basic model is shown in Figs. B.7 and B.8.

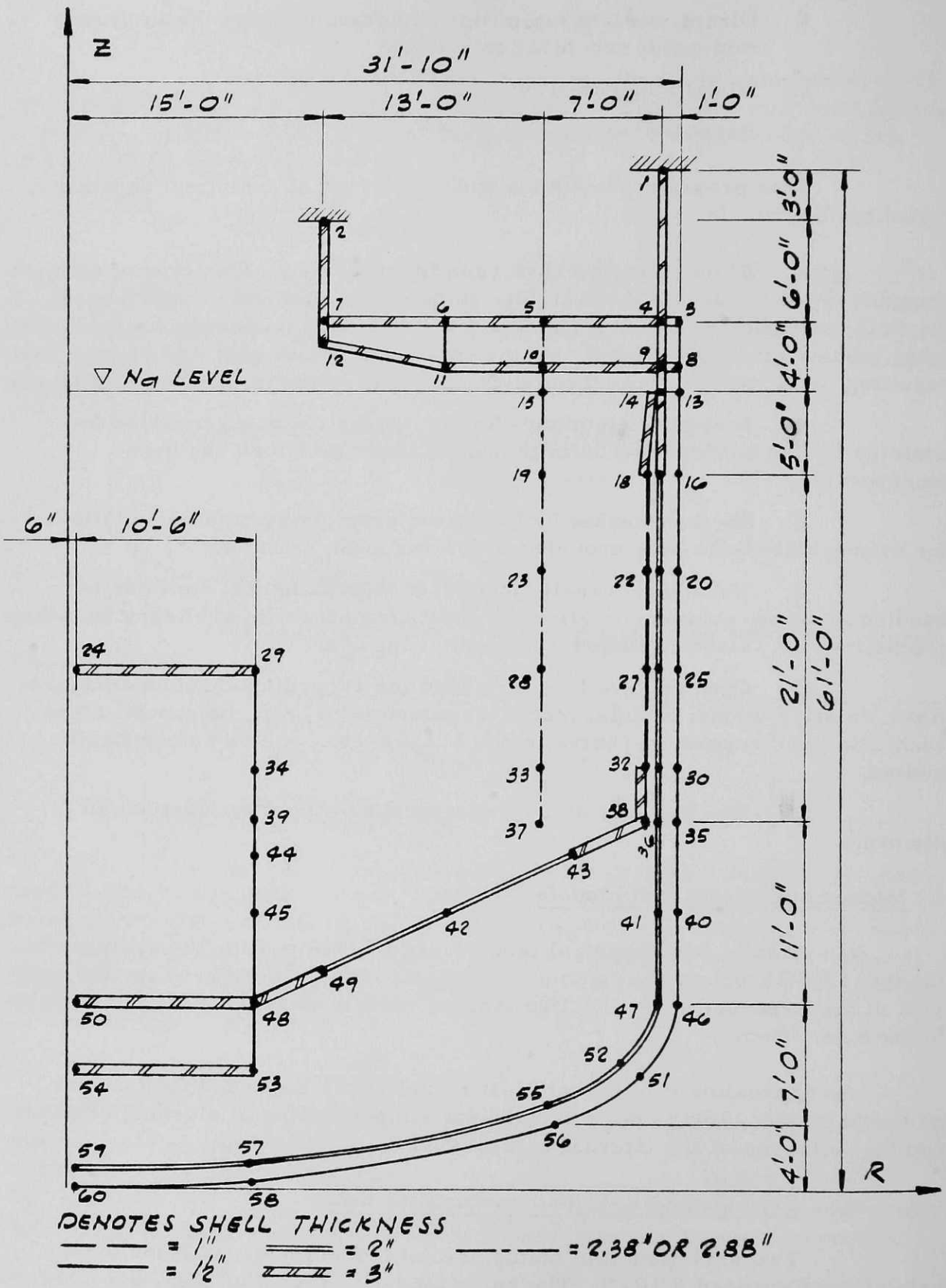

Fig. B.7. Expanded EBR-II Mathematical Model Nodes and Element Thicknesses

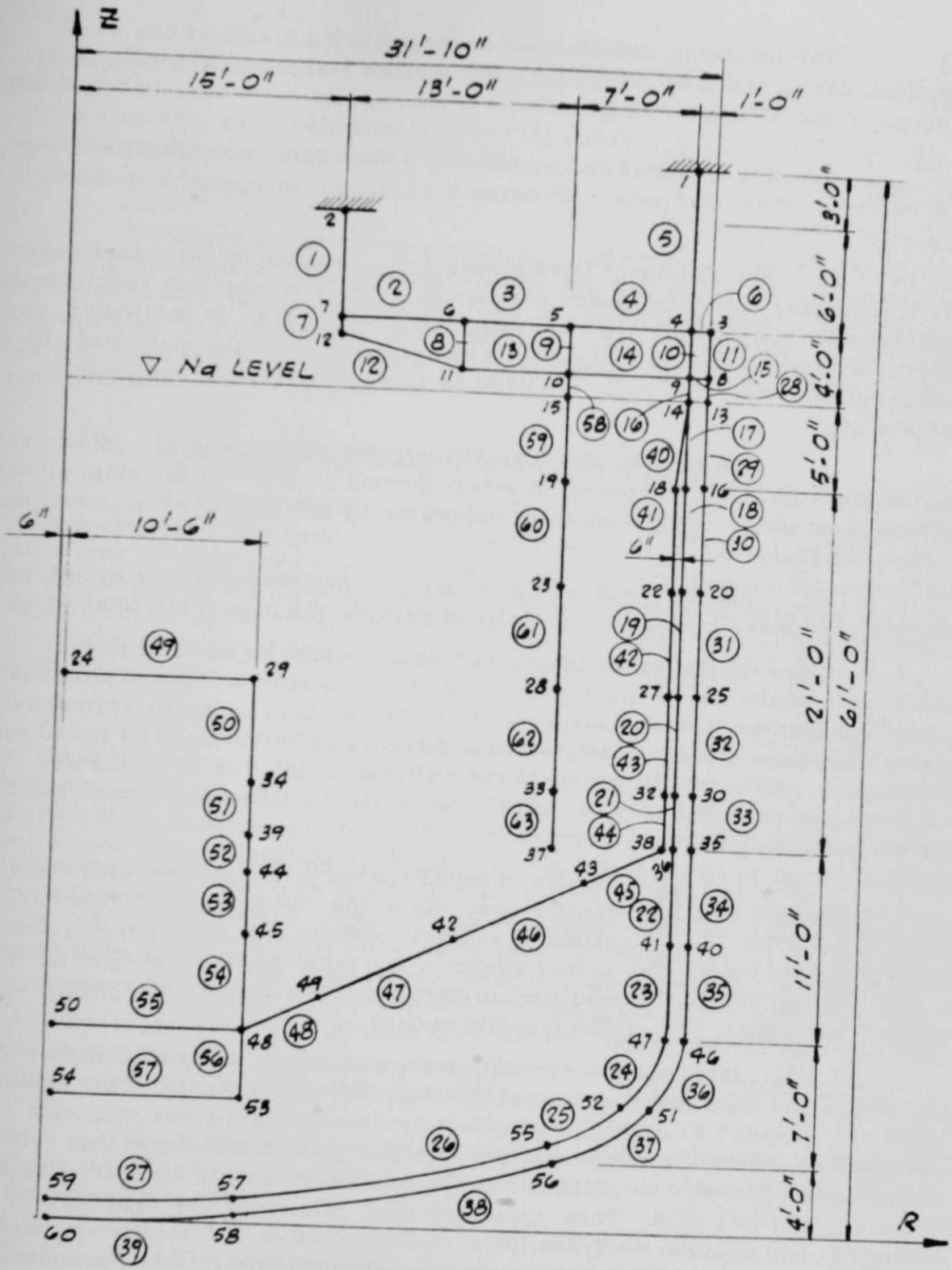

Fig. B.8. Expanded EBR-II Mathematical Shell Elements

The following assumptions and associated idealizations were made in order to describe adequately the various structural and equipment elements of the primary system:

1. All elements, except the fluid elements, were idealized with finite-element shell elements with defined thickness, stiffness, and mass properties.

2. Shell element thicknesses were selected to represent reasonable engineering estimates based on structural need or erection requirements. These are not necessarily the final required thicknesses. In particular, plate thicknesses in excess of 2 inches were used to represent the stiffening effects of nearby components or were required to represent rigid or semirigid components.

For the primary tank cover plate a thickness of 3 inches was used to approximately represent the transformed sections of the steel girders. Fifty percent of the equipment mass supported by the primary tank cover was used in the analysis.

3. The core and its substructure were modeled as a cylindrical tank with the mass lumped in the walls to provide the necessary total weight.

4. The primary tank cover was modeled as two flat plates, which were stiffened, or tied together, by the main tank and guard tank, the equipment (pumps, heat exchangers, etc.) and the inner primary system support. The plates were spaced the same distance as in the tapered radial support beams. Although the approximate stiffness of the radial beams was achieved, the analysis does not address directly the individual beams and the forces which they carry.

5. The two supports of the (Expanded) EBR-II are continuous skirts embedded into the massive concrete of the shield deck above the primary tank cover. It was assumed, as illustrated in Fig. B.7, that the supports are fully fixed. The actual supports are most probably partially fixed. It was felt that for the current preliminary analysis, fully fixed supports (radial, tangential, and vertical) would suffice.

6. It was felt to be important to include a simplified model of the pumps and heat exchangers into the overall system model. This equipment was modeled as a rough equivalent cylindrical cantilever with side walls at the center line of the equipment. The plate thickness of this cylinder was calculated to provide stiffness properties only roughly similar to those of the actual equipment. This calculated plate thickness was different for the horizontal and vertical analyses because the effective stiffnesses are unequal for the two cases. Partial liquid flow was allowed through this continuous, but actually perforated, cylinder.

7. Piping and other such "secondary" elements were neglected for these preliminary analyses.

8. The fluid elements were used only in the regions where the liquid sodium is entrapped by concentric shells. Otherwise, the masses of the liquid sodium were lumped appropriately to the adjacent nodal points. These elements were effective only for the horizontal earthquake; for the vertical earthquake, they were replaced by solid elements with insignificant stiffness.

Figure B.8 shows the basic configuration used in the analysis. Two slightly different versions were used--one for the horizontal and one for the vertical direction. Because the fluid is essentially incompressible, it was felt sufficient to assume that little motion of the fluid would occur in the vertical direction, and the mass could be distributed to the various supporting shell elements (nodal points).

b. Simulated SuperPhenix Mathematical Models

Three axisymmetric models were developed for the analysis of the SuperPhenix. The basic models are shown in Figs. B.9 and B.11. The guard tank was analyzed separately because it is structurally separated from the primary tank. In these models, the pumps and heat exchangers are supported directly by the shield deck. The dynamic properties of the equipment were included in the model to account for coupling effects due to the fluid-solid interaction.

The hot-center-pool is a cantilevered structure supported by the bottom of the primary tank near the core supports. Thus, it cantilevers from the bottom of the primary tank, a double (two-mass) pendulum effect. Two axisymmetric models were necessary to evaluate the performance of the hot center pool walls. In Model 1, the pool is supported partially by the 3/4-inch divider along the curved portion of the primary tank (elements 19 to 25 of Fig. B.10). In Model 2, element 25 was removed so that the hot-center-pool was supported by its 1-inch-thick walls near the reactor core. In addition, a new member was added, tying the hot-center-pool to the primary tank near its support. Originally, it was intended that the hot-center-pool in Model 2 be bottom-supported only. Preliminary stresses, particularly from the vertical component of the earthquake, indicated over-stressing for this simpler concept. Thus the concept was not analyzed in detail.

The following assumptions and associated idealizations were made in order to describe adequately the various structural and equipment elements of the primary system:

1. The reactor core and the reactor core support structure were idealized as a group of cylindrical and horizontal plates. The desired 1600 kip weight was lumped into these plates by adjusting the mass densities.

2. The 3/4-inch plate shrouds of the pumps and the heat exchangers (which encase the equipment so that the equipment does not participate in the interaction between the shrouds and the nearby vessels) were

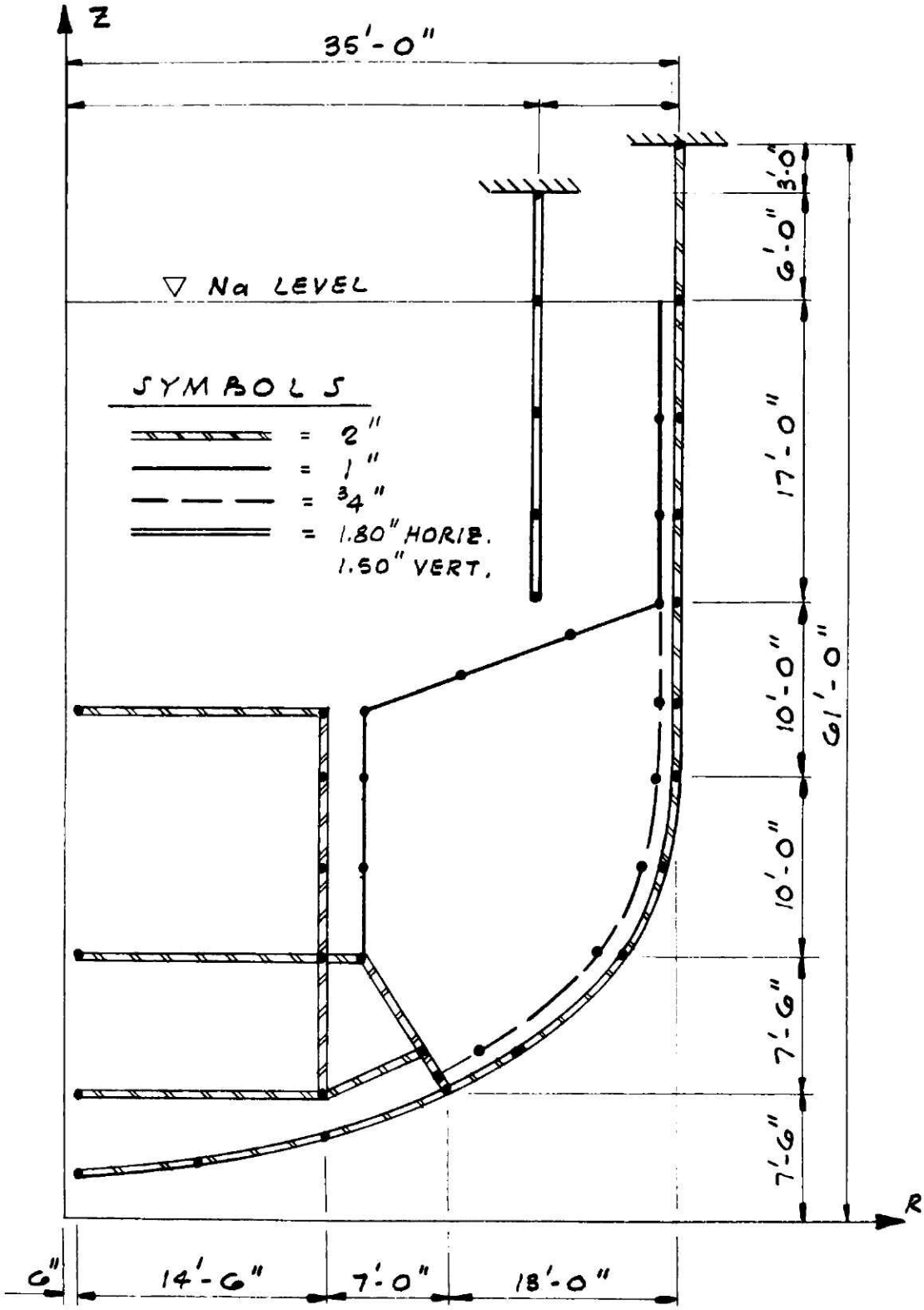

Fig. B.9. SuperPhenix Mathematical Model Nodes and Element Thickness

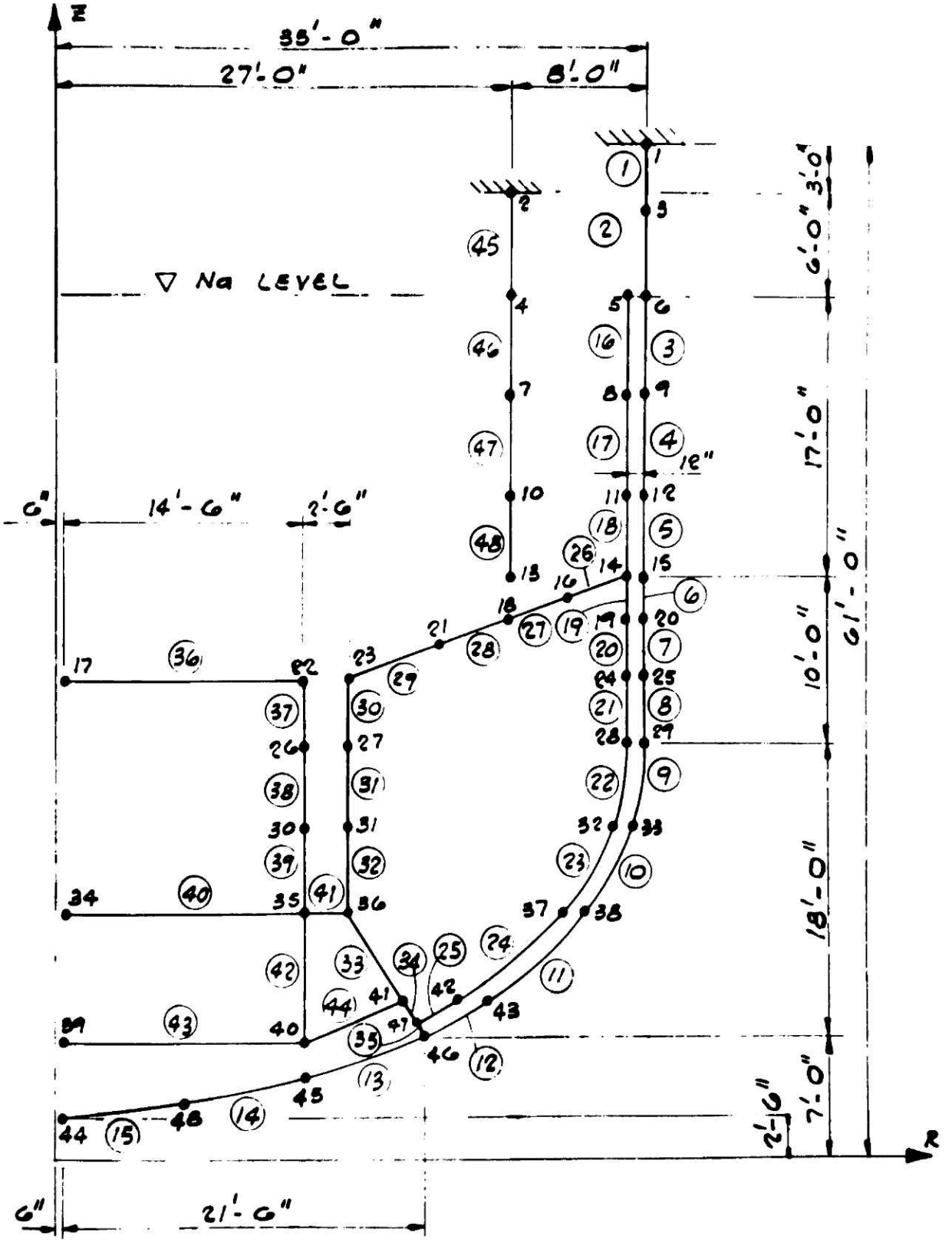

Fig. B.10. SuperPhenix Mathematical Model 1 Shell Elements

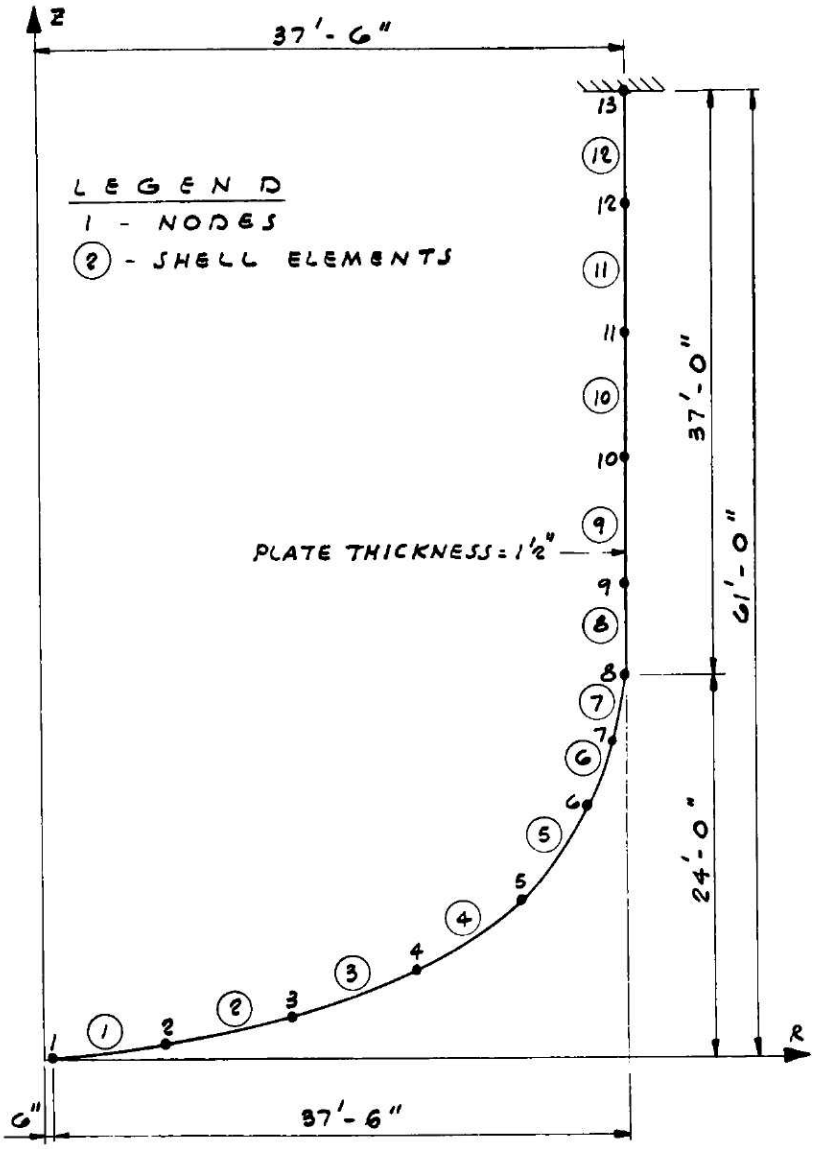

Fig. B.11. SuperPhenix Mathematical Model Guard Tank

modeled as a continuous cylinder with stiffness and mass equivalent to that of the assumed equipment. Partial liquid flow was allowed through this continuous, but actually perforated, cylinder.

3. The baffle structures were simplified for these models. The 3/4-inch-thick baffles were attached to the hot-center-pool walls, to the primary tank (as described above), and to the reactor core support structure.

4. Several other elements, such as numerous baffles, equipment near the reactor core, etc., were omitted from the models because they are not expected to modify significantly the overall structural behavior of the primary system.

5. All elements, except the fluid elements, were idealized with finite elements with defined thickness, stiffness, and mass properties.

6. The support of the primary tank was modeled as a continuous skirt embedded in the massive concrete of the shield deck. It was assumed, as illustrated in Fig. B.9, that the supports are fully fixed.

7. The main piping, which interconnects the primary pumps, the reactor core was neglected for these analyses.

8. The fluid elements were used only in the regions where the liquid sodium is entrapped by concentric shells. Otherwise, the masses of the liquid sodium were lumped appropriately to the adjacent nodal points. These elements were effective only for the horizontal earthquake; for the vertical earthquake, they were replaced by solid elements with insignificant stiffness.

c. Test Model of Expanded EBR-II

An approximate 1:70 scale rigid model of the Expanded EBR-II primary system was constructed. The reactor core and the core support structure were modeled, and the gross effects of the pumps and heat exchangers were modeled using equivalent cylinders. The structural stiffnesses of the various components were not modeled because the test model was constructed to determine the effects of the contained equipment and supports on the sloshing modes and frequencies of the contained liquid sodium.

The model assumed that the fluid pressures on the actual, flexible tank may be approximated by the fluid pressures on a geometrically similar rigid tank. This assumption was based on the following observations. The fluid pressures can be decomposed into a part associated with the inertial force of the fluid mass and a part associated with sloshing of the fluid. Preliminary investigations indicated that the frequencies would be short with respect to the structural frequencies of the tank. Therefore, little interaction was expected between tank vibrations and sloshing vibrations. Furthermore, sloshing amplitudes were expected to be large with respect to the structural deformations. Therefore, it was assumed that the sloshing amplitudes and pressures for the actual, flexible tank could be well approximated by the sloshing amplitudes and pressures for a geometrically similar rigid tank.

5. Dynamic Response of the Structures

a. Expanded EBR-II

(1) Model

The following analyses were completed for the Expanded EBR-II concept:

Horizontal Direction

1. Primary System with Sodium

2. Primary System without Sodium

Vertical Direction

3. Primary System with Sodium

4. Primary System without Sodium

Member forces and stresses were calculated for all of the above runs for the stated SSE response spectra. The analyses of the primary system without sodium were not essential; the results were used to check the validity of the models and the influence of the fluid and its interaction with the structure. The alterations of the natural frequencies of the empty system were consistent with the increased masses from the sodium. In addition, new modes were introduced, which are attributed to fluid-structure coupling effects.

Table B.1 summarizes the frequencies for the four analyses. Table B.2 summarizes in more detail the dynamic characteristics of the primary system with sodium under horizontal excitation, including the participation factors of the more important modes. Seven of the fifteen extracted modes contribute most of the inertial forces. These seven modes are basically the major structural modes of the system, with the fluid coupling effects included. The approximate frequency of the empty guard and primary tanks is 20 Hz. For the full system, the frequency is about 13 Hz. The frequency of the core support structure is similarly lowered from about 11 Hz to about 8 Hz.

TABLE B.1. Expanded EBR-II Frequencies (Hz)

Mode No.	Horizontal Direction		Vertical Direction	
	without Na	with Na	without Na	with Na
1	5.68	2.03	3.49	2.53
2	6.45	2.37	3.68	3.33
3	11.3	3.38	3.85	3.81
4	19.5	5.31	6.14	4.87
5	19.7	5.36	8.68	4.95
6	20.0	7.36	-	8.04
7	23.6	8.63	-	11.0
8	25.2	12.3	-	11.2
9	26.0	15.0	-	14.0
10	26.5	18.0	-	14.2
11	-	18.8	-	-
12	-	19.5	-	-
13	-	21.7	-	-
14	-	25.2	-	-
15	-	25.7	-	-

TABLE B.2. Expanded EBR-II Participation Factors--Major Modes Only (Horizontal Direction with Na)

Mode No.	Frequency, Hz	Participation Factor, %	Active Structural Components
4	5.31	11.2	Pumps Supports and Core Area
5	5.36	14.5	Pumps Supports and Core Area
6	7.36	13.0	Core and Core Support
7	8.63	14.3	Core and Core Support
8	12.3	12.3	Primary Tank and Core Support
9	15.0	7.4	Primary Tank and Core Support
11	18.8	9.6	Guard and Primary Tanks
		Total = 82.3%	

The presence of the liquid sodium, and to some extent the presence of unconnected structures and equipment items, causes closely spaced modes. These are modes with frequencies that are within 10% of each other. For the horizontal direction analysis with sodium, modes 4 and 5 (f = 5.31 and 5.36 Hz) are closely spaced. When the frequencies are not closely spaced, the stresses (and/or forces, displacements, accelerations) are combined by the Square Root of the Sum of the Squares (SRSS) method. Closely spaced modes are combined by the Absolute Summation method. In these analyses, all stresses were combined by the SRSS method. Generally, this method is conservative but may result in less conservative stresses in a few instances. It was felt that for the purposes of these preliminary studies, the SRSS method would be adequate.

(a) Displacements

Figures B.12 and B.13 show the SRSS displacements of the entire system for the horizontal and vertical earthquake components, and Table B.3 summarizes the maximum displacements for each major structural component of the structure. The radial displacements of all tanks are less than 0.26 inch and the vertical displacements are less than 0.49 inch. The primary tank cover deflects about 0.94 inch in the vertical direction for the assumed configuration and approximate equipment masses. None of the displacements appear sufficient to cause pounding between adjacent structures.

(b) Stresses

Figures B.14 and B.15 summarize the maximum longitudinal stresses, σ_m, for three cases:

1. Horizontal earthquake
2. Vertical earthquake
3. SRSS of the above

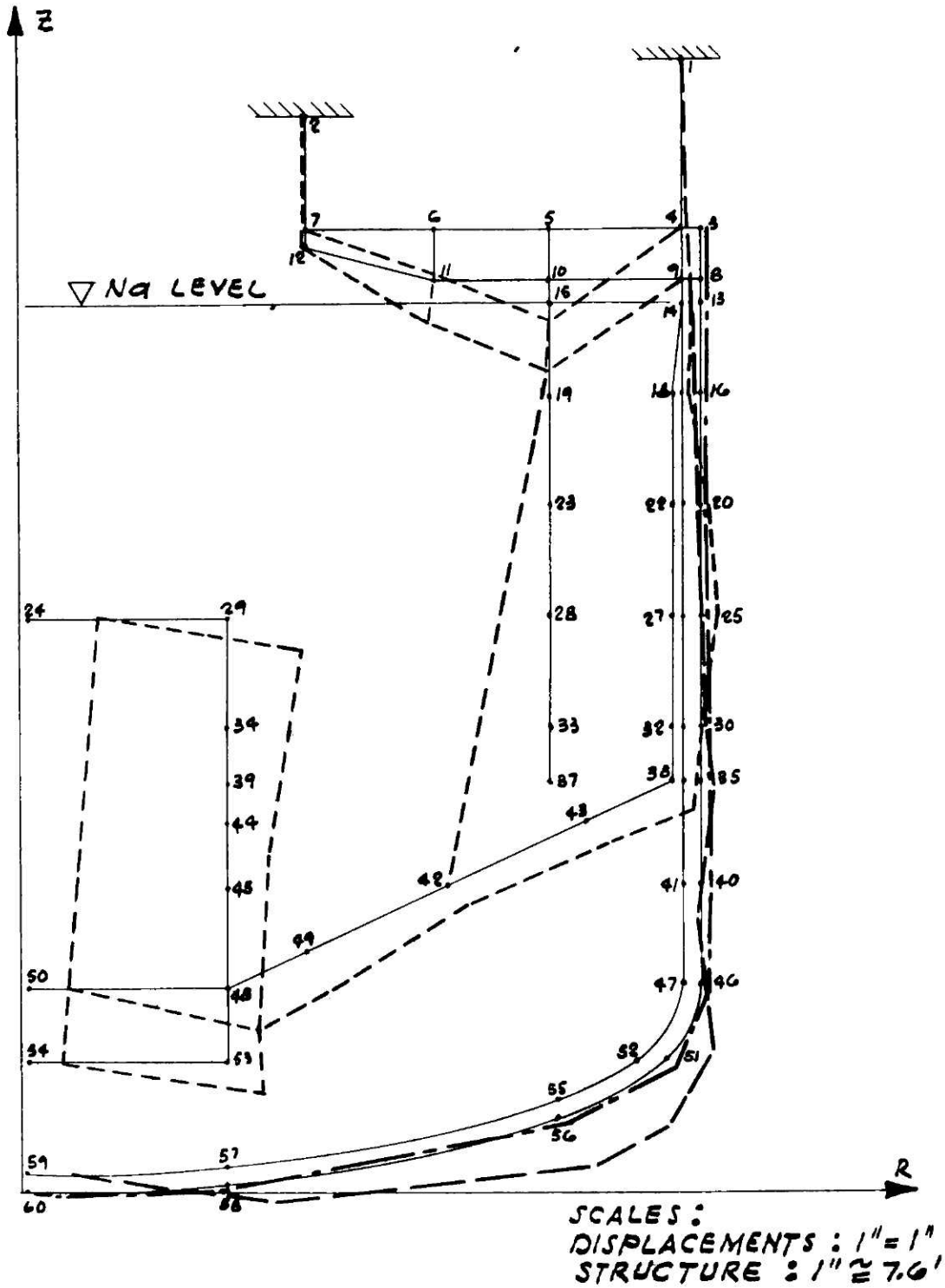

Fig. B.12. Expanded EBR-II SRSS Displacements--Horizontal Earthquake

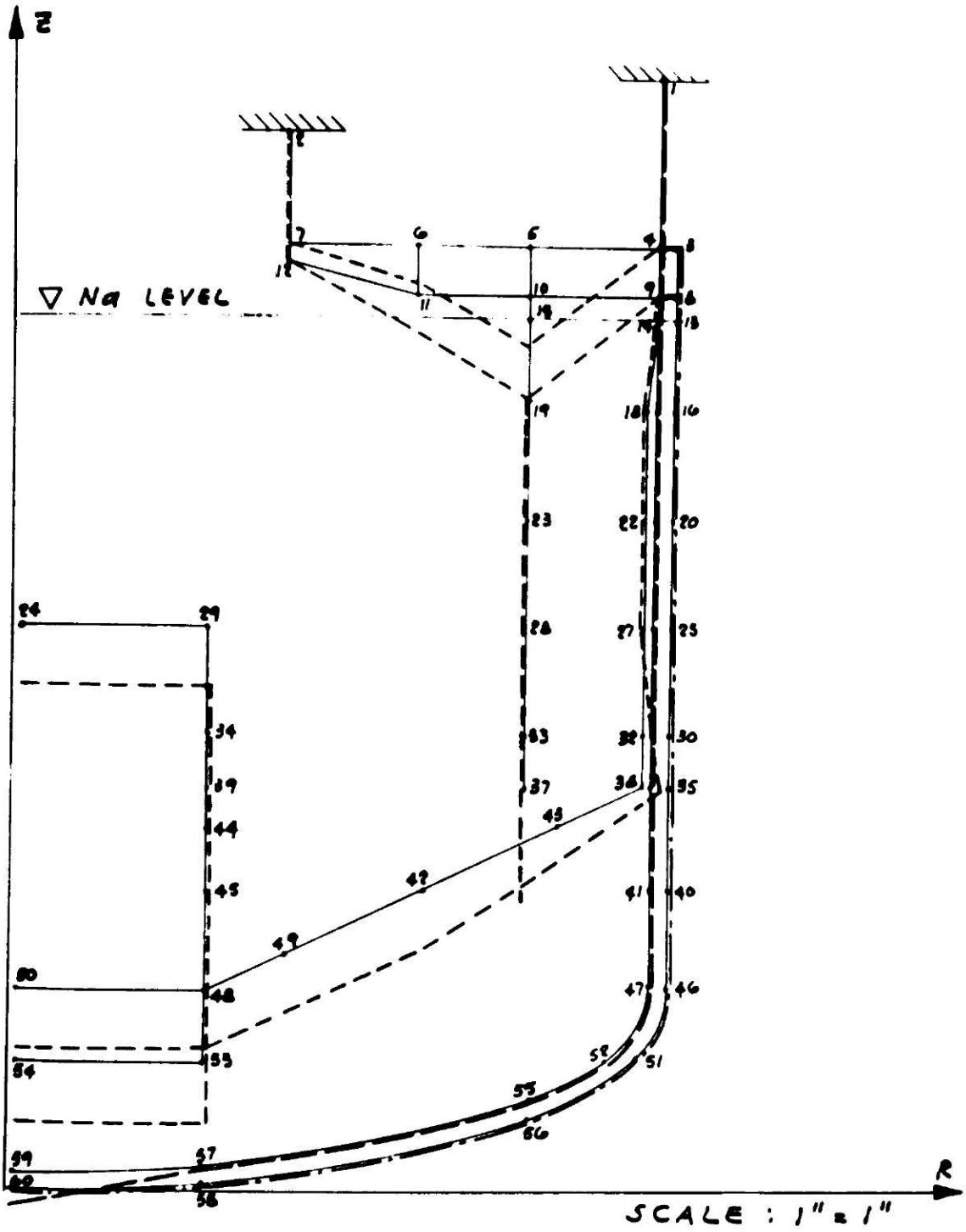

Fig. B.13. Expanded EBR-II SRSS Displacements--Vertical Earthquake

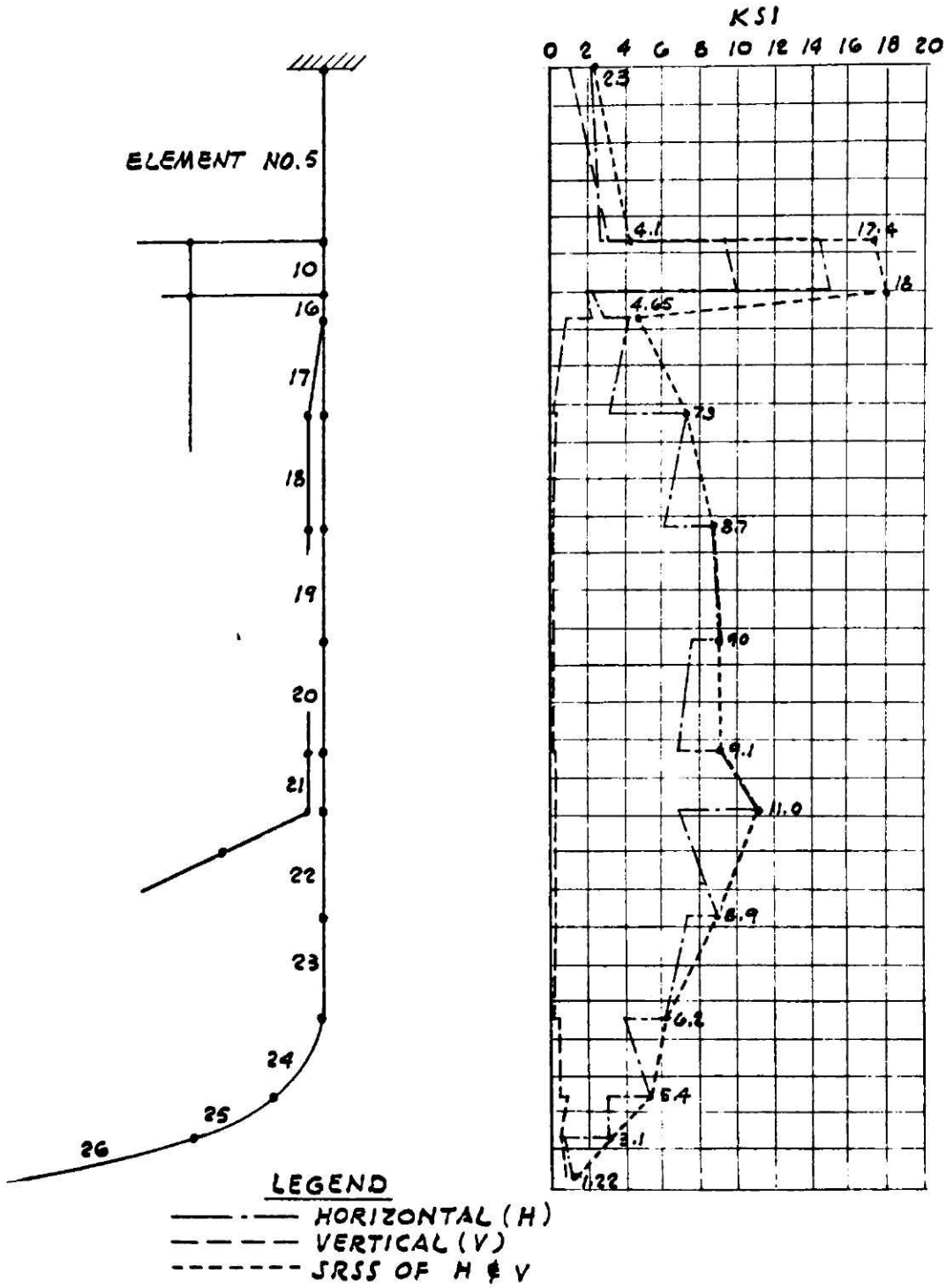

Fig. B.14. Expanded EBR-II-σ_m--SRSS Maximum Longitudinal Stress--Primary Tank

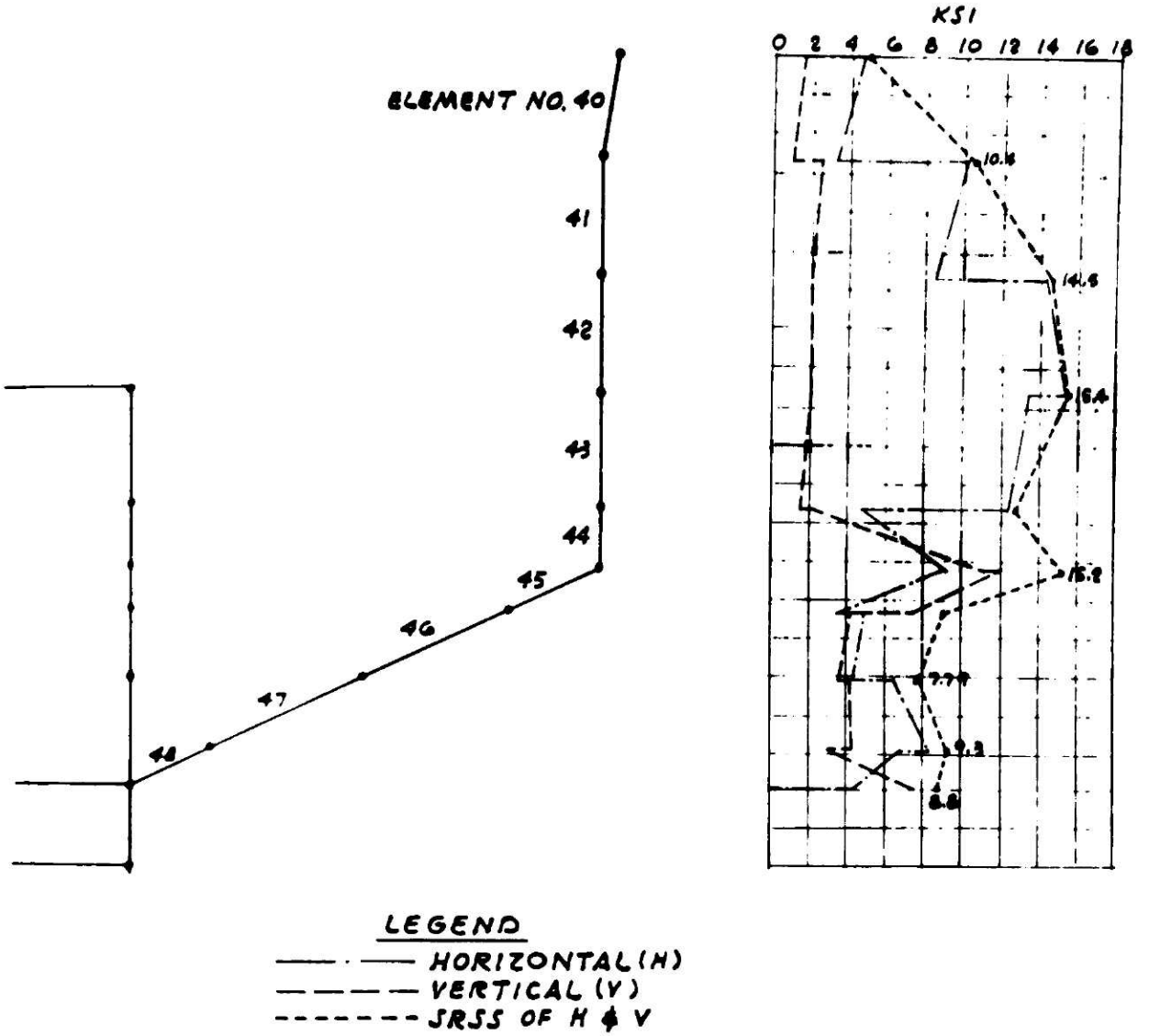

Fig. B.15. Expanded EBR-II-σ_m—SRSS Maximum Longitudinal Stress—Core Support Structure

TABLE B.3. Expanded EBR-II Maximum SRSS Displacements (Inches)

Component	Nodal Point	Horizontal		Vertical		SRSS	
		R	Z	R	Z	R	Z
Primary Tank Cover	10	0.051	0.625	0.023	0.697	0.056	0.936
Guard Tank	46	0.041	0.011	-	0.005	0.041	0.011
Primary Tank	55	0.255	0.451	0.008	0.021	0.255	0.451
Core Support	48	0.198	0.237	0.008	0.423	0.198	0.485

Primary Tank. In this analysis, the primary tank liner plate thickness is mostly 2 inches, except at the support, elements 5, 10, 16, and 17, where it is 3 inches.

At the support skirt (element 5) the SRSS stresses are as follows (see Sec. 3.4 for the stress nomenclature):

$$\sigma_m = 4.1 \text{ ksi}$$

$$\sigma_t = 1.8 \text{ ksi}$$

$$\sigma_{mt} = 1.4 \text{ ksi}$$

Thus, the assumed 3-inch thickness is unnecessary and a thinner skirt, perhaps 2 inches, may suffice.

σ_m at the primary tank cover connection is about 18 ksi. σ_t is about 7.6 ksi. These local stresses are due to discontinuity effects introduced by the cover.

Below the core support connection, σ_m varies between 1.0 and 11.0 ksi. The hoop stress, σ_t, is of a comparable magnitude.

The shear stress, σ_{mt}, is less than 2.0 ksi for the structure.

In summary: for the assumed structural configuration, equipment, supports, and masses, the primary tank is highly stressed in the vicinity of the primary tank cover support and for much of its height. Attention to design detail, however, should reduce these stresses to acceptably low levels.

Core Support Structure. For the cylindrical section σ_m exceeds 10.0 ksi. σ_t is generally somewhat higher than σ_m in the higher stress areas. Otherwise the two stresses are similar. σ_{mt} reaches 5.1 ksi in element 47, near the core. Elsewhere, the in-plane shears are below 3.5 ksi.

In summary, the thin cylindrical section is highly stressed. Thickening of the section may lower the stresses. It would also stiffen the entire structure and reduce the sway of the core. Rounding of the cone-cylinder connection may also lower the local stress concentrations.

Guard Tank. The stresses in the guard tank are low. At the connection to the primary tank (element 11) the SRSS stresses are:

$$\sigma_m = 6.2 \text{ ksi}$$

$$\sigma_t = 4.0 \text{ ksi}$$

$$\sigma_{mt} = 1.0 \text{ ksi}$$

Below that connection, the individual stresses quickly fall off to less than 1.0 ksi.

Primary Tank Cover. In this model, the cover supports the equipment. It appears that the connection of the cover to the primary tank causes local stress concentrations. It may be necessary to provide a separate skirt for the cover, which would carry most of the equipment loads directly into the shield deck thus minimizing the seismic stresses in the primary tank.

b. Simulated SuperPhenix

(1) Model 1

The following analyses were completed for the simulated SuperPhenix, Model 1:

Horizontal Direction

1. Primary System with Sodium
2. Primary System without Sodium

Vertical Direction

3. Primary System with Sodium
4. Primary System without Sodium

Member forces and stresses were calculated for all the above runs for the stated SSE response spectra.

Table B.4 summarizes the frequencies (in Hz) for the four analyses. Table B.5 summarizes in more detail the dynamic characteristics

TABLE B.4. SuperPhenix Model No. 1 Frequencies (Hz)

Mode No.	Horizontal Direction		Vertical Direction	
	without Na	with Na	without Na	with Na
1	4.79	1.45	1.51	0.86
2	11.9	3.97	1.54	1.55
3	13.3	5.63	4.11	4.07
4	13.4	6.27	6.51	4.84
5	14.1	7.03	15.7	6.05
6	24.3	7.16	23.3	8.23
7	26.1	8.25	27.1	8.88
8	34.3	9.15	34.1	14.2
9	36.0	10.6	36.1	15.5
10	36.6	12.6	37.3	17.7
11	-	13.3	-	-
12	-	13.4	-	-
13	-	16.0	-	-
14	-	18.4	-	-
15	-	19.2	-	-

TABLE B.5. SuperPhenix Model No. 1 Participation Factors--Major Modes Only (Horizontal Direction with Na)

Mode No.	Frequency, Hz	Participation Factor, %	Active Structural Components
1	1.45	11.3	Hot-Center-Pool, Divider, Core
2	3.97	7.3	Hot-Center-Pool, Divider, Core
5	7.03	18.3	Primary Tank, Hot-Center-Pool, Divider, Core
6	7.16	14.0	Primary Tank, Hot-Center-Pool, Divider, Core
7	8.25	16.3	Primary Tank, Hot-Center-Pool, Divider, Core
9	10.6	19.3	Primary Tank, Hot-Center-Pool
10	12.6	6.0	Divider, Hot-Center-Pool
		Total = 92.5%	

of the primary system with sodium under horizontal excitation, including the participation factors of the more important modes. Seven of the fifteen extracted modes contribute most of the inertial forces. These seven modes are basically the major structural modes of the system, with the fluid coupling effects included.

For the horizontal direction analysis with sodium, modes 5 and 6 (f = 7.03 and 7.16 Hz) are closely spaced. In these analyses, all stresses were combined by the SRSS method. Generally, this method is conservative, but may result in some less conservative stresses in a few instances.

(a) Displacements. Figures B.16 and B.17 show the SRSS displacements of the entire system for the horizontal and vertical earthquake components, and Table B.6 summarizes these displacements for each major structural component of the system. The largest displacements, 1.27 and 1.13 inches in the horizontal and vertical directions, respectively, are near the top of the hot pool walls. These displacements indicate that the baffles in the area will see concentrated interaction forces and may be overstressed. The maximum displacements in the primary tank are small, 0.25 inch and 0.53 inch in the horizontal and vertical directions, respectively. None of these displacements appear sufficient to cause pounding between adjacent major structural components.

(b) Stresses. Figures B.18 to B.20 summarize the maximum longitudinal stresses, σ_m, for three cases:

1. Horizontal earthquake
2. Vertical earthquake
3. SRSS of the above

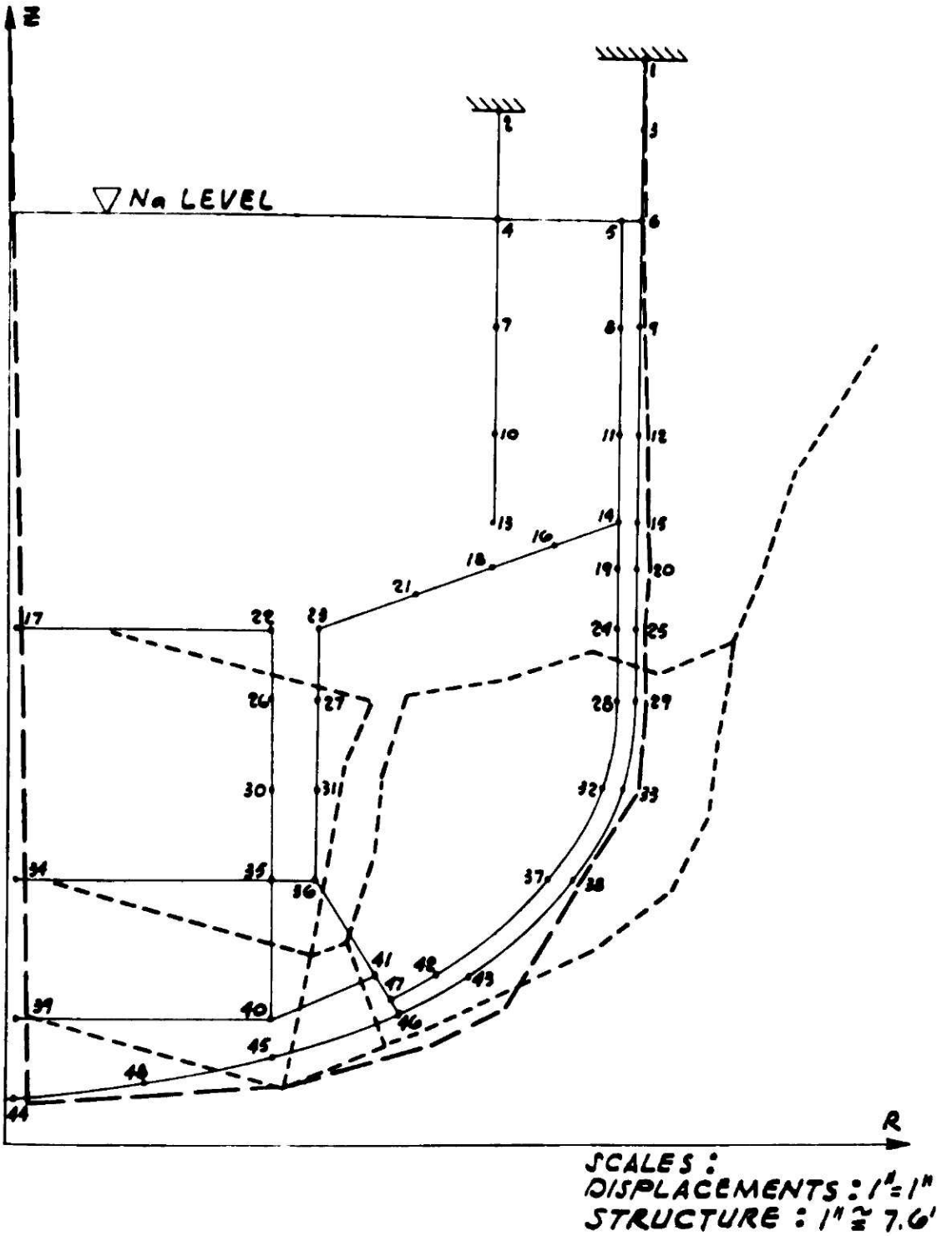

Fig. B.16. SuperPhenix, Model 1 SRSS Displacements--Horizontal Earthquake

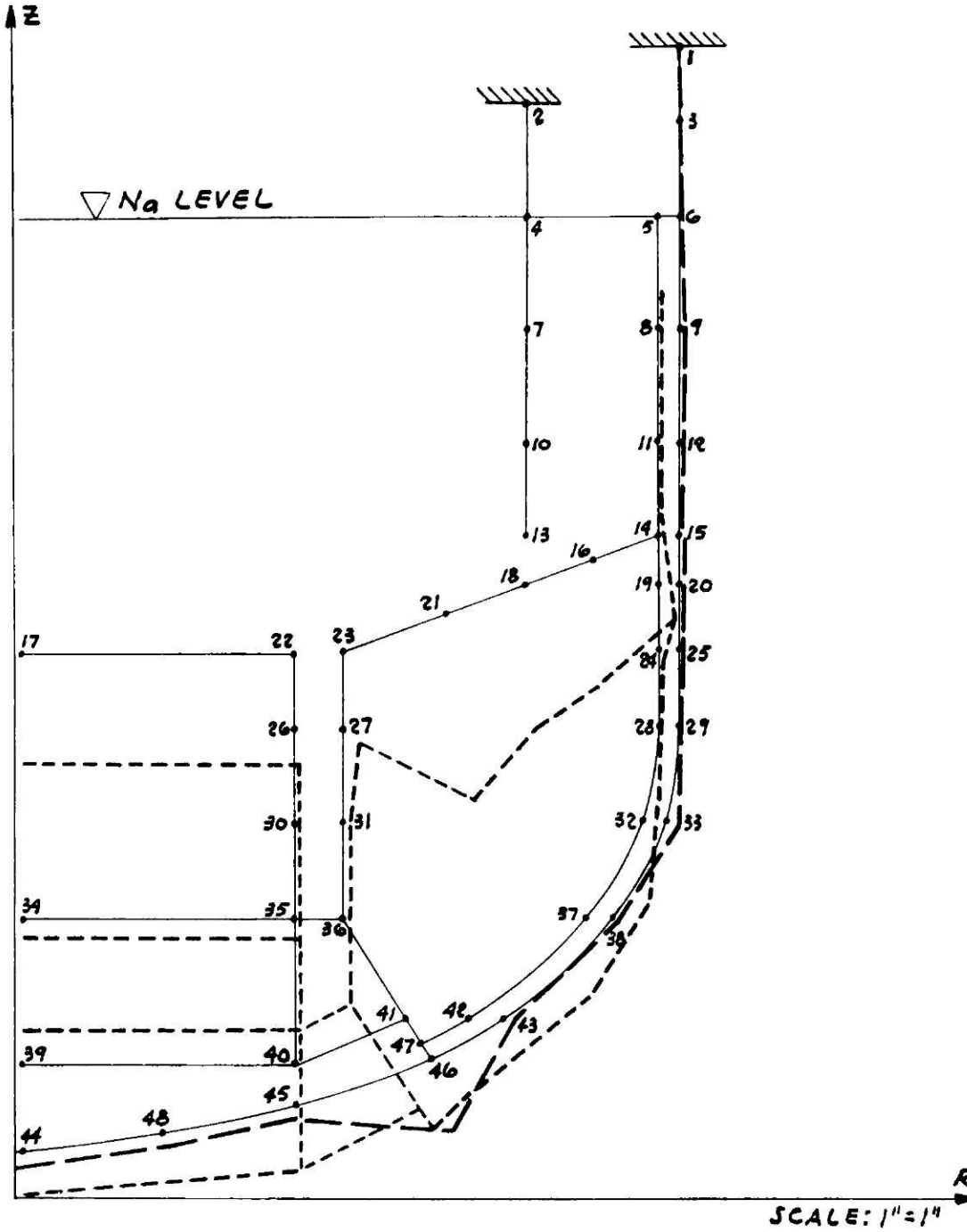

Fig. B.17. SuperPhenix, Model 1 SRSS Displacements--Vertical Earthquake

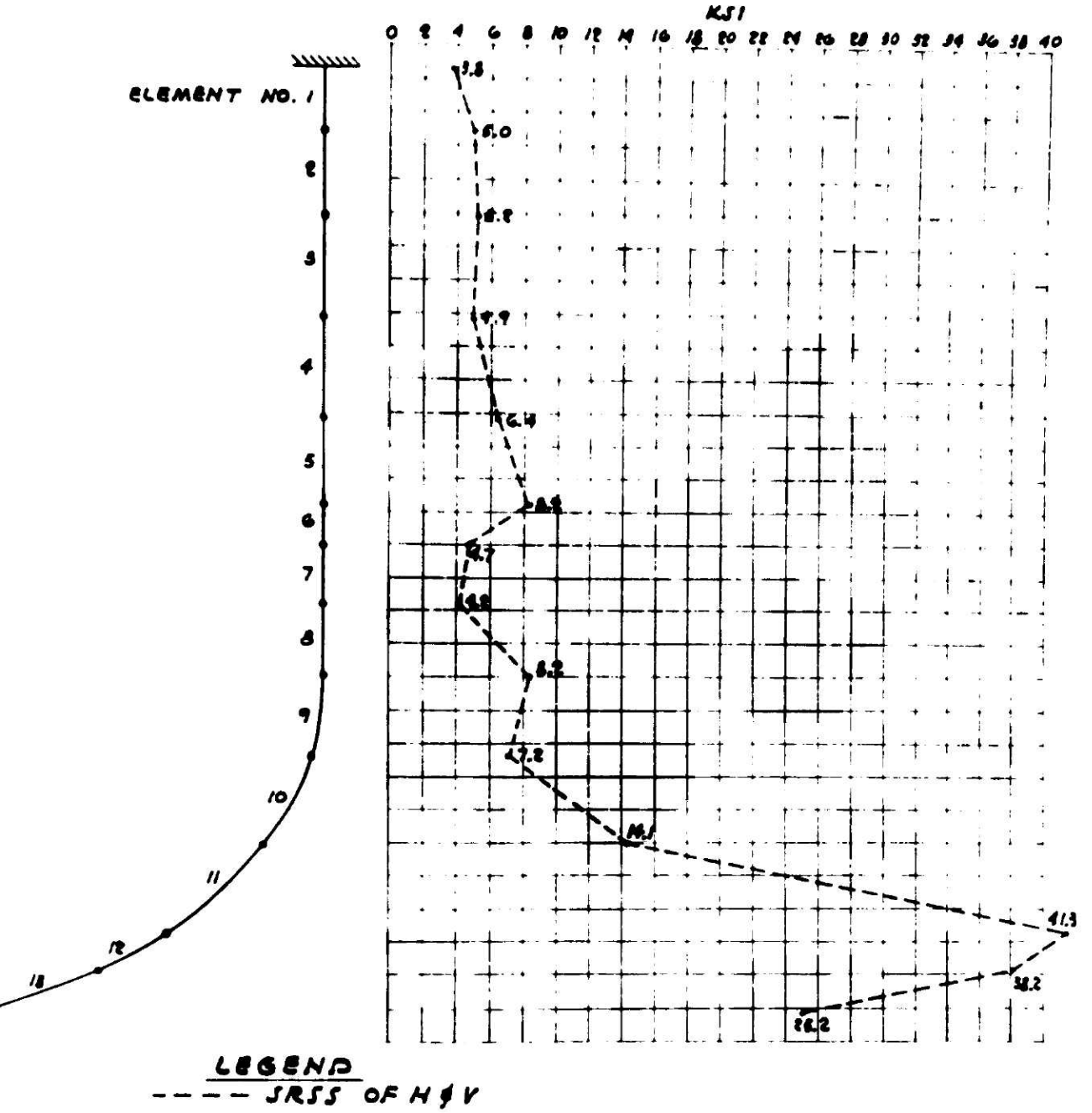

Fig. B.18. SuperPhenix, Model 1 SRSS Maximum Longitudinal Stress—Primary Tank

TABLE B.6. SuperPhenix Model No. 1 Maximum SRSS Displacements (Inches)

Component	Nodal Point	Horizontal R	Horizontal Z	Vertical R	Vertical Z	SRSS R	SRSS Z
Primary Tank	29	0.107	0.023	0.019	0.027	0.109	0.035
Primary Tank	46	0.210	0.212	0.130	0.486	0.247	0.530
Hot Center Pool	8	1.265	0.980	0.000	0.560	1.265	1.129
Hot Center Pool	14	0.781	0.856	0.111	0.562	0.789	1.024
Hot Center Pool	36	0.269	0.483	0.003	0.599	0.269	0.769
Divider	32	0.569	0.792	0.049	0.559	0.571	0.969
Core Support	35	0.270	0.500	0.003	0.760	0.270	0.910

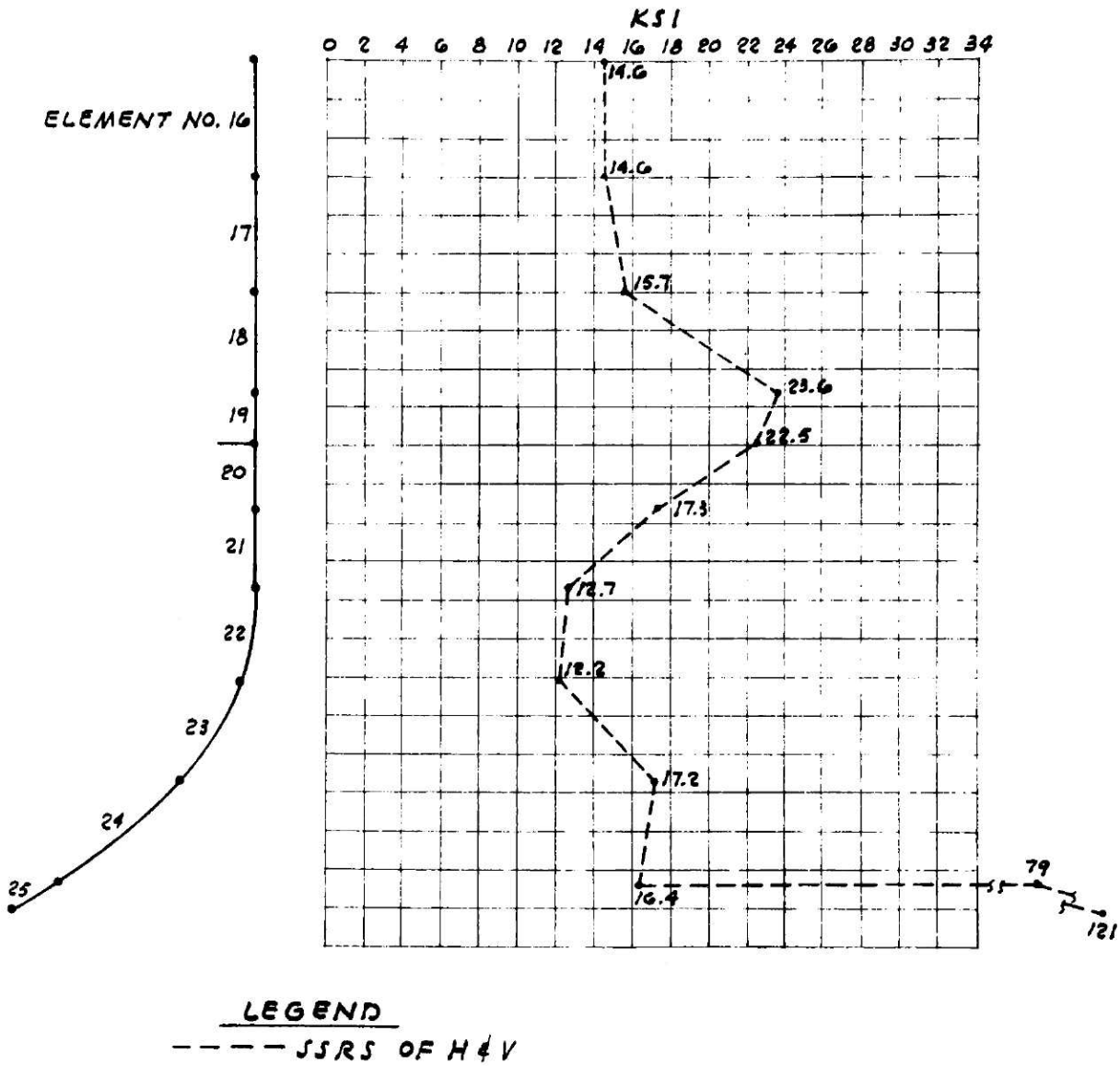

Fig. B.19. SuperPhenix, Model 1 SSRS Maximum Longitudinal Stress—Hot Center Pool and Divider

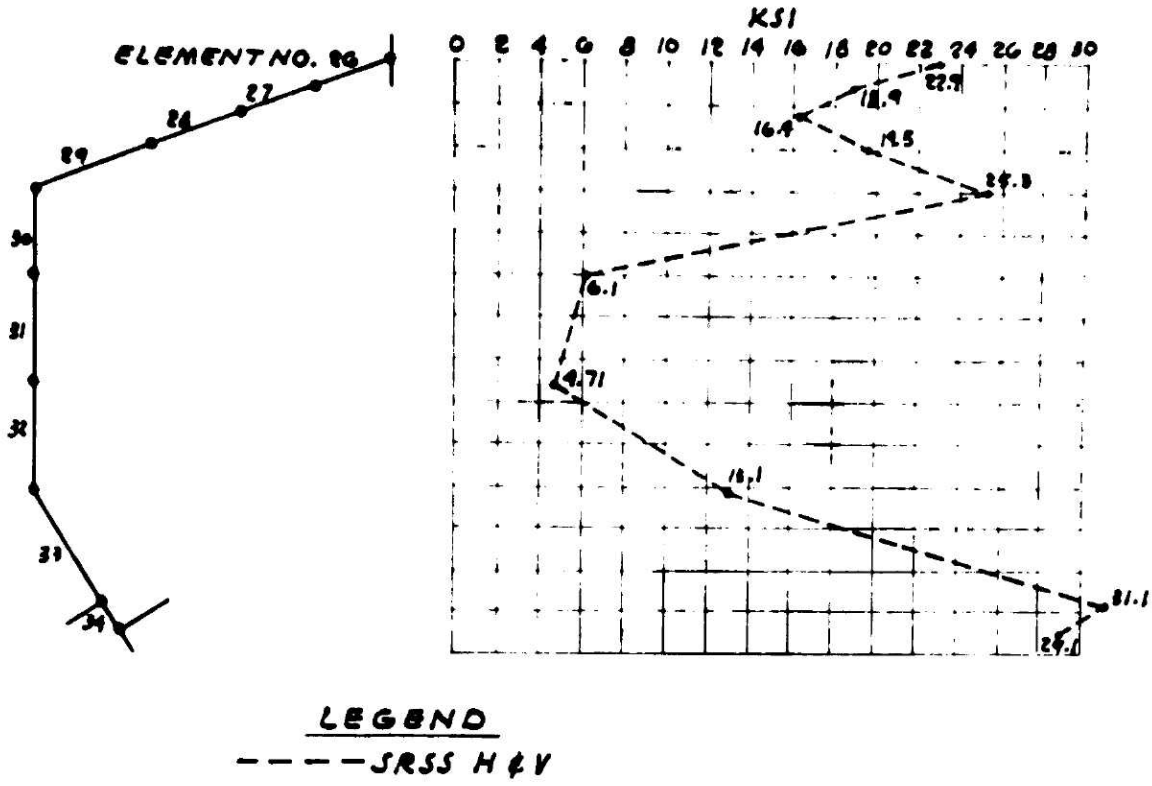

Fig. B.20. SuperPhenix, Model 1 SRSS Maximum Longitudinal Stress—Hot Center Pool and Core Support

Main Tank. In this analysis, the main tank plate thickness is 2 inches.

At the support skirt (element 1) the SRSS stresses are as follows:

$$\sigma_m = 3.8 \text{ ksi}$$

$$\sigma_t = 1.2 \text{ ksi}$$

$$\sigma_{mt} = 1.9 \text{ ksi}$$

σ_m for the cylindrical portion of the main tank is generally below 8.2 ksi, usually about 5.0 ksi. The stress increases dramatically near the discontinuities introduced by the connections between the core support structure and the primary tank.

The hoop stresses, σ_t are similar and generally somewhat lower than the longitudinal stresses.

The shear stresses, σ_{mt}, are below 2.7 ksi for the cylindrical section, increasing to a maximum of 5.4 ksi near the stress discontinuity of σ_m (nodal point 46).

In summary, the 2-inch-thick main tank appears adequate in the cylindrical section. However, the supporting structure for the core and the hot-center-pool needs to be redesigned (locally, probably) to lessen the present stress discontinuity introduced into the main tank wall as well as into the core support structure.

Hot-Center-Pool. The cylindrical upper portion of the hot center pool is 1 inch thick, and the divider below is 0.75 inch thick. The remaining pool walls are 1 inch thick. Figures B.18 and B.20 show the high stresses in large portions of the pool walls (only σ_m is shown). σ_m generally exceeds 10.0 ksi, often by factors of 2 to 3 near stress discontinuities at connections.

The hoop stresses, σ_t, at similar in magnitude to those for σ_m, whereas the shear stresses, σ_{mt}, are generally below 4.0 ksi except at the curved section of the divider, where they reach 7.4 ksi.

The hot center pool and the divider walls are highly stressed for the present plate thicknesses. Thicker walls would be required. The discontinuities between the cylindrical and the conical sections of the pool (joints 14 and 23) should be minimized to lower the locally high forces. If possible, the pool should be self-supported by a skirt from the shield deck--top supported, rather than bottom supported, as in the present analysis.

Connections Near the Core Supports. High stresses are present at the connections and supporting plates below the core, particularly at the connections between the core support structure, the main tank, and the hot-center-pool.

Guard Tank. The independently suspended guard tank is adequate. With a fundamental frequency of 21.3 Hz in the horizontal direction and 18.9 Hz in the vertical direction, the tank experiences maximum SRSS deflections of less than 0.027 and 0.011 inch in the horizontal and vertical directions.

The stresses are low, generally below 1.0 ksi, reaching a maximum of 2.3 ksi at the skirt.

(2) Model 2

As described in Sec. 4.b, SuperPhenix Model 2 was constructed to improve the seismic performance of the system. Figures B.21 to B.23 summarize the stresses for this alternative.

In summary, the stresses through the system are lowered significantly, usually by about 50%. Dramatic differences are indicated at the various higher stress areas of Model 1. This brief analysis indicates that a

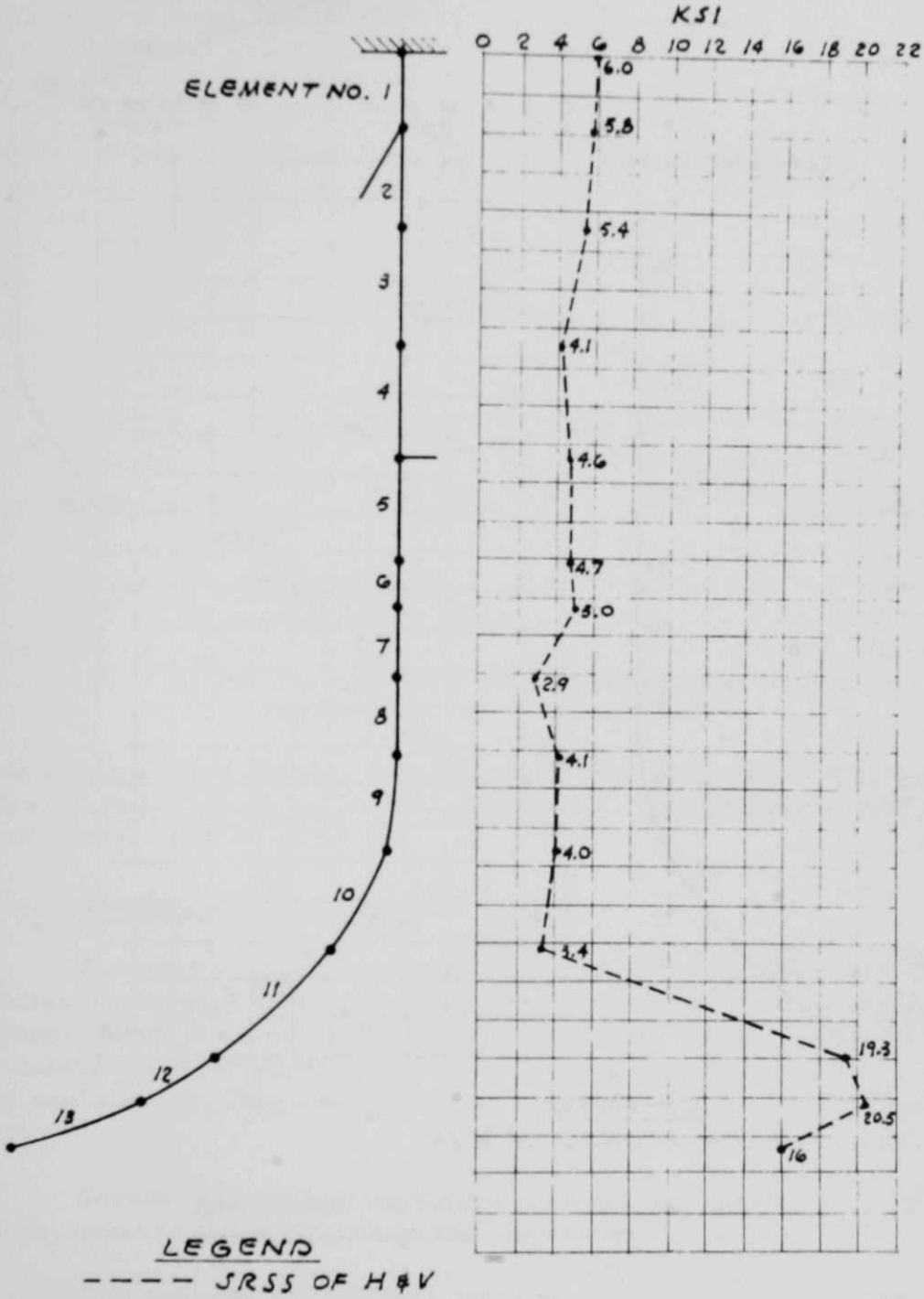

Fig. B.21. SuperPhenix, Model 2 SRSS Maximum Longitudinal Stress--Primary Tank

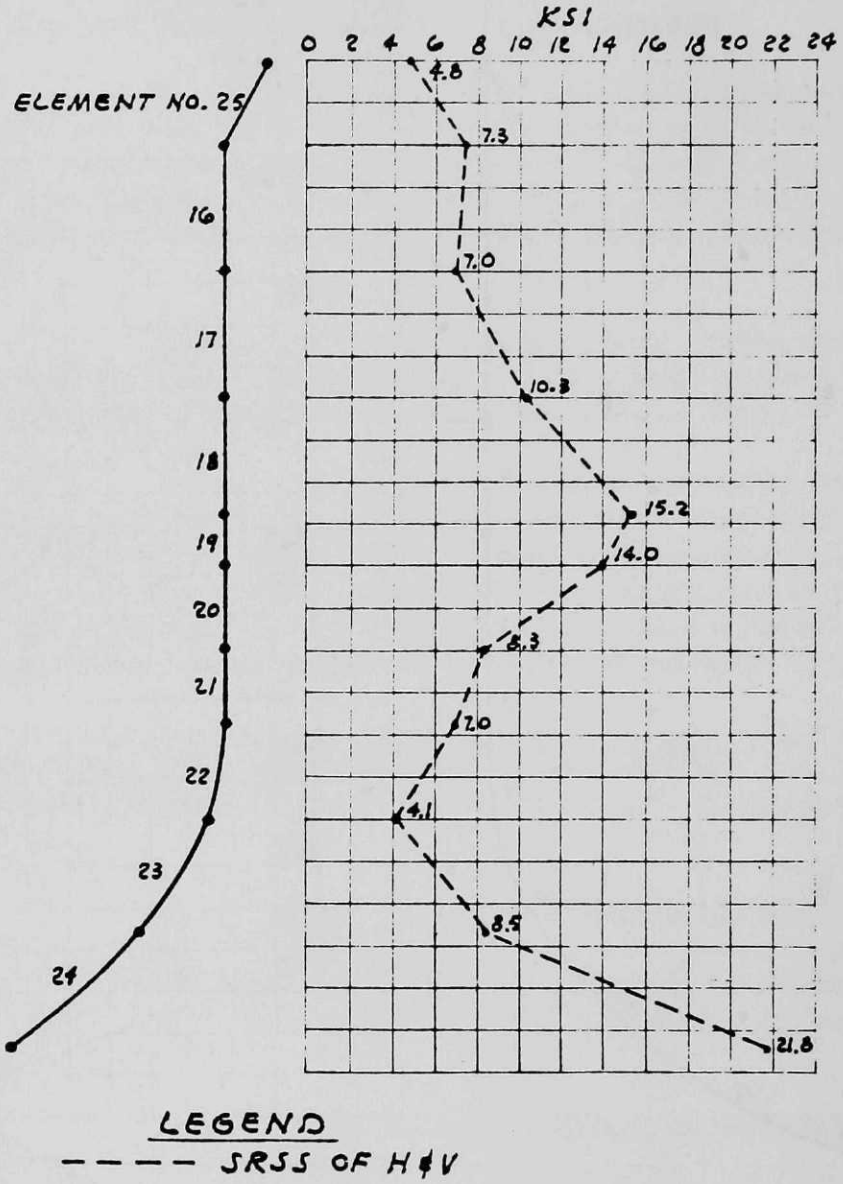

Fig. B.22. SuperPhenix, Model 2 SRSS Maximum Longitudinal Stress--Hot Center Pool and Divider

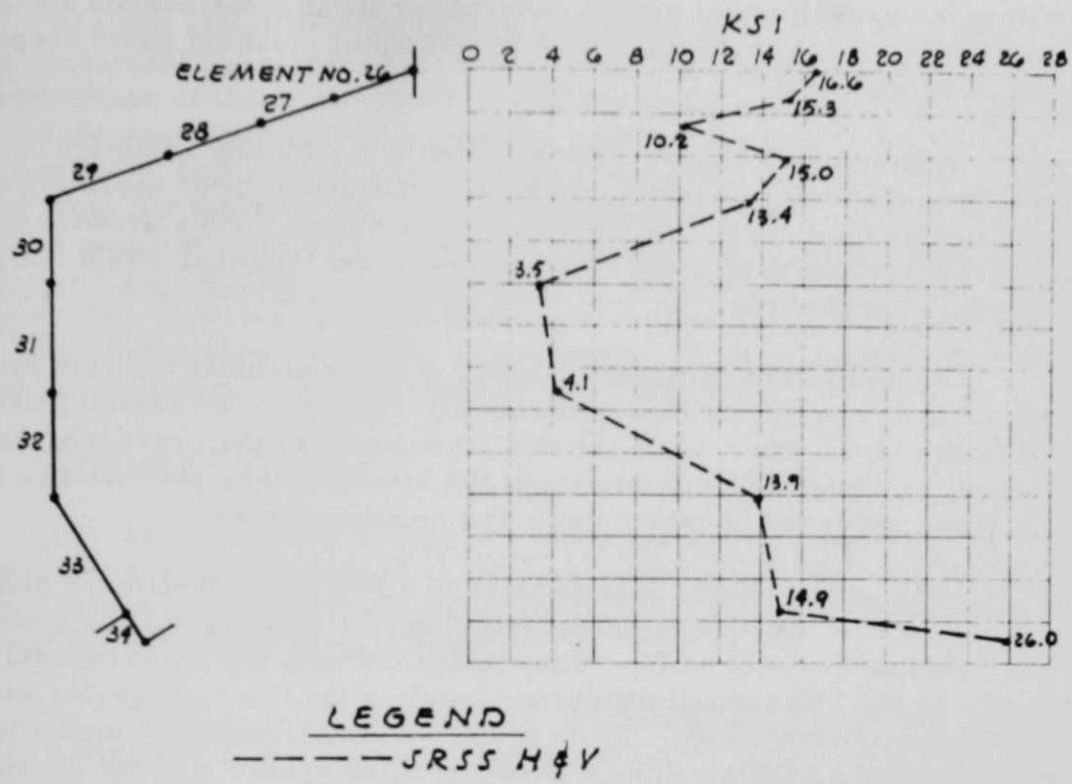

Fig. B.23. SuperPhenix, Model 2 SRSS Maximum Longitudinal Stress--Hot Center Pool and Core Support

separate suspension of the hot center pool would seem to make this system more desirable for the stated seismic conditions. Thickening of some plates would, of course, also be required.

c. Testing and Fluid Sloshing Results

A series of experiments were performed to measure the natural frequencies of sloshing with and without the structures and equipment within the Expanded EBR-II main tank (hereafter called the inner works). The inner works include the core support structure, the reactor core, the pumps, and the heat exchangers. The main tank was modeled as a flat-bottomed, circular cylinder.

Several experiments were carried out in different horizontal directions, because the inner works are not symmetric.

The average fundamental sloshing frequency with all inner works and for all horizontal directions was found to be 1.59 Hz.

The average fundamental sloshing frequency without the inner works (i.e., the primary tank full of sodium, without the inner works) was found to be 1.67 Hz. The difference between the two frequencies is expected

to be within the experimental error. Except for local disturbances about the pumps and the heat exchangers, the fundamental sloshing mode shapes appeared to be identical.

Higher mode shapes were not obtained with any accuracy or consistency. It is difficult to sustain any higher-frequency mode vibration without sophisticated equipment. Furthermore, the higher modes produce small surface irregularities within the central area of the tank that would not produce significant pressures.

From the experimental results, it was concluded that the presence of the inner works produces no significant alterations in the sloshing behavior of the primary tank. Therefore, for this investigation, the pressures due to sloshing can be computed separately and the stresses they produce can be added to those produced by other static and seismic forces.

The experimental results were verified using Housner's Sloshing Model analysis.[39-41] The fundamental frequency of sloshing for the model tank was computed to be 1.73 Hz. That compares with the experimental frequency of 1.67 Hz. The small difference between the two is probably within the experimental error range. Thus, it appears that Housner's model is sufficient for computing the sloshing frequency of the system and the approximate wave height.

The Housner model was then applied to the full-size primary system for the (Expanded) EBR-II concept. The results also apply to the simulated SuperPhenix primary system. The fundamental frequency of the primary system sloshing is: $f = 0.204$ Hz (i.e., a period of 4.9 seconds).

The maximum sloshing wave height based on Housner's model, at 0.6 g peak horizontal acceleration, is between 10.3 and 13.1 feet. Because the freeboard above the sodium level is about 4 to 5 feet, the main tank cover will deflect the wave. The pressures exerted by the wave increase the stresses in the main tank wall by approximately:

$$\sigma_m = 1.6 \text{ ksi}$$

$$\sigma_t = 1.8 \text{ ksi}$$

$$\sigma_{mt} = 0.0 \text{ ksi}$$

6. Conclusions and Recommendations

The general structural features of two concepts of pool-type LMFBR vessels and selected elements of their primary system were analyzed to determine their performance under significant seismic excitation. The analyses, preliminary in nature, were intended to generate information for further conceptual design improvements of the vessels and their interconnection and supports.

Various simplifications and approximations were made with respect to the structural characteristics of the various components of the system and the interaction of these components with the liquid sodium. The effects of some of the major equipment within the primary system were modeled very approximately, and other equipment and the various piping systems were excluded from the models. The following general conclusions were reached from the reported analyses. They are preliminary in nature, and considerable caution and engineering judgment should be exercised in their interpretation and use. The reader is advised to become thoroughly familiar with the seismic input assumptions, e.g., the nature of the assumed support motions and the major structural features adopted in the analytical models.

Expanded EBR-II

1. The computed displacements of the major structures and their components are small, less than one inch in the orthogonal directions. Thus, contact between the primary and the guard tanks during an earthquake is unlikely for the configuration studied.

2. The connection between the main tank and the primary tank cover causes a significant stress concentration at that location. Redesign of the connection (or stiffening of the cover) would significantly alleviate the stresses in the main tank. If the cover could be suspended independent of the main tank, one would expect significant reduction of these locally high stresses.

3. The core support structure liner plate thickness needs to be increased, especially at the edge connections. Rounding of the cone-cylinder connection might also alleviate the high stresses at that location.

SuperPhenix

1. The computed displacements of the major structures and their components are small, slightly more than one inch in the orthogonal directions. Thus, contact between the primary and the guard tanks during an earthquake is unlikely for the configuration studied. The various baffles between the hot center pool and the primary tank are specifically excluded from the analysis and would require further studies.

2. The stresses in the cylindrical section of the main tank are low; however, the tank walls are overstressed in the general vicinity of the core support connection because of local stress concentrations. This connection requires more detailed attention. The structural supports for the core support structure, and particularly the hot center pool, need to be evaluated in more detail.

3. The plate of the hot-center-pool walls probably need to be thickened. The discontinuity between the cylindrical and conical sections

should be minimized to decrease the local stress concentrations. If possible, this inner pool should be suspended separately from the shield deck.

General Conclusions

1. Sloshing of the liquid sodium at its free surface (for the SSE) appears sufficiently high to warrant consideration in the design of the tank and its cover.

2. The tests conducted on the Expanded EBR-II test model indicate that the presence of equipment inside the primary system does not significantly alter the fundamental frequency of sloshing.

Areas Suggested for Further Study

1. As is the case for LWR nuclear reactor plants, the effects of the site geological conditions and the structural configuration of the reactor building on the seismic response of the LMFBR vessels and primary system require detailed investigations. Parameter studies are needed to establish more accurately the floor response spectra at the supports of the primary system. These studies should account for possible variations in the site soil conditions (i.e., the presence or lack of soil-structure interaction) for possible embedment of the reactor building and should consider the structural characteristics of the reactor building itself. At a peak ground acceleration level of 0.3 g, the seismic forces appear to govern much of the structural design of the primary system. These forces are particularly sensitive to changes in the design spectra at the operating floor level.

2. An experimental program should be designed and carried out in conjunction with the detailed seismic analysis; scaled models of the primary system should be built, instrumented, and tested to verify the analytical results, particularly for fluid-structure interaction.

3. Fluid-structure interaction is an important aspect of the seismic response and requires a further and much closer examination.

4. The critical damping of the fluid-structure system requires further study; preferably the study should be supported by experimental evidence at appropriate acceleration levels.

5. The connections between the various tanks (guard tank, main tank, etc.) should be reviewed, and wherever possible, kept to a minimum or eliminated entirely to avoid local stress concentrations.

6. The effects of some of the equipment--neglected in the present analysis--including the various major piping systems that interconnect the larger equipment, should be included in any future, more detailed studies.

7. Other structural features may be vulnerable to seismic forces and need to be examined in detail. These include the various baffles, the core supports, etc.

8. More refined models of the core and of the major equipment should probably be included in any detailed analysis of the entire system.

9. The elastic and creep buckling behavior of compression members, such as the core-supporting members, under both static and dynamic loads, should be studied.

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