TID-26156

1000 MWe MOLTEN SALT BREEDER REACTOR CONCEPTUAL DESIGN STUDY

FINAL REPORT—TASK I

SUBCONTRACT NO. 3560
with
UNION CARBIDE CORPORATION
NUCLEAR DIVISION
OAK RIDGE NATIONAL LABORATORY
OAK RIDGE, TENNESSEE

under

U.S. ATOMIC ENERGY COMMISSION
CONTRACT NO. W.7405-eng-26

February 1972

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Chapter 1

Introduction

1.1 Origin of Industrial Interest in MSBRs

For a number of years Ebasco Services Incorporated (Ebasco) has had an interest in advancing the technology of molten salt reactor systems as a natural consequence of its continuing search for promising ways in which to serve its principal client, the Utility Industry. This Industry, in its service to the public, constantly seeks to produce reliably, and at lowest cost, the energy needed in the domestic and industrial sectors of the economy. The desire to conserve and, if possible, extend the energy resources of the world to reduce the environmental impact of energy generation, and to present the lowest hazard to the public are also strong motivations among the electric power companies.

In the summer of 1969 Ebasco announced the formation of a Molten Salt Group bringing together all major industrial capability needed to conceive, design, manufacture, and construct molten salt reactor systems for the utility industry and with utility companies participating financially. The utility participants are:

Dallas Power & Light Company
Houston Lighting & Power Company
Kansas Gas and Electric Company
Middle South Services, Inc.
Minnesota Power & Light Company
Northeast Utilities Service Company
Texas Electric Service Company
Texas Power & Light Company
Union Electric Company

Industrial participants in that group are:

Ebasco Services Incorporated Management, technical direction, nuclear design, power plant technology.

Babcock & Wilcox Reactor vessel, primary heat exchangers, general reactor technology, steam generators.

Continental Oil Co., Inc. Chemical processing, chemical engineering.

Union Carbide Graphite technology.

Cabot Corporation Hastelloy-N, special metal alloys, materials technology.

Byron-Jackson Fused salt pumps, general pump technology.

The industrial companies supplied senior technical personnel at their own expense to work as a team under the management and technical direction of Ebasco to evaluate Molten Salt Reactor Technology.

On September 30, 1970, Union Carbide Corporation, Nuclear Division, operators of the Oak Ridge National Laboratory for the USAEC, issued a request for a proposal for an independent Molten Salt Breeder Reactor Design Study. Ebasco and its group of industrial companies responded with a proposal which was accepted by the Union Carbide Corporation and the USAEC with Ebasco Services Incorporated as principal subcontractor to Union Carbide Corporation and with the Molten Salt Group members as sub-subcontractors to Ebasco. The official commencement date of this contract was March 8, 1971, and is expected to run 30 months.

1.2 Project Organization

The Molten Salt 1000 MWe Breeder Reactor Conceptual Design Study is under the Corporate Cognizance, in Ebasco, of L. F. C. Reichle, Vice President - Nuclear. The Nuclear Division (Figure 1.1) is responsible for the MSBR design study. This study is under the technical direction of D. R. deBoisblanc, Ebasco's Chief Nuclear Consulting Engineer and Manager of Research and Development. Figure 1.2 shows the organization including the sub-subcontractors.

The organization chart (Figure 1.2) shows five major divisions: Systems and Components, Technical, Reactor Engineering, Plant Design, and Instrumentation and Control. The Systems and Components Group is responsible for the conceptual design work on the major functional components such as heat exchangers, reactor vessel, pumps, etc., in the primary salt system and also for the development of the flow sheets and conceptual design detail for the various subsystems.

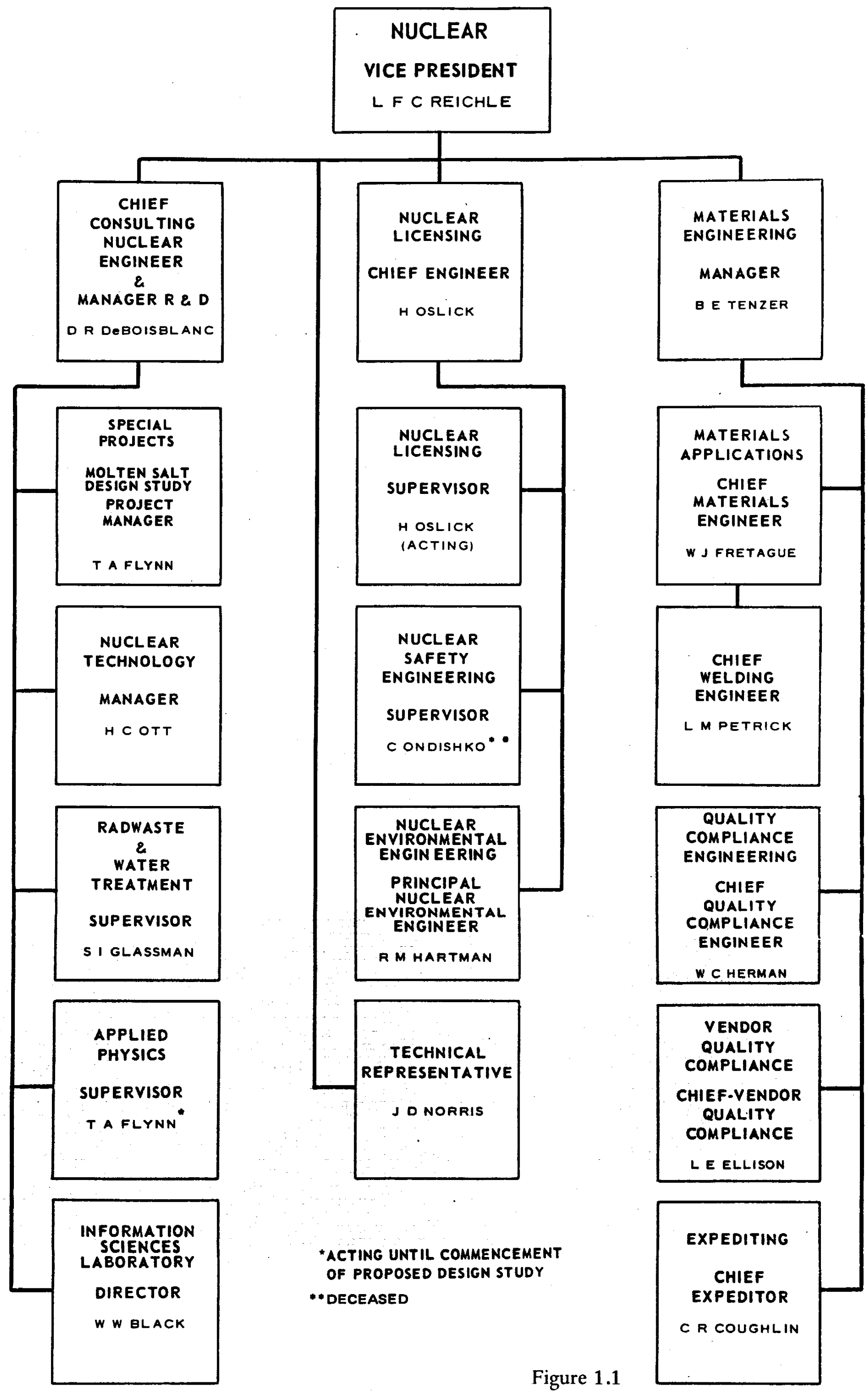


Figure 1.1
Ebasco Nuclear Organization

Figure 1.1: Ebasco Nuclear Organization.

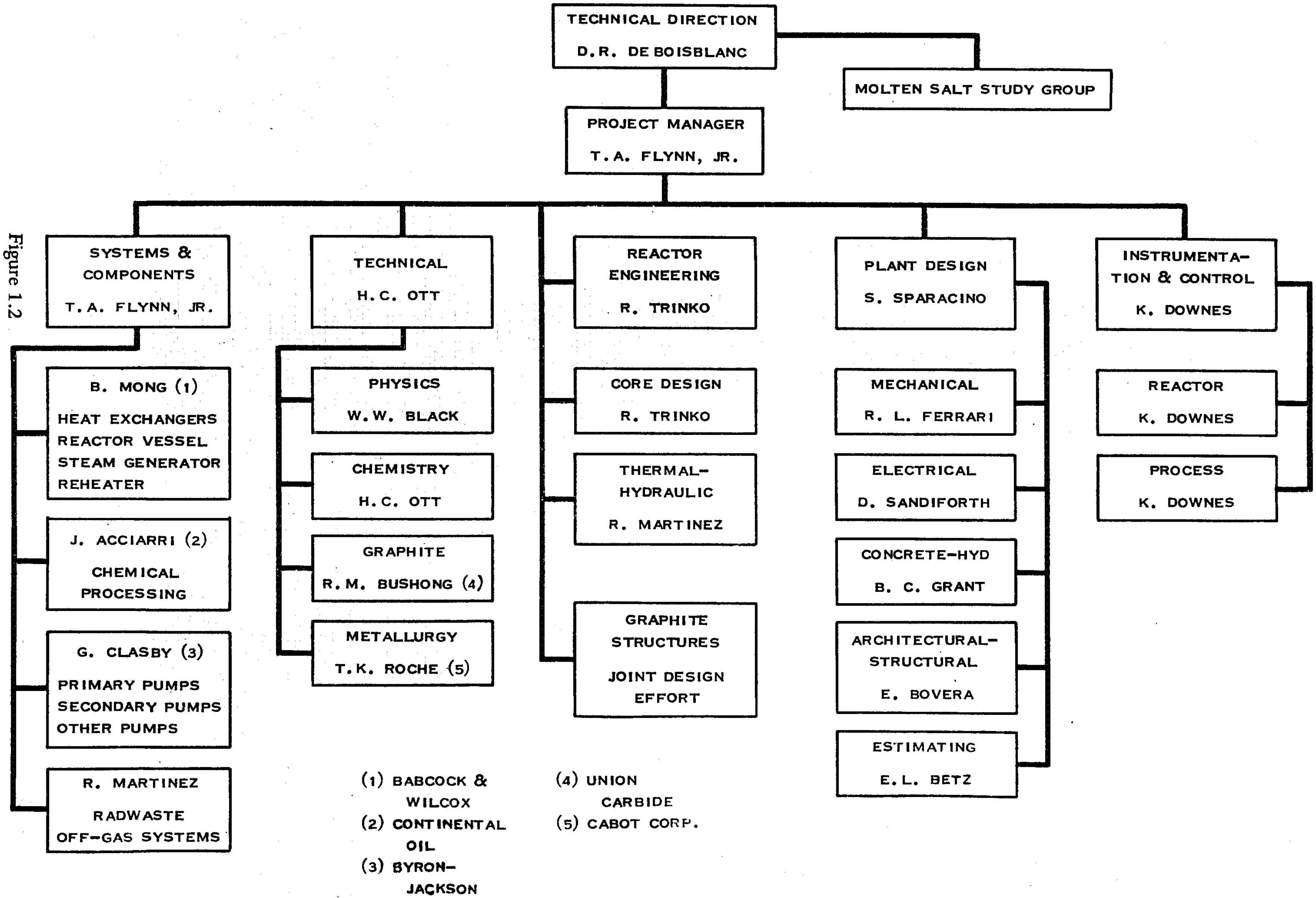


Figure 1.2
MSBR Study Project Organization

Figure 1.2: MSBR Study Project Organization.

The Technical Group provides the special consulting and conceptual input in the areas of physics, chemistry, metallurgy, and graphite technology.

The Reactor Engineering Group is responsible for the overall nuclear engineering design of the reactor including the reactor physics, thermal-hydraulics and the specifications of the geometry of the graphite structures.

The Plant Design Group brings in the traditional power plant design disciplines of mechanical engineering, electrical engineering, concrete-hydraulics engineering, architectural and structural engineering and estimating.

The Instrumentation and Control Group will be responsible for the conceptual design of all instrumentation systems for the reactor and process systems in the plant. These groups report to the Project Manager who in turn reports to the Technical Director.

Permission was obtained from the utility sponsors of the Molten Salt Group for a continued activity of this Group until utility funds were expended. In addition, the Technical Representatives of the industrial members of the group serve as an informal advisory panel from time to time during the course of the design study. The availability of this additional MSR effort within Ebasco and this advisory panel greatly enhances the overall management and technical direction of the project so as to maintain it within a commercial framework which is one of the main goals of the design study.

The sub-subcontractors participating in the design study with Ebasco provide design effort for a substantial portion of the overall MSBR Plant. Babcock & Wilcox provides the design of heat exchangers, steam generators, and the reactor vessel. Babcock & Wilcox has been active in the atomic energy business supplying these components to the Atomic Energy Commission and to the Nuclear Navy since 1944 when they manufactured equipment for the Manhattan Project. In addition, the company is currently designing and fabricating 13 nuclear steam supply systems for the nuclear industry.

Continental Oil Company, Incorporated is engaged in the design of the fuel salt chemical processing system. CONOCO is among the world's ten largest energy companies selling over S2.5 billion worth of goods every year. CONOCO is involved in all forms of the energy business; petroleum, coal, nuclear, chemical, and plant foods.

Stellite Division, Cabot Corporation, provides assistance in the design of advance Hastelloy materials. Cabot Corporation is a diversified producer of performance chemicals, energy, and engineered products. Its Stellite Division produces many types of high quality alloys developed to resist different conditions of wear, heat, and corrosion. They maintain a staff of fully qualified chemical and metallurgical engineers.

Union Carbide Corporation, Carbon Products Division, provides assistance in the design of graphite core structures and other graphite components. The Carbon Products Division has a 75-year background in the manufacture of carbon, graphite, and related ceramic, metal, and composite materials products. Carbon Products Division is the only producer in the industry fabricating all varieties of carbon and graphite products. Its production plants are supported by a completely integral technical center at Parma, Ohio.

Byron-Jackson, Division of Borg-Warner, provides assistance in the design of salt-circulating pumps. Byron Jackson is the leader in supplying pumps for the LMFBR Sodium Reactor Program, and a leader in supplying pumps to the water-cooled nuclear industry. Byron Jackson conducts research on pump technology.

Chapter 2

Purpose and Scope of Task 1

2.1 Purpose

Task 1 had three main purposes:

a) to generate the bases for selection of an MSBR reference concept,

b) to develop an MSBR plant concept from these bases and the criteria specified in the contract,

c) to identify other concepts meriting further study.

The reference concept developed in this study is one which the industrial team can recommend to utilities for future construction. The plant should be licensable and operable on a utility system with safe, economic, and reliable performance. This concept assumes accepting advances in technology anticipated over the next fifteen years.

2.2 Scope

Emphasis of this study was limited to:

a) systems or components vital to MSBR success even though their cost may not be significant, and

b) systems or components having high cost even though technical feasibility may be reasonably well assured.

Based on these considerations, efforts were limited to the primary system (including reactor and vessel), secondary system, steam system, off-gas system, fuel-salt drain tank system, building and general arrangements, and design of a chemical process plant based on the ORNL flow sheet.

Several graphite element designs and methods of replacement were examined. Several reactor vessel concepts were examined in some detail. Several primary system piping arrangements were laid out and stress analyzed. Several component arrangements were examined. Design concepts for steam generators, reheaters, and primary heat exchangers were developed. A supercritical steam system with 700F feedwater was analyzed and designed. A layout of buildings and structures was developed and seismically evaluated. The performance criteria of the off-gas system were re-examined. An alternate concept was proposed, and an engineering design of the ORNL concept was developed. The electrical distribution system was analyzed. Several fuel-salt drain tank systems were explored.

The initial effort in Task 1 was to establish what each system could and should do as an integral part of the power plant. After establishing criteria (some of which were borrowed from the Molten Salt Group Reports), we attempted to define a conceptual design which would satisfy the criteria. In some cases, our design reached a high degree of completion, e.g., the moderator element and the steam system. In other cases, this report represents several alternates which must be evaluated in Task 2. In all cases, however, the alternates are such that the overall concept resulting from the combination of each system is a total power plant concept.

Chapter 3

Reactor System

The reactor system is composed of the reactor vessel, the reactor control rods and drives, the reactor internals, the circulating fuel salt, the fuel salt pump, the intermediate heat exchanger, the fuel salt drain tank, and the piping system connecting these components. Power is produced in the reactor when the fuel salt, 7LiF-BeF_2-ThF4-UF_4 (71.7-16-12-0.3 mole percent) is made critical by the moderating properties of graphite structures through which it flows. The fuel salt is composed of low neutron absorption cross section materials blended to produce a liquid having fluid properties suitable for developing a breeder reactor, a high temperature reactor, and a reactor with inherent safety characteristics. The fuel salt operates above its liquidus temperature of 930F. The nominal reactor inlet temperature of 1050F provides sufficient margin to avoid concern over freezing of the fuel salt in the system. The salt is quite viscous and so the highest ΔT is established at 250F, and the reactor outlet temperature is 1300F, the upper limit being based on the available mechanical properties of the piping systems as a function of temperature and by a contractual requirement for Task I.

The pump cooling consideration results from the lined pipe concept in that the bypass flow that cools the vessels experiences a rise of 50F to 1100F at the top of the core. This stream flows in the outlet pipe liner at a rate of 30.5 gpm. The flow in the liner rises 50F in passing through the outlet pipe and up around the pump bowl. This allows us to maintain the certerline injection temperature in the pump at 1150F (as in the ORNL 4541 pump design).

The combination of the high operating temperatures and the extreme corrosiveness of the molten fluoride fuel salt severely restrict the materials which may be used in the system. The high moderating power of carbon (which is determined by the low neutron absorption cross section of carbon coupled with its low atomic weight which renders neutron scattering collisions efficient in slowing down neutrons) and the excellent high temperature structural properties of graphite make this material uniquely suitable for the reactor internal structures. The use of graphite in nuclear reactors has a history extending from the same original point as that of uranium, the Stagg Field pile. The behavior of graphite under radiation is complex in that normal graphite materials behave differently along their crystallographic axes, i.e., they are anisotropic under radiation. This behavior is even more complex from a mechanical design standpoint in that a cross section first exhibits a linear shrinkage after exposure to a fluence of 1.5 $\times 10^{22}$ nvt and then reverses the direction of change and swells. The studies at ORNL of graphite's radiation induced properties have led to a definition that the useful core lifetime of graphite is reached when the regrowth of the originally contracting material equals the original dimension. For graphites having suitable properties for MSBR applications this occurs at 3 $\times 10^{22}$ nvt at a temperature of 700C or about 4 years of operation at an 80 percent plant factor for the physics conditions leading to satisfactory breeding. Our design study has initially accepted the ORNL physics design of the reactor (ORNL 4541) as a basic assumption for Task I. We plan to run benchmark and survey calculations on physics uses in mid-FY72.

A family of high nickel alloys, known as Hastelloys, has been developed which are capable of containing the fluoride salts under pressure. Hastelloy was successfully employed as the structural material for the MSRE. The high temperature radiation endurance of this material has been established as a ground rule for the Task I design study. Curves giving design stresses as a function of temperature have been supplied for Task I use. Modified Hastelloy-N has been defined as having a value of Sm of 3500 psi at 1300°F. Since 3Sm is defined by the nuclear power piping code as the maximum value of primary and secondary stress acceptable for conservative design, the design study was originally restricted in the hot leg portion of the plant to stress values of 10,500 psi for the combined sum of pressure, weight, seismic and thermal forces. This restriction has had a very pronounced impact on the mechanical and structural design of the reactor vessel, the fuel salt pump and the hot leg piping system. It led to the Task I design effort to develop a lined and/or jacketed primary system. The curve supplied by ORNL for Sm as a function of temperature shows a very sharp rise in strength with decreasing temperature, e.g., Sm = 13,000 psig at 1100F. Accordingly, the premium associated with reducing the temperature of the pressure boundary is quite high. We found the thermal shock problem in the reactor outlet line so severe that we concluded a bare system in this leg was only marginally feasible. Surprisingly, the reactor inlet leg also required a protective measure when we considered the thermal shock associated with loss of secondary cooling pump power. The temperature in the inlet leg rose rapidly to 1300F with a loss of strength and severe shock occurring. We decided to avoid detailed design effort on this case by adopting a lined (or jacketed) inlet line.

A lined system is defined as one in which the pressure boundary of the fluid system is insulated from rapid changes in temperature by either a stagnant or laminar region of the contained fluid. A jacketed system is defined as one in which the pressure boundary of the fluid system is separated from the contained fluid by a second fluid. The second fluid (different from, but compatible with, the fuel salt) is maintained at a higher pressure than the fuel salt so that leakage always occurs from the jacket fluid into the fuel salt across the metallic "O" Ring seals.

The entire primary system which contacts the fuel salt is made of either Hastelloy or graphite. The use of Hastelloy is dependent on its corrosion resistance and the ability of the metal to maintain its ductility under neutron bombardment. All Hastelloys contain some boron as an impurity. The boron tends to precipitate at the grain boundaries. Under neutron irradiation, helium gas, via the ^{10}B$(n, \alpha)^7$Li reaction, is produced near and generates bubbles at the grain boundaries. The presence of bubbles in the grain boundaries is the principle cause of the loss of ductility in standard

Hastelloy-N. To overcome this problem ORNL has developed modified alloys containing additives (both titanium and hafnium appear promising) that have greatly improved rupture life and ductility under irradiation. This material is assumed to have adequate ductility (defined as 4 percent minimum strain to rupture) as a groundrule for Task I. Grain boundary attack was observed in the MSRE. Its cause has not yet been firmly established. Oak Ridge is actively investigating this intergranular attack. As a Task I groundrule, we assume that both the corrosion problem and the embrittlement problem have been solved.

In the following section we shall describe the reasoning which led to our Task I conclusions. The first consideration shall be the design of the graphite moderator, reflector and internal structure of the reactor.

3.1 Graphite Moderator Element Design

The fabrication techniques and the properties of graphite for high temperature radiation service are generally well understood. The excellent service of the graphite for the life of the MSRE has led to confidence in the ability to design these elements. The condition of breeding imposed on the MSBR design study, however, may make it necessary to have a graphite that will not absorb xenon gas from the fuel salt stream as it flows over the graphite surfaces. A contractual ground rule for the design study specified that the xenon poison fraction be no greater than 0.005 neutrons lost to xenon per neutron absorbed in fissile material (0.5 percent poison fraction).This ground rule has been adopted as a criterion for the Task I study effort. There are a number of ways of reducing the loss of neutrons to xenon. We have studied the bubble generator system proposed by ORNL 4541 and, as an alternate, have investigated a spray system. The results obtained in the salt simulation loop at ORNL seem to indicate the pump's action on the bubble rich fluid breaks the bubbles into a very much smaller size range than anticipated. The effect on mass transfer of the smaller bubble size has been calculated by several models and it appears to enhance xenon removal due primarily to increased surface area. The spray chamber appears interesting because the ratio of gas to fluid for a given surface is very much greater than in the bubble method. The high gas to liquid ratio of the spray chamber appears to offer potential for removal of tritium as well as of xenon.

The penetration of xenon into the graphite can be prevented by the application of a pyrolytic carbon or graphite coating on the exposed surfaces of the graphite element. The technology of applying exterior pyrolytic coatings to graphite is available today. The technology of applying pyrolytic coatings to interior surfaces in long pieces of graphite is less well developed. It is reasonable to expect that coating materials that will successfully seal graphite against xenon intrusion will be developed within the 15 year projection of the design study. Demonstration of the ability of exterior coatings to seal against gas penetration for the 3×10^{22} nvt radiation lifetime of the graphite can be performed by test irradiations of relatively small graphite structures. Demonstrations of the seal for interior coatings can only be performed by tests that utilize long pieces. No tests of this nature are currently planned. The integrity of the coating under thermal cycling and mechanical stresses in full sized pieces will not be demonstrated by extrapolating tests on small pieces. The full verification of large scale pyrolytic coating under neutron irradiation will require a considerable extension of the current testing programs which are not currently planned. Thus, we establish for Task I the following criteria for graphite element design:

1) Pyrolytic sealing materials are available.

2) Exterior coatings that will resist handling friction, stress, vibration and temperature changes are available.

3) Interior pyrolytic coatings should be avoided if possible.

3.1.1 Mechanical Design

The Molten Salt Group critique of the ORNL conceptual design led to several serious questions with respect to the mechanical design presented in ORNL 4541. These objections can be briefly summarized here as follows:

1) The practicality of lifting a large weight remotely under conditions of very tight clearance on a nonroutine basis (i.e., every 4 years).

2) The desirability of using an unshielded machine with the risk of a cask malfunction causing severe problems in recovery.

3) Graphite elements positioned outside of the central core zone achieve only a small fraction of their allowable service life.

4) The feasibility of applying and inspecting a high integrity pyrolytic graphite coating in small diameter holes running through long graphite elements may be beyond a reasonable extension of technology.

Based on this critique, we established functional objectives for the design of a moderator element:

1) Moderator elements shall have only external surfaces.

2) The maximum heat path from the interior of the log to the salt should not exceed 0.7 inches. This results in a temperature-damage relationship at least equal to the ORNL reference design element.

3) The salt/graphite ratios for each core zone shall be maintained at the ORNL conceptual design values to preserve the physics model to the greatest extent practical. (Task I ground rule established by Ebasco.)

4) The elements shall be sized through a trade-off between ease of handling and the number of moves required to service a core. It was decided not to reinsert underexposed elements because that would require, 1) reorificing highly contaminated assemblies, 2) possibility of seal

damage and 3) variations in coefficient of friction during service life. While a hexagonal geometry was selected (due to its self-standing characteristics), the results of this trade-off apply approximately to other geometries.

5) Raw graphite pieces shall not exceed the current maximum dimensions for special graphite production. Manufacturing techniques are available that can produce acceptable graphite uniformity and density if the distance from any point of the section to the atmosphere of the graphitizing furnace does not exceed 9 inches. For cylindrical shapes this implies a maximum diameter of 18 inches. The lengths available are unlimited for our purposes.

The geometries examined in the design study are presented in paragraphs 3.1.2 to 3.1.5. Concurrently with this geometric study, a unit element size selection study was conducted. The results of the sizing study are presented in paragraphs 3.1.6 to 3.1.7.

3.1.2 Arrays of Cylinders

Cylinders were considered as the elements for the core pieces because of ease of fabrication and the uniformity of heat removal across a section of the rods. The diameter of the rods would have to remain small, 1-3/8 in. or less, in order to keep the graphite lifetime in the range required by the contractual ground rules for the reference design. The large number of these relatively long and slender rods does not appear attractive from the standpoint of handling and fragility. It would be possible, of course, to use a central hole (1 in. ID x 3 in. OD) and have both ease of handling and long lifetime, but the sealing problem is not much different from the ORNL design.

Control of the salt fraction and improvement of the salt flow in the space between stacked rods[1] could be achieved through use of raised helical rib, say 1/32 in. high machined onto the surface of the graphite.

Unitizing of bundles of rods could be done by banding together hexagonal arrays of rods at the tops and bottoms of their lengths with Hastelloy-N bands or with graphite. In the case of the Hastelloy bands the coefficient of thermal expansion differential would allow loosening of the bundle at operating temperature which would be undesirable. In addition, there is some question whether the bundle could be held tightly enough to permit handling as a unit without additional pinning or other means of fastening at the ends.

A possible unit considered consists of a hexagonal array of rods contained in a hexagonal graphite box. Commercial production of graphite pipe in 10-12 in. diameter is state of the art, and box-like extrusions in the same size range have also been produced. In this configuration control of salt flow could be accomplished by cutaway portions of the rods and box in the plenum regions.A cross section of a core unit of this nature is shown in Figure 3.1.

[1]Experimental Investigation of Velocity Distribution and Flow Resistance in a Triangular Array of Parallel Rods, W. Eller and R. Wigsing (EURATOM) Nuclear Engineering and Design 5, 1967, North Holland Pub. Co., Amsterdam.

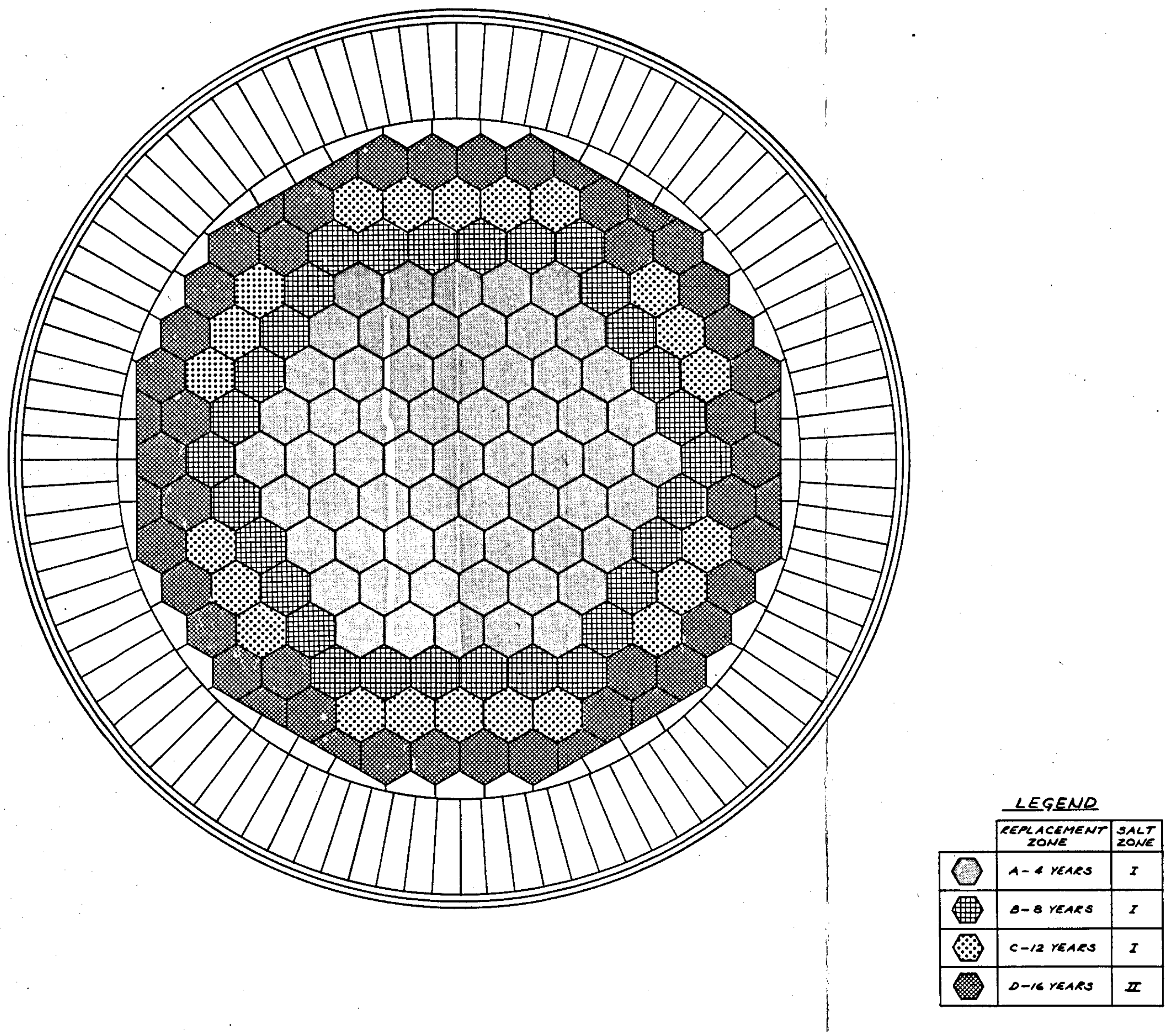


Figure 3.1: Reactor Cross Section.

3.1.3 Solid Blocks

The use of machined solid graphite blocks was considered as an obvious method of unitizing core sections into convenient size and weight units for replacement. For purposes of this study it was assumed that molded blocks or hexagonals in sizes of about 18 in. across flats by about 36 in. length could probably be produced in an MSBR grade, and that extrusions 18 in. diameter by 15 ft long in MSBR grade were probably not feasible. As a first approach to this design, a hexagonal block 9 in. across the faces with relatively large holes for salt flow was considered (a sketch of this unit is shown in Figure 3.2). In this sketch the 13 volume percent salt was obtained by using 19 holes approximately 1-5/16 in. in diameter plus a spacing of 1/4 in. between adjacent hexagonal blocks. It is shown that much smaller holes or slots would be necessary in order to avoid large distances for heat transfer from the inner graphite to salt (and hence high graphite temperatures and reduced life). It is thus shown that difficult and costly machining would be involved and that sealing of the large number of small interior surfaces would be much more troublesome than in the reference design.

In addition, the necessity of several vertical segments to make up the core height entails careful indexing at each joint and throughout the graphite machining operations to insure continuity of the salt flow and poses a serious design problem to avoid stagnant salt or ill defined salt flow regions at the joints.

The size effect on radiation damage is a further drawback. Although the incremental dimensional changes, creep effects, etc., could possibly be kept in the same level, the leverage of size is certain to cause breakage at some point in radiation dosage. This in itself would not necessarily be catastrophic, but would complicate the replacement process which we are trying to improve.

For these reasons the use of machined solid blocks in the core was considered undesirable.

3.1.4 Iris Array of Curved Slabs

It would be desirable from the radiation damage (and power generation) standpoint to have all graphite receive uniform dosage within any individual graphite unit. This is, in practice, impossible, but an array of concentric cylindrical units built up of curved slabs in an iris pattern was considered which appears to have some desirable features from the symmetrical uniformity of radiation loading. A sketch is shown in Figure 3.3 which depicts a section through two units of an iris array of curved slabs which would be unitized by attachment to graphite plates at the top and bottom. This configuration presents a gradually increasing thickness of graphite (and salt channel) as the radiation intensity falls off and thus a leveling of graphite temperature and lifetime. Hydraulically, this configuration requires further design enhancement before becoming acceptable.

The curved slabs would be considerably more difficult to manufacture than straight sided slabs but would be feasible to produce. Replacement of the central one or two cylindrical units would appear

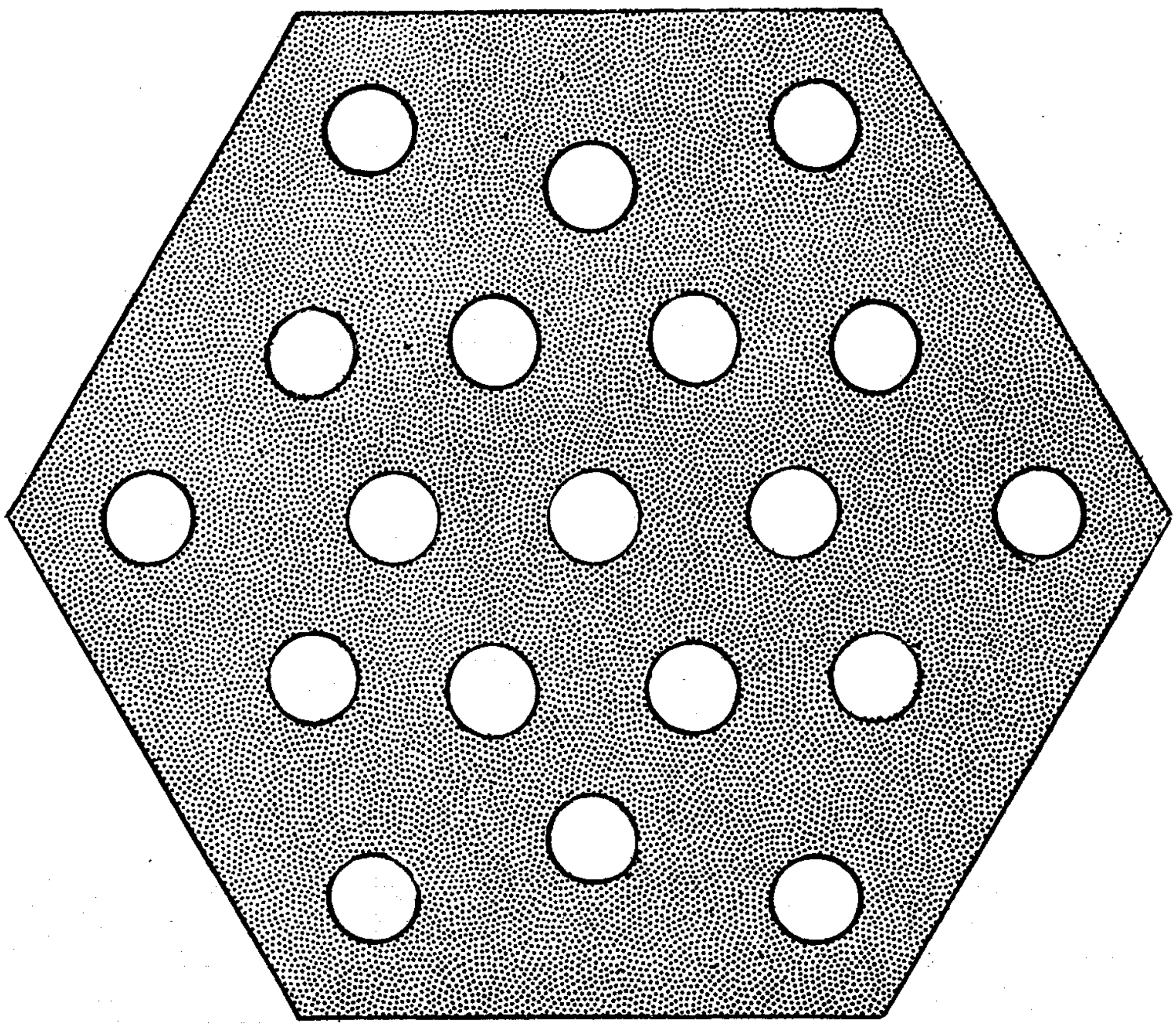


Figure 3.2: Solid Block Graphite Element.

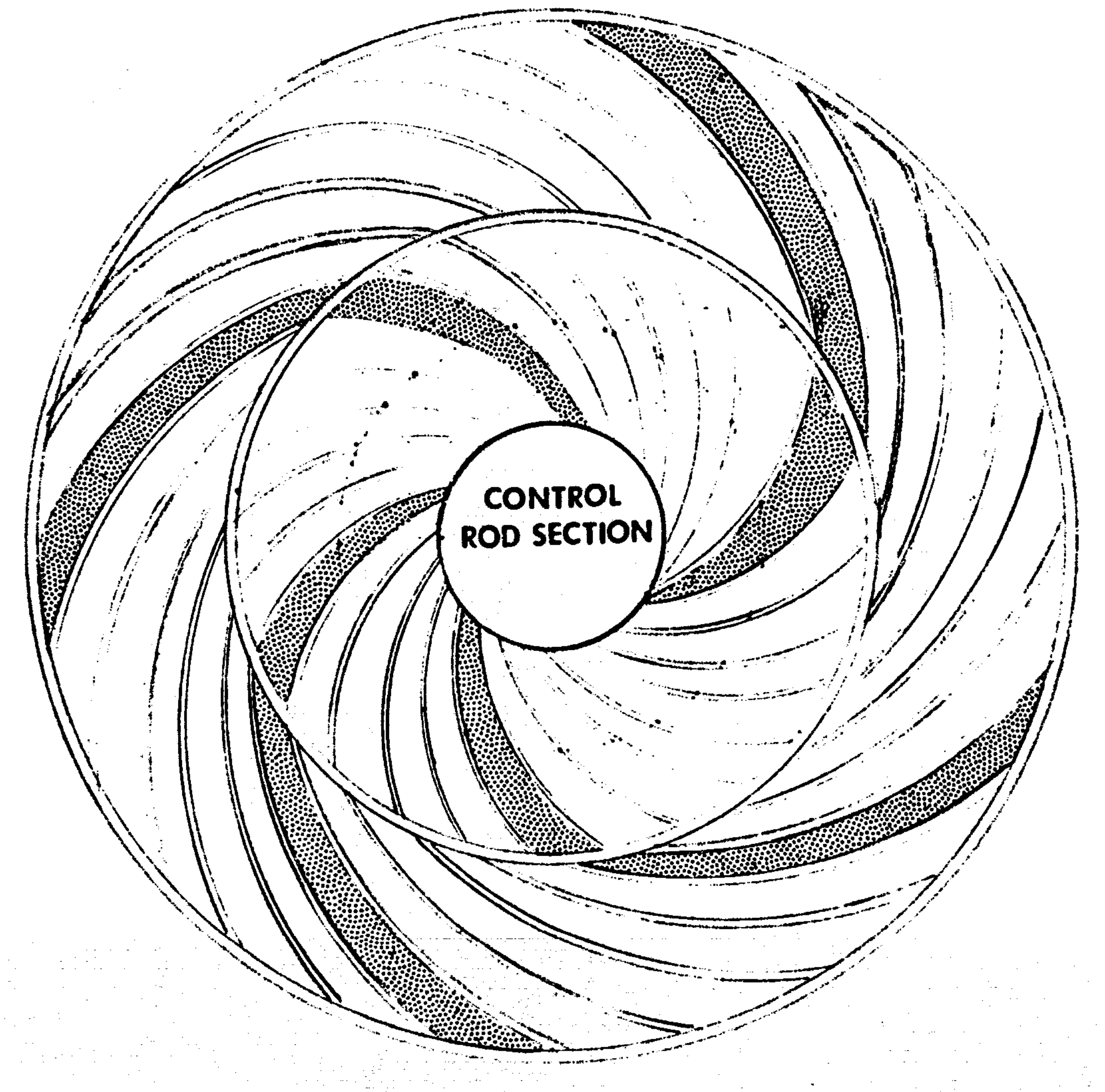


Figure 3.3: Graphite Elements Iris Array.

to be feasible but the outer rows would have to be handled as segmented rings and replacement of odd shaped and sized units is unattractive.

3.1.5 Slab Arrays

The important consideration of short heat flow paths from all areas of graphite in the core is accommodated very well by use of relatively thin (up to perhaps 2 in. thick) slabs or "boards" of graphite separated by salt flow channels. The manufacturing technology of graphite plates is well established and needs only to be extended to MSBR grades. Sizing considerations need only be consistent with reactor requirements and ease of handling (very thin plates would be too fragile). A maximum size of plate for study purposes was taken as 2 in. x 12 in. in cross section by full reactor height (approximately ~15 ft).

1) Square
   The simplest array considered was a square arrangement of plates sized to the central control section (sketch shown in Figure 3.4). This array would not be very symmetrical in a circular cross section but might be entirely practical. The square cross section would require some holding fixtures during replacement to maintain stability of the remaining core elements after removal of the first few units.

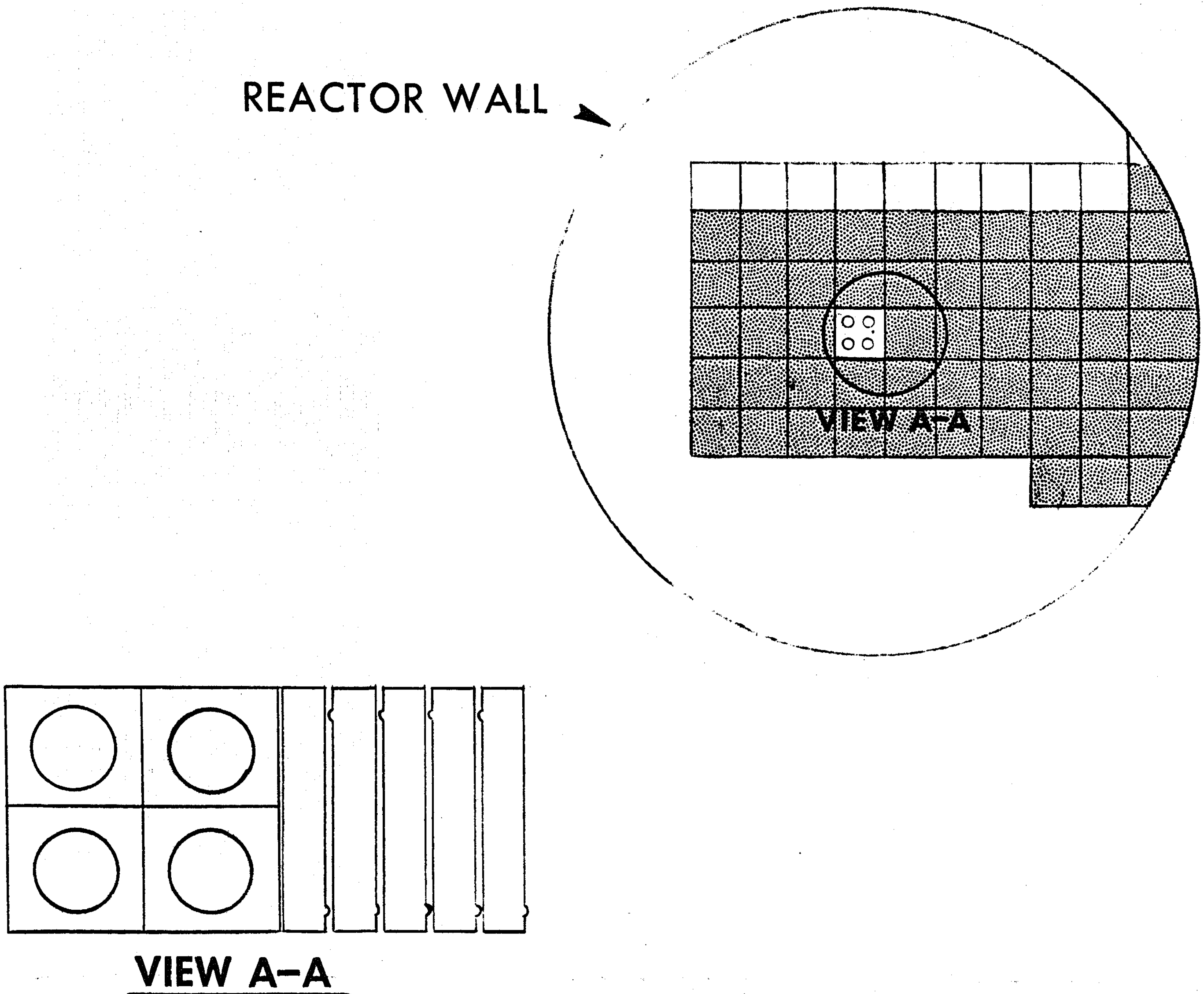


Figure 3.4: Square Array of Plates.

2) Hexagonal (Task I Reference Selection)
   The basic design selected for the Task I reference concept is shown in Figure 3.5-3.8. This concept has all the desirable features mentioned above, and in addition, is compatible with the overall dimension for moderator element unit size which was being determined independently. The element is composed of flat plates for which the manufacturing and coating techniques are certain and with nubs or ribs to provide the separation for salt channels. The Y-shaped yoke forms the main structure both for normal operating conditions and for removal. Flow control may be achieved by orificing in a bottom end plate, a top end plate, or by a pronounced tapering of the flat moderator plates at the inlet and/or outlet. Assembly of the plates into the hexagonal units is made by cementing and doweling. All cemented joints are made in low flux zones.

   The detailed design developed for a typical Zone 1 moderator element is shown in Figure 3.9-3.11. The basic slabs are extruded and are machined to final shape. The Y Yoke slabs are then cemented. The Y Yoke assembly is inserted in the bottom end box which acts as a retaining jig during the element assembly. The ring is free to slide on the milled shoulder in the yoke for 1 in. Dowels inserted through the 5 in. support legs on the bottom of the yoke pieces hold the retaining ring on the element during lifting. While in service the ring will float against the milled shoulders of the slabs. The moderator slabs are then added starting with the closest to the Y Yoke and working outward to the edge slabs. The top retaining and support ring is then added. Both rings (top and bottom) are molded graphite pieces formed out of graphite filament stock and machined to fit up dimensions. The top ring is then doweled to the moderator plates and the edge plate is doweled to its inner neighbor. The element is then placed in a furnace along with other completed elements and the cemented joints are fired. After firing, the dimensions are checked and any final machining of outer surfaces performed. The thickness of the moderator plates is varied for the central zone elements to accommodate the changing salt fraction required to provide the axial blanket and plenum regions. The central core region is composed of 13 percent salt and 87 percent graphite and has a height of 13 ft. The blanket region has a volume of 37 percent salt, 63 percent graphite and a height of 6 in. both top and bottom. The plenum has a volume of 85 percent salt, 15 percent graphite and a height, top and bottom, of 6 in. for an overall graphite element height of 14 ft, 11 in. There is one-inch clearance at the bottom of the element. The distance from the top reflector bottom surface to the top of the bottom reflector is 15 ft. The blanket - moderator elements which surround the core radially have a uniform 37 percent salt fraction for 14 ft, a top plenum of 6 in. and a bottom plenum of 5 in. of 85 percent graphite and 1 in. bottom clearance of 100 percent salt.

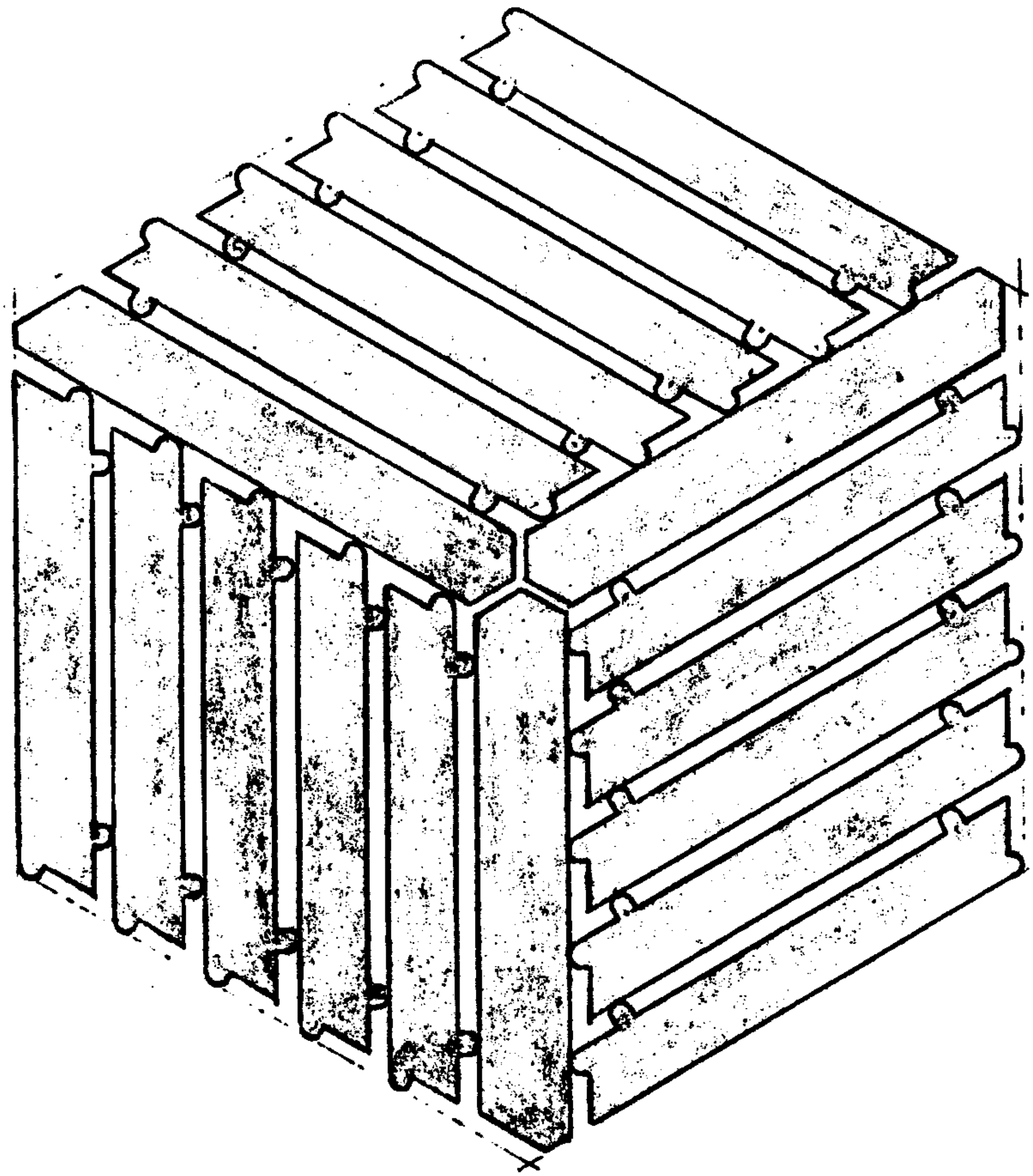


Figure 3.5: Graphite Moderator Element Section @ Midplane, Zone I.

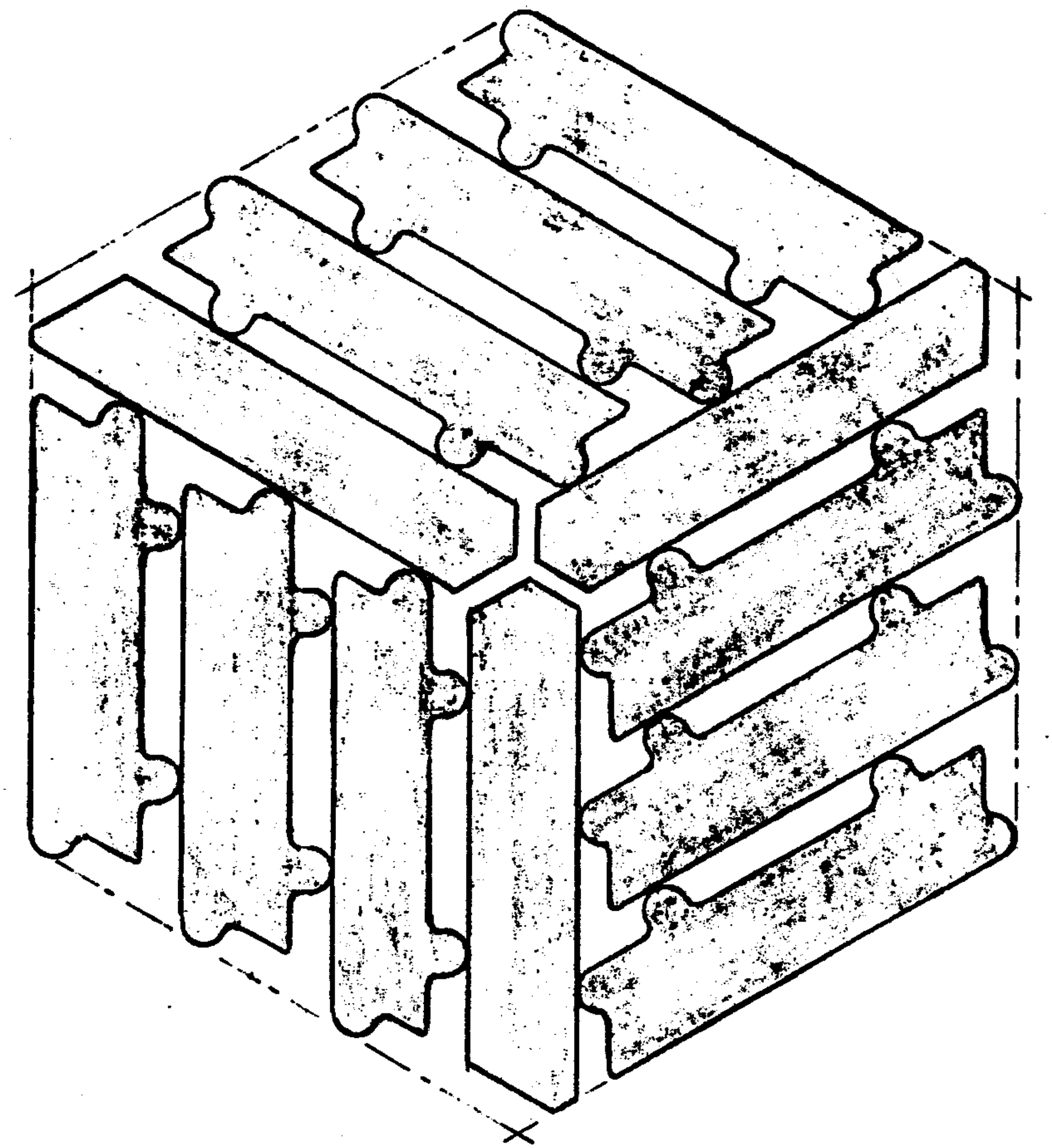


Figure 3.6: Graphite Moderator Element Section @ Midplane, Zone II.

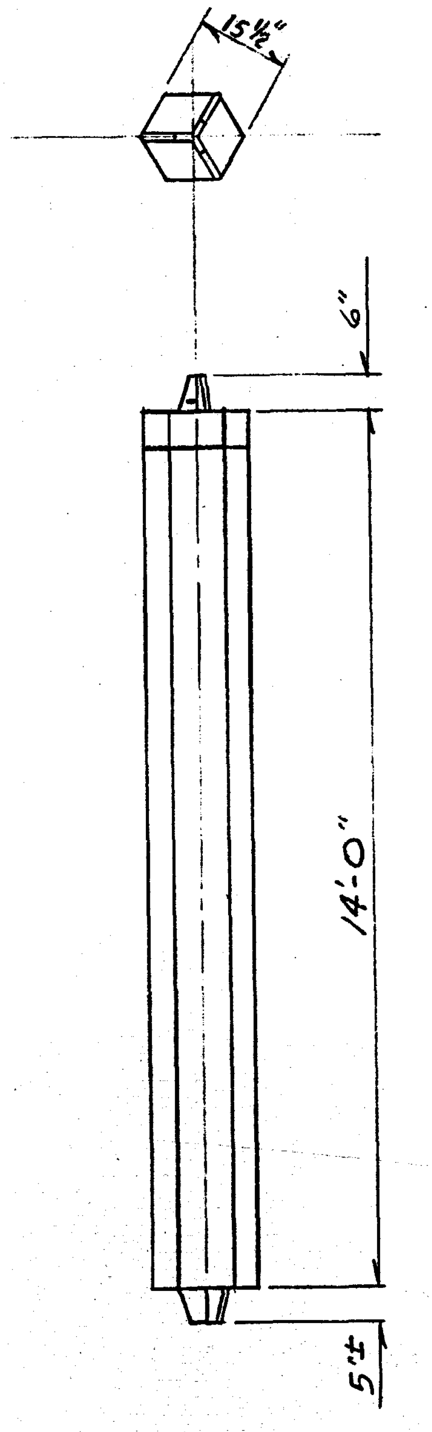


Figure 3.7: Graphite Moderator Assembly - Elevation.

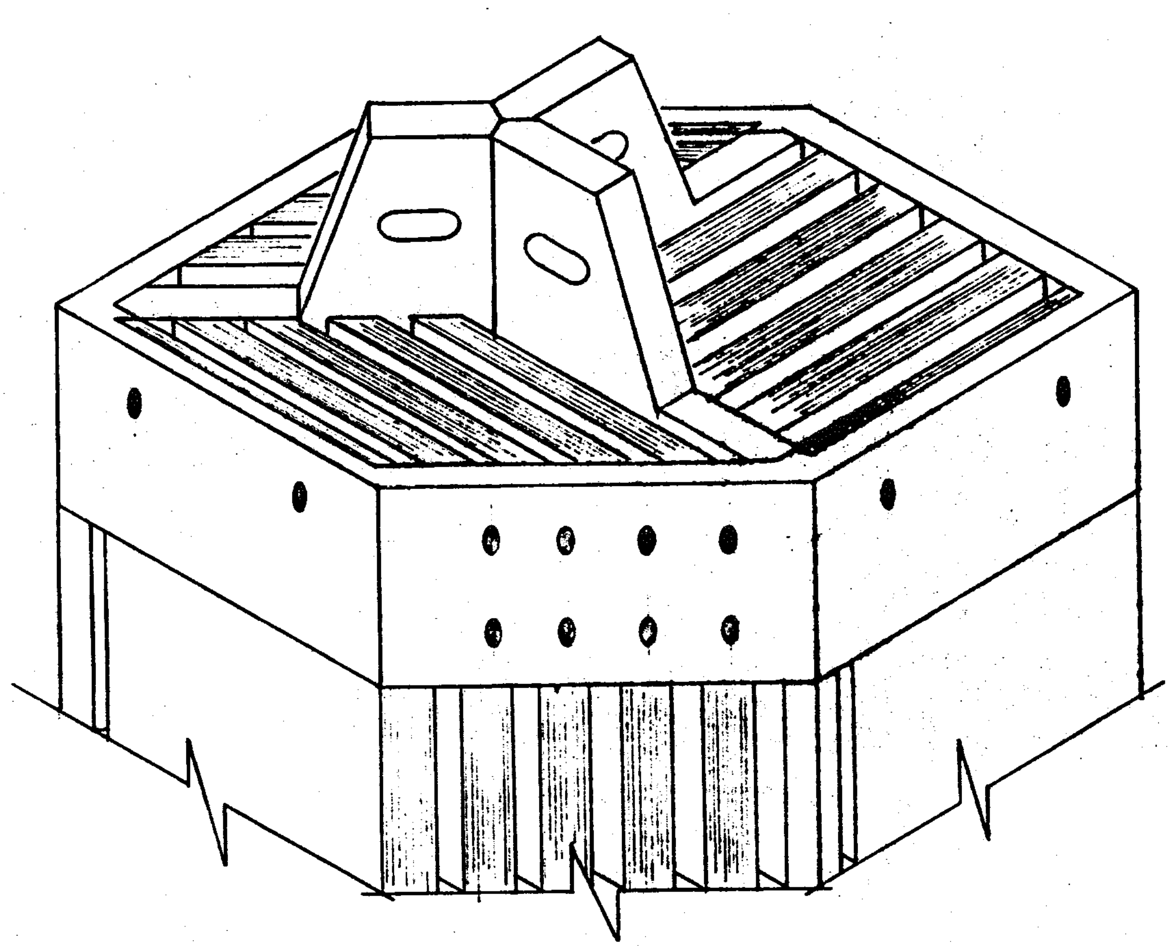


Figure 3.8: Graphite Moderator Assembly Plenum-Lifting Head Detail.

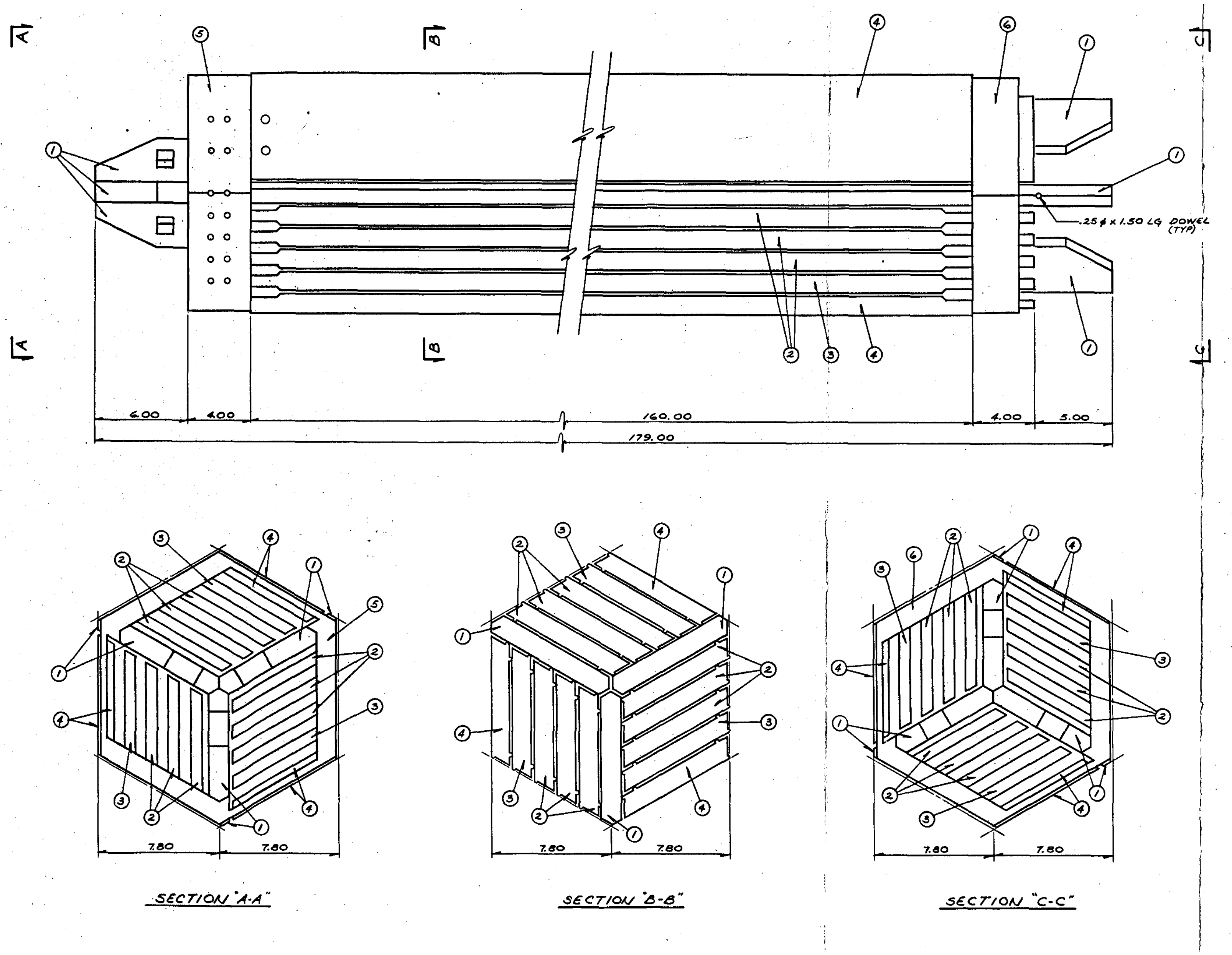


Figure 3.9: Zone I - Moderator Element.

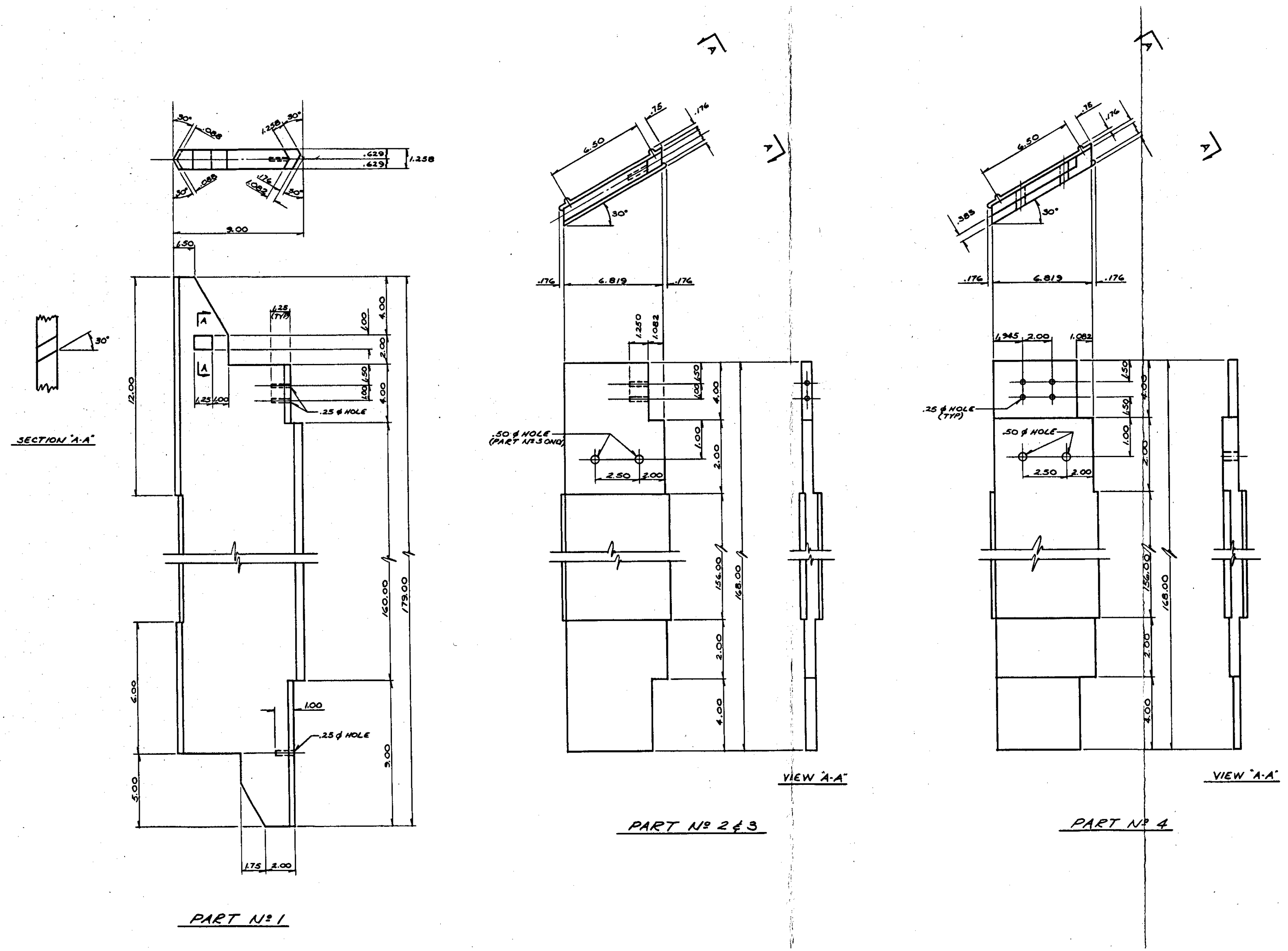


Figure 3.10: Zone I - Moderator Element Parts.

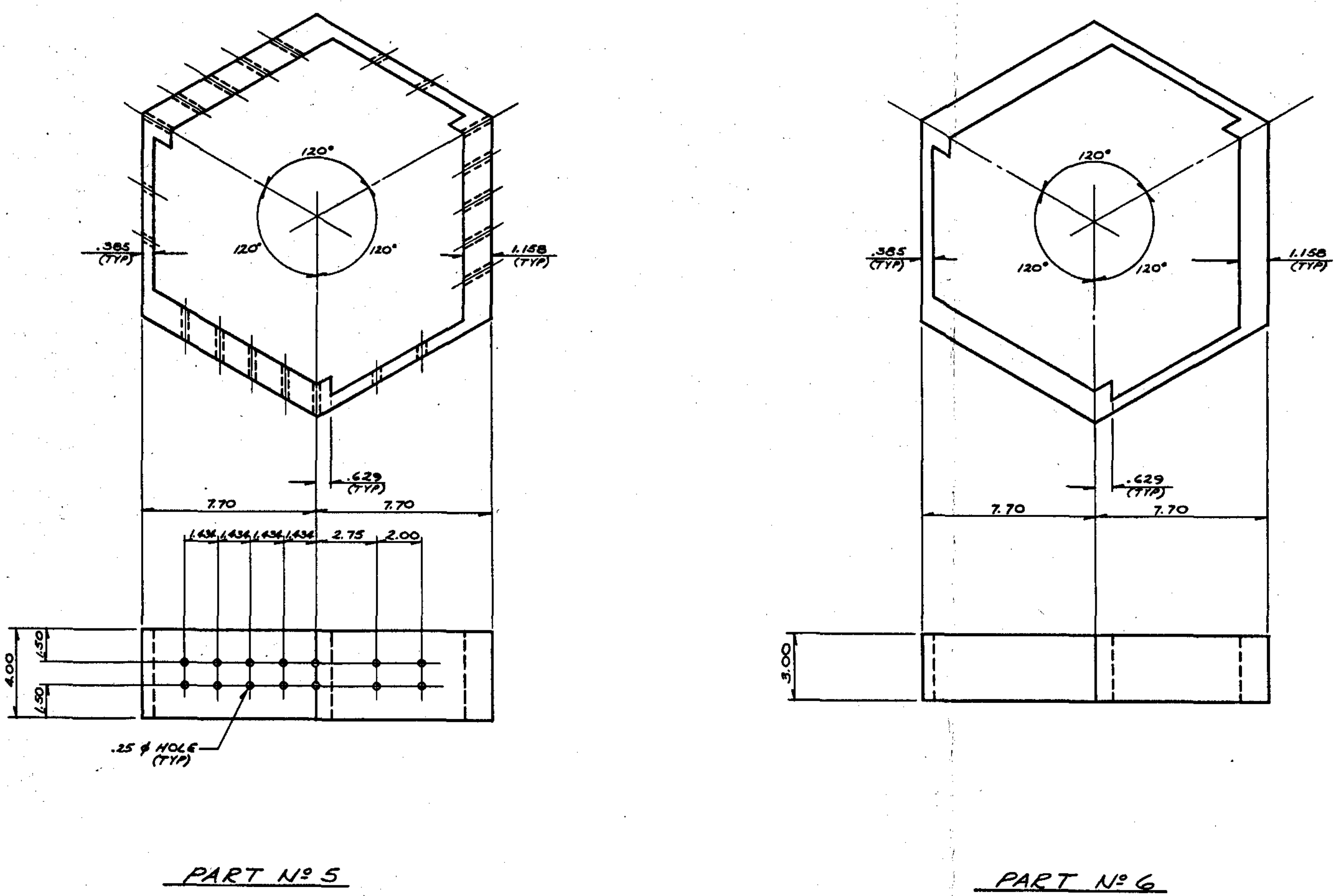


Figure 3.11: Zone I - Moderator Element Parts.

This design produces 6 in. plena at both exit and entrance to the reactor core. These plena are entered by four transition nozzles which convert the circular geometry in the exit and entrance piping to a rectangular geometry compatible with the plenum's thickness. In addition, these transition nozzles provide balancing points for the several sidestream flows to originate and return to the main stream. The sidestream at the inlet transition nozzles provides for:

1) Bottom head cooling during drain tank dump flow.

2) Reactor vessel cooling flow.

At the outlet nozzle this sidestream divides to provide for:

1) Flow into the nozzle liner-pipe annulus.

2) An attemperation flow into the outlet pipe liner.

The entrance conditions for flow into and between the hexagonal elements are most constant over the element's life at the bottom. The element never reaches the regrowth portion of the graphite cycle due to low flux and low temperature. This makes the bottom plenum especially attractive for flow control orifices. It may be necessary to interrupt the nubs along the length of the element to prevent channeling and stagnation regions.

Handling of the moderator blocks has not been considered in detail other than to consider the lifting requirements and their influence on the mechanical design. Normally the moderator elements are lifted by rotating a three-pronged tool through the holes provided in the top plenum spacer lugs on the Y Yoke. These holes provide a total of almost 4 sq. in. of lifting area or a load of 500 psi on the lug material. The graphite in the lug should easily maintain sufficient strength to accommodate 2000 psi loading. Thus even with two defective lugs the element will lift.

Conceptual sketches have been made of a backup tool consisting of three thin plates which can slip down between the flow channels in the element, rotate and engage all plates from below. This tool is considered a backup for mechanically damaged elements.

3.1.6 Graphite Utilization

The basic time interval selected for evaluating moderator handling requirements was the 4-year interval between major turbine-generator overhauls. The basic trade-off evaluation was between moderator element size and the number of element moves required to service the core. The conclusion of the handling study, adjusted to the hexagonal configuration developed in the geometry study, was that the "15 in." element led to an optimum total number, size, weight and number of moves for the element. Results of this study were presented in the progress report dated June 1971.

The core configuration resulting from these studies is shown in Figure 3.1, which is a cross section view through the midplane of the reactor. The central element is used for reactor control and safety rods. There are 162 full hexagonal elements in six full hexagonal rings and a partially filled seventh ring (24 full and 12 half hexagons out of a possible 42 elements). The reflector consists of 96 slightly wedge-shaped slabs 30 in. deep by 14 ft long with an average width of 8 in and a 2 in. taper over the width. The reflector is connected to the array of full and 1/2 hexagons by 18 triangular and 24 flat wedge-shaped spacers. The vessel is protected by a 2 in thick neutron shield made either of borated graphite or a Hastelloy-N honeycomb structure filled with boron carbide. One purpose of this shield is to attenuate the thermal neutron flux by a factor of 1000 thus reducing the He generation rate in the reactor vessel. In addition, the thermal neutron shield forms a flow baffle to divert a stream of inlet salt for the purpose of cooling the reactor vessel. The cooling salt annulus is approximately 1/2 in thick. The reactor vessel is assumed to be 2 in thick and is designed for a nominal pressure of 75 psi like that proposed by ORNL in ORNL 4541. Salt enters the reactor vessel through four inlet nozzles, each of which is protected from rapid thermal transients by a liner similar to the top transition nozzle liner configuration. The cold (1050°F) salt from the heat exchanger forms a thermal sleeve to protect the inlet pipeline from rapid changes due to transients in the secondary salt system. This salt stream provides the normal and the emergency cooling flow for the bottom head.

3.1.7 Future Thermal - Hydraulic Considerations

In Task I of the design study the reactor outlet temperature has been specified as a maximum of 1300°F. The inlet temperature is chosen as 1050°F. These choices were based on reasoning similar to ORNL's, stemming from the high melting point of the primary salt, the high pumping power required by the viscous salt, the allowable stress values in Hastelloy-N material and our desire to preserve the ORNL reference design physics model for Task I. The basic objective of the orificing provided is to maintain a 250°F temperature rise across the core. The hexagonal moderator concept developed in our design study utilizes a similar hydraulic control concept to the ORNL prismatic element, viz., unorificed flow in the flow channels between elements and orificed flow in internal channels. As we progress further into the moderator element design, we must resolve several important questions on hydraulics.

1) The number of orifice zones required. In our configuration we would anticipate that no more than seven zones (seven rings of hexagons) would be required.

2) The need for interrupted spacer ribs to promote interchannel mixing. If studies in Task II show undesirable hydraulic effects, then the ribs can be interrupted. Also, holes can be drilled through the plates to allow interchannel communication.

3) The actual method of providing the flow control. Control can be provided as part of the end boxes (either top or bottom) or as an actual change in sections of the moderator plates at entry or exit. Top orificing would permit hydraulic compaction. Our design allows us maximum freedom of choice based on the results of future studies.

4) The flow perturbations resulting from dimensional change of the graphite. These changes are functions of position, temperature, and power history. Since the lowest temperatures and lower

power history will be at the inlet, our current effort is directed at orificing there.

5) The hydraulic implications of the change in flow area required in the central region element to switch from 37 percent salt to 13 percent salt and back to 37 percent salt again as the flow rises through the core, thus forming the axial blanket region.

6) An analysis of the heat removal requirements for the radial reflector blocks. Our present concept involves the use of large graphite blocks stacked to form a free-standing structure during handling operations (including initial loading). Design details must be completed to show that adequate cooling flow between all blocks is provided.

7) An analysis of the heat removal requirement of the top and bottom reflectors. Our design provides a 6 in. plenum at the top and bottom of the core to: a) improve the flow path from the inlet to the element orifices and b) to reduce radiation damage to the reflector since we desire both reflectors to last the plant life. The present top head concept makes it possible to replace top reflector blocks. The replacement of the bottom reflector would be a major operation. To allow for this we are considering the use of a 2 in. thick inlay of graphite over the central 11 ft radius of the bottom reflector. Since the bottom reflector graphite runs cold (1050°F), it can take much more fluence than the top reflector graphite. Thus it is unlikely that the bottom reflector will require replacement. This inlay would be replaceable during the replacement of the four-year-life graphite elements.

8) The effect on the salt flow area of the 2 percent change in graphite linear dimensions. At its maximum shrinkage, the interior portion of the element experiences a change in graphite area to 83.5 percent from 87 percent. The corresponding change in salt area is from 13 percent to 16.5 percent. This effect is experienced in the ORNL design also, but in that design the effect is absorbed by an overall compaction in the outer flow channels. In our hexagonal design the bridging effect of the radial rings strengthens the outer flow channel structure and prevents the general compaction. The effect could possible lead to a Task II physics optimization that has a central region less than 13 percent.

3.2 Reactor Vessel

The basic concept of the reactor vessel for the MSBR design study is shown in Figure 3.12- 3.14. In each of these figures the same configuration of vessel is used with a different head closure presented for each case. The vessel is a right circular cylinder 2 in. (nominal) thick, having straight sides. The vessel is closed by a bottom dished head of 3 in. (nominal) thickness forming a continuous membrane with the cylindrical vessel. The vessel top head is a dished section of 2 in (nominal) thick Hastelloy-N.

Three methods of closing the top head onto the cylindrical vessel have been developed on Task I and will be further evaluated in Task II. The first closure involves the use of a support skirt for the head extending from the top of the graphite reflector into the accessible areas of the reactor

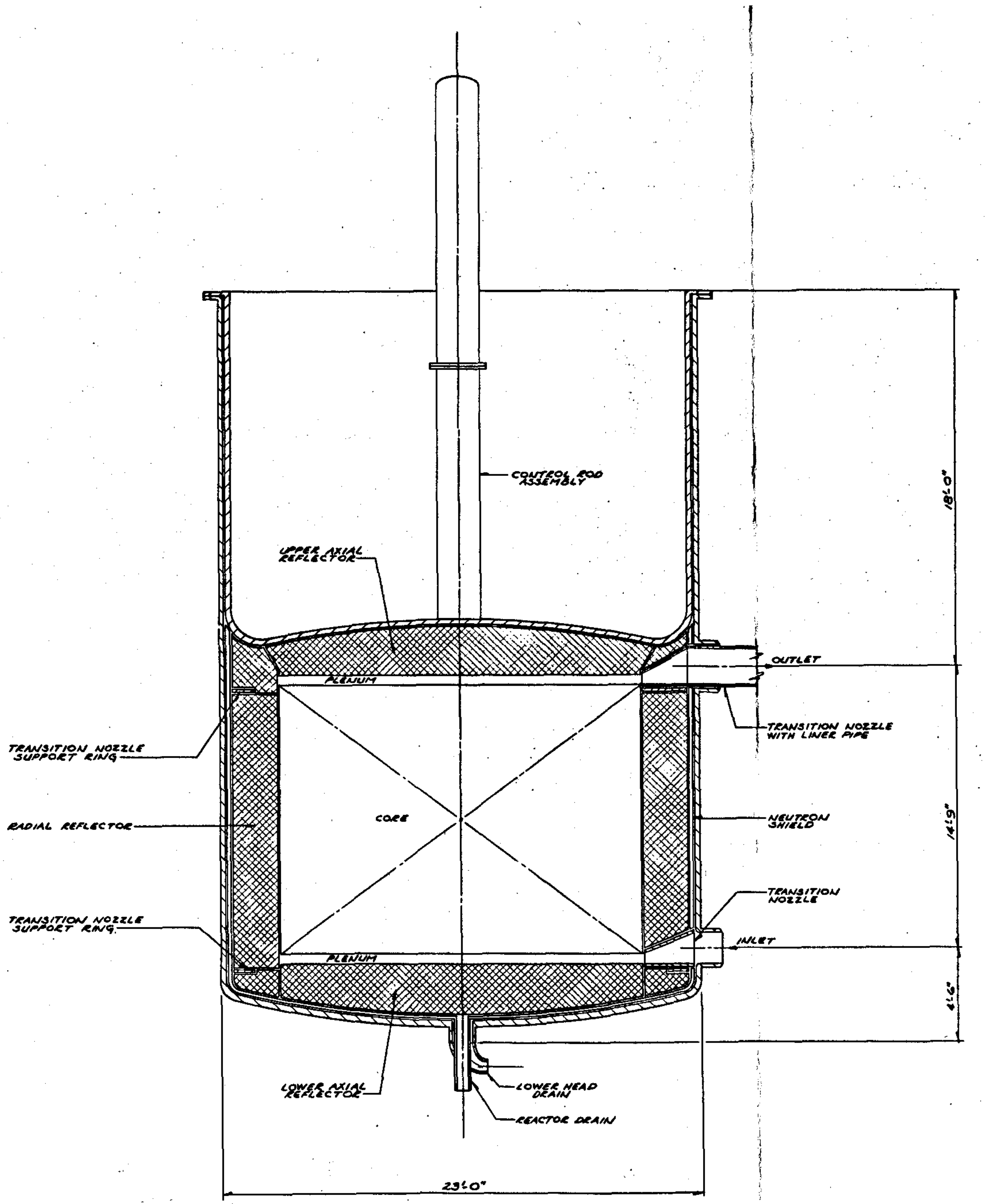


Figure 3.12: Reactor Vessel - Section.

operating floor. Both vessel and head support skirt are bolted together in a clean, cold area This design requires the use of a large skirt for both the vessel and the head, which are fabricated of Hastelloy-N. This design may be unattractive for economic consideration. However, its design simplicity and the ability of effecting a closure at low temperature in a non-radiation environment are tangible operating benefits. A cooling flow would be introduced between the vessel and the head support skirts to give a programmed temperature gradient in the distance from the top of the reactor to the operating floor. The second head closure involves the use of a support skirt for the head and a vessel extension skirt again reaching from the top of the reactor to the operating floor, and is shown in Figure 3.13. This more complicated geometry may be required to avoid streaming problems in the annular space formed by the 2 in. vessel, the 2 in. head support skirt and the 1/4 in. void between them. The third head support scheme shown is Figure 3.14, which involves the use of a flanged bolted head. A problem exists in this design in that the space required between the outlet nozzle of the reactor and the flange on the vessel produces an unnecessarily large top reflector region. To avoid the penalty of filling this space with graphite, which would serve no useful neutronic purpose, we have provided a graphite support structure which would hold the top reflector.. We have several concepts for this top structure. In one concept, it consists of an independent member which is removed separately from the top head. In another it consists of a structure integral with the top head consisting of bars on which notched graphite members are assembled and pushed into position. The void above the structure supporting the top reflector would be filled with inert gas, and possibly with injection tubes for salt, to cool the reflector.

There are four symmetrical inlet nozzles.The inlet nozzles terminate in a liner which is located inside of the pressure bearing pipe coming to the reactor vessel to the head exchanger outlet. The purpose of the liner is to protect the pressure bearing pipe from the rapid changes in temperature associated with the loss of secondary coolant pumping power. The loss of secondary coolant pumping power would produce a rapid rise in the temperature of the salt exiting from the heat exchanger, and would constitute a serious shock to the inlet pipeline and the reactor inlet nozzle. The liner protects the inlet pipe and the reactor inlet nozzle from this shock by virtue of the fact that salt in laminar flow is an excellent insulator and restricts temperature changes to very slow variations even when the fluid inside the inlet pipe liner experiences a rapid temperature change.

The inlet fluid enters the reactor at 1050°F. It flows through a 6 in. plenum containing 85 percent salt and 15 percent graphite formed by the 5 in. support legs. A small stream of salt is diverted down from the inlet to cool the bottom graphite reflector. This heated bypass remixes with the main stream in the bottom plenum. A very small salt stream passes outside of the thermal neutron shield to cool the bottom head and is dumped to the drain tank by the lower head drain line. The reactor vessel cylinder wall is cooled by a bypass flow of 140 gpm which is diverted from the inlet nozzles and flows up in the gap between the thermal shield and the reactor vessel wall. The vessel cooling flow rises 50°F to 1100°F as it is collected at the outlet nozzles. The 1100°F fluid is divided into four streams of 35 gpm each, which is passed between the outlet nozzle and the transition nozzle liner extension. This extension passes the cool stream of salt that has washed the vessel down to the pump in an annular laminar flow to protect the outlet nozzle and pipe from the thermal shock of a reactor scram. Our calculations indicate that following a scram the fuel salt

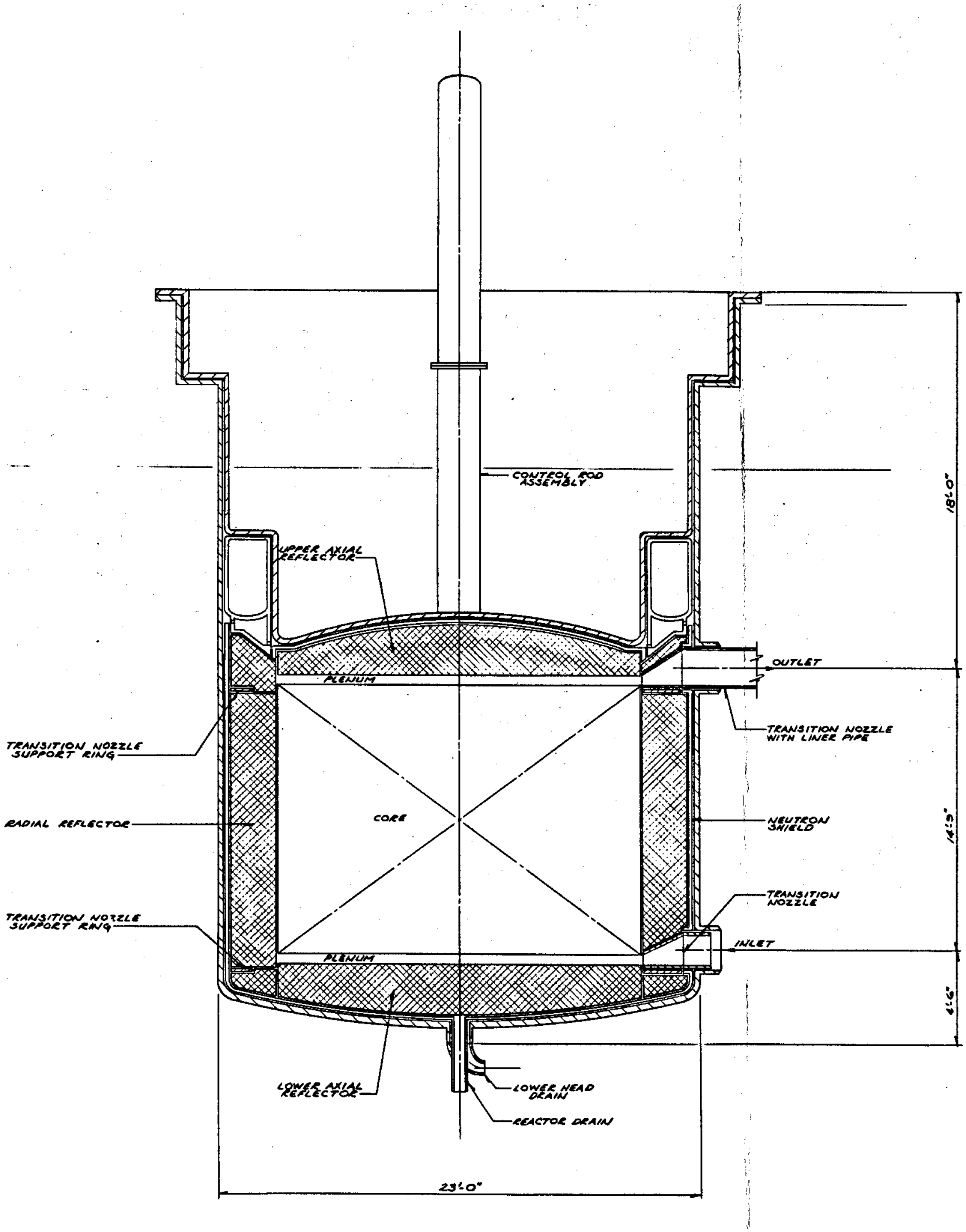


Figure 3.13: Reactor Vessel - Section.

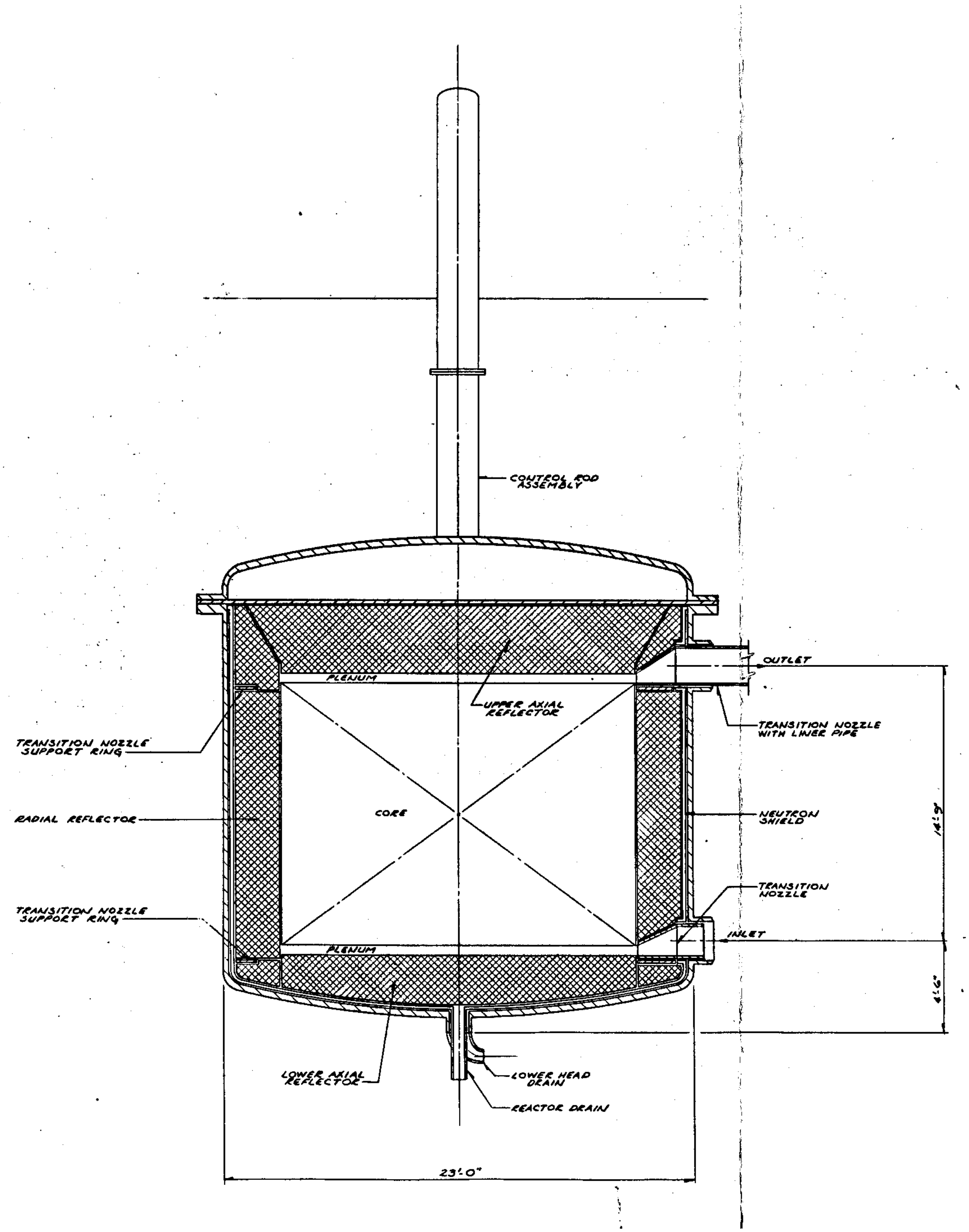


Figure 3.14: Reactor Vessel - Section.

outlet temperature falls from 1300°F to 1050°F in less than 10 seconds. For the thickness of pipe in the primary system, this is equivalent to a step change of 250°F. The top reflector graphite is cooled by a stream of 1050°F salt injected through the control rod drive penetration.

3.3 Reactor Drain System

3.3.1 Drain Tank and Reactor Lining

Fuel salt is taken from the primary system both intermittently and continuously (Figure 3.15). The intermittent drains from the primary system occur through the reactor drain line, which is controlled by a freeze valve and located at the lower-most point of the reactor vessel bottom head, and secondly in the event of a pipe break from the catch basin, which is located beneath the primary system.

The catch basin is isolated from the drain tank by a rupture disk and a check valve. The catch basin drain line is tied to the reactor drain line and penetrates the reactor cell containment in a common penetration. The drain tank is situated in a separate containment cell directly below the reactor containment.

The continuous drains from the primary system are:

1) The reactor vessel bottom head drain line.

2) The four pump overflow lines.

3) The four heat exchanger drain lines.

4) The chemical plant return feed line.

The reactor drain line and the heat exchanger drain lines introduce fuel salt to the reactor cell batch tank at a temperature of 1050°F. The chemical plant feed return line is also at relatively low temperature (probably 1100°F). The pump overflow lines are introducing fuel salt at a temperature of 1300°F. The volume of salt from the pump overflows will be held to a low value, just sufficient to keep the line hot and in service. This line is available to allow overflows of larger quantities of salt in the event surges occur in the pump tank. The major flow of salt will be at the lower temperature, 1050°F.

The ratio of hot to cold salt will be adjusted to produce a mixed-mean temperature in the reactor cell batch tank of 1100°F. During surges this temperature will be higher.

The off gas removed from the system by the gas stripping device will provide an intense source of decay heat. This gas will be introduced into the reactor cell batch tank where it will mix with the 1100°F fuel salt and be carried to the drain tank through the reactor cell batch tank drain line. Since the fuel salt will be substantially below the 1300°F limit, which we are using in the design

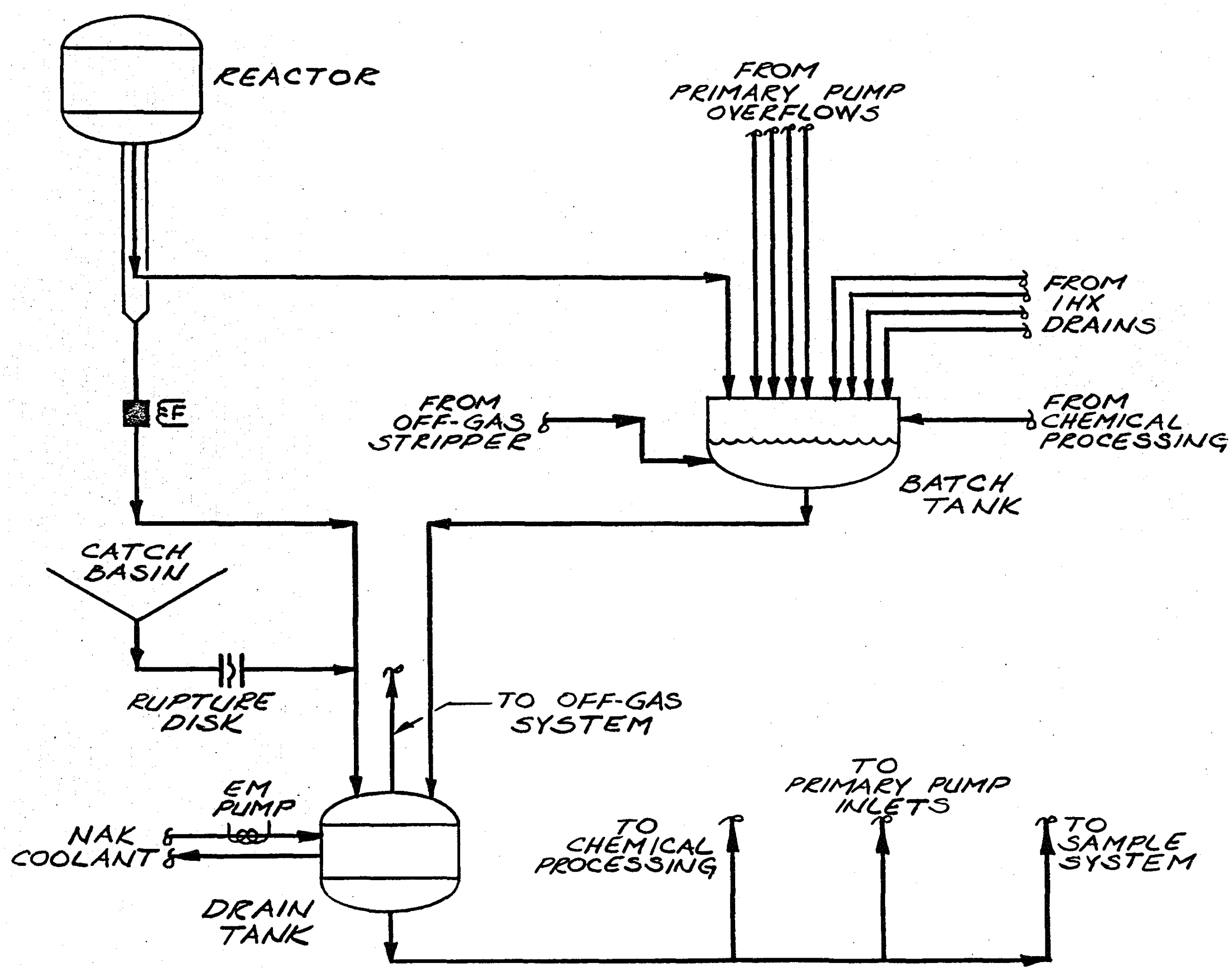


Figure 3.15: Reactor Drain System.

study for Hastelloy service, there will be sufficient heat capacity in the nominal 600 gpm flow rate to contain the off-gas decay heat without requiring external cooling of the line.

Fluid will be drawn from the drain tank by means of jet pumps and supplied to:

1) The chemical plant feed line.

2) The primary system coolant pump suction line.

3) The chemical sampling system.

A possible alternate on the return of the cooled fluid from the drain tank occurs in the fully lined piping system. The concept described in the Task I report uses a stream of fluid leaving the pump to serve as a thermal barrier between the turbulent inner stream and the pressure containing pipe. Since this nominal layer of salt in drawn from the pump bowl, its temperature is 1300°F, and the allowable Sm is only 3500 psi. Reinjection of the return flow by a booster pump into the jacket between the hot salt and the pressure pipe on this leg would allow a substantial upgrading in the stress capability of the Hastelloy. Only a small portion of the return flow of approximately 150 gpm per loop would be required for this jacketing function. A nominal value of 35 gpm has been used in Task I calculations. The remainder would still return to the pump suction via the jet pump.

In designing the reactor vessel bottom head for a dual drain system, i.e., one drain for the bottom head space between the thermal neutron shield wall and the reactor vessel wall. and a second drain for the primary salt to leave the reactor region and go directly to the drain tank, we have provided flexibility that will enable a drain accident to be accommodated without overheating of the reactor bottom head. In the drain, the fluid will leave the reactor vessel passing through the bottom graphite space, enter the separate concentric reactor drain tank line, and flow to the drain tank. The back flow from the heat exchangers will pressurize the fluid in the plenum and also cause a flow to enter the space between the reactor vessel and the thermal neutron shield, thus continuing to provide cool fluid to the reactor vessel head in this region. The flexibility of this design allows us, if it proves necessary, to incorporate into the system with minor modifications a third fluid complete jacketing system except in regions where the Hastelloy can be kept in net compression or easily replaced. In the present concept we have lined the system using fuel salt at reactor inlet temperature and allowed the temperature of the liner fluid to rise 50°F in passing up to the inlet nozzle liner and a further 50°F in passing from the nozzle linear to the pump bowl. In the full jacket system a separate source of salt, possible L2B or a uranium-free salt mixture, would be injected into a sealed liner. Seals would be located at the junction of the transition nozzles and the thermal neutron shield reactor vessel at the outlet nozzle, thermal neutron shield junction, in the outlet nozzles and at the primary pump piping bowl transition piece, and in the inlet line to the heat exchanger. These seals would not be designed to be absolutely tight since the jacket salt would be selected on the basis of its compatibility with the fuel salt and would be maintained at a higher pressure than the fuel salt. Small inleakages could be tolerated. A total seal length for all seals required to fully jacket the system would be approximately 120 ft. Metallic or ring seals can be expected to perform in this dimension to restrict inleakage to approximately 1 percent a day. In laying out the primary cooling system (we have assumed that we would use the largest components feasible based on preliminary estimates by the Babcock & Wilcox Company), it appears that the entire intermediate heat exchange function can be conveniently provided in four units. Accordingly, we have selected four loops to the reactor vessel. The piping studies performed attempted to minimize the volume of primary salt contained in the piping while at the same time accommodate the thermal expansion requirements of the system. The first configuration studied is shown in Figure 3.16 (detail A), and this is a minimum inventory system where the heat exchanger and the reactor vessel and the pump are connected by the shortest possible pipe runs. The inlet and outlet nozzles to the vessel do not fall in the same vertical plane but are displaced from each other so that the piping system is categorized as the strut configuration.

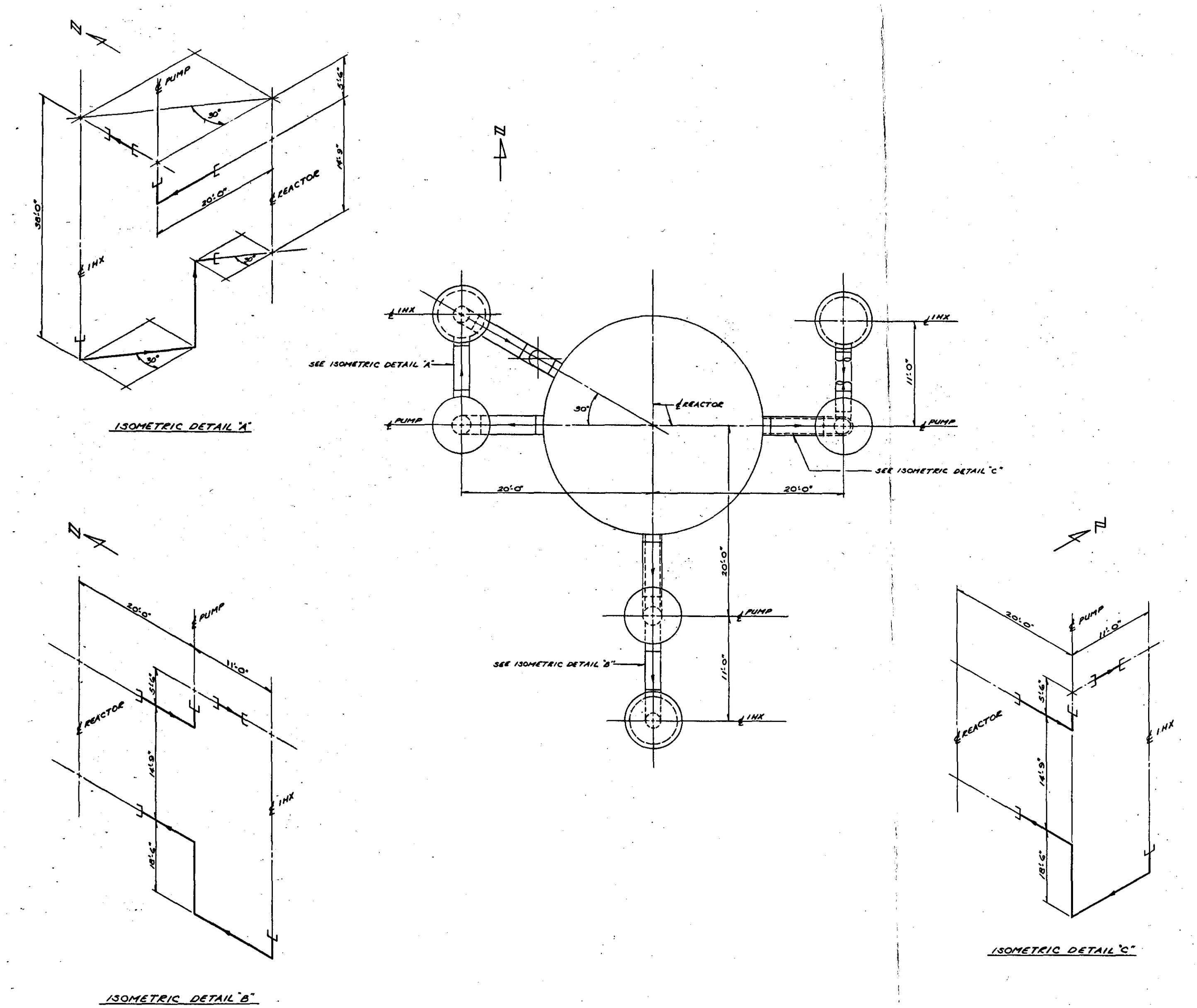


Figure 3.16: Primary System Arrangements.

3.3.2 Piping and Component Arrangement

In developing our approach to the Task I piping for the reactor system, two design considerations became paramount:

a) The length of the intermediate heat exchangers far exceeded the length of our reactor vessel.

b) The available high temperature strength of Hastelloy-N was urgently needed for thermal transients and could not be spared for resisting large seismic forces.

The structural implications of the support requirements are more fully discussed in Section 6.0. We will only summarize the implications of these studies as they affect the reactor piping system in this section.

a) A decision was made to eliminate top support of the reactor and heat exchanger in part to reduce the seismic loads.

b) Even with the reduced magnification of the imposed seismic load due to lowering the building loading point by 50 ft, the piping could not stand the differential response of the nonsynchronous reactor with the four heat exchangers.

c) A decision was made to provide a rigid horizontal coupling between the reactor and the four heat exchangers and to anchor this intertie into the building so that the building, the reactor, and the four heat exchangers respond within the same forced horizontal motion, thus eliminating any component loading on the pipe, i.e., the pipe must withstand only its own and its contained fluids accelerations.

d) The provision of a continuous metal support path from the bottom of the containment through the top of the reactor and heat exchangers greatly increased the thermal growth of these vessels with respect to the operating deck and its fixed support points. (The pump mountings and the vessel flange).

The decay of reactor power following a reactor scram imposes the most severe transient the plant can produce on the piping system at the point of highest temperature and lowest allowable strength — the reactor outlet line. Scram will produce an outlet temperature line change of 250°F in less than 10 seconds. We were not able to restrict the secondary stress produced by this thermal shock to allowable limits. To solve this problem we adapted a pipe liner to provide a thermal barrier between the pressure containing pipe and the hot turbulent salt leaving the reactor. The liner has two basic functions: a) it insulates the pressure pipe from the 1300°F outlet salt by trapping a stream of cool salt that has bypassed the core to cool the reactor vessel; this salt is heated from 1050°F to 1100°F at its flow rate of 30.5 gpm per loop; it constitutes a large thermal resistance to a radial transfer of heat, and b) it restricts the outer pipe wall temperature thus making this wall stronger. In Task I we have not utilized this second factor in the design since the actual thermal and hydraulic performance of the liner requires further study.

The thermal transient produced in the system by a loss of secondary coolant pumping power is nearly as severe for the reactor inlet line. If we assume no natural circulation or bulk mixing of the secondary salt, this transient heats the salt leaving the heat exchanger to 1300°F (from 1050°F) in under 30 seconds. To protect against this transient we have provided a similar low velocity laminar flow through a pipe liner.

It should be noted that all the transients discussed here are assumed to proceed in their most odious fashion. There has been no corrective action assumed. For example, scram has been assumed to occur by firing all safety rods into the core at the highest velocity possible. No coastdown was assumed for the secondary salt pumps. There are a great number of corrective measures that will improve the plant's ability to accept transients, e.g., programmed scram rates, pump flywheels, and pony motors. However, if the plant can accept the unmodified transient, it cannot be rendered unsafe by the failure of a corrective measure, and so our design effort has been directed against deliberately chosen severe conditions.

The inventory in the primary coolant piping system in the strut configuration was lowest for the configurations studied. The pipe stress calculations performed on this configuration resulted in extremely high values of both primary and secondary stresses and bending moments, because the configuration proved to be extremely stiff. Accordingly, this configuration was abandoned as being not practical. Calculations for the hot leg used a value Sm of 3500 psi. (This is the value at 1300°F.)

The next configuration studied is shown in Figure 3.16 (detail B). In this configuration (which we called the in-line arrangement) the primary heat exchanger, the pump and the reactor centerline all fall in a common vertical plane. The piping is necessarily longer than in the strut configuration which results in a higher salt inventory penalty. Hot salt leaves the top reactor nozzle, flows in the bottom of the pump, is circulated through the pump discharge line to the heat exchanger plenum, and returns to the reactor through the bottom heat exchanger discharge line entering the reactor at the inlet nozzle. The inventory penalty in this configuration is equivalent to the distance from the centerline of the pump to the reactor inlet nozzle in both horizontal and vertical dimension. This configuration also proved to be extremely stiff and most of the hot leg from the reactor through the pump to the heat exchanger inlet was considerably overstressed.

After studying this arrangement we undertook a third configuration shown in Figure 3.16 (detail C). This is called the dog-leg configuration. The hot salt leaves the reactor outlet nozzle, enters the pump casing which is located on a plane in common with the centerline of the reactor, is rotated through the pump casing, exits from the pump in a direction at 90 deg from its entry, passes to the heat exchanger, flows into the heat exchanger top plenum down through the heat exchanger tubes, is collected in the heat exchanger outlet plenum, and is returned to the reactor through a pipeline that passes back under the pump at 90 deg and enters the reactor in the same plane as the reactor inlet nozzle and pump centerline. In other words, the return line is everywhere below the exit line from the reactor, making a right angle bend underneath the pump. This configuration has an even higher inventory of salt than in the in-line configuration. The total linear distance traversed in the hot line is approximately the same. The total linear distance traversed in the return line is again approximately the same, but there is a considerably greater quantity absorbed in the elbow than in the straight reactor pipeline. This penalty is small in comparison to the increase of the in-line arrangement with respect to the strut arrangement. The stresses in this configuration again proved to be over the allowables, that is over 3Sm in the hot line. However, they were closer to allowable than they had been in any of the preceeding runs by virtue of the increase in flexibility accounted for by the hot-leg configuration. It had been noted that changes in elevation of the pipeline caused by changes in the support configuration below the reactor produce substantial variations, variations in fact that were larger than those produced by the change in piping configuration in the pipe stresses. Accordingly, it was decided to lower the reactor with respect to the heat exchanger. This would provide more flexibility in the hot line which was the one that was most severely overstressed and make the cold line or leg more rigid. A reduction of 12 ft was effected, and a substantial improvement in pipe stresses resulted. The degree of overstressing present in the system with this configuration was within the margin allowable in a Task I study, i.e., 3Sm of 10,500 psi. The resulting stresses were less than 3 percent greater than the allowable. This was considered satisfactory, and the configuration was selected for the Task I concept. Additional details of stress analysis are presented in Appendix A.

Chapter 4

Secondary System

4.1 General

It is necessary to redundantly isolate the highly radioactive fission products contained in the circulating fuel salt from the potentially dispersive forces that could be caused by the high pressure contained in the steam system. This protection must be afforded without interfering with the transport of heat from the fuel salt to the steam system. To accomplish this decoupling an intermediate coolant salt system is provided.

The advantages of using a coolant salt loop are:

1) To provide a redundant barrier for
    a) protecting the steam system from fission products and
    b) protecting the primary system from steam system pressures.
2) To bridge the temperature gap between the fuel salt melting point and the steam system feedwater temperature.
3) To compress the tubes of the intermediate heat exchanger to
    a) insure inleakage of secondary salt to the primary system and
    b) to keep the Hastelloy-N tubes in compression to prevent cracking associated with intergranular corrosion.

A flow diagram of the coolant salt system is shown in Figure 4.1. The coolant salt system is comprised of four independent loops each loop consisting of shell side of the intermediate heat exchanger, the shell side of the steam generator and the reheater, a coolant salt pump, a blowout device for overpressure protection, a coolant salt melt tank for adding salt, a coolant salt storage tank, a coolant salt filter, a cover gas system, coolant salt piping and valving. The selection of the

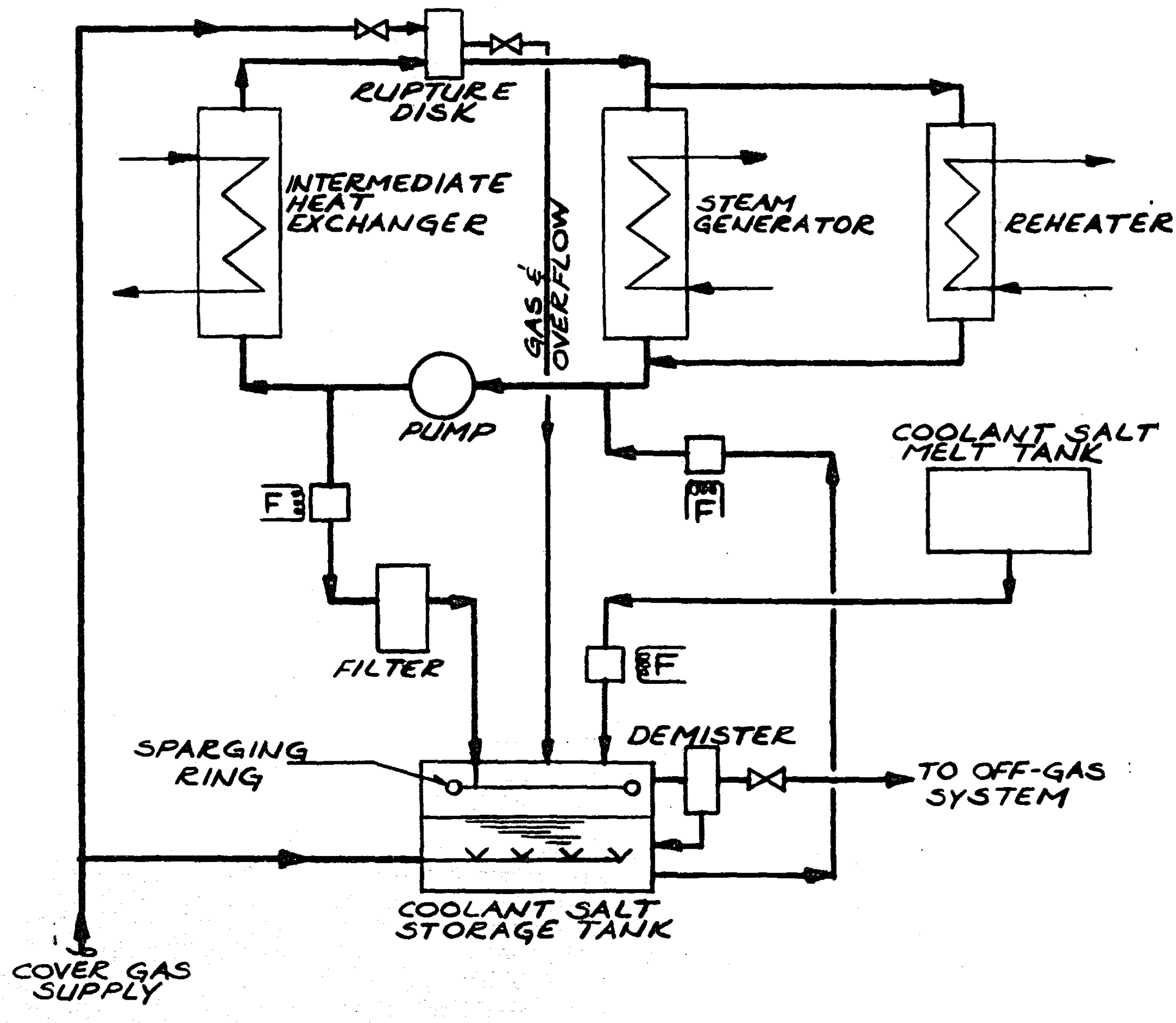


Figure 4.1: Coolant Salt Flow System.

coolant salt for the MSBR design study is confirmed by a study by H. C. Ott for the Molten Salt Group which evaluated:

a) Cost of inventory and makeup

b) Cost of heat transfer surface

c) Cost of pump and pumping power

d) Cost of fuel salt inventory in the Intermediate Heat Exchanger (a function of heat transfer area)

e) Chemical stability and vapor pressure (at temperature)

f) Corrosion characteristics

g) Compatability with the fuel salt in the event of leakage into the fuel salt system

h) Effect on the migration of tritium to the steam system.

Based on this study, Ebasco has selected the eutectic of NaBF4-NaF as the Task I reference coolant salt. A possible alternate selection for Task II study is He based on the use of a fluidized bed heat exchanger. The use of He gives the potential of using a simple chemical purification for tritium recovery. The increase in overall heat transfer coefficients derived from the use of the fluidized bed decreases the size and design pressures required of a helium coolant system to the point of practicality.

4.2 Steam Generator—Reheater Concept Selection

4.2.1 Background

This section establishes concept recommendations to Ebasco for the steam generators for use in the 1000 MWe MSBR reference concept. Based on experience, the formulated basic design criteria establish the basic design features such as vertical orientation, counter-current flow, and other preferred characteristics of the various designs that have been studied. The section lists and discusses the selection criteria which were used to evaluate the entire range of feasible designs.

After the designs are compared, four are selected for detailed evaluation. These four concepts are arranged in order of preference from the component designer's viewpoint, and recommendations are made for further evaluation in conjunction with the system designer.

The design development, evaluation, and selection for the steam generator are also applicable to the reheater. Section 4.2.6 deals with a study of the reheater.

4.2.2 Basic Design Criteria

In the context of this report, design criteria represents the procedures and methods used by the component vendor to meet the requirement of the MSBR system. When developing the design criteria for future steam generators, we used experience gained through design, fabrication, and operation of past units. Based on this experience with high-temperature, high-pressure steam generators, several basic requirements have been established. The following features provide the simplest, most economical, and reliable units:

1) The units will be basically shell-tube designs with the high-pressure supercritical fluid inside the tubes and the coolant salt on the shell side.

2) The heat transfer surface shall be arranged for counter-current, once-through flow between coolant salt and water; when using parallel (co-current) or mixed parallel and counter-current flows, the high-temperature difference existing between the coolant salt inlet and supercritical fluid inlet would impose severe thermal stresses on the tubes due to the large temperature gradients across the tube wall. The thermal effectiveness of the heat exchanger is also increased if counter-current flow is used.

3) The steam generator should be arranged so that the heated fluid flows upward in the region of heat transfer, and the cooled fluid flows downward. This arrangement minimizes chances of unstable operation.

4) All boundaries containing coolant salt are made of Hastelloy-N or modified Hastelloy-N, but other boundaries containing water may be made of Croloy-2-1/4 or stainless steel.

Most standard materials like Croloy and stainless steels are impractical for high corrosion resistance on the coolant salt side. Hastelloy-N, which does have good corrosion resistance, is very costly compared to the standard materials; therefore, for the concept selection the use of Hastelloy-N has been restricted to the coolant salt boundaries. The cost of Hastelloy-N in the future may be comparable to standard material after sufficient industrial use, and/or a less corrosive coolant may be developed.

After establishing the common design factors, it is necessary to formulate the various possible designs for identifying the major parameters that can be varied. The design parameters that were considered for this preliminary study are as follows:

1) Tube Geometry

   a) Straight tube (including sine-wave tube, C tube, hockey-stick tube)

   b) U tube

   c) Return bend tube (plattens)

   d) Helical-coil tube

2) Size of Units

a) Large (4 units)

b) Module (8 or 16 units)

3) Unit Orientation

a) Vertical

b) Horizontal

Using evaluations performed in conjunction with Ebasco and evaluations based on the preliminary study of thermal- hydraulic performance and cost evaluation (discussed in Appendix B), we selected four large units for the reference 1000 MWe design. Selecting four full-size units rather than eight units was for plant simplicity; otherwise, no significant differences in performance, cost, safety, or reliability are evident.

Vertical units are preferred to horizontal units because the horizontal units are more prone to stratification into temperature layers of coolant salt in the shell side. (One exception is the U tube, U shell with the hot leg vertically above the other leg.) This condition affects the performance of the supercritical fluid side and may cause reverse flow and instability. Although it is possible to reduce this instability by orificing and baffling, it is generally not practical; therefore, horizontal units are restricted to the U tube and U shell (with hot leg above cold leg) types.

The combination of the design parameters and the limitations previously mentioned provides several practical and feasible designs, from which four concepts that are based on selection criteria are recommended.

4.2.3 Selection Criteria

The selection criteria that were applied to various feasible steam generator designs are themal stresses; thermal-hydraulic performance; manufacturing; and inspection, maintenance, and repair.

4.2.3.1 Thermal Stress

The thermal stress criteria for the steam generator are similar to those for the IHX. The major problems in steam generators are thermal stresses (particularly during transient operation) combined with the mechanically induced stresses caused by the high-pressure supercritical fluid. The design of the tube bundle is most important; however, unlike the IHX, the steam generator tubes, tubesheets, and plenums are designed for very high water-steam pressures. These sections are quite thick, are associated with slower transient response, and have nonlinear temperature gradients and higher thermal stresses. A combination of tube geometry, tubesheet, shell, and plenum designs arranged for low combined stresses is desirable.

4.2.3.2 Thermal-Hydraulic Performance

It is important to study the thermal-hydraulic performance of each of the steam generator designs because the density difference between supercritical fluid inlet and outlet conditions ranges from 35 to 5 lb/ft3. A thermal system having such a large density difference should have reasonably well-balanced flow through the system. A large flow imbalance impairs the heat transfer performance, causes large temperature differentials in different circuits, and enhances the possibility of flow and pressure oscillations. The problem is more severe at low loads when the frictional pressure losses are considerably reduced with the reduced mass flow rates, and when the pressure differences due to density differential become predominant. The steam generator should have a stable operation at full load and at expected minimum loads. The details of flow stability and reversibility are explained in Appendix C.

4.2.3.3 Manufacturing

Each design was studied to determine the degree of difficulty in manufacturing the components. For the IHX the assembly of the tube bundle and of the tube supports was a problem because most of the preferred designs were variations of the straight-tube concept. The manufacturing problems with straight-tube variation concepts and (to some extent) with U tube concepts are discussed in section 4.3. The return-bend (Platten) concept and (to some extent) the helical-coil concept are not as well established as the conventional heat exchangers. The degree of difficulty when manufacturing these units can only be studied qualitatively.

For the steam generator the tube configuration of several varieties should be studied further. Each of the configurations requires an entirely different manufacturing approach; experience has proved that all these are feasible to manufacture but are manufactured with some degree of difficulty, which is often directly related to in-service reliability.

It may be concluded that the manufacturing of the concept design should be simple, and for reliability the final assembled unit should be accessible for inspection.

4.2.3.4 Inspection, Maintenance, and Repair

Selecting these criteria is probably the most difficult because the state of the art in 1980 is very difficult to assess due to rapid developments in the industry.

The inspection and maintenance of the steam generator are much simpler than those of the IHX and reactor internals because the supercritical fluid side is not radioactive. The tubes are easily accessible from the manholes in supercritical fluid-side plenums with little delay for the radioactive decay. If required, major repairs could also be accomplished with the unit in place. Due to this relatively high degree of confidence the steam generators are usually designed to have non-removable tube bundles, and this condition enables the designer to offer simpler designs.

It is expected that the various designs formulated earlier could use identical inlet-outlet plenums. The major difference among the various designs is in the tube bundle region, which is fixed between the tubesheets and surrounded by shell. Since the routine inspection and maintenance areas are outside this fixed region, the inspection and maintenance criteria are not significant when selecting a steam generator concept.

4.2.4 Concept Evaluation

Several steam generator concepts are evaluated. Some of the units that show obvious drawbacks when subjected to these selection criteria are eliminated; for the remaining concepts that may be used, all the merits and demerits in each category of selection criteria are tabulated. The discussion that follows is in the sequence of tube geometry formulation.

4.2.4.1 Straight Tube and Variations

Figure 4.2 shows four different concepts based primarily on the straight-tube principle. The straight-tube bundle (Figure 4.2) is the simplest to manufacture and is probably the most reliable from a manufacturing viewpoint, but from the thermal-stress viewpoint, the straight-tube bundle is expected to experience severe transients and is eliminated.

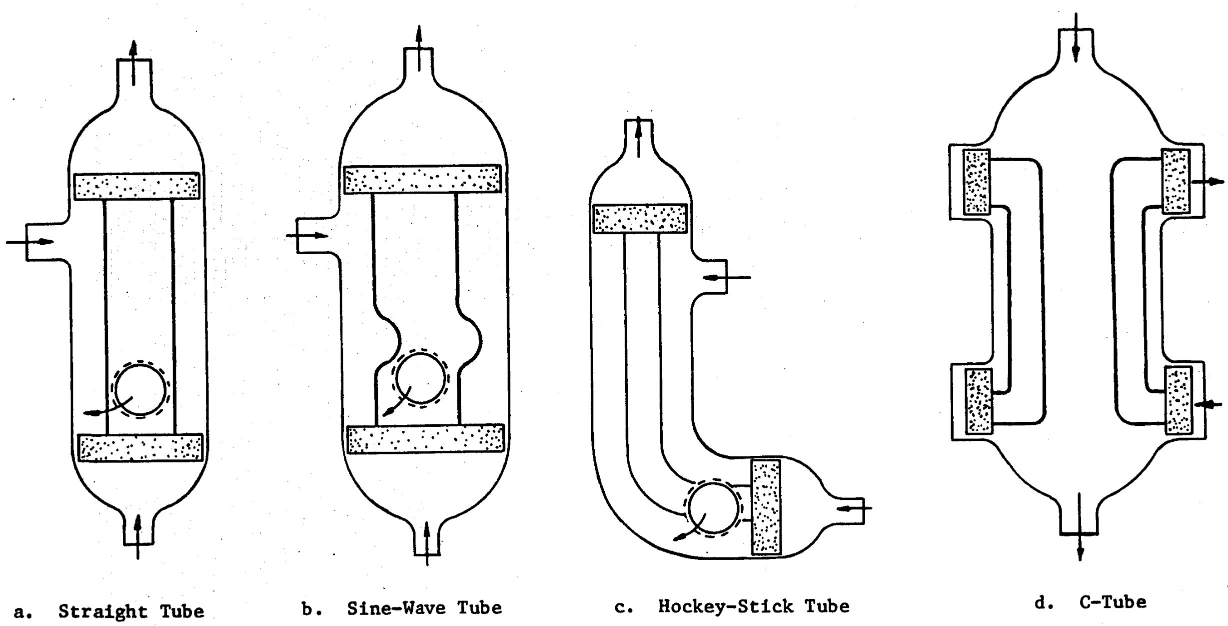


Figure 4.2: Straight Tube Concept Variations.

To withstand large thermal differential during transient operation, more flexible tubes are needed. The sine-wave, hockey-stick, and C tubes, which are treated as variations of the straight tube, are shown in Figure 4.2 (b, c, and d). Each of these variations can be selected for the practical steam generator design; however, there are manufacturing problems with the hockey-stick and the C tube bundle — these are the shell of the hockey-stick tube bundle and the support baffles of the C tube bundle. With the increasing use of sine-wave tube bundles currently, major advances in the manufacturing of these units are expected.

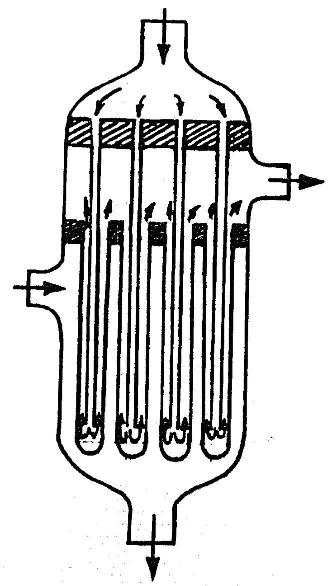


Figure 4.3: Bayonet Tube Arrangement.

Figure 4.3 shows the bayonet tube arrangement, which is also a variation of straight-tube bundles. Each tube consists of two concentric tubes fastened to separate tubesheets at the top. The inner tube is open at the bottom, and the outer tube is sealed at the bottom. Coolant flows down the inner tube and returns up the annulus. This arrangement increases the size of the tube and the unit due to extra tube thickness required in the larger-sized outside tubes. Compared to other straight-tube variation, the unit is somewhat thermally inefficient; however, experience with this concept is quite limited and it will not be pursued further.

Based on the previous discussion, we have selected the sine-wave tube bundle for further evaluation.

4.2.4.2 U Tube Concept Variations

Figure 4.4 shows four different concepts using the U tube geometry. The arrangement shown in Figure 4.4a has one serious disadvantage, which is the close proximity of the tube-side inlet and outlet, which gives rise to very high thermal gradients. This is a very poor arrangement when faced with high-temperature units and is eliminated.

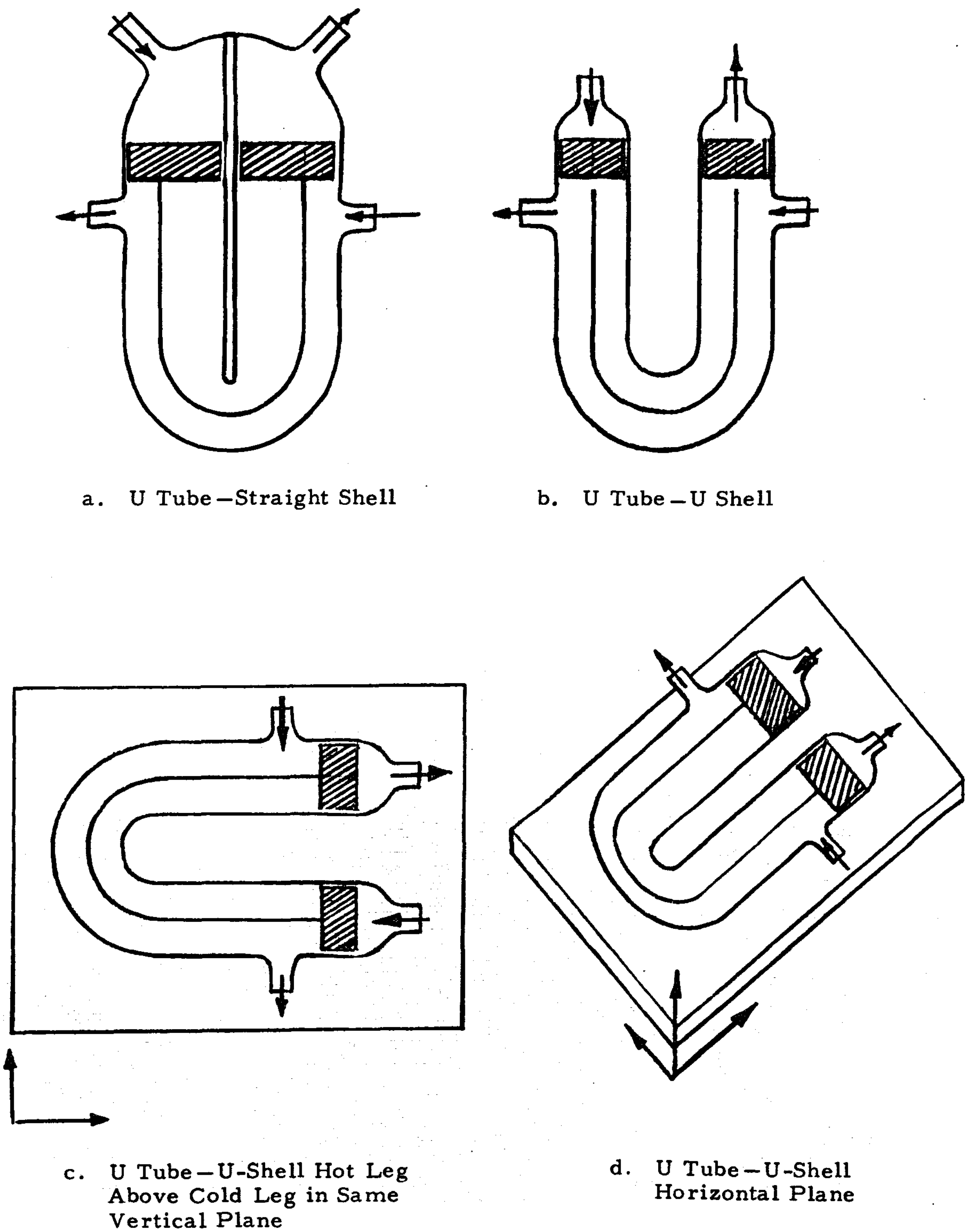


Figure 4.4: U Tube Variations.

The vertical U tube arrangement (Figure 4.4b) is generally avoided because it is potentially instable. The desirable feature is to have the heated fluid flow up and the cooled fluid flow down. With two equal legs this criterion is violated in one half of the unit. This arrangement also causes maldistribution and is eliminated.

The horizontal, U tube, U shell unit (Figure 4.4d) has both legs in the same horizontal plane. This arrangement is prone to stratification due to temperature layers of coolant salt in the shell side; however, this condition may be reduced considerably by providing several baffles for mixing on the shell side. This unit is not pursued further.

The horizontal, U tube, U shell (Figure 4.4c) has the hot leg above the cold leg in the same vertical plant. This unit has potentially better flow stability characteristics and many other advantages of the U tube; it is further evaluated.

4.2.4.3 Return-Bend Tube (Platten)

The return-bend tube (platten) concept meets all selection criteria requirements and is evaluated further. (See Figure 4.7.)

4.2.4.4 Helical-Coil Tube

The helical-coil tube concept meets all selection criteria requirements and is evaluated further. (See Figure 4.8 and 4.9.)

It may be concluded that all four steam generator concepts selected previously are suitable for the 1000 MWe MSBR reference design. The discussion that follows indicates the advantages, the disadvantages, and some special comments for each of these concepts.

4.2.5 Detailed Evaluation of Selected Concepts

4.2.5.1 Vertical, Once-Through, Straight Tube With Sine-Wave Bend (Figure 4.5)

Advantages

1) The unit is the simplest to manufacture.
2) Much less flow modeling and experimentation are necessary. Experience in the operation of the sine-wave tube bundle unit ensures stable thermal and hydraulic performance (for supercritical operation).
3) The degree of difficulty of performing major repairs may be considerably reduced because the experience with straight-tube bundles may be utilized.

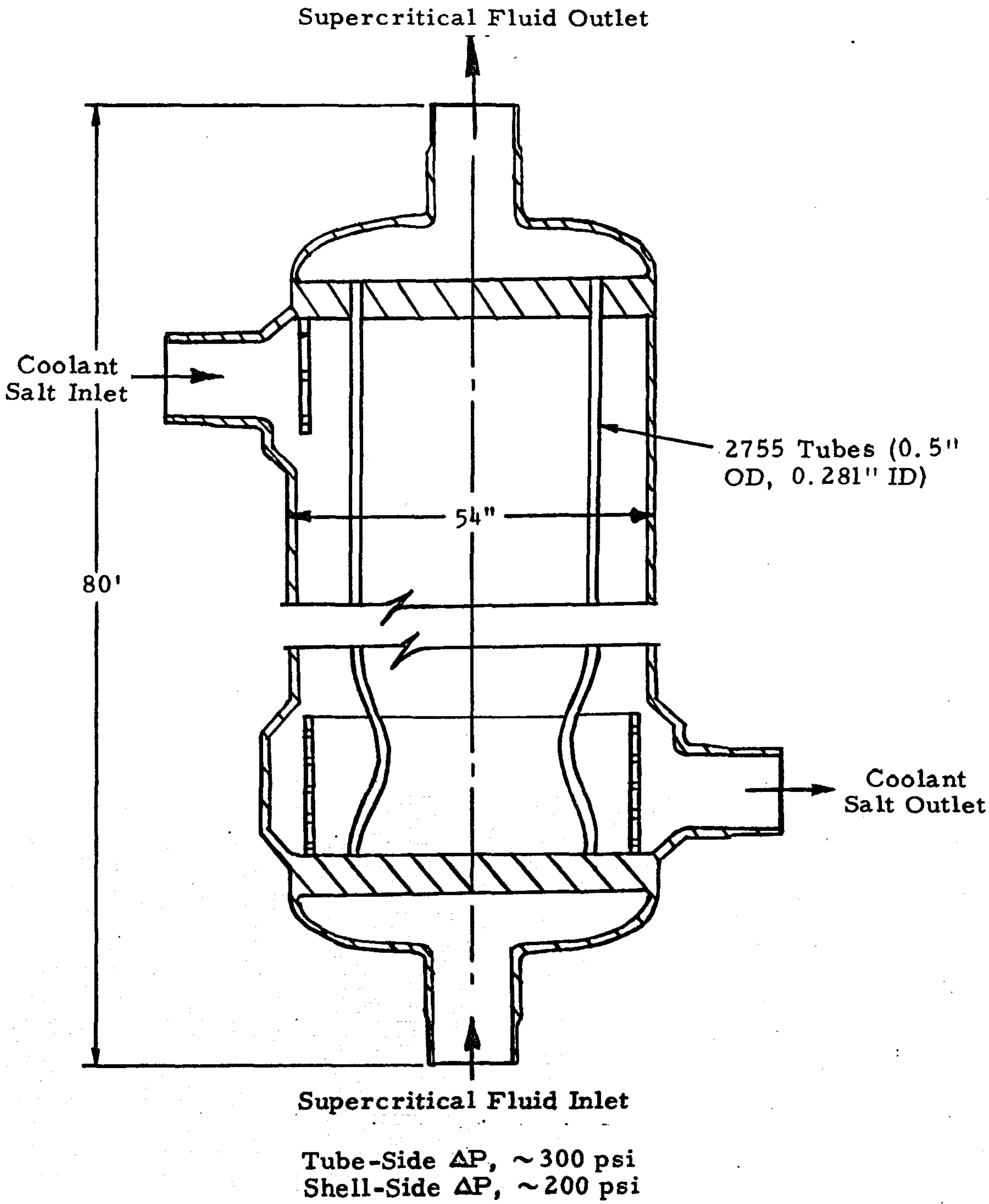


Figure 4.5: Straight Tube With Sine-Wave Bend Concept.

4) Thermal efficiency is increased because small tubes which have small wall thicknesses can be used.

Disadvantages

1) Designing sine-wave tube bundles for suitable envelopes is difficult; preliminary calculations indicate that to achieve stable operation the length of the steam generator should be about 65 to 75 ft. This length may not match properly with the system arrangement. It is possible to provide several orifices to increase stability without adding length, but this approach is not efficient and would require further cost evaluation.

2) A large sine-wave bend is necessary to withstand large transients usually associated with the steam generators. There are some problems in providing tube-support baffles in this large, sine-wave bend region.

Comments

If the large envelope for the steam generator is acceptable to the system designer, further evaluation of the sine-wave tube bundle concepts is recommended.

4.2.5.2 Horizontal, Once-Through U Tube (Figure 4.6)

Advantages

1) The thermal efficiency is increased because small-size tubes can be utilized.

2) Stable units can be designed by providing large numbers of baffles for better mixing of the coolant salt on the shell side. The propagation of coolant salt flow disturbances to the water-steam side may be avoided.

Disadvantages

1) Large U bends and large clearances are required to withstand steam generator transients.

2) The average temperature difference between the hot and cold legs of the U-shell is generally very high. A complex support arrangement is required to avoid high thermal stresses.

3) The tube-support baffles in the U bend region have proved to be a major problem in recent designs.

4) The shipment of the large unit may be quite difficult.

5) The manufacturing of the U bend region of the shell has to be in several pieces and is complex.

4.2.5.3 Vertical, Once-Through, Return Bend (Platten) (Figure 4.7)

Advantages

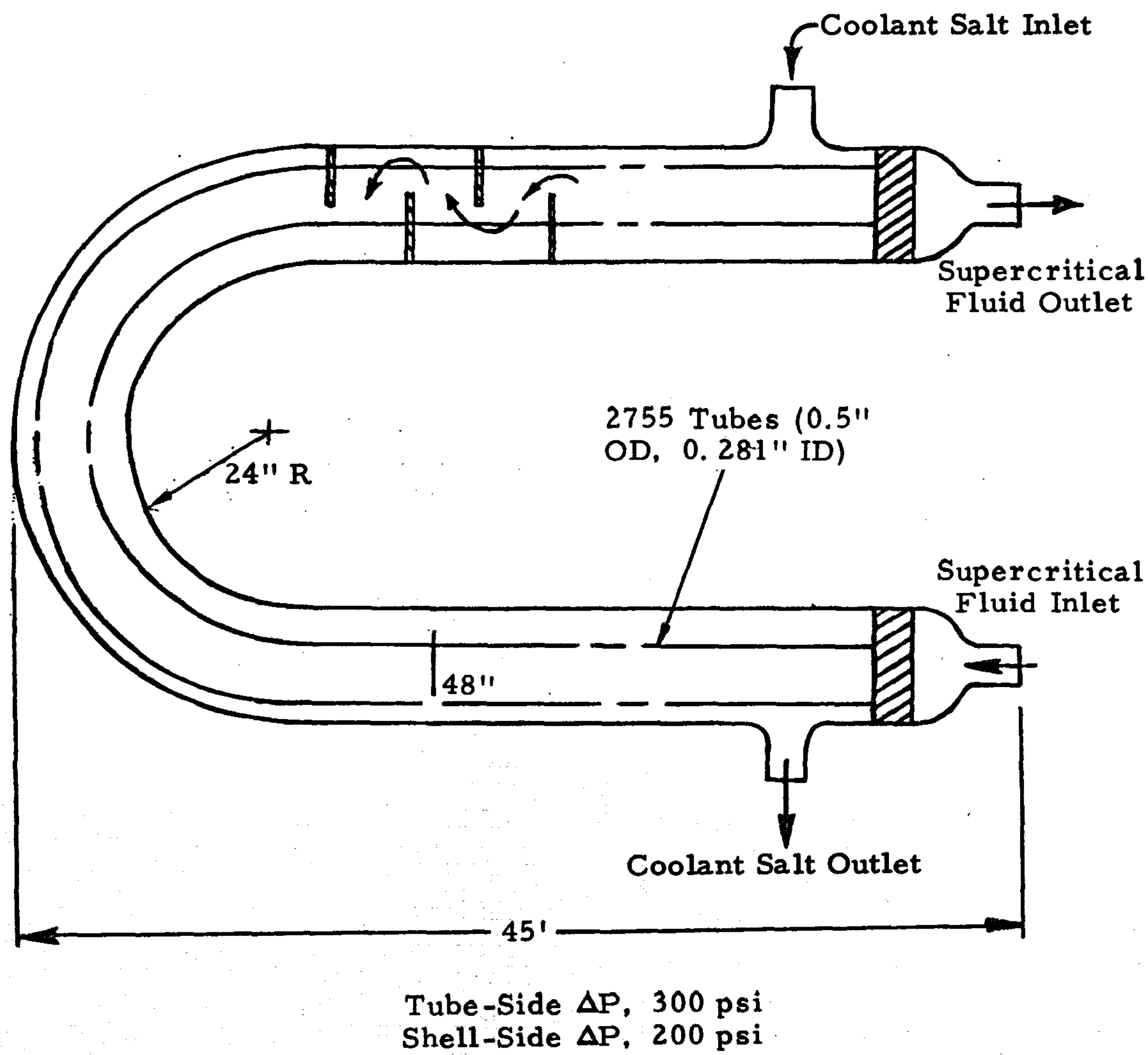


Figure 4.6: U Tube, U-Shell Concept With Hot Leg Above Cold Leg.

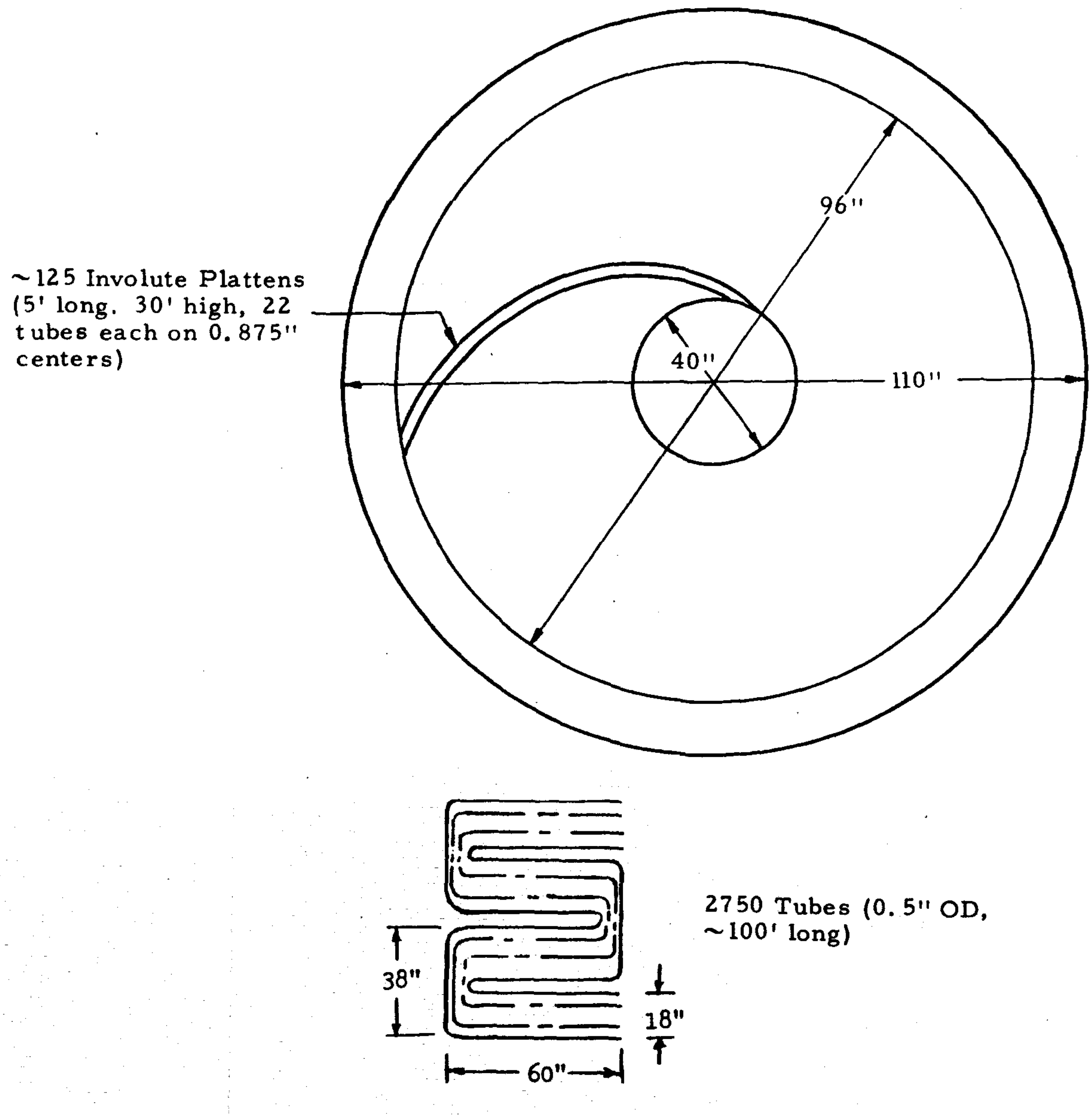


Figure 4.7: Involute Return-Bend Tube Bundle.

1) The tube bundle is quite flexible; therefore, it can withstand large thermal transients.

2) Small-size thermally efficient tubes can be used.

3) The steam generator can be designed with large varieties of envelopes.

Disadvantages

1) The manufacturing of the tube bundle is complex and more expensive.

2) The manufacturing and operating experience is limited.

3) The shell-side flow is virtually axial from top to bottom of the steam generator; therefore, the circumferential mixing on the shell side is negligible. Should there be any major shell-side flow disturbance, the entire length of the platten would be affected.

Comments

From experience it is possible to design plattens based on the results of the use of the helical-coil or straight-tube bundle. The results also indicate that the obvious trend is to use large numbers of thermally efficient small tubes; this use increases the manufacturing cost.

4.2.5.4 Vertical, Once-Through, Helical-Coil (Figure 4.8 and 4.9)

Advantages

1) The tube bundle is quite flexible; therefore, it can withstand large thermal transients.

2) With the additional parameter (helix angle), it is possible to design the steam generator to fit any reasonable envelope.

3) The chances of experiencing coolant salt flow disturbances of large magnitude in some tubes are quite limited. The tubes are coiled parallel to the shell; therefore, the water, flowing inside the tube, experiences the coolant-salt flow at various locations along the circumference. The coolant-salt flow disturbance is shared by most of the tubes.

4) Manufacturing and operating experience is rapidly increasing.

Disadvantages

1) Due to manufacturing difficulties, large numbers of thermally efficient small-size tubes cannot be used.

2) The coiling of the tubes and the manufacturing of the tube bundle are difficult.

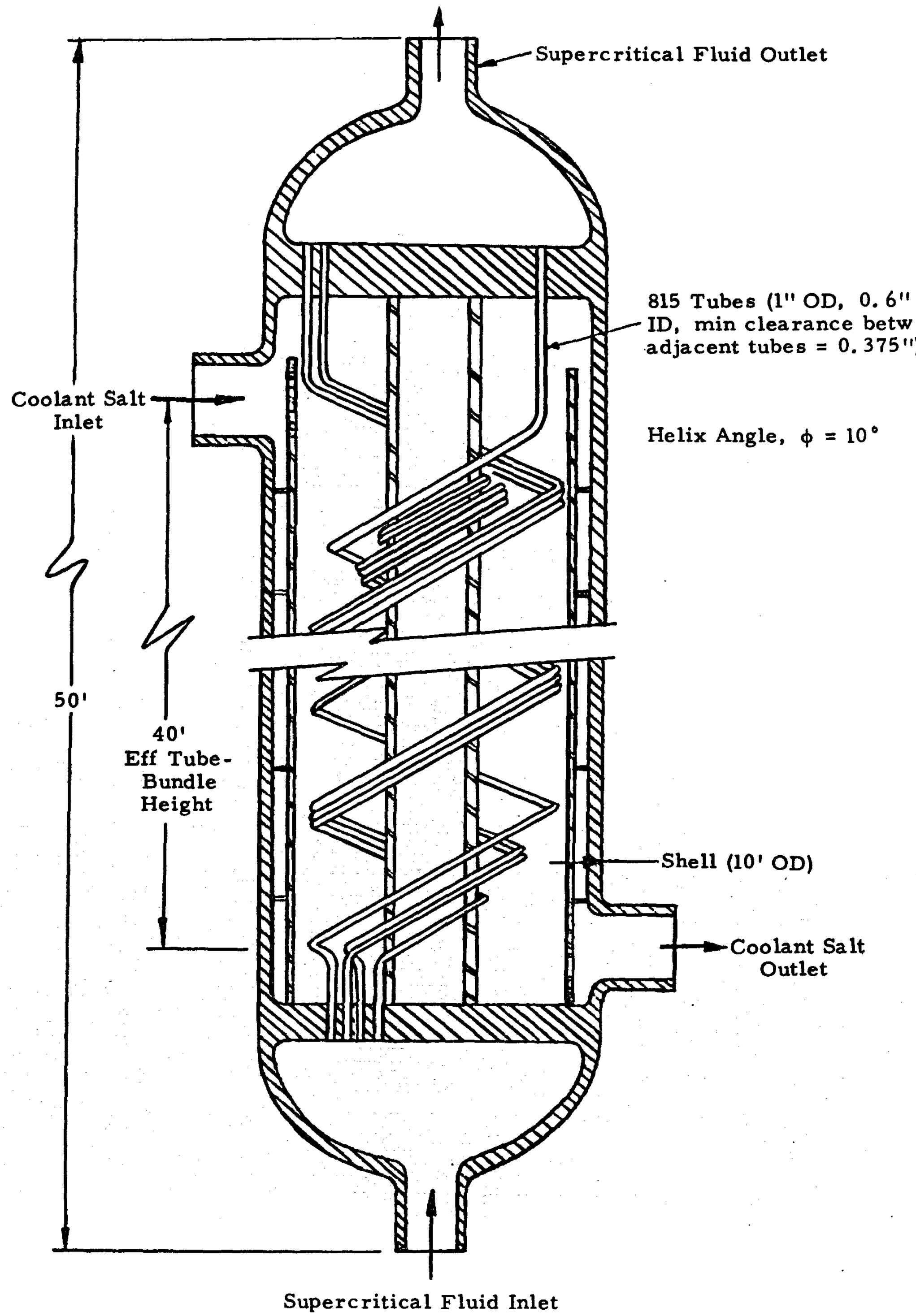


Figure 4.8: Helical Tube-Bundle Alternate Concept.

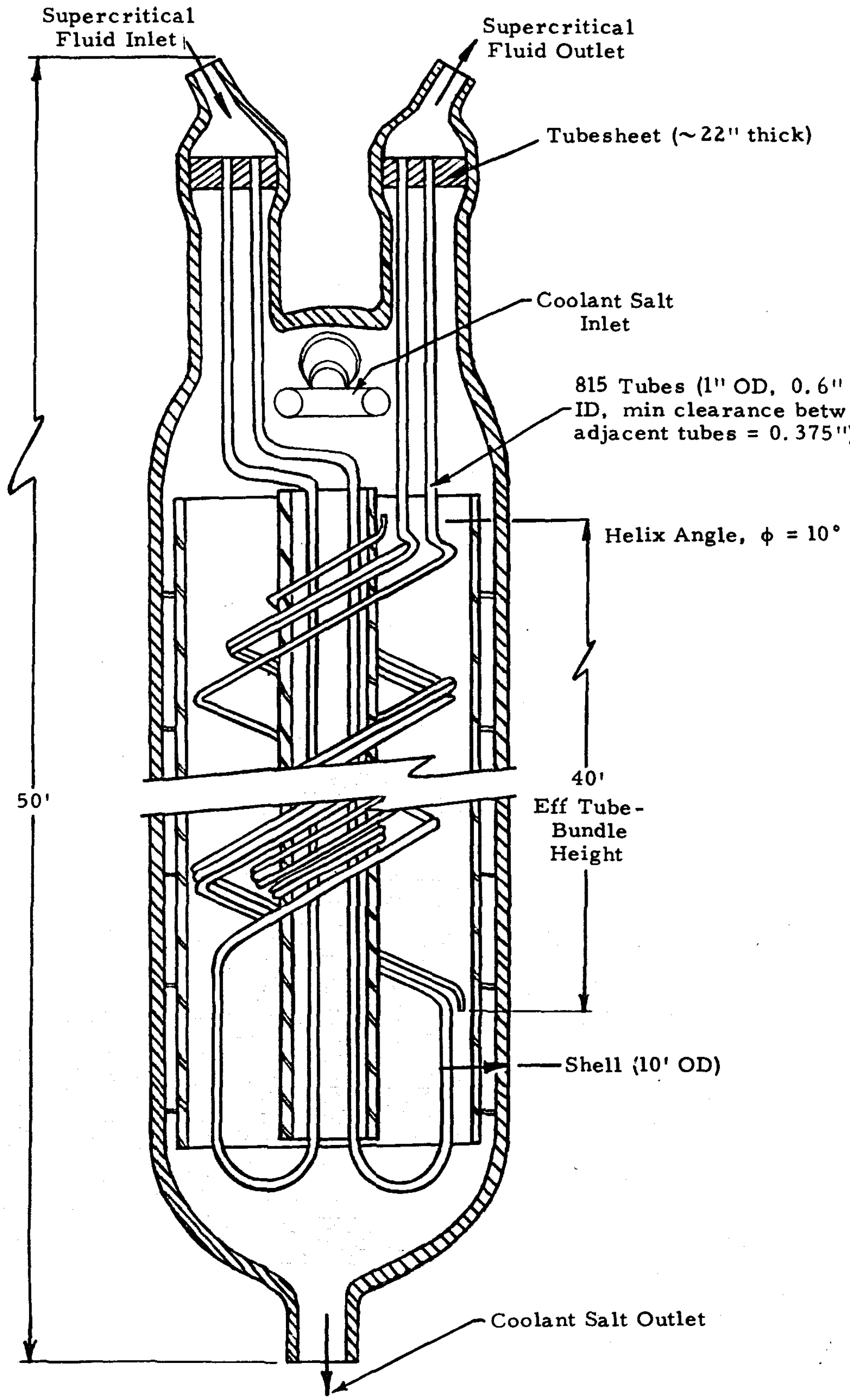


Figure 4.9: Helical Tube-Bundle Concept.

4.2.6 Reheater

The recommended reheater arrangement is identical to the steam generator except for size; it was selected for the following reasons:

1) Costs and time spans for delivery will be reduced because similar design and fabrication processing can be used for the steam generator and the reheater.
2) Development of welding mockups, stress modeling, and flow modeling will be minimized.
3) The helical-coil arrangement permits versatility in the final selection of the number of reheater units.
4) The helical-coil unit appears to be the most attractive because the selection criteria of the helical-coil unit are nearly the same as those for the steam generator.

A conceptual sketch of the selected reheater arrangement is shown in Figure 4.10. To determine the basic size of the reheater, a parametric study was performed.

A primary objective when selecting a reheater design is to provide an arrangement with a low steam-side pressure drop at minimum cost. For this study Ebasco recommended that the tubeside (steam) pressure drop be limited to 20 psi. The shell-side (salt) pressure drop was set in the range from 10 to 20 psi.

Other design parameters which were evaluated by tradeoff studies were the Hastelloy-N volume, the number of units, and the unit height and diameter. When identical designs were compared, the Hastelloy-N volume was virtually independent of the number of units. For a straight-tube unit with full-diameter tubesheets (chosen as a base for parametric study), the amount of material was constant at about 210 tons for one unit or 25 tons for each of the eight units. The helical-coil unit utilizes significantly smaller tubesheets because fewer tubes are required for a given heat transfer surface. For example, the recommended concept uses four 10 in. thick tubesheets in lieu of a single 28 in. thick tubesheet; however, this weight saving is counterbalanced by additional weight caused by using a large tube diameter (due to manufacturing considerations) and a larger shell diameter. The estimated weight range of the helical-tube reheater is 220 to 240 tons.

Four units were selected for the design because of the following factors:

1) Desirability for plant arrangement.
2) Insignificant Hastelloy-N material volume differences.
3) Operational flexibility.
4) Performance requirements met with reasonable dimensions.

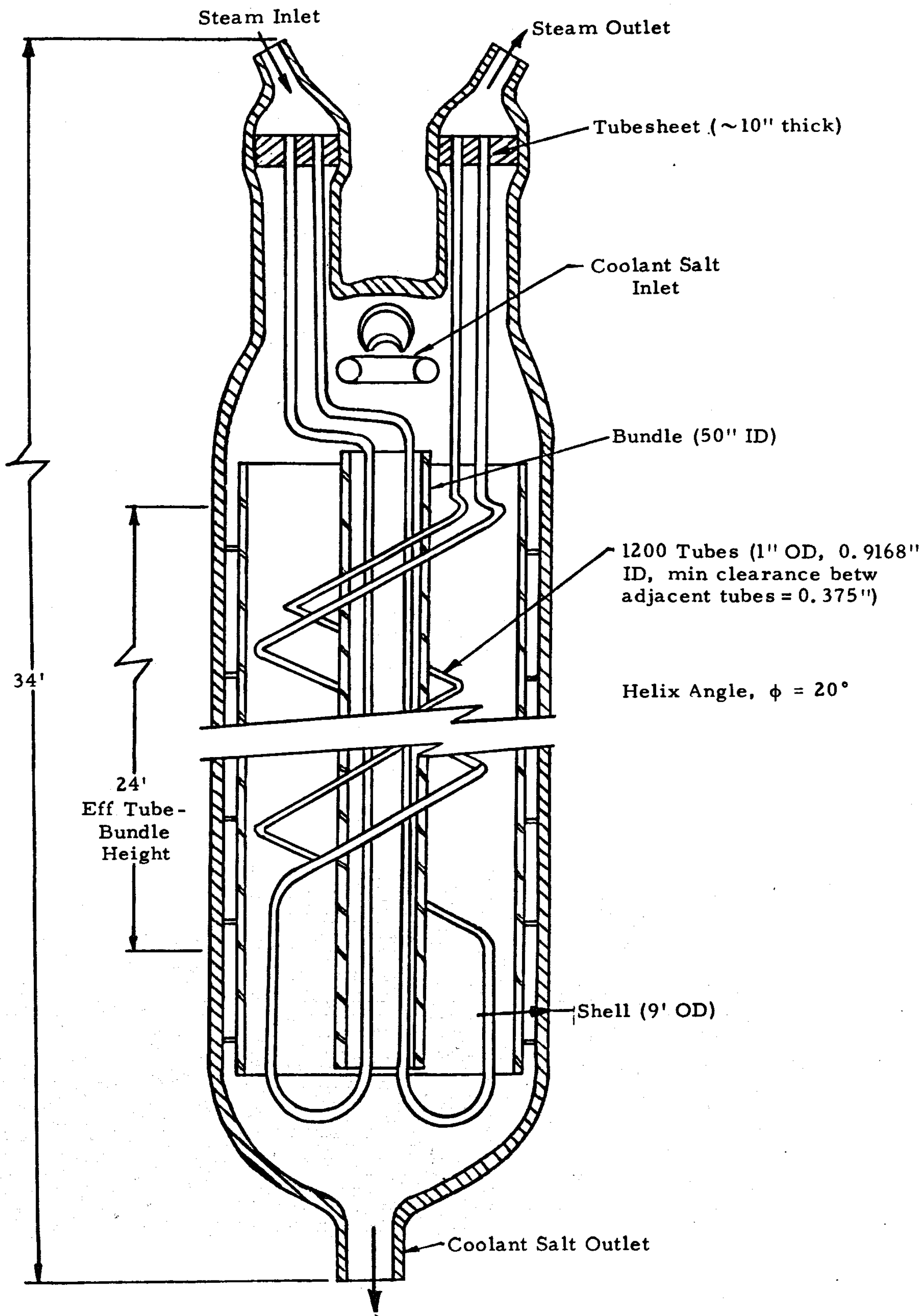


Figure 4.10: Helical Tube-Bundle Reheater Concept.

4.2.7 Conclusions and Recommendations

From an overall view of the design development, the evaluation, and the selection of several feasible steam generator concepts for the 1000 MWe MSBR reference plant design, four different concepts have been established. The first choice is the vertical, once-through, helical-coil unit, which should be developed in subsequent tasks of the design study. The concepts in order of preference are as follows:

1) Vertical, once-through, helical-coil.

2) Vertical, once-through, straight tube with sine-wave bend.

3) Vertical, once-through, return bend (platten).

4) Horizontal, once-through, U tube unit with hot leg above cold leg but in the same vertical plane.

It is recommended that these concepts be evaluated further by the designers of the system and the components to establish a mutual choice.

4.3 Primary Heat Exchanger Concept Selection

4.3.1 Background

This section discusses the criteria by which a primary heat exchanger (IHX) concept is to be judged and presents a large number of standard and nonstandard concepts with evaluations based on the criteria. The IHX concepts are evaluated according to the following breakdowns:

1) Removable Tube Bundle (Primary fuel salt in the tubes)

2) Nonremovable Bundle (Primary fuel salt in the tubes)

3) Primary Fuel Salt on Shell Side

4) Advanced Concepts

Concepts 1 and 2 are relatively standard compared, to those generally studied for LMFBR heat exchangers except that the hot primary fluid is inside rather than outside the tubes. Another difference is the smaller size of the primary plenums, which causes problems of flow distribution. The extent of the problem of the even distribution of primary fuel salt around a cylindrical tubesheet could not be determined. One (undesirable) solution for better flow distribution is additional inlet and outlet nozzles in each IHX plenum.

A much more detailed manufacturing evaluation is required to properly rate the concepts with primary salt on the shell side. Only a brief evaluation is made of the advanced concepts since these have major effects that should be evaluated by a systems engineer.

The criteria by which the concepts are to be evaluated are discussed below. The parametric charts used to select an optimized straight-tube and helical-tube bundle are included. The selected tube-bundle sizes are indicated on the concept sketches in section 4.3.3.

4.3.2 Selection Criteria

The selection criteria that were applied to various IHX concepts are primary salt inventory; thermal stresses; thermal-hydraulic performance; manufacturing; and inspection, maintenance, and repair.

4.3.2.1 Primary Fuel Salt Inventory

Designs with low primary salt volumes appear highly desirable due to the design requirement to minimize fuel inventory. In most conventional designs of heat exchangers the ratio of tube-side volume to shell-side volume is low (Figure 4.11). When using these conventional designs, the arrangement of primary fuel salt on the tube side is mandatory. For example, the reference tube bundle design chosen has 3/8 in. diameter tubes and 5/8 in. pitch (or 1/4 in. between tubes). Figure 4.11 shows that the ratio of shell-side volume to tube-side volume is about 4. The chart is based on a straight-tube bundle, but the results can be applied to most tube bundle designs.

Figure 4.12 shows a typical arrangement of parameters for a straight-tube-bundle IHX. This chart shows the tradeoff among length, bundle diameter, tube size, tube spacing, and tube-side and shell-side pressure drop. For easy comparison rough tube-side volume numbers, corresponding to each tube size, are given. The advantage of the smaller sized tubes can be easily noted when envelope and primary volume are considered.

For most tube designs, such as the sine-wave, the C tube, the hockey-stick, and the J tube, the envelope of the heat transferring zone is about the same as that for a straight bundle. There will be small differences, but most designs are assumed to have 3/8 in outside-diameter tubes on a 5/8 in. pitch, and the salt volume penalty will depend on the bundle ends and not on the main heat transferring region.

The helical-tube bundle is different. It would be extremely difficult to use 3/8 in. diameter tubes, as the bundle height/diameter ratio would be quite small. Accordingly, the helical-tube design is heavily penalized on a volume basis.

4.3.2.2 Thermal Stress

The stress criterion, which is used as a measure of the adequacy of the various IHX designs, is the magnitude of the tube stresses caused by differential thermal expansion. It is not immediately obvious why this condition exists. When stress analysis is performed, tubesheets, shells, nozzles,

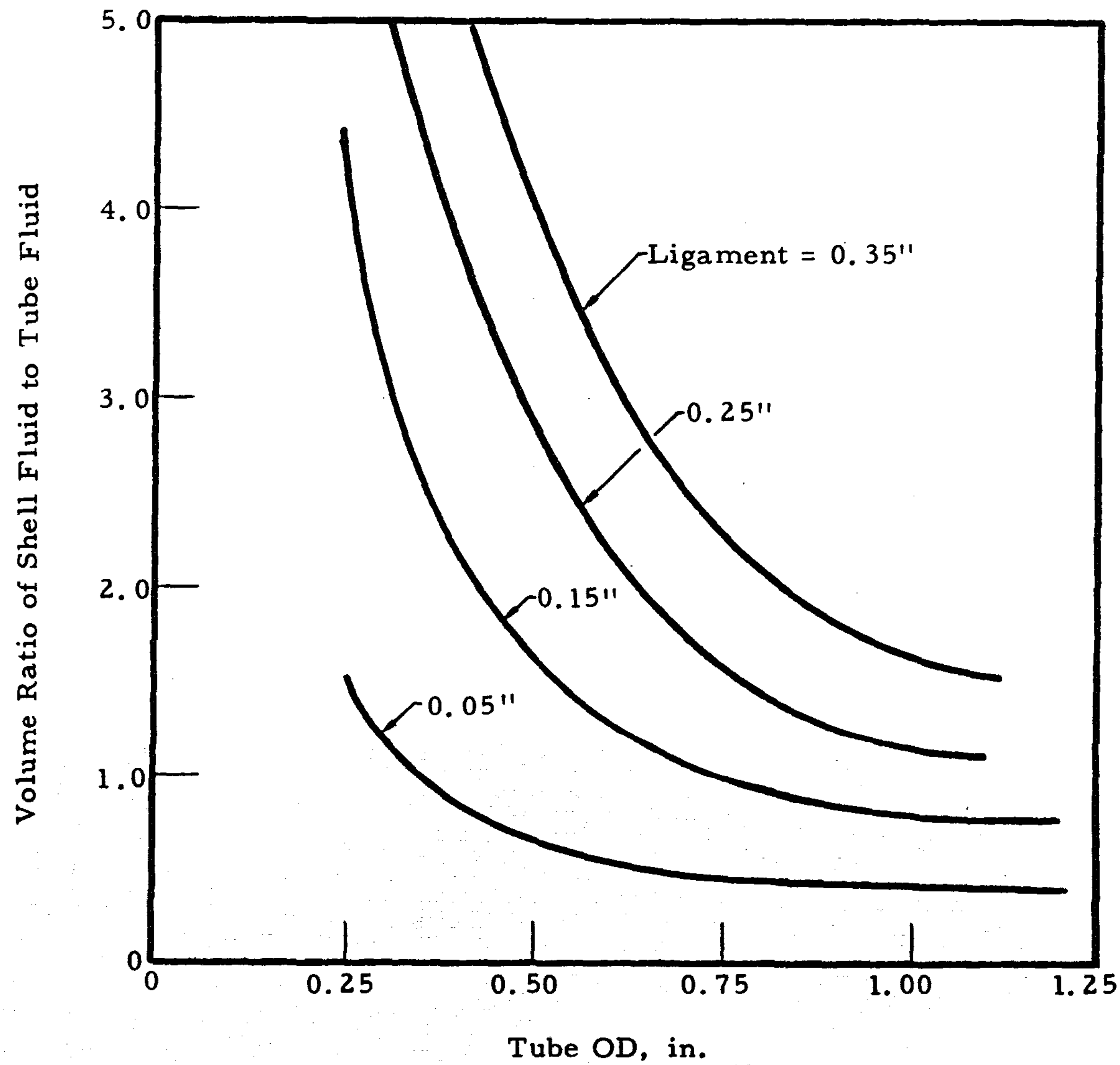


Figure 4.11: Straight-Tube Bundle, Ratio of Shell-Side Volume to Tube-Side Volume.

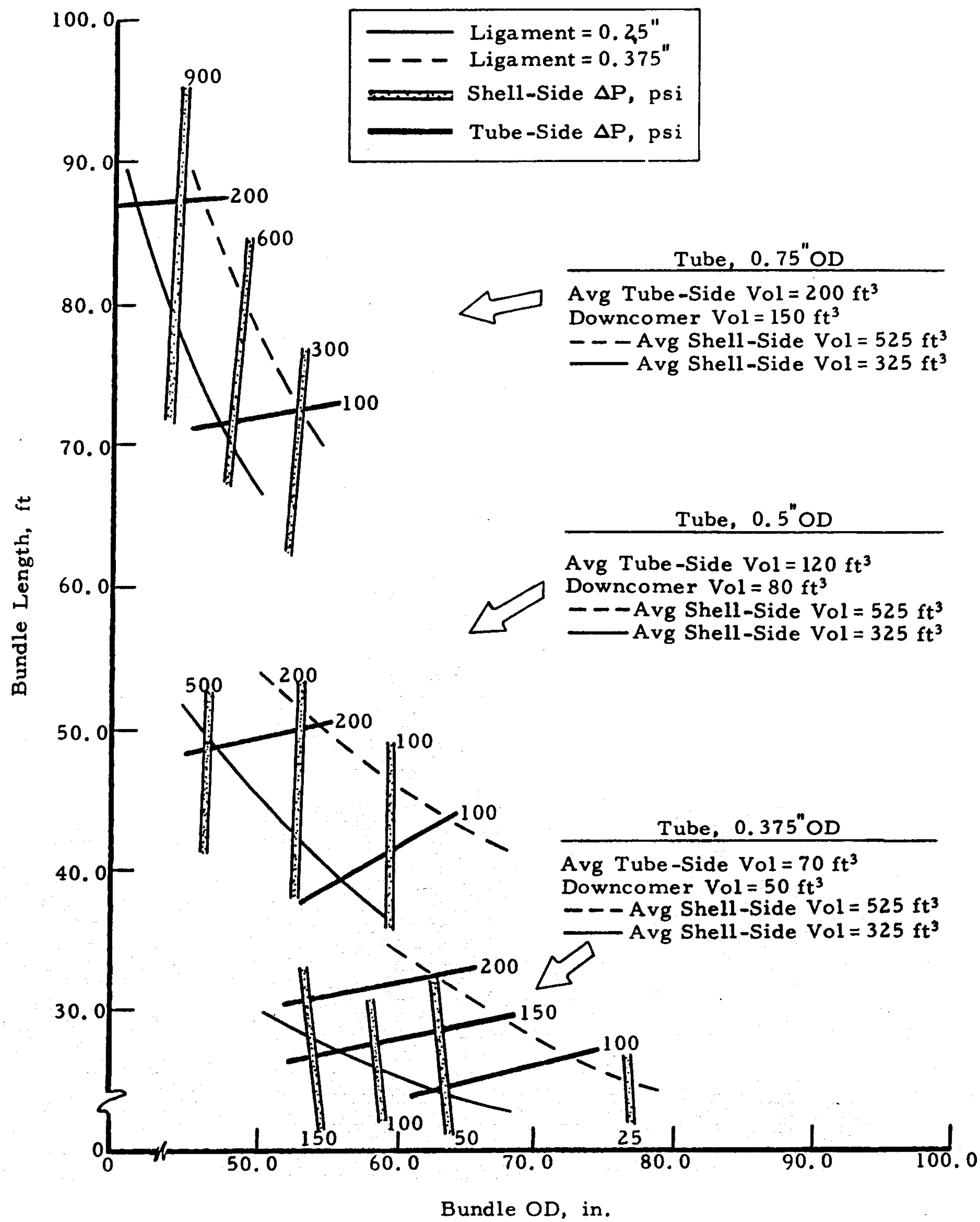


Figure 4.12: Straight-Tube Bundle IHX, Parametric Chart.

and heads should also be considered. Preliminary analyses of these components indicate that the thicknesses, diameters, and shapes of these are within state-of-the-art technology. Flat, circular tubesheets will be relatively thick (9 to 13 in.) to reduce stresses, but they are not a major factor in limiting the basic IHX design. Shells, nozzles, and headers will not be overly thick (1 to 3 in.), despite the high design temperature, because the design pressures are low. Thick sections should be avoided because they are subject to high radial temperature gradients during transients, and the number of cycles of these transients fatigue the metal.

Supports are not a problem. Support skirts for bottom or top support are easily designed; hanger rods for top or middle support are also possible, and all these can be used to isolate the structure around the IHX from the 1300°F temperature of the unit.

The stress problem, which has not been solved easily, is that of secondary stresses in the tubes. These are caused by differential thermal expansion between the tubes and the shell joining the tubesheets. If we assume that the primary fuel salt flows in the tubes to minimize primary inventory, the average temperature of the tube wall will be between the average temperature of the primary salt (1300 + 1050)/2 = 1175°F, and that of the secondary salt (1150 + 850)/2 = 1000°F. If a linear axial gradient is assumed, the average temperature of the tube wall will be (1175 + 1000)/2 = 1087.5°F. In most designs the shell that joins the tubesheets is insulated on one side; so its average temperature is the temperature of the shell-side fluid, which is 1000°F. Calculations show stresses of $E\alpha\Delta T \approx (26.3 \times 10^6)(8.4 \times 10^{-6})(87.5) = 19250$ psi in a straight-tube bundle when the tubes are 87.5°F hotter or colder than the shell. The allowable stress intensity range for Hastelloy is 10500 psi (3Sm at the 1300°F, maximum tube metal temperature). Consequently, the steady-state thermal stress or stress intensity is almost twice the amount which is allowed for all thermal conditions combined. Note that the ASME Code Section III definition of differential thermal expansion stresses as secondary stresses is used. Mechanically induced stresses, such as pressure and vibration stresses, are not negligible, but these are not as significant as the thermally induced stresses.

The straight-tube design with opposed tubesheets, either fixed or floating, is not feasible under the given steady-state temperature differences; however, there are a few basic solutions applicable to the IHX to reduce tube stresses. There are three methods commonly used to allow for differential thermal expansion between the tubes and the shell joining the tubesheets. These are discussed as follows.

The two methods that seem applicable are flexible tubes and bypassing-flow along the shell. Some flexible tube shapes which are considered are the U tube, the sine-wave, the C tube, the helical, the J tube, and the hockey-stick. The tube shapes which appear useful to the 1000 MWe MSBR, their sizing, and their manufacturing problems have been discussed previously. Using these tubes would produce smaller secondary stresses than those produced in straight tubes.

The second method generally involves bypassing a small percentage of the tube-side fluid along the surface of the shell that connects the upper and lower tubesheets. The rate and amount of flow are controlled so that the axial temperature gradient in the shell nearly matches that in a typical tube. The object is to minimize the difference between the thermal growth of the shell and that of the tubes; therefore, the tubes need not be very flexible to absorb the small differential growth that does occur. Straight tubes in a fixed bundle with this bypass can be considered if the connnecting cylinder responds to thermal transients as fast as the tubes respond. In comparison to bent tubes, straight tubes are desirable due to the manufacturing simplicity and size economics.

The third method, one that has been used in the past but is not developed in any detail for the MSBR component design study, is the use of the flexible shell. The most obvious design for a flexible cylinder is to include bellows, such as those used in the shells of the Hallam Nuclear Power Facility IHXs. Currently, the major objection to the use of bellows is their reliability and maintainability relative to the alternative designs mentioned previously. Experience has shown that the bellows are susceptible to loss of integrity if they are cleaned improperly or if the surface is not entirely free of scratches and dents. For lack of other options, bellows are sometimes used in nuclear plant applications, but even then they are generally small-diameter bellows located in accessible areas where their failure would not produce a major safety hazard.

We recommend a straight-tube IHX design to suggest what might be done if large, reliable bellows could be obtained in the next several years, although it appears that the advantage gained would not be a great one. There are other problems, such as remote maintenance methods and high-temperature material behavior, that are more worthy of immediate development.

Some tentative conclusions can be made about the thermal stresses expected in the various tube shapes which were considered. The steady-state thermal stresses have been calculated for several of these and are presented in Appendix D. An assumption made for each tube shape is that the total desplacement of the tube is the same as the free differential thermal expansion of the shell.

The types of tubes considered are the straight, the sine-wave-bend, the hockey-stick, the C tube, and the hockey-stick/sine (ORNL IHX concept).

The straight tube without flow bypass is overstressed due to differential thermal expansion, and this condition alone is sufficient to remove it from consideration. Although the transients have not been specified for the MSBR, it appears that the straight-tube bundle with flow bypass could not survive the transients because of the low allowible stress at 1300°F.

The stress levels in the other tubes are roughly the same, but the hockey-stick, the C tube, and the hockey-stick/sine are disadvantageous because they need a vessel with a significantly larger diameter to accommodate the tubes.

Stress analysis of the tube during transients can be done using standard techniques; however, the stress analysis of tubes with a C, J, or hockey-stick shape will be difficult at the bend region due to the uncertain effect of tube support systems.

4.3.2.3 Thermal-Hydraulic Performance

Experience with heat exchangers operating at high temperatures where flow maldistribution could cause severe temperature maldistribution and high stresses has led to the development of vertical shell-and-tube heat exchangers. Other advantages of the system also contribute to the selection of the vertical arrangement and are mentioned elsewhere. As with the steam generator, a basic criterion is the hot primary fuel salt flowing downward over the heat transfer surface in countercurrent to the secondary coolant salt flowing upward. This type of flow arrangement ensures that flow and temperature imbalances, which may occur at part-flow conditions, will be minimized.

Other design criteria which are related to the thermal and hydraulic performance of the heat exchanger are as follows:

1) Provision for tube support designs that permit adequate support against flow-induced vibration and ensure good flow distribution.

2) Provision for inlet plenums or special baffles to aid in initial distribution of primary and secondary salt.

To evaluate general trends in unit performance, it is best to start with the straight-tube bundle because it is geometrically simple. The straight-tube bundle, in a sense, represents one end of a scale. At the other end we have the helical unit, which represents almost pure crossflow as opposed to the straight-tube axial flow. The general trends can be formulated for all other bundles by evaluating these two types of tube bundles.

A general parametric study for the straight tube is shown in Figure 4.12. Each of the three regions represents one tube size with ligaments of 0.25 in and 0.375 in., respectively. Across the regions are plotted the lines of constant pressure loss for the tube side and the shell side. To facilitate the establishment of general trends, some values of salt inventory have been averaged over small ranges. Obviously, the tube-side salt inventory changes with bundle diameter; but for the range of the plots they were averaged within 25 percent of actual values.

The smaller tube sizes offer large reductions in bundle length while maintaining similar ΔP values. This condition reduces both shell-side and tube-side volumes significantly. Another significant feature is that the minimum bundle diameter is fairly well fixed at about 55 in. To provide for clearances and shell thickness, the minimum shell OD should be about 65 in. The maximum bundle OD is about 70 in.; this makes a shell OD of about 80 in.

Currently there does not appear to be any reason for exclusion of the small-diameter tubes (a final economic tradeoff would determine exact values); therefore, bundle lengths smaller than 30 ft are reasonable, and this would imply an overall unit length of about 40 ft. The large number of smaller tubes may cause high manufacturing costs plus maintenance problems; therefore, B&W considered the possibility of 0.5 inch diameter tubes, which would add about 20 ft, thus offering overall lengths of about 60 ft. This unit was later found to be too long and to add significantly to the primary salt inventory.

Some potential problems with small tubes are supports, buckling, and tube plugging. First, the number of supports may be determined by the natural vibration frequency required. Figure 4.13 shows that small-diameter tubes require smaller support lengths, which increase costs and pressure losses. Second, the buckling characteristics of small tubes (say, during a thermal transient) usually require very close support spacing. This requirement may be a tighter restriction than that of the support spacing for natural frequency evaluation. Third, remote tube-plugging methods have not been developed for the tube ODs less than 0.625 in.

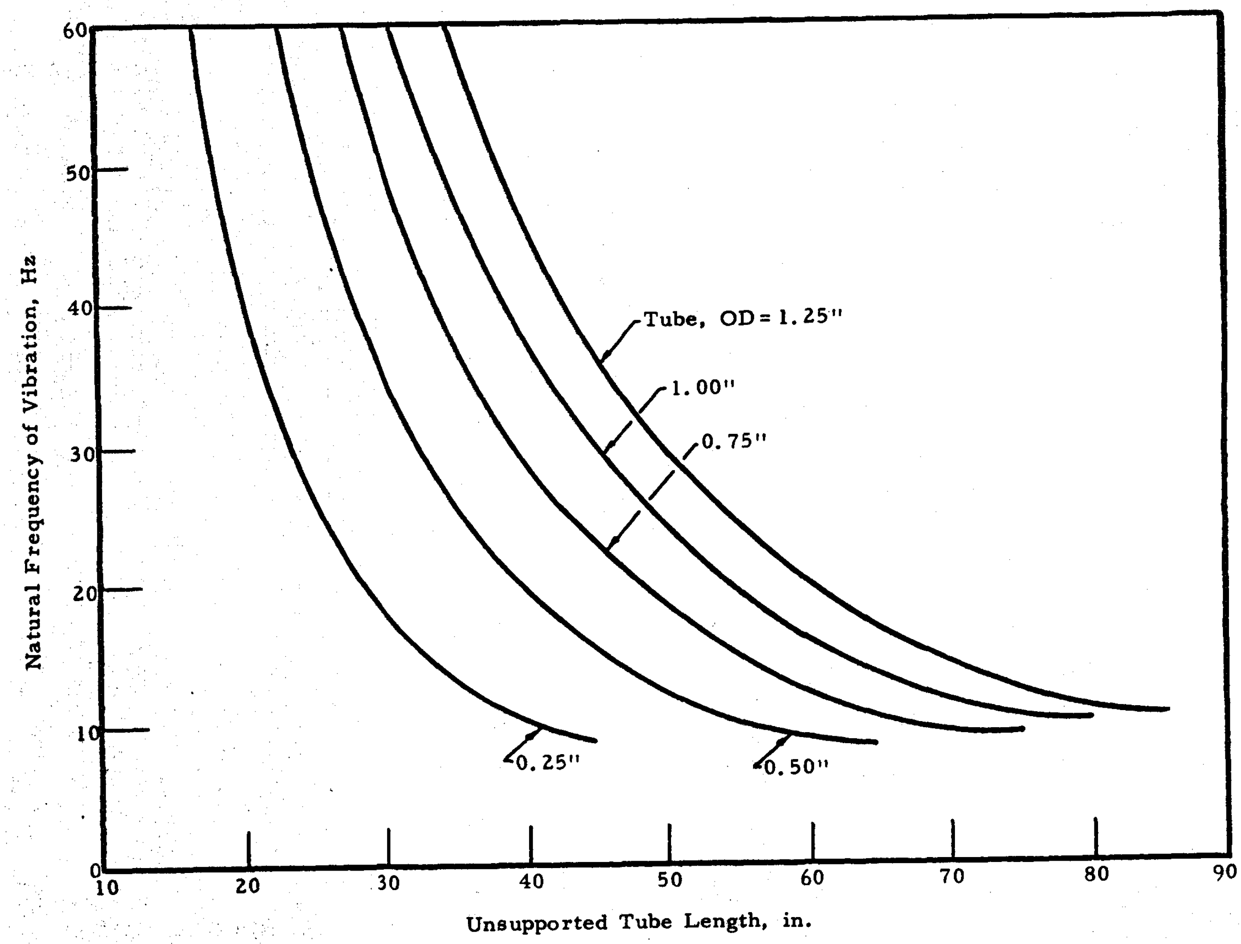


Figure 4.13: Span Length of Tubes vs. Natural Frequency.

A parametric study for the helical bundle (which is the only arrangement with significantly different performance from the straight tube) is shown in Figure 4.14. It is difficult to extract general conclusions due to the complex relationships within the bundle; however, with experience some generalizations can be made.

The major problem is that the tube-side pressure loss is relatively high even at the large diameters. Smaller tube sizes would decrease the length but not the diameter. The graph is drawn with the constraints of the minimum distance between tubes in the bundle being 0.375 in and the helix angle being 20 degrees. The diameter of the tube bundle could be reduced by possibly packing the coils closer. The tube ΔP could be decreased by adding more tubes, and the diameter could be decreased by increasing the helix angle. Only a more complete tradeoff study could delineate

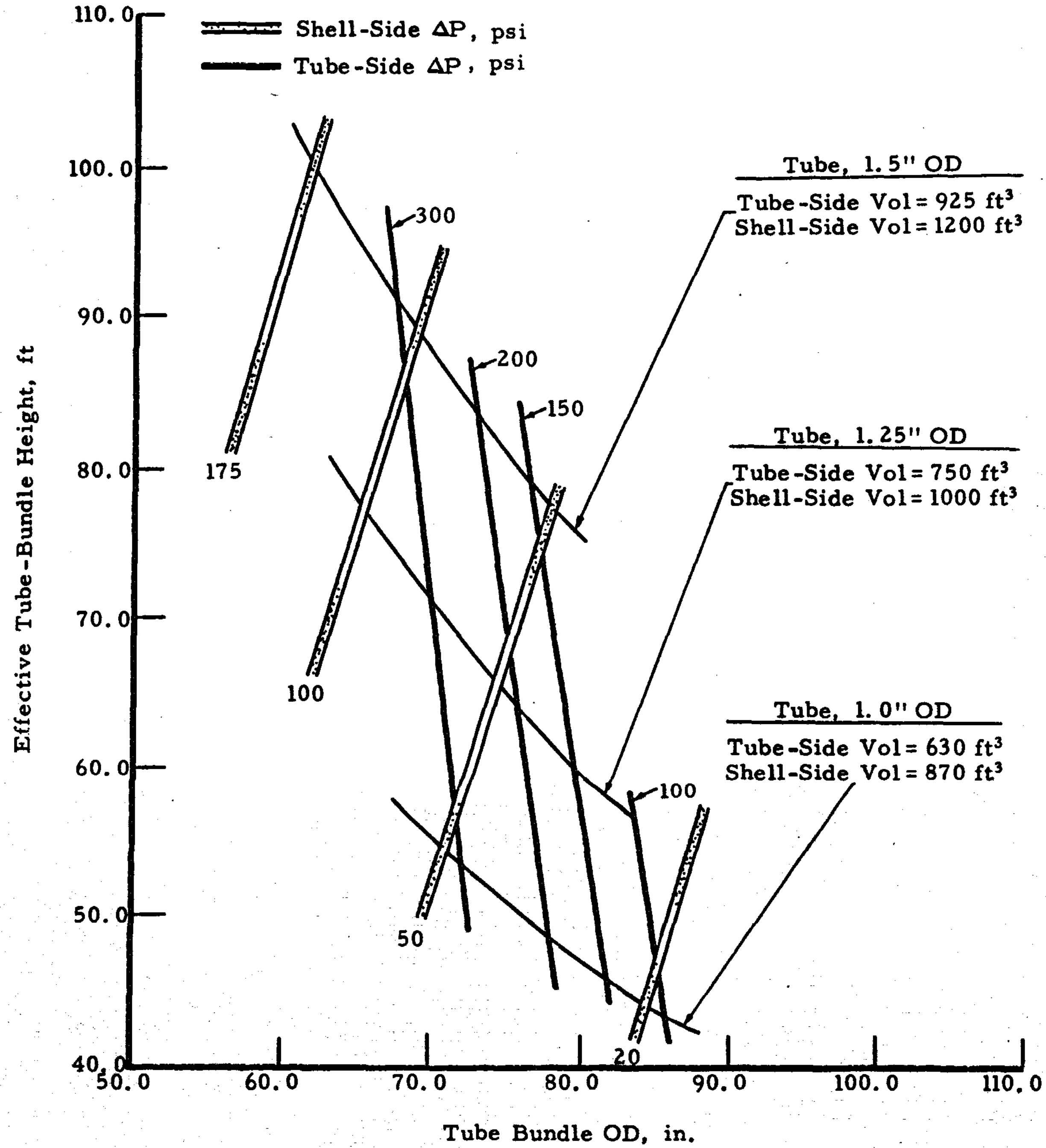


Figure 4.14: Helical Tube-Bundle IHX, Parametric Chart.

these effects.

Methods of improving heat transfer have not been actively pursued at this design stage. The use of studs or small fins could improve the heat transfer, but the pressure losses could influence the results greatly. Since these features can be used for any tube design, it would seem prudent to pursue the analysis at a later stage.

4.3.2.4 Manufacturing

Each design has been studied to determine the degree of difficulty in manufacturing the components. In nearly all cases the problems are the assembly of the tube bundle and the method of tube support.

One major question has been the smallest possible tube size which can be assembled and welded. Currently the smallest tube size used at B&W is 5/8 in.; however, it is felt that with development this size could be reduced to 3/8 in. A further reduction may be possible (say, to 1/4 in.) by 1980. Due to the large reduction of unit volume, the units have been sized on the basis of 3/8 in. OD tubes.

The method of tube support preferred by B&W for the more conventional types of tubes is the broached plate or drilled plate shown in Figure 4.15. This arrangement provides support to each tube and is easily made. The results of preliminary laboratory tests of the vibration/wear characteristics are highly promising. This tube support method has also been adopted in PWR steam generators.

In many instances the plate cannot be used; for example, in the assembly of a sine-wave tube bundle the tubes cannot be supported with plates in the region of the sine-wave, although they may be used elsewhere along the tube. The sine-wave section may be supported by wrapping a metallic band around each row of tubes.

There are potential problems with the C tubes, the hockey-stick tubes, the J-tube bends, and the hockey-stick/sine tubes. The C tubes can be pushed into the tubesheets, but there is no obvious method of support. Currently it appears that the best technique is placing the tubes in circular spacing and banding each ring of tubes as it is placed. The J-tubes and the hockey-stick tubes have a bend support problem similar to the sine-wave tube bundle. The hockey-stick/sine tubes can be assembled by stacking the support plates on the bottom tubesheet and bending the tubes to fit into the top tubesheet (cantilevered from the bottom tubesheet). After all tubes are in position, the support plates may be raised (possible assisted by vibration) until they are spaced properly.

4.3.2.5 Inspection, Maintenance, and Repair

The impossibility of direct contact maintenance at any time after the reactor has been in operation, and related requirements of inspection and repair, are major considerations in developing and se-

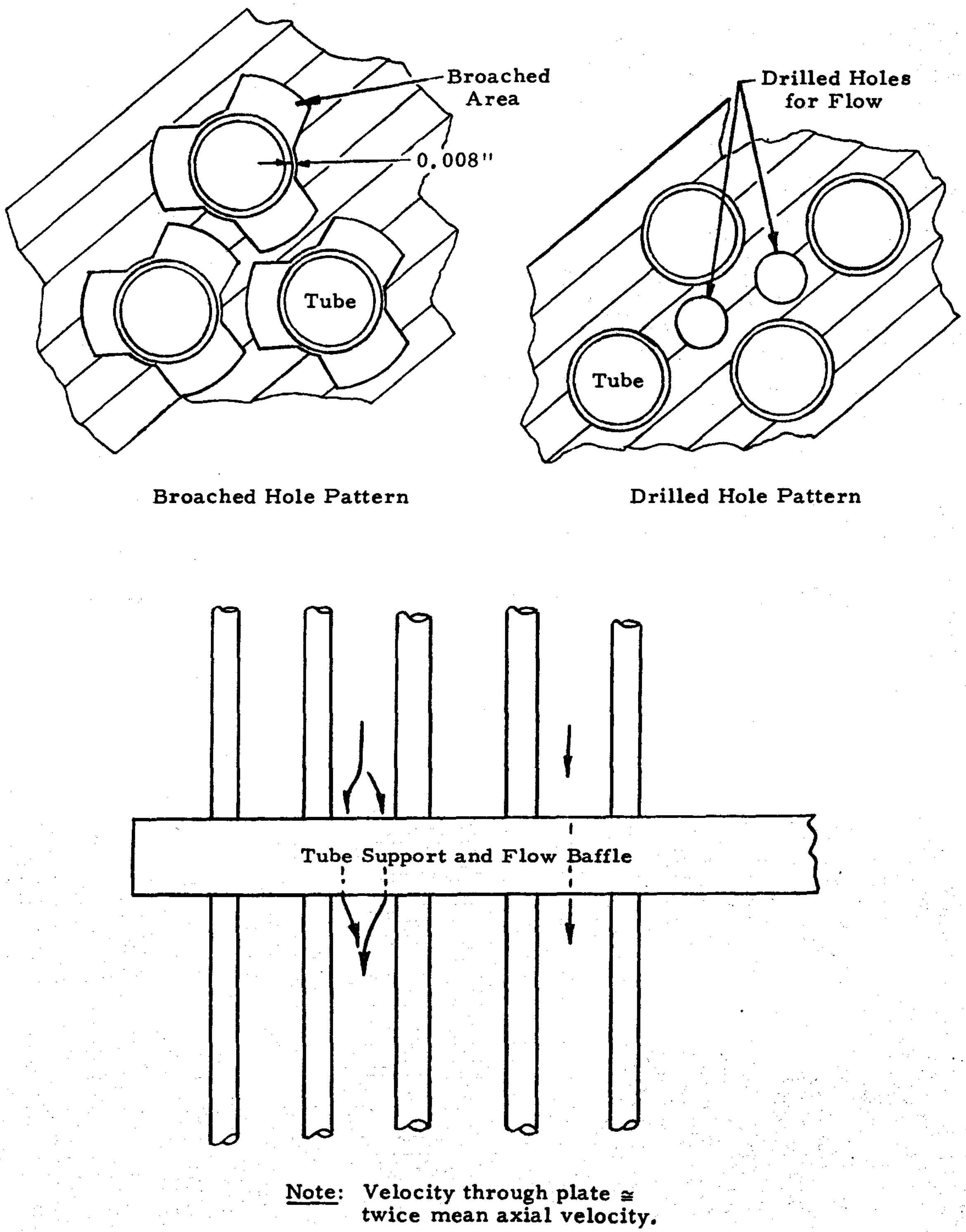


Figure 4.15: Broached or Drilled Plate Baffle.

lecting the recommended IHX designs. Maintenance can be performed on the IHX in one of three following ways:

1) The unit or tube bundle can be removed to a hot cell facility which has been set up for maintenance on various components of the reactor plant.

2) The heat exchanger or tube bundle can be removed to a decontamination facility; it can be decontaminated and cleaned, and maintenance can be performed using standard techniques.

3) In-place remote maintenance can be performed if access is provided to the areas which are likely to be inspected or maintained at some time during the plant design life.

Currently the trend in the light water reactor industries for heat exchangers and steam generators is not to remove the tube bundle but to provide hand holes and access space to ensure repair, maintenance operations, or inspections with minimum exposure to radiation. The trend for the liquid metal fast breeder reactor (LMFBR) plants (either already constructed or in the design stage) is removable tube-bundle designs for heat exchangers and steam generators. This method has been made possible by some inherent advantages of the LMFBR design because the components are not as thick and massive as those used in water reactors. These components are not anywhere near the weight of those used in the large PWR steam generators; therefore, provision for tube-bundle removal is not difficult.

The reliability of the first generation plants with high design temperatures, potential sodium-water reactions, and unproven heat exchanger designs is of concern; however, the technology required for the MSBR heat exchanger design is an extrapolation which can be easily made as more heat exchangers are designed and operated in high-temperature environments. With this expected experience, providing for a removable tube-bundle design may not be necessary because by 1980 a fixed tube bundle will be a conservative design arrangement due to technological changes and advancements. The component manufacturer must realize that prospective purchasers of this equipment may prefer a removable tube-bundle design, for high reliability and availability can be achieved perhaps with the use of a spare tube bundle. Current estimates of the cost of outage of a 1000 MWe reactor plant range from $100,000 to $200,000 per day depending on the location of the utility in this country. These figures are based on the 1971 dollar value; at these rates the cost of a spare tube bundle may be insignificant to the plant operator.

B&W does not believe that the choice between a removable or non-removable tube bundle is clear-cut, and consequently the major goals for the removable and non-removable bundle designs were considered. The considerations for maintenance were applied to each design concept during the concept evaluation, and the criteria for judging each design are as follows:

1) Provisions for meeting the requirements of the ASME Code Section XI — In-Service Inspection of Nuclear Reactor Coolant Systems — although this code presently applies only to light-water-cooled and moderated reactor systems.

2) Provision for access to flanges, seal welds, or other joints which must be broken for maintenance on the heat exchanger.

3) Provision for convenient access to the tube ends and tubesheets.

4) Provision for removal of tube bundle with minimum amount of piping or shell cuts.

5) Considerations for access in plenums, downcomers, or other areas where remote tooling may be utilized for maintenance or inspection.

6) Consideration of remote tube-bundle inspection and plugging equipment.

7) Provision for arrangement of the tube bundle to ensure convenient draining of the radioactive primary fluid and also the secondary fluid, if necessary.

Maintaining a radioactive system has been demonstrated successfully for the Molten Salt Reactor in the MSRE. During the operation of this reactor it was determined that remote maintenance was feasible by using fairly simple tooling. The equipment manufacturer should work closely with the system designer to accommodate simple tooling or any specialized fixtures that would be required to perform remote maintenance. In some cases, however, the method of remote maintenance, utilizing the techniques that will be developed from now to 1980, may be limited by some of the design goals for the IHX. In each concept developed with primary fluid on the tube side, one major design goal has been to minimize the tube diameter consistent with low primary salt inventory. The trend of smaller and smaller diameter tubes necessitates a re-evaluation of the potential of remote-tube-plugging techniques which are currently being developed in the industry. A specific technique is the explosive-tube-plugging method being developed by B&W and other component manufacturers. This method entails the detonation of an explosive plug within the tube to seal off any leaks between the tube and the tubesheet or within the tube itself. The detonation causes the tube to weld to the tubesheet or to the plug, depending on which area is being sealed off; to accomplish proper welding there must be a sufficient amount of explosive charge in the tube. Presently, it is questionable if this technique can be used with the small-diameter tubes which fulfill the design objective of minimizing the primary salt inventory.

All designs being considered are envisioned to provide good access to the tubesheets; however, some designs are inherently more flexible in accommodating different maintenance techniques. For example, the cylindrical tubesheets, which could be utilized to conserve the amount of Hastelloy material, are very difficult to reach for plugging tubes or for performing any inspections. In other cases it is necessary to require a long reach of remote tooling, which then bends upward to gain access to a lower tubesheet. This approach is more difficult than an approach where the tubesheet can be reached directly through a seal-welded hand hole.

For further discussion on the IHX design concepts, see section 4.3.3.

4.3.3 Concept Evaluation

1. Removable Tube Bundle

The following data pertain to the optimized straight-tube bundle selected by B&W for this study of the MSBR IHX, and the numbers apply to all concepts (except the helical-tube bundle and the ORNL IHX concept) when the primary is in the tubes.

Number of tubes	7000
Tube OD, in.	0.375
Tube ID, in.	0.259
Pitch (in heat transfer area), in.	0.625
Bundle OD, in.	
With no downcomer pipe	55.5
With 18-in. downcomer pipe	59.0
Effective heat transfer length, ft	26
Primary ΔP, psi	125
Secondary ΔP, psi	75

Some units require 5 to 20 percent more volume in the tubes because of tube shape, and this requirement is discussed for each unit. The minimum amount of primary salt inventory which is required is as follows:

Tubes, ft3	65
Two plenums, ft3	55 to 60
Downcomer (if used for primary fluid), ft3/ft of length	1.75

The ratings of the five main removable concepts are presented in two ways: first, a full-size sketch follows a list of the advantages, disadvantages, and conclusions for each concept; second, the evaluations are reduced to letter ratings in a simple table (see Figure 4.22). The same process is used in the following section; however, the ratings of the two sections are not intended to be compared. The J-stick and the hockey-stick shapes are included in that section because these are not removable types. A concept is selected for each group.

The evaluations should indicate that a choice is not simple for either of the two groups; however, after considering all criteria, we recommend the ORNL IHX design from the removable tube bundle design group.

4.3.3.1 Removable Straight-Tube IHX With Flow Bypass (Figure 4.16 and 4.17)

Advantages

1) The design and the construction are simple.
2) The tube thermal stresses are zero at steady state.
3) Easy access to the bottom tubesheet is possible (through tubes or downcomer) for repair, inspection, or plugging.

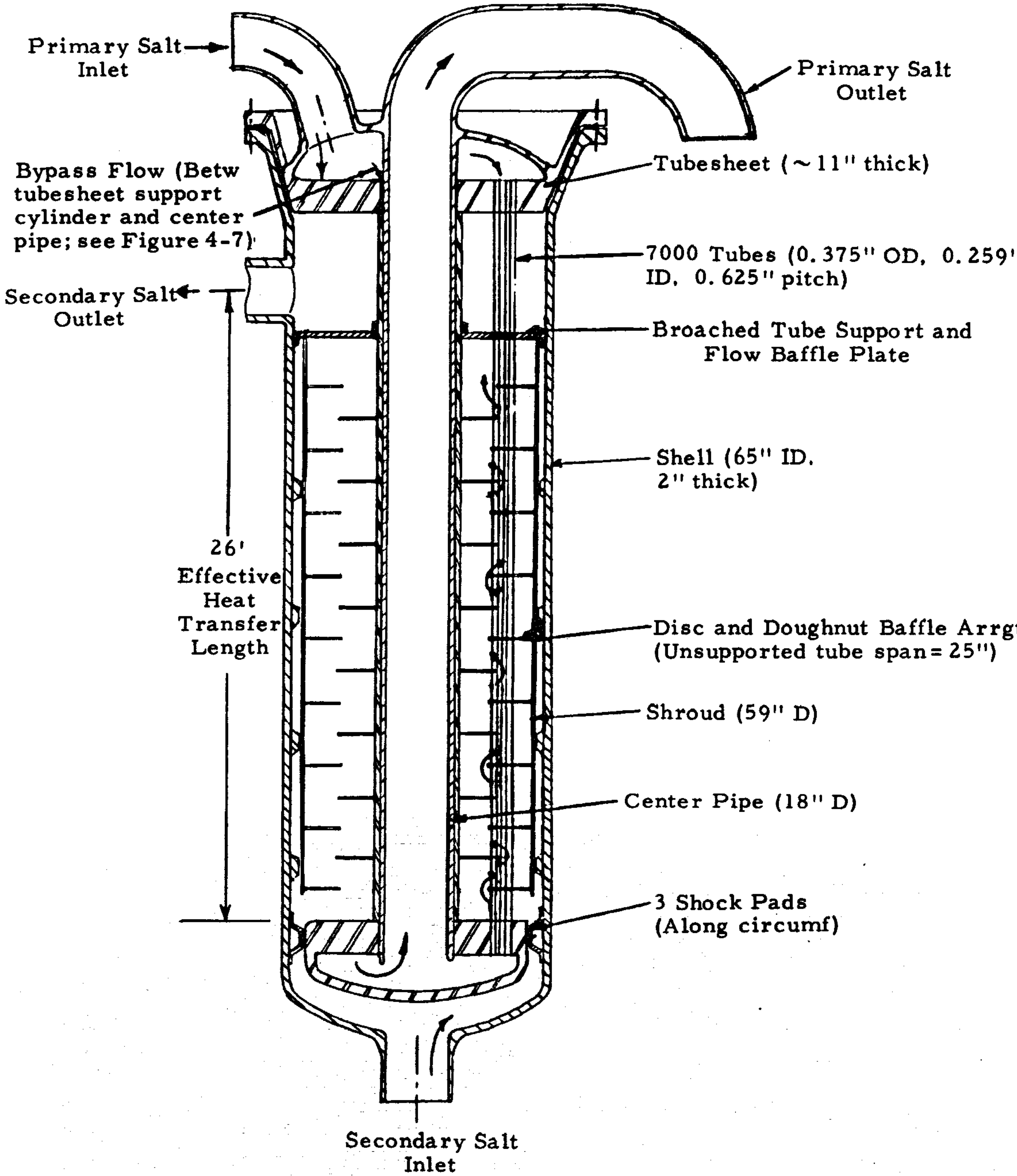
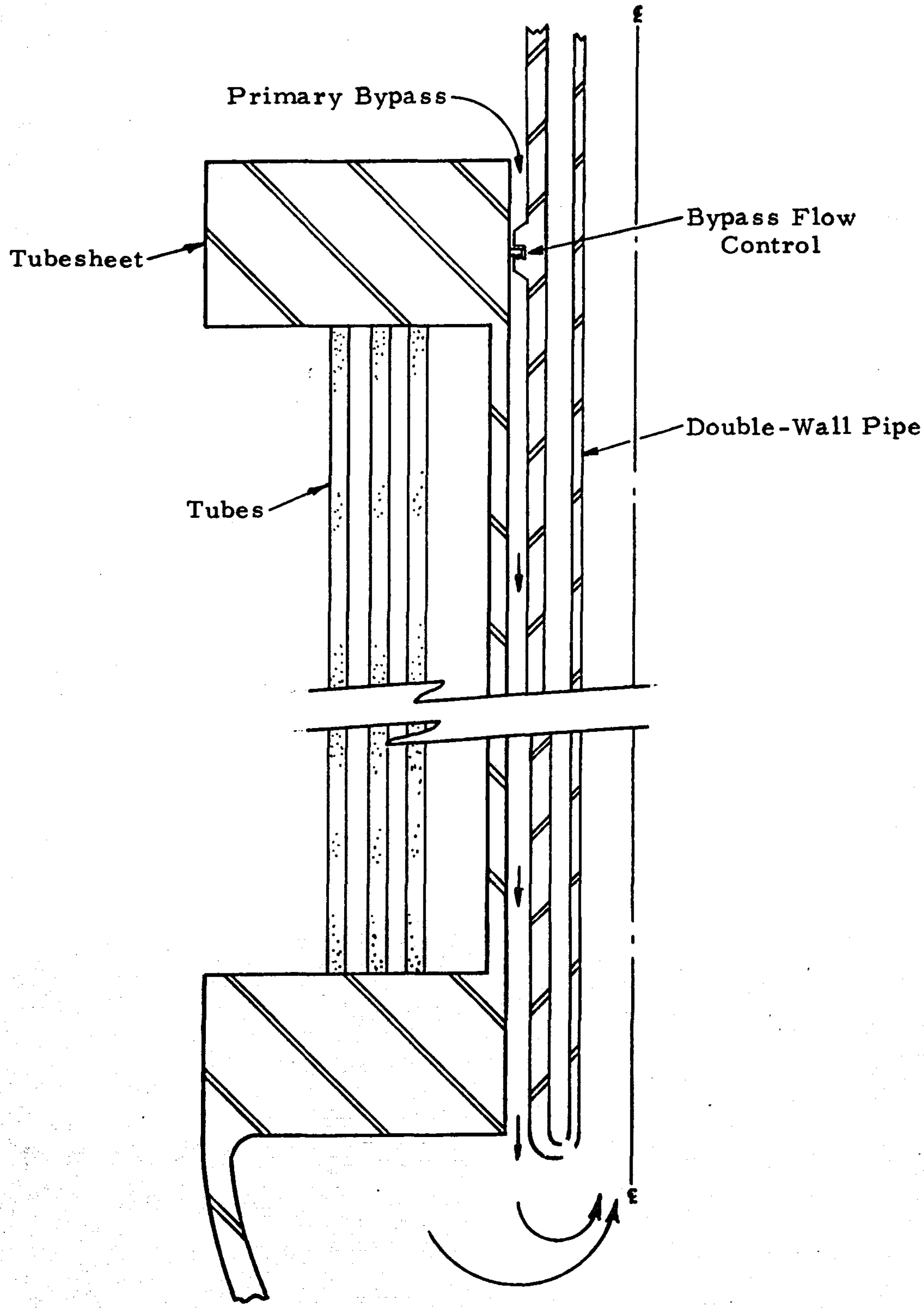


Figure 4.16: Removable Straight-Tube IHX With Flow Bypass.



Note: Bypass maintains average temperature of support cylinder equal to average temperature of tubes. At steady state bundle operates with zero thermal stress.

Figure 4.17: Bypass Flow - Design Details.

Disadvantages

1) The response of the support cylinder to thermal transients may be slow, even with the effect of the bypass flow; this condition causes high tube stresses.

2) The location of the primary salt outlet at the top adds significant inventory in the center pipe and outside the return pipe (up to 90 ft extra).

Conclusion

If the extra primary salt inventory is acceptable to the designer of the system, this concept should not be deleted; although realistically the extra inventory will be very costly for four units. When information on thermal transients is available, the response of the support cylinder should be investigated. If the design passes both conditions, it should be considered further because of the advantages offered.

4.3.3.2 Removable Sine-Wave Tube IHX (Figure 4.18)

Advantages

1) Maximum tube stresses occur at the cooler end of the unit where the sine wave is located and where the allowable stresses are higher.

2) Access to the bottom tubesheet is possible through the tubes or the downcomer.

3) This tube shape has been used previously on several LMFBR heat exchangers.

Disadvantages

1) The downcomer and the return pipe add significant inventory, as in the removable straight-tube concept.

2) The assembly of tubes in the tube bundle and clamping tubes are complicated compared to those of the straight-tube bundle.

3) The tube shapes require extra manufacturing time.

Conclusion The large amount of extra primary salt inventory makes this concept costly, as is the straight-tube concept.

4.3.3.3 Removable Helical-Tube IHX (Figure 4.19)

Advantages

1) The tubes are flexible.

2) The 360 degree flow in the tubes eliminates thermal maldistribution of shell-side fluid.

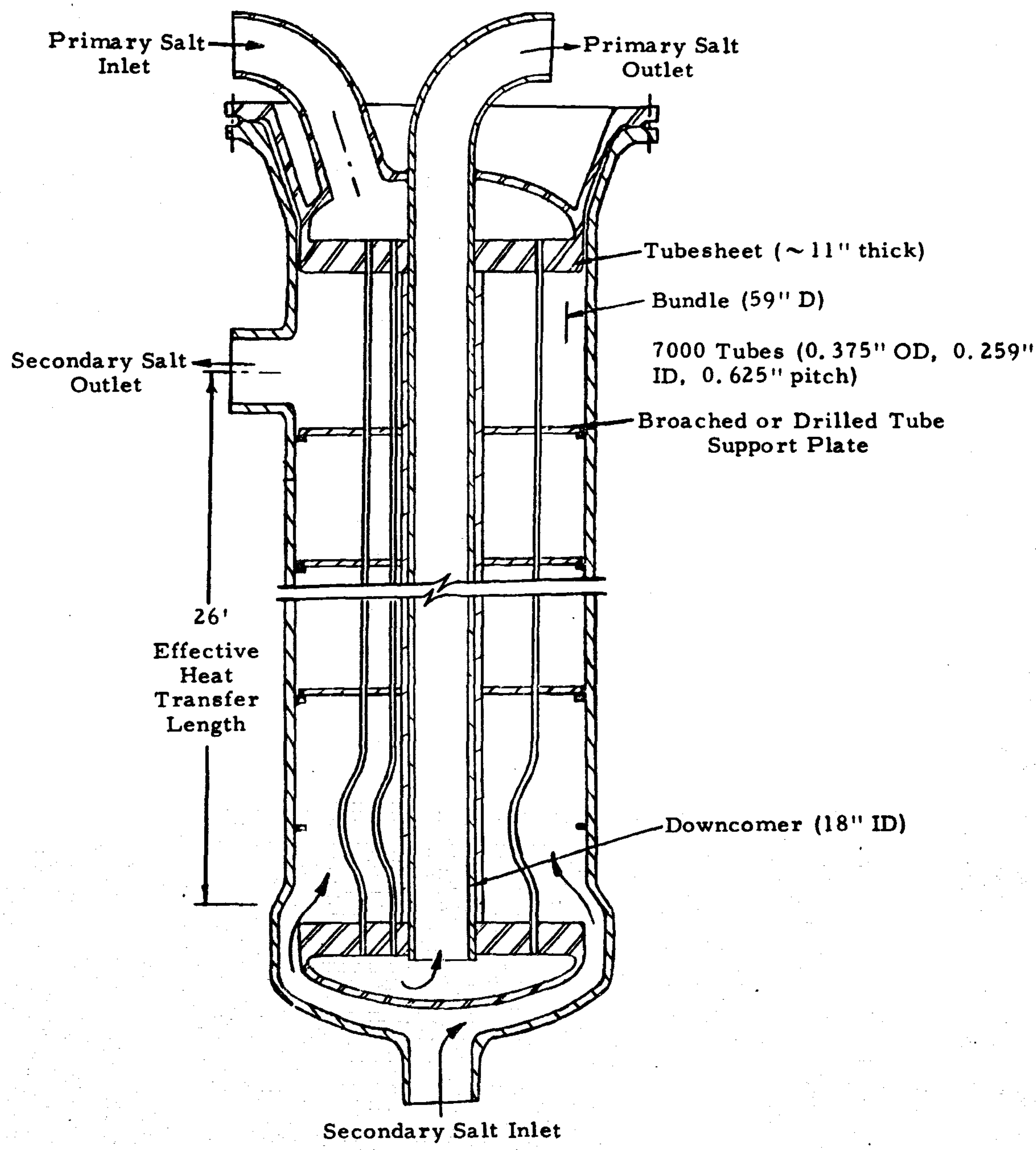


Figure 4.18: Removable Sine-Wave Tube IHX.

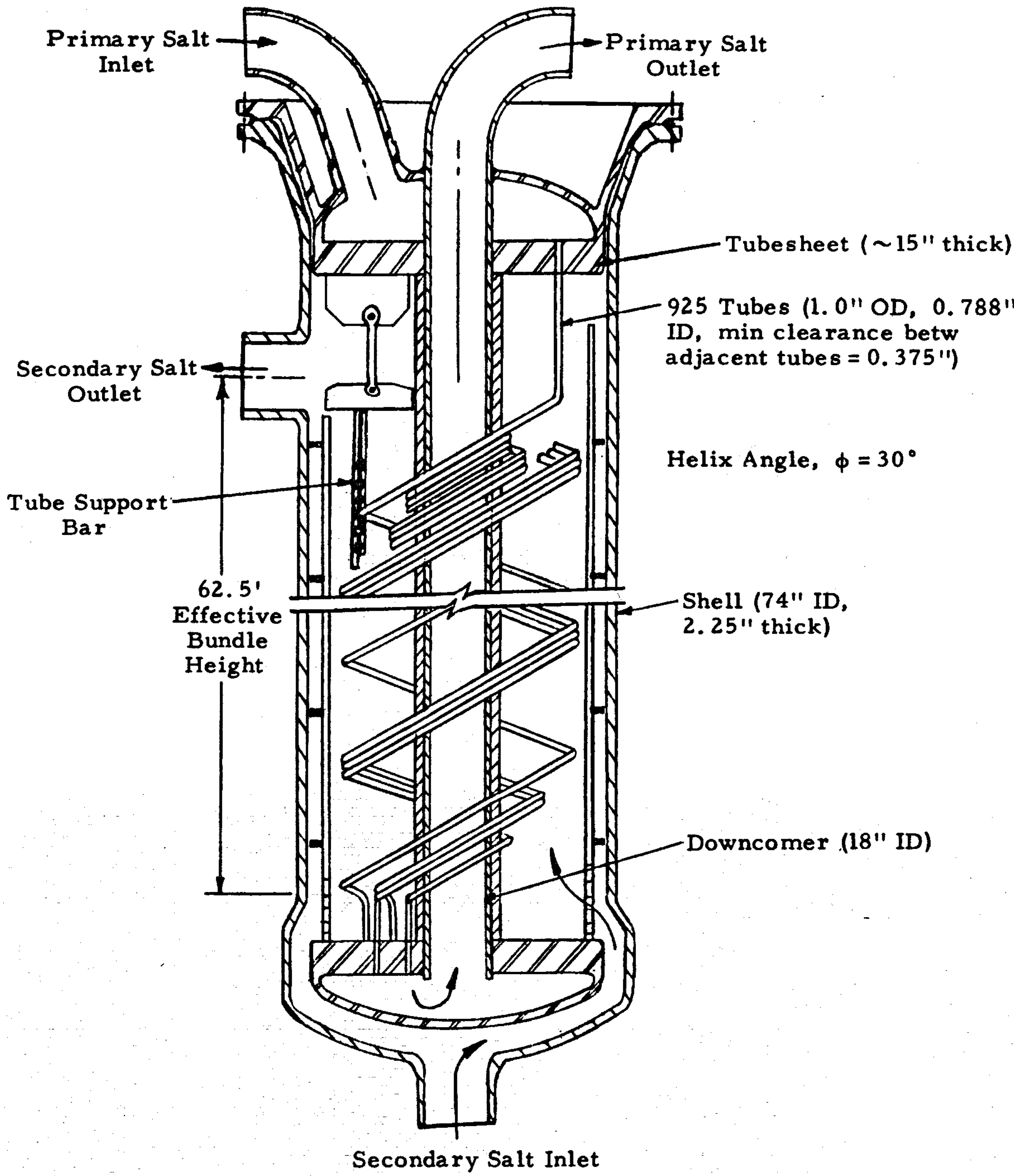


Figure 4.19: Removable Helical-Tube Bundle IHX.

3) The large OD tubes make remote explosive tube plugging easier.

Disadvantages

1) The extra tube-bundle length, the center exit pipe, and the return pipe give this unit the highest primary salt volume requirement of the concepts.
2) Tube coiling and tube support are complicated manufacturing processes.

Conclusion Excessive primary salt inventory makes this unit very undesirable.

4.3.3.4 Removable C Tube IHX (Figure 4.20)

Advantages

1) Cylindrical tubesheets are much thinner than flat tubesheets.
2) The primary salt inventory is very close to the minimum amount because annular plenums reduce the volume considerably.
3) The tube thermal stresses are low.
4) The tubes can be close-packed while maintaining large ligaments in the tubesheets.

Disadvantages

1) The tube layout and the tube shapes are complicated.
2) Tube support and flow baffles throughout the bundle will be special items. The use of broached tube-support plates or disc-and-donut baffles does not seem possible because of the tube shape. Alternative supports, such as banding circles of tubes, still leave the bend areas unsupported.
3) Without removing the bundle, cylindrical tubesheets and small plenums make tube-end access difficult for inspection or repair.

Conclusion

The difficulties of remote maintenance, tube support, and tube-bundle assembly could be overcome with careful design, more expensive manufacturing, and the use of a shielded cask for tube repair on a removed bundle. If these additional costs are acceptable, a unit using low primary inventory with flexible tubes is realized.

4.3.3.5 ORNL IHX Concept (Figure 4.21)

Advantages

1) The primary salt inventory is a minimum.
2) The tube thermal stresses are low.

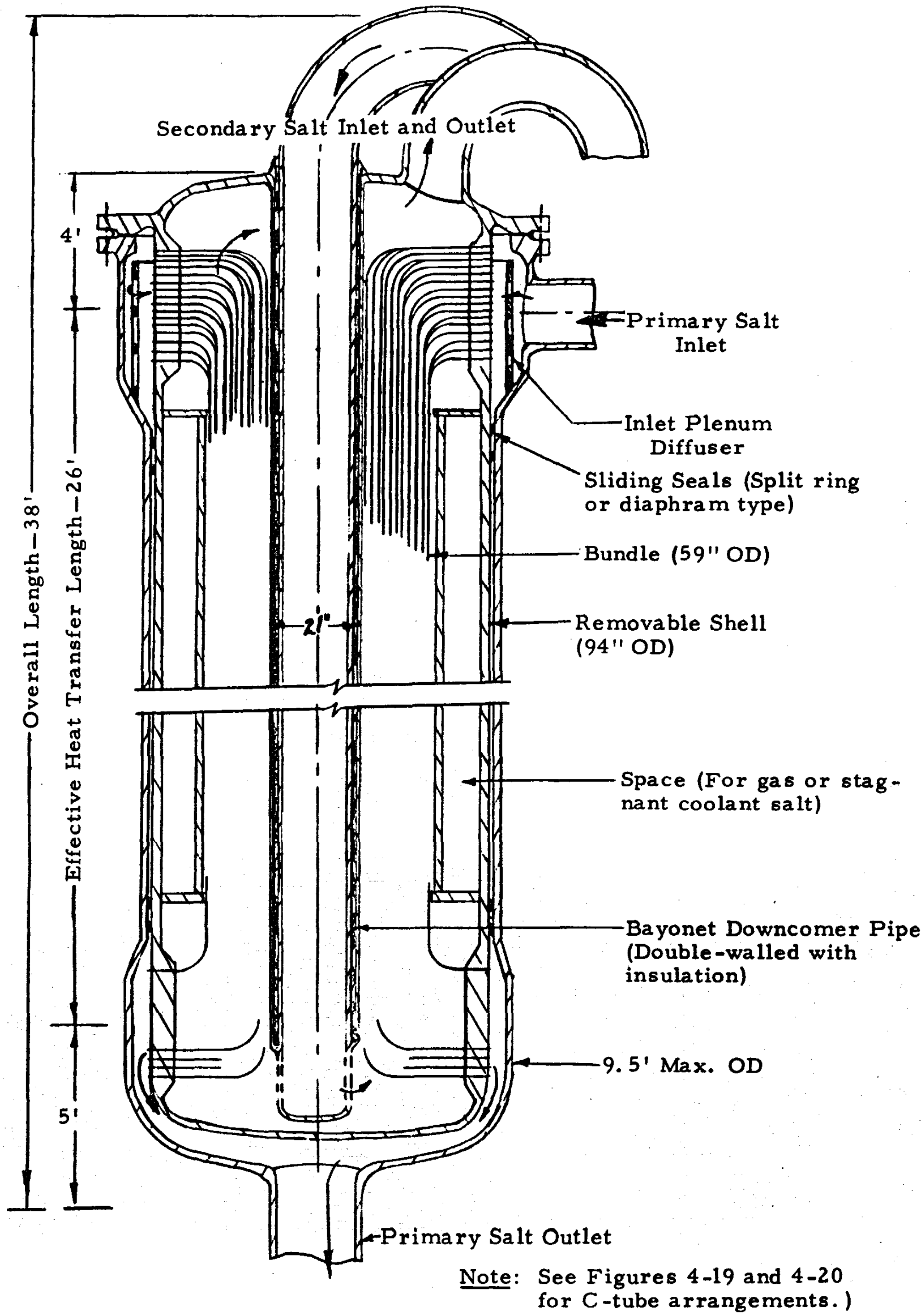


Figure 4.20: Removable C Tube IHX.

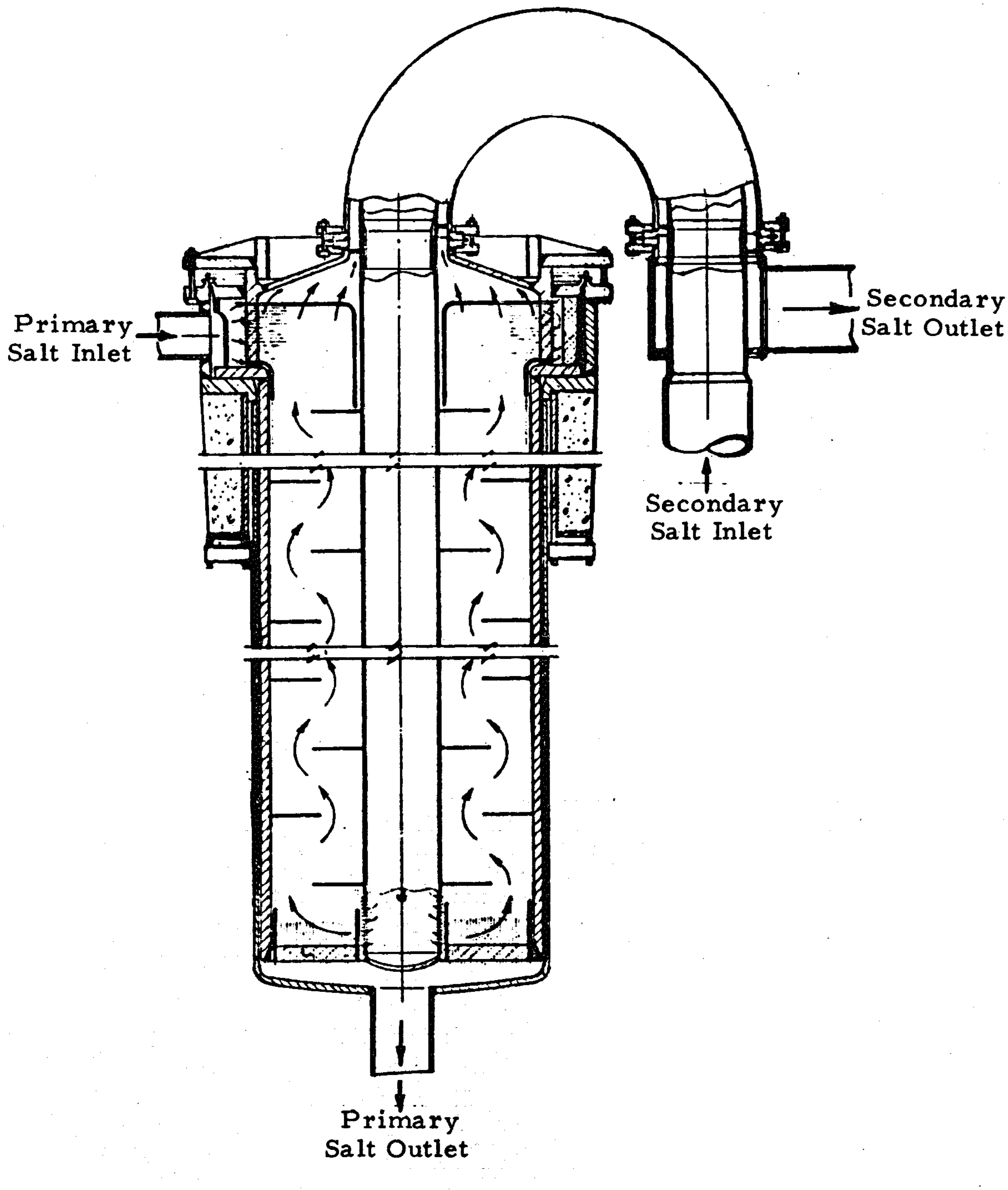


Figure 4.21: ORNL IHX Concept.

3) A cylindrical tubesheet at the hot end is thinner than a flat tubesheet would be.

Disadvantages

1) The tube dimensions change with each circular row as in the sine-wave tube.

2) The tube layout is a complicated radial pattern as in the C tube.

3) Tube support and flow baffling in the 6 ft long sine-wave bend region will require special designs.

4) The secondary salt inlet and outlet are at the same end.

5) The upper tubesheet is not readily accessible due to its cylindrical shape.

Conclusion Even though the disadvantages cause some concern about the manufacturing and design details of this concept, solutions to these problems appear to be available in the near future. Because of its minimum primary salt inventory and low thermal tube stresses, this unit is recommended as a first choice from the removable tube bundle designs. An estimate of the additional maintenance costs related to the inaccessibility of the upper tubesheet is required.

2. Nonremovable Tube Bundle

As mentioned in the previous section, the evaluation is presented in two ways: first, a sketch follows a list of advantages, disadvantages and conclusions for each of the six fixed-bundle concepts; second, the comments are related by letter ratings in a table for each of the four major criteria (see Figure 4.32). From this evaluation the fixed sine-wave-bend tube bundle is recommended as the best nonremovable concept. Again, the choice is not clear-cut, and a change in the emphasis on one or more of the criteria could alter the selection. For example, if remote maintenance becomes a very important item because of AEC regulations, the selection could change. When considering tube thermal stress, the reader should refer to Appendix D, which presents stress graphs of several basic bent-tube shapes. Since all of the tube shapes could be made flexible enough to exhibit low thermal stresses, a qualitative decision was made to determine the maximum size of the bundles. This decision limited the size of the various tube bends and gave an approximate indication of relative thermal stress levels.

4.3.3.6 Fixed Sine-Wave-Tube IHX (Figure 4.23)

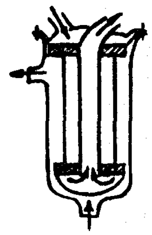
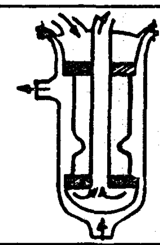
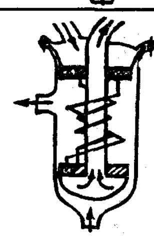
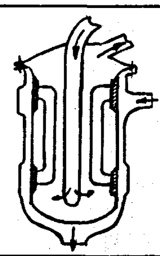
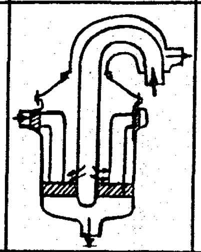
Advantages

1) The primary salt volume is a minimum.

2) Tube thermal stresses are low and can be located at the cooler end of the unit where the allowable stresses are higher.

3) This tube shape has been used on several LMFBR heat exchangers in recent years.

4) The tubesheets are relatively accessible for inspection and for remote tube repair.

Concept		Primary Salt Volume Ratio*	Thermal Stress	Manufacturing	Maintenance
Straight Tube (with bypass)		2.1	B^-	A	A
Sine-Wave Bend Tube		2.1	B^+	B^+	A
Helical Tube		5.5	B^+	B	B
C-Tube		1.0	A	B^-	C
ORNL		1.0	B^+	B	C

Legend:

A = Very Good
B or B^+ = Good
B^- = Fair
C = Poor

*Primary salt volume ratio based on minimum volume of 120 ft^3.

Figure 4.22: Removable Tube-Bundle IHX Criteria Rating.

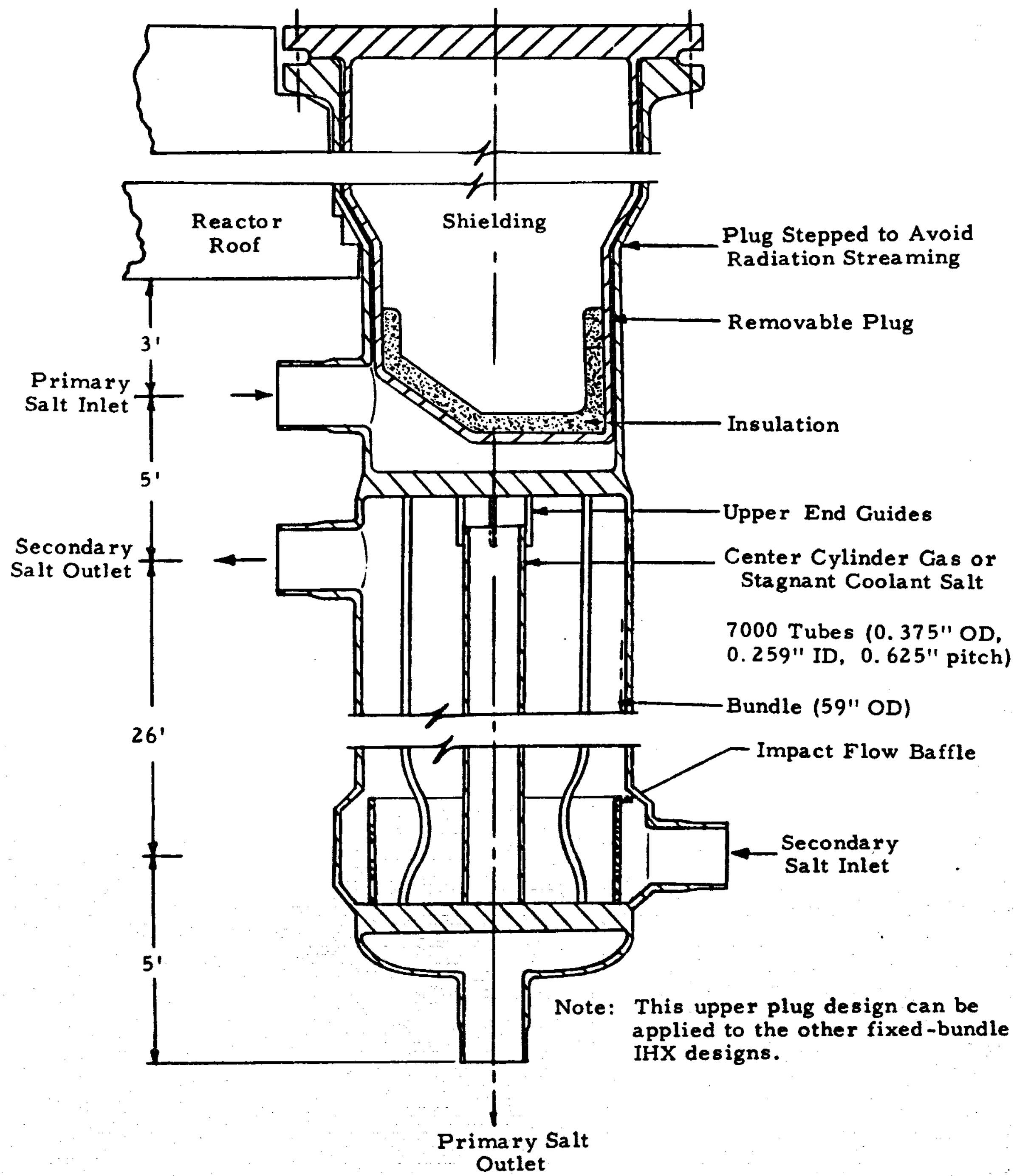


Figure 4.23: Fixed Sine-Wave IHX With Tubesheet Access.

Disadvantages

1) The tube-bundle assembly is complicated as compared to that of a straight-tube bundle.

2) Tube supports and flow baffles in the sine-wave-bend region require special attention to avoid vibration problems.

3) Tube shaping is expensive since a double curvature is used, and the circumferential bend varies with each tube circle.

4) Maintenance must be performed remotely, but it is no more difficult than that performed on other fixed-bundle designs.

Conclusion For a cost of some solvable manufacturing and tube-support problems, a nonremovable unit can be fabricated with minimum primary salt inventory and relatively low thermal-tube stresses during steady-state and transient operation. Of all nonremovable designs, this concept appears to be the most attractive.

4.3.3.7 Fixed Helical-Tube-Bundle IHX (Figure 4.24)

Advantages

1) Thermal stresses are negligible due to small differential expansion between the shell and the tubes.

2) The large tube size simplifies remote explosive tube plugging.

3) The flow pattern eliminates major flow maldistribution possibilities.

4) A very small number of tubes is required; consequently, tubesheets can be thinner.

Disadvantages

1) The primary-salt inventory is very high, or about four times that of the straight-tube or sine-wave tube bundles.

2) The tube manufacturing process is complicated, and the tube supports may be costly.

3) The unit length means that more secondary salt volume is required; the containment cell must be deep, and transportation will be difficult.

4) Shortening the unit would require a significant increase in the primary salt inventory, which is already high.

Conclusions

Even if the extra primary salt inventory is acceptable, the length of this unit makes it highly undesirable. If the tube-bundle effective height is shortened from 62.5 to 35 ft while maintaining

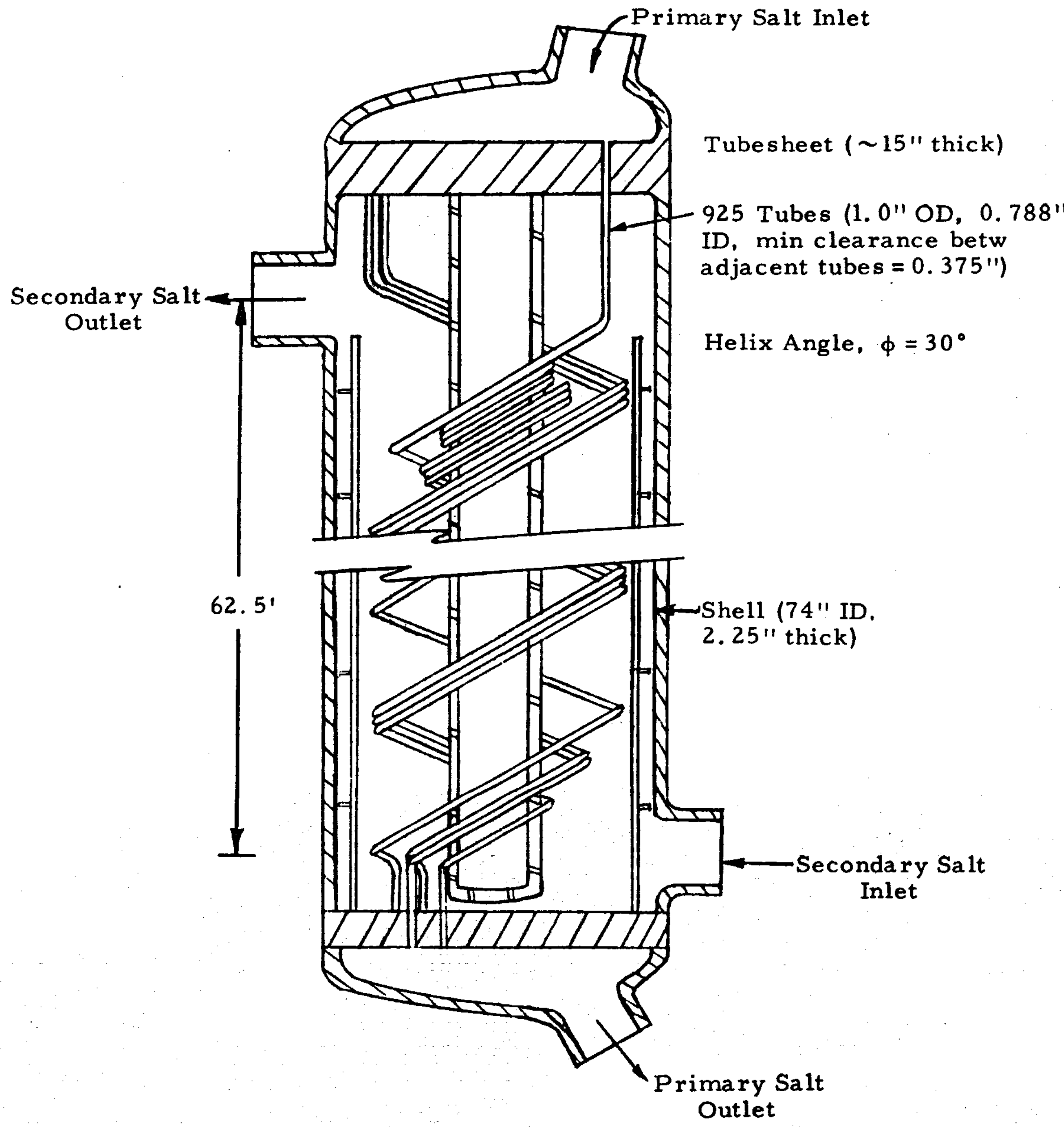


Figure 4.24: Fixed Helical-Bundle IHX.

the same primary side pressure drop, the helix angle decreases, and the primary salt inventory increases from approximately 450 to 750 ft3.

4.3.3.8 Hockey-Stick IHX (Figure 4.25)

Advantages

1) Compared to the other bent-tube shapes, the hockey-stick is easy to form.
2) Tube thermal stresses are generally lower than those of the sine-wave or C shapes, and the maximum stresses are located at the cool end of the IHX.
3) The lower tubesheet is accessible through the tubes or through the primary salt outlet plenum.
4) Simple broached or drilled tube-support plates may be used in the straight portion.

Disadvantages

1) The insertion of tubes into the tubesheets presents problems similar to those of the sine-wave.
2) The shell shape at the bottom bend requires many fit-ups and welds.
3) In the bend region at low-load operation, uncertainties about the flow-baffle design, the tube-support design, and the means of effective heat transfer exist.
4) Although the hockey stick is similar in shape to the J-tube, it requires about 15 percent more primary salt inventory than the sine-wave or straight-tube designs require.

Conclusion

Due to the shell manufacturing and tube-support problems, this concept appears to be less desirable than the fixed sine-wave-tube IHX concept.

4.3.3.9 J Tube IHX (Figure 4.26 and 4.27)

Advantages

1) Inserting tubes into tubesheets and into broached or drilled tube-support plates is simple.
2) The lower tubesheet is accessible for direct but remote tube plugging due to its inverted orientation.
3) Tube thermal stresses are low.
4) When the short return pipe to the reactor is considered (see Figure 4.27), this unit offers a minimum primary salt inventory comparable to or less that that of the sine-wave, depending on the location of the reactor inlet.

Disadvantages

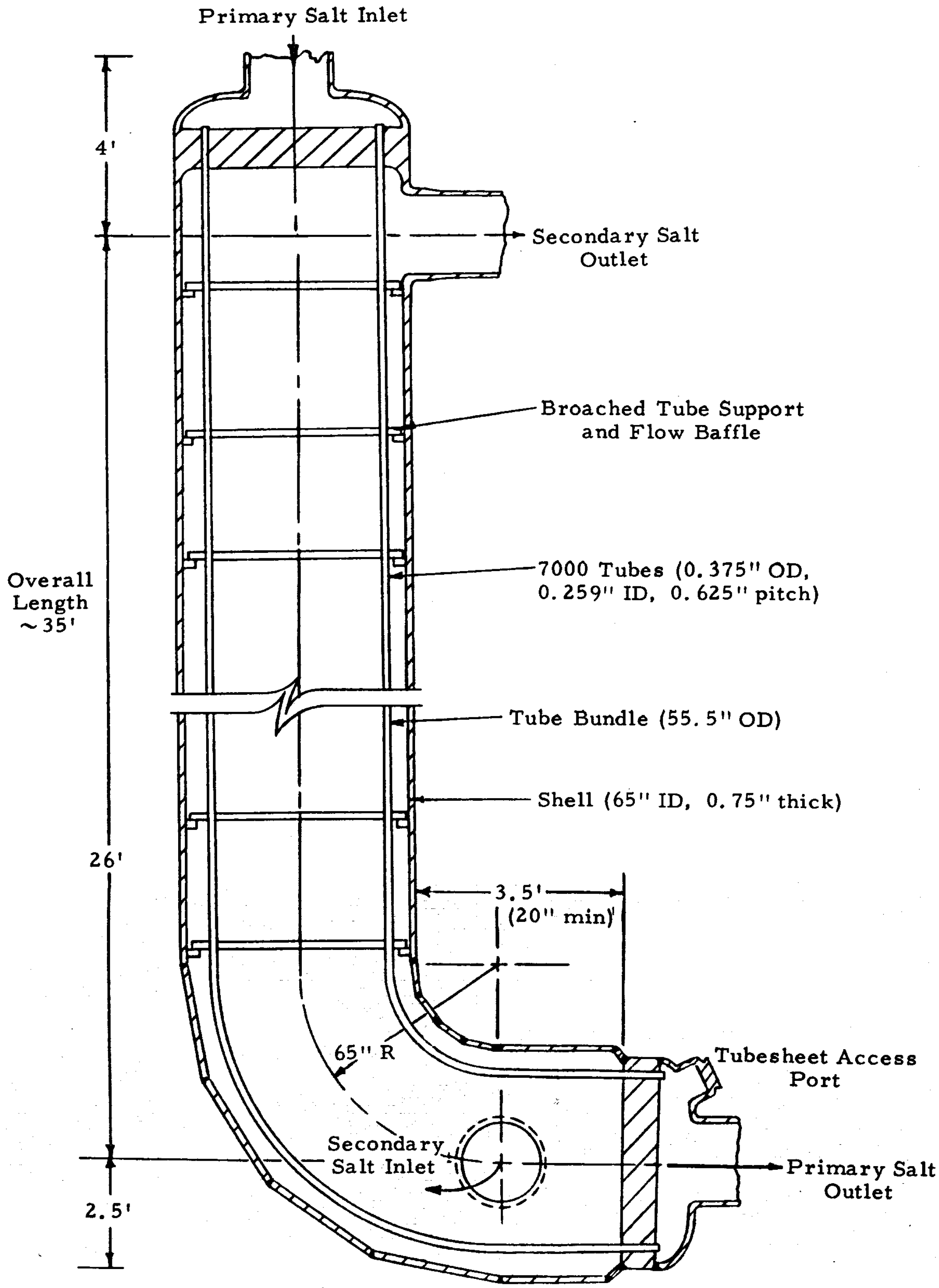


Figure 4.25: Hockey Stick Tube IHX.

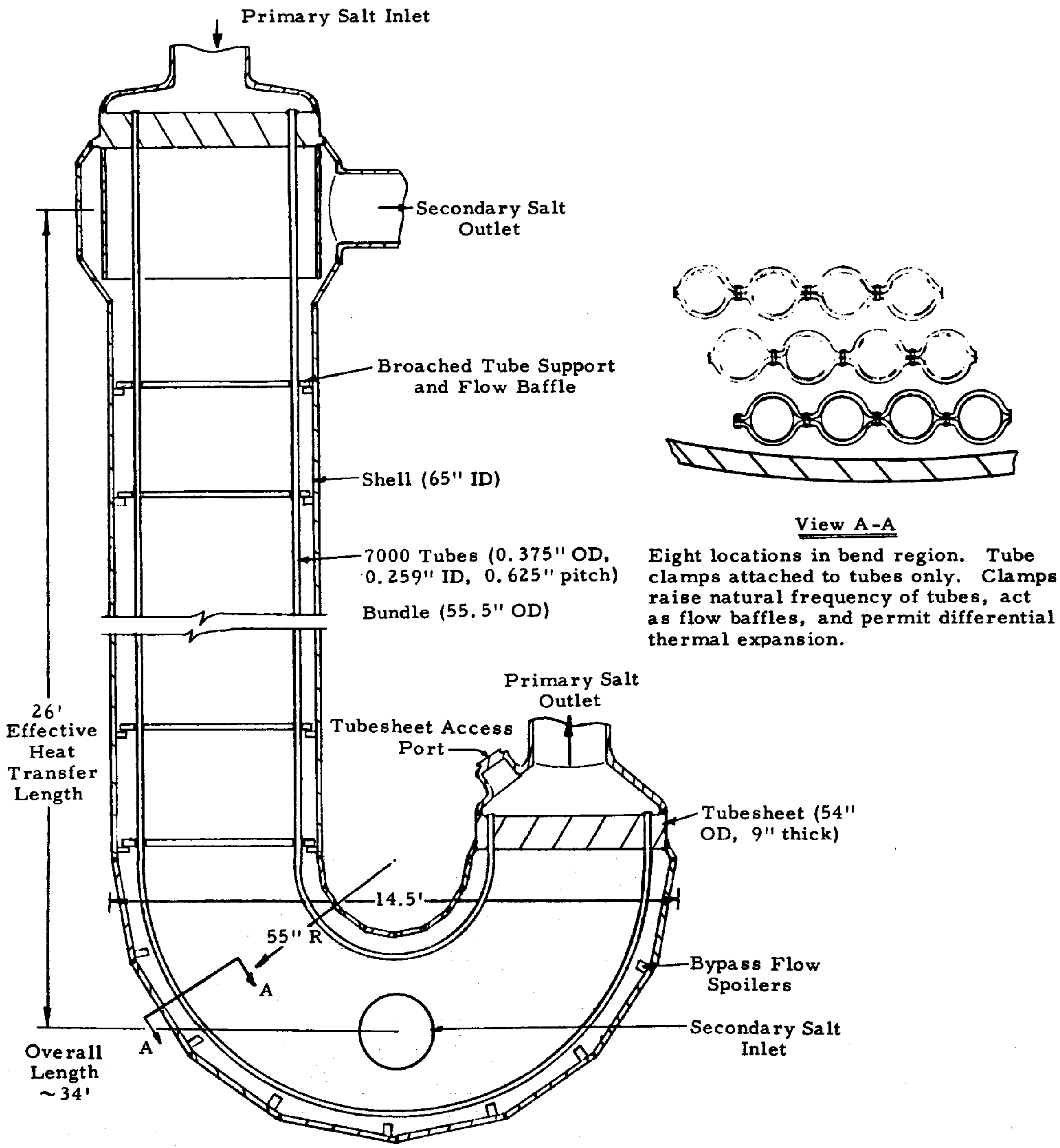


Figure 4.26: J Tube IHX.

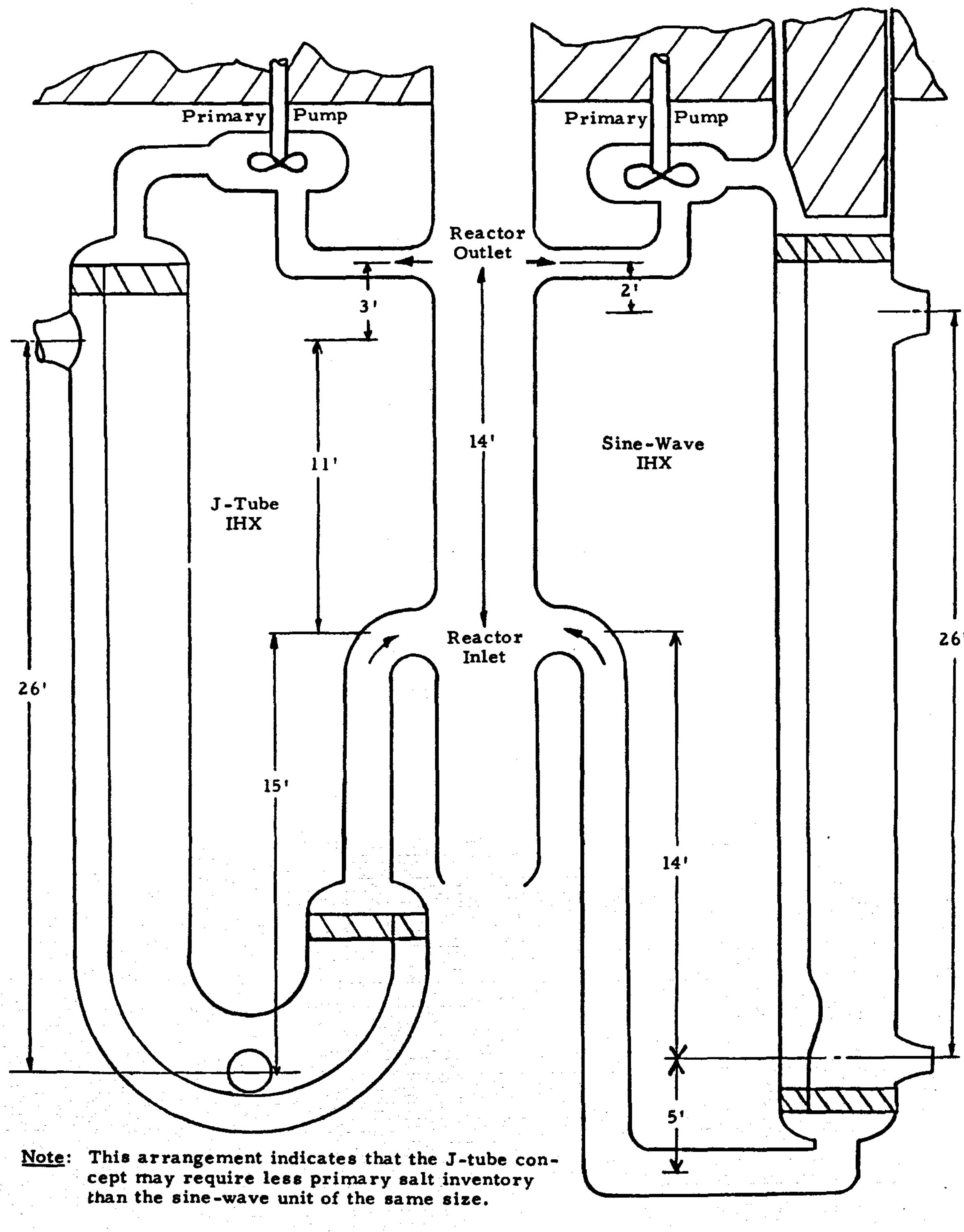


Figure 4.27: Primary Piping Arrangements for Two Fixed Tube-Bundle IHX Concepts.

1) Supporting tubes and controlling flow in the bend region are difficult.

2) The complicated shell shape requires many welds and fit-ups.

Conclusion

In most areas the features of this unit are comparable to the best features of several other concepts; this concept would be comparable to the sine-wave concept except for the complicated shell shape.

4.3.3.10 C Tube IHX (Figure 4.28, 4.29 and 4.30)

Advantages

1) The tubes are flexible.

2) Cylindrical tubesheets can be thinner than flat tubesheets, and the tube-hole spacing can be larger than the pitch in the bundle.

3) The annular plenum shapes help to minimize primary salt inventory.

Disadvantages

1) The tubes are difficult to support in the bend region.

2) Tube access for repair or inspection is very difficult because of the 360 degree tubesheets.

Conclusion

Major problems are foreseen with manufacturing, tube layout, and maintenance; this design is not recommended.

4.3.3.11 ORNL Concept, Fixed-Bundle Version (Figure 4.31)

The comments are the same as those for the fixed C tube IHX; the conclusions are not significantly different.

3. Primary Fuel Salt on Shell Side

From the viewpoint of maintenance, inspection, and repair, it is advantageous to locate the primary (fuel) salt on the shell side. The surface of the tubesheets will not be highly radioactive as it is in the previously discussed concepts with primary salt in the tube; this location of primary salt simplifies shielding during remote maintenance on the tubes and tubesheets. This section discusses concepts with requirements and the methods used to achieve this unit at a reasonable cost penalty.

To avoid a primary salt volume penalty (a primary-to-secondary ratio of 1.0 indicates no inventory penalty), the tubes should be closely spaced (see Figure 4.33). A ligament between tubes of about 0.050 in. would make the shell-side volume about equal to the tube-side volume. This

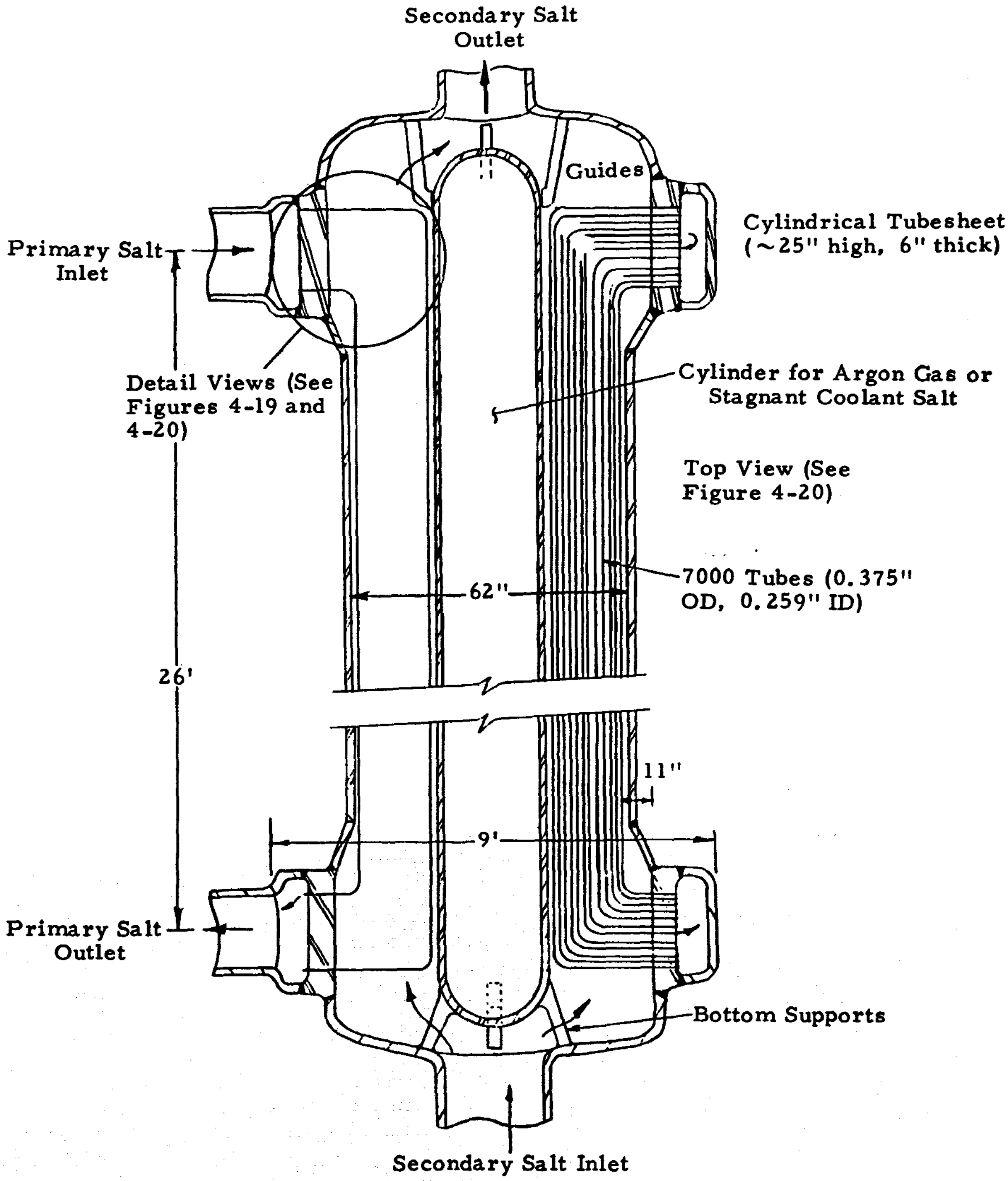


Figure 4.28: Fixed C Tube IHX.

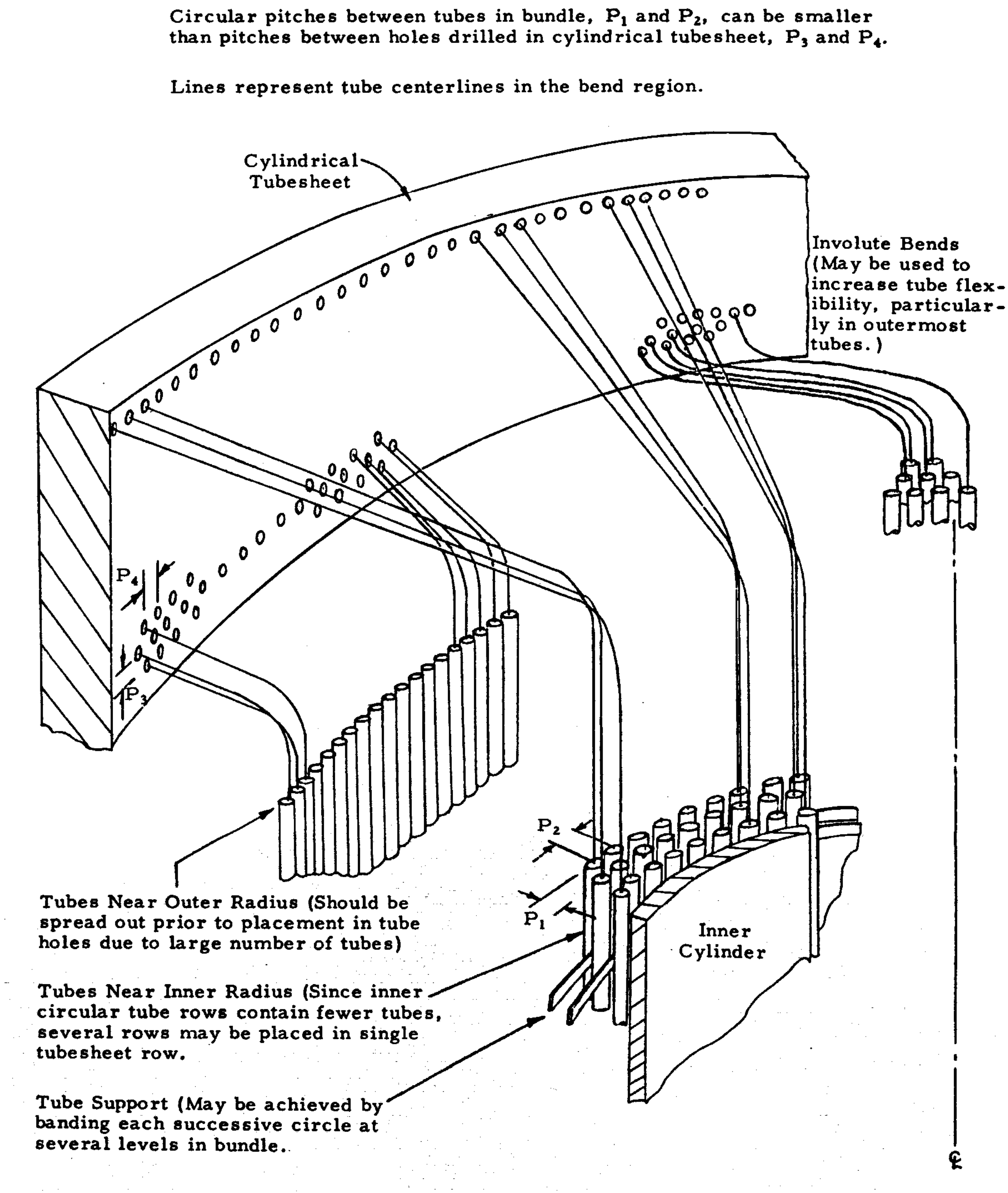


Figure 4.29: C-Tube Arrangement, Circular Pitches.

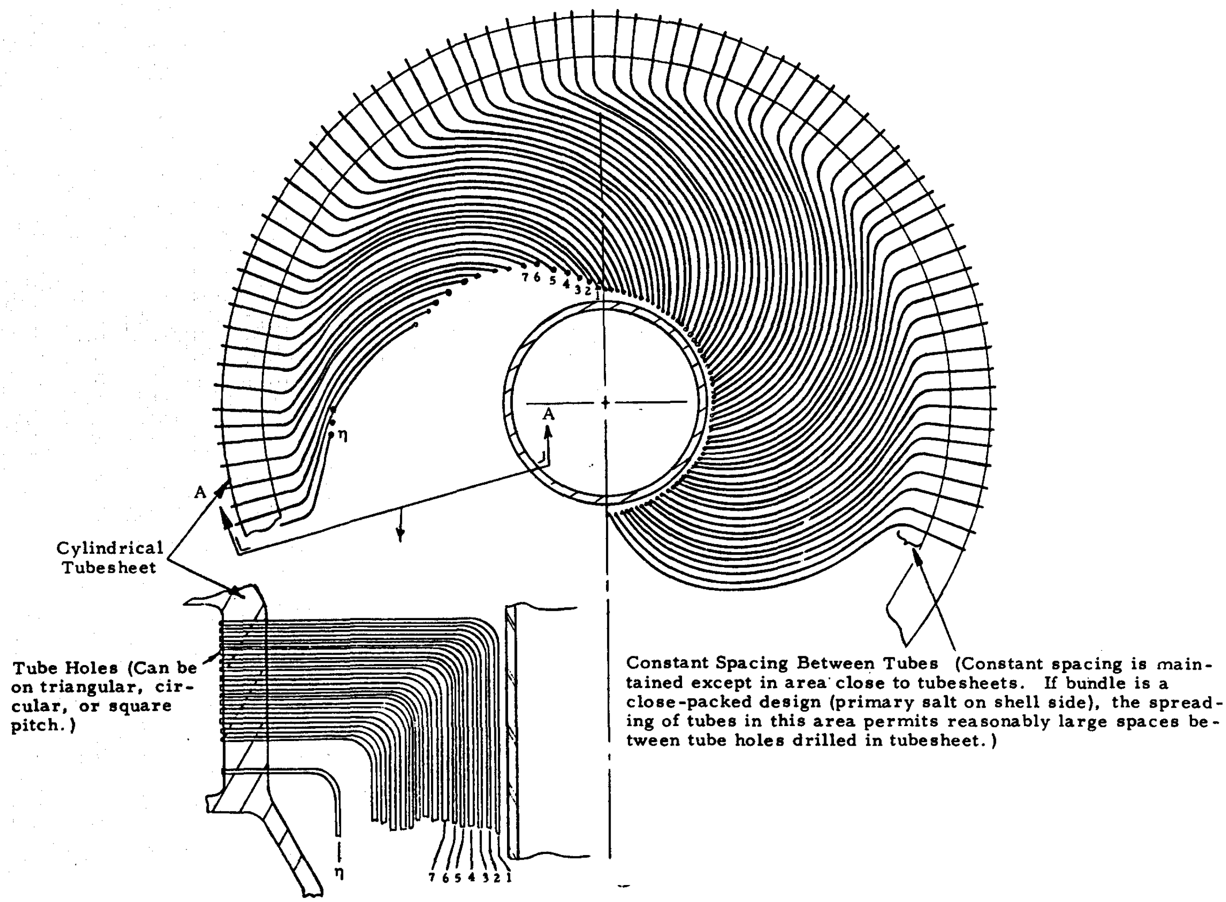


Figure 4.30: Involute Tube-Bundle Array, Top View.

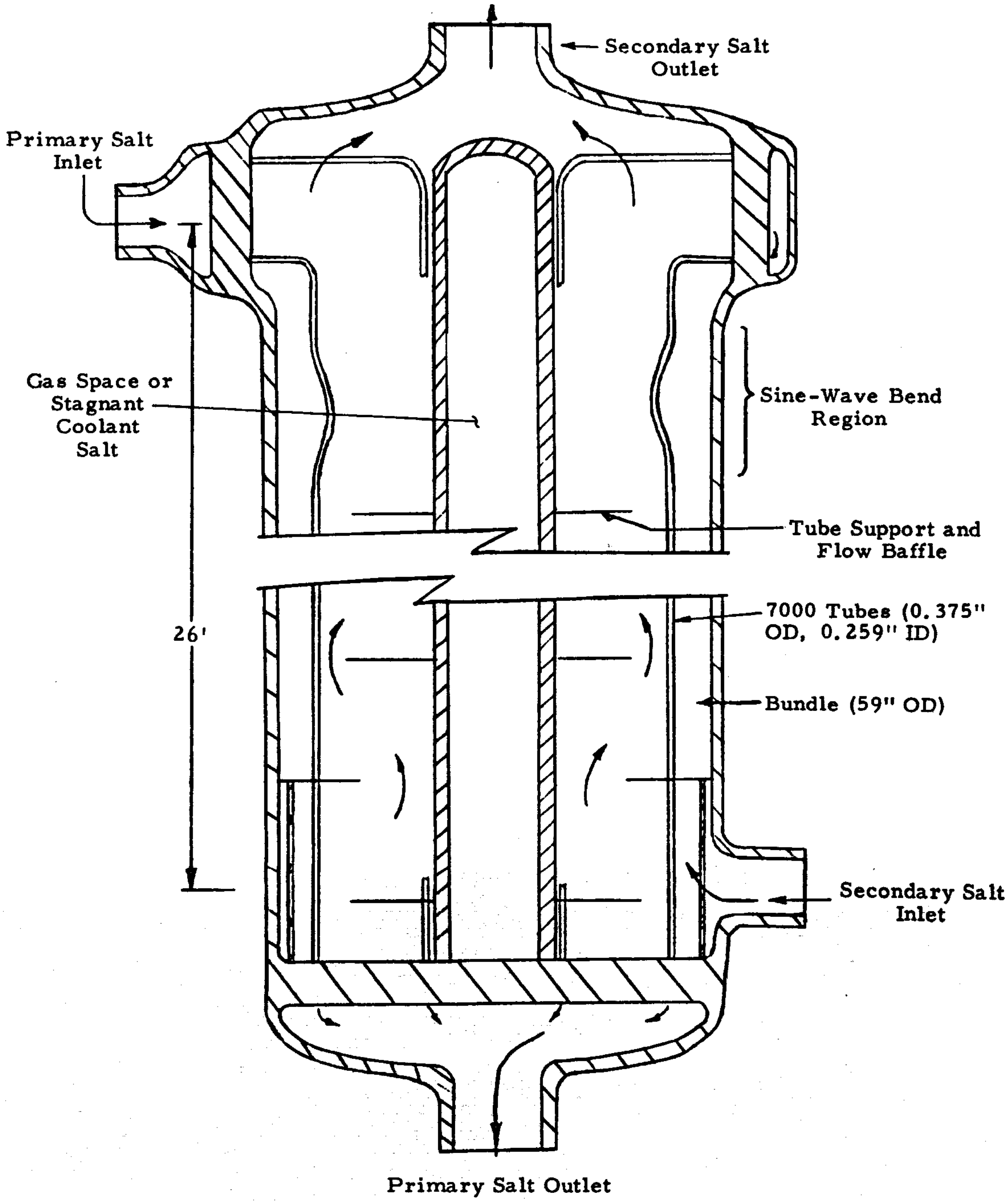
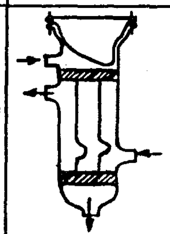
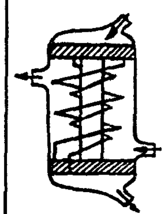
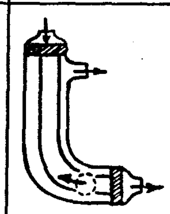
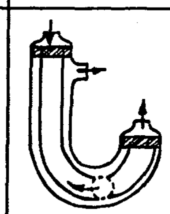
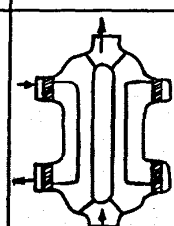
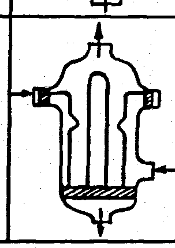


Figure 4.31: ORNL Design, Fixed Tube-Bundle Version.

Concept		Primary Salt Volume ratio*	Thermal Stress	Manufacturing	Maintenance
Sine-Wave Bend Tube		1.0	A	B^+	A
Helical Tube		4.5	B^+	B^+	A
Hockey-Stick Tube		1.15	A	B	A
J-Tube		1.15	B^+	B	A
C-Tube		1.05	A	B^-	B
ORNL		1.05	B^+	B	B

<u>Legend:</u> A = Very Good B^- = Fair
B^+ or B = Good   C = Poor

*Primary salt volume ratio is based on the minimum volume of 120 ft^3.

Figure 4.32: Fixed (Nonremovable) Tube-Bundle IHX Criteria Rating.

close spacing immediately introduces two problems—support of the tubes and thermal-hydraulic aspects.

Currently the only method that is envisioned for tube support is to wrap thin metal bands (approximately 0.050 in. wide) around the tubes. This method would require that the tubes be in a circular array as shown in Figure 4.34.

The effect of tube spacing on thermal-hydraulic performance is shown in Figure 4.35. Placing the primary salt on the shell side (line A-B) requires a volume penalty of 20 ft3 (85 minus 65 ft3). The volume in the IHX plenums and pipes is approximately the same regardless of the location of the primary salt. To achieve the correct tube-side (secondary salt) pressure drop, probably 7000 to 8000 tubes are required.

For comparison, point C is shown; this is a typical design with primary on the tube side and 1/4-in. spacing between tubes. The primary salt volume in the bundle region is about 65 ft3 as opposed to 85 ft3 of the other concepts. Figure 4.35 is intended to show a rough comparison of tube-side versus shell-side concepts. For example, it is assumed that the designs are straight tubes, that the bundles with primary salt on the tube side have disc-and-donut baffles, and that the bundles with primary salt on the shell side have no baffles but are banded as shown in Figure 4.34. Obviously, the actual method of support should be studied before definite conclusions are made on these trends; however, despite the small primary salt volume penalty of 20 ft3, the use of close-packed bundles appears highly promising.

Several methods of close packing the tubes have been devised. Tube holes cannot be drilled within 0.050 in. in the tubesheets. Figure 4.36 shows a removable C tube IHX with the primary salt on the shell side, and the following section lists the advantages, disadvantages, and conclusions. Spreading the tubes out as they bend to enter the cylindrical tubesheets achieves a reasonably large ligament between tube holes while keeping the tubes closely packed in the vertical heat transfer region. The penalties seem to be manufacturing (long cylinders with tube welds inside) and some excess primary salt volume.

4.3.3.12 Removable C-Tube With Remote Repair Feature

<u>Advantages</u>

1) With controlled bypass past the ring seals, the steady-state operating temperatures of the tubes and tubesheet support cylinder are equal. Stresses due to differential thermal expansion are nearly zero at steady-state operation.

2) Manufacturing the IHX is relatively simple except for the tube bundle.

3) Easy and remote access for maintenance to both tubesheets is possible through the center.

<u>Disadvantages</u>

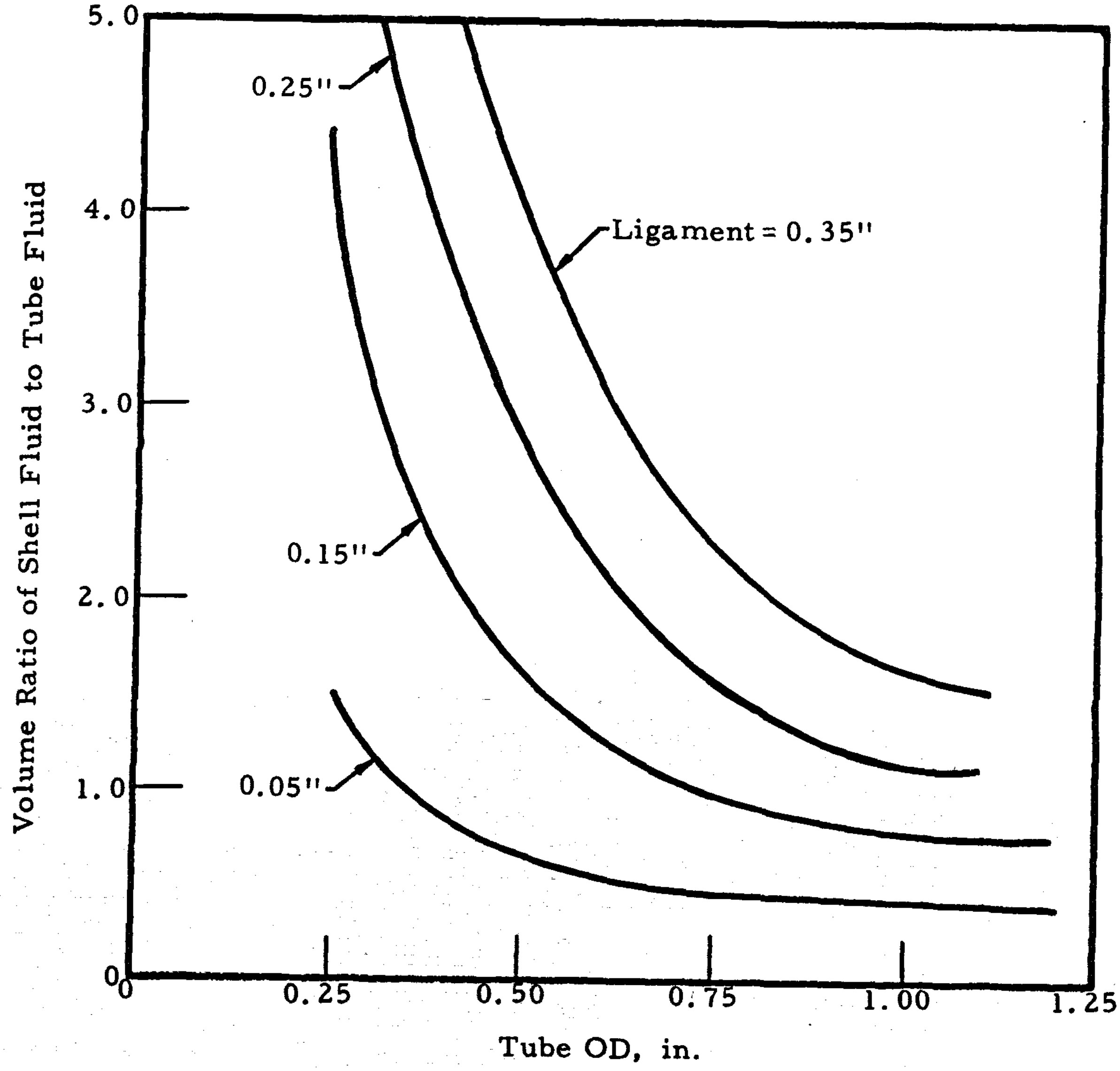


Figure 4.33: Straight-Tube Bundle, Ratio of Shell-Side Volume to Tube-Side Volume.

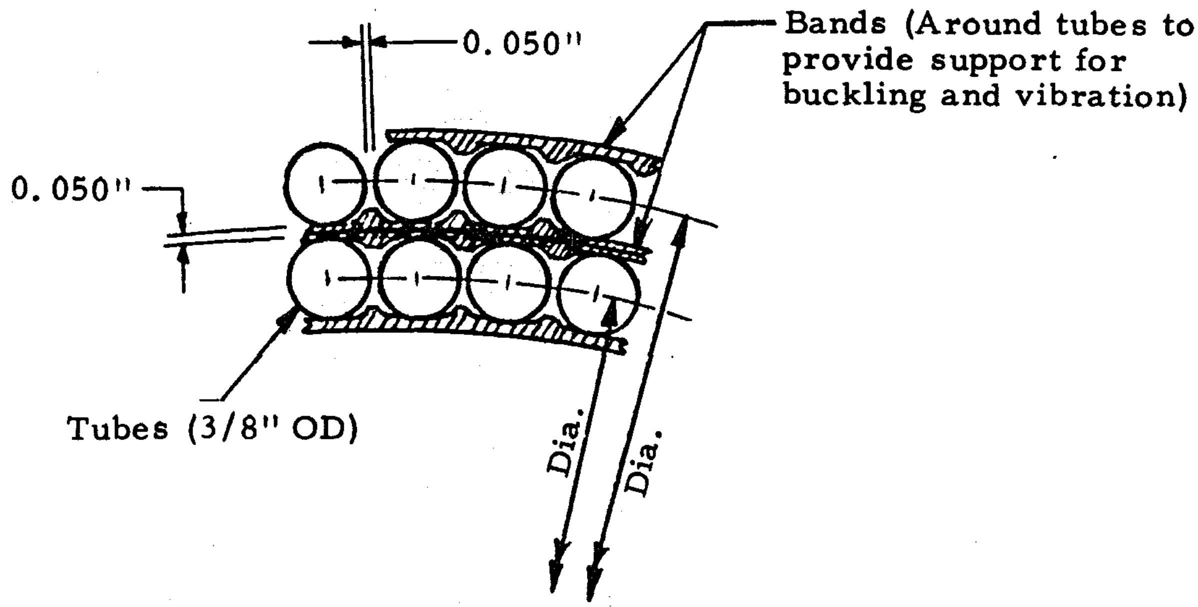


Figure 4.34: Close-Packed Tubes on Circular Pitch.

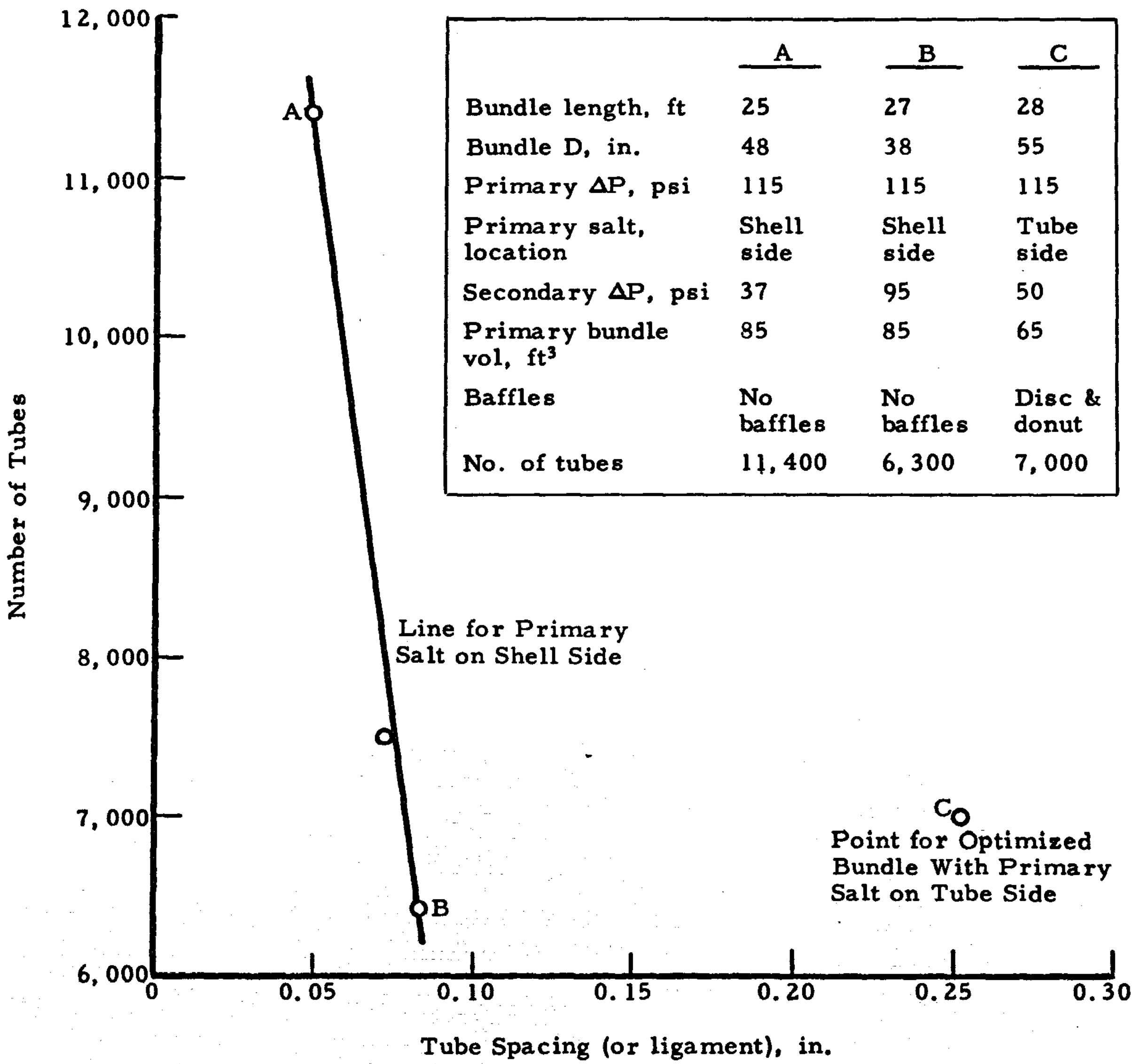
1) To achieve low primary salt inventory, the tubes will be placed about 0.050 to 0.080 in. apart. This placement gives support problems; however, one possible method is to wrap bands around each row of tubes (circular pitch).
2) More tubes are required to meet the ΔP requirements, say about 8000 as opposed to 6000 to 7000 for other units.
3) This design will have an increase in primary salt volume of about 20 ft3 relative to the designs with primary salt on the tube side.
4) This C tube arrangement requires a complicated tube arrangement and has several variable dimensions.

Conclusion

The major problems are the tube arrangement and supports. If these problems can be overcome, the design has good possibilities.

A concept for reducing shell-side volume with displacement cylinders or rods in a straight-tube bundle is shown in Figure 4.37. This concept allows normal tube pitches for the holes in the tubesheets and can be applied to a sine-wave, J tube, or hockey-stick tube bundle with some chance that there will be problems when displacing volume in the bend region.

These two methods, spreading the tubes into tall, cylindrical tubesheets or using volume displacers where there are flat tubesheets, locate the primary salt on the shell side in all of the concepts of sections 1 and 2 with the possible exception of the helical-bundle concepts. The helical-tube bundle



	A	B	C
Bundle length, ft	25	27	28
Bundle D, in.	48	38	55
Primary ΔP, psi	115	115	115
Primary salt, location	Shell side	Shell side	Tube side
Secondary ΔP, psi	37	95	50
Primary bundle vol, ft^3	85	85	65
Baffles	No baffles	No baffles	Disc & donut
No. of tubes	11,400	6,300	7,000

Figure 4.35: Effect of Tube Spacing.

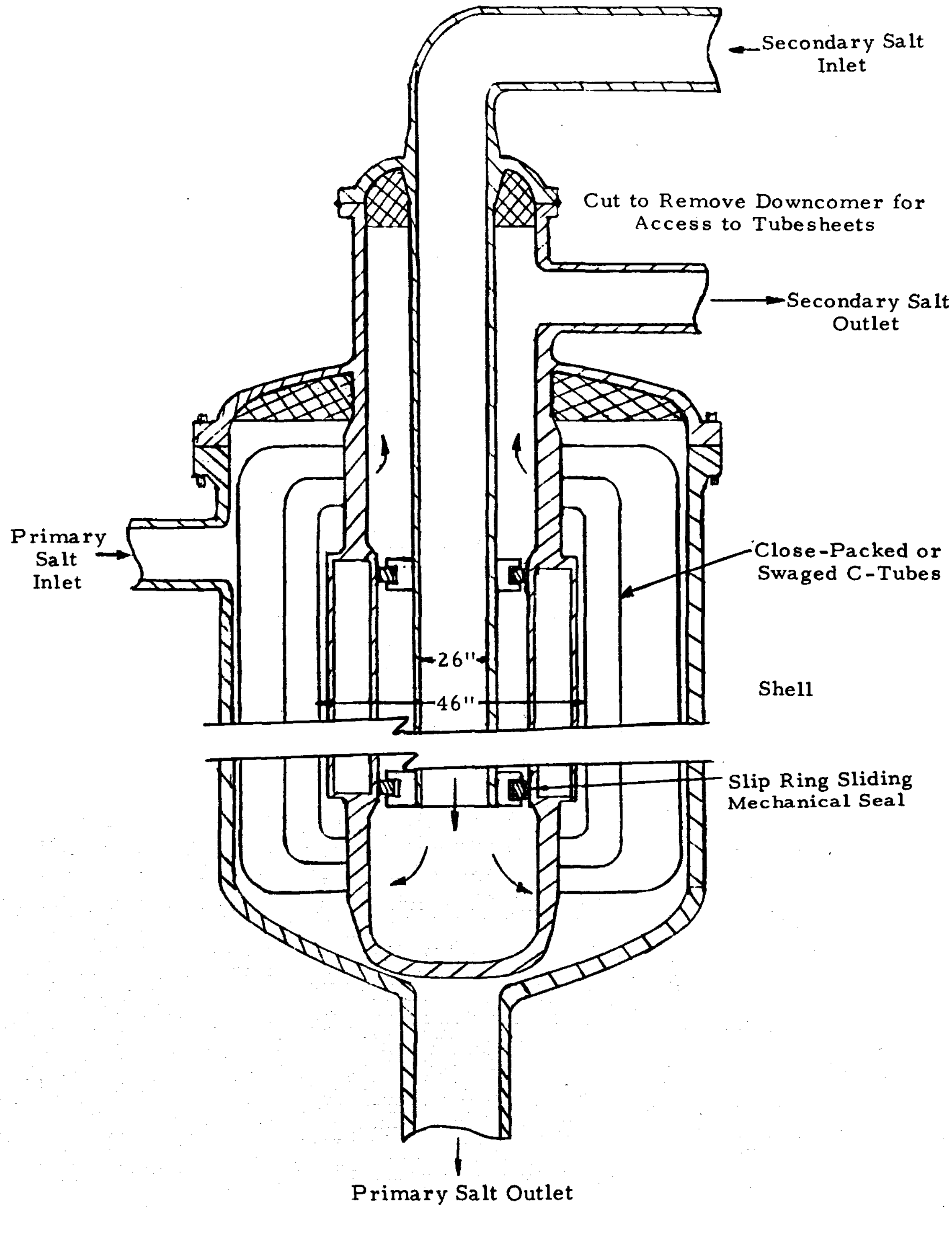


Figure 4.36: Removable C Tube IHX With Remote Repair Features.

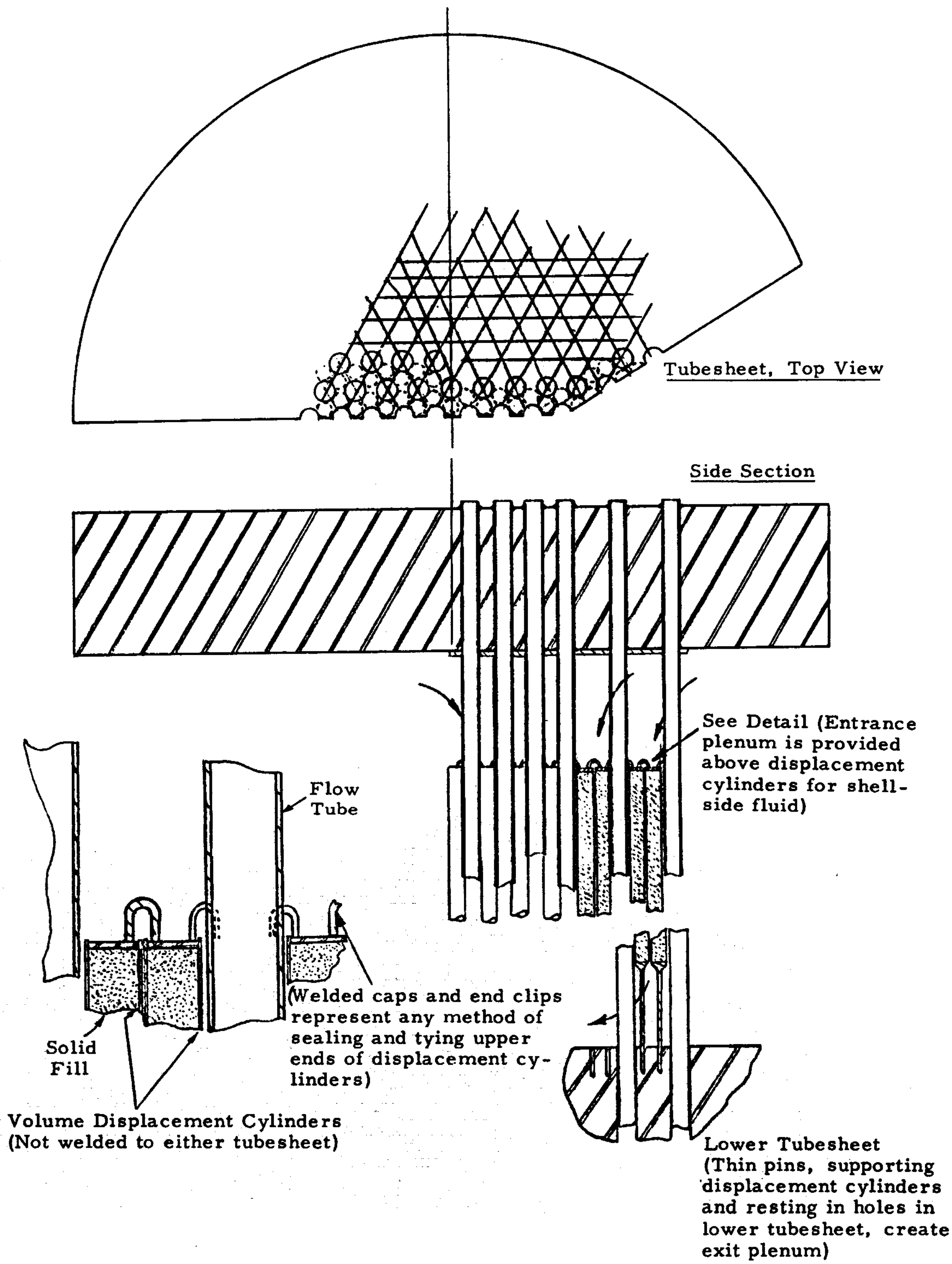


Figure 4.37: Conventional Tube Bundle With Volume Displacement Cylinders.

concepts have been rejected because of excessive primary salt inventory, and locating the primary salt outside the tubes will not alter this conclusion.

4. Advanced Concepts

In some respects the concepts that locate primary salt on the shell side could be called advanced concepts. There is essentially no experience to which to refer, whereas most of the concepts in sections 1 and 2 represent types of units that have been built or studied for the LMFBR program.

As mentioned in section 4.2, a method, allowing for differential expansion between the tubes and the shell, includes a large-diameter, flexible bellows in the shell. Figure 4.38 shows this method used with a straight-tube bundle. The major disadvantage, as discussed, is the unreliability and maintenance difficulty of large bellows. Since the LMFBR industry still discourages the use of bellows after more than a decade of development, this concept may not be attractive even with the additional 1980 technology.

Since minimizing the primary salt inventory is a major design goal, three advanced designs were sketched; these eliminate part or all of the primary piping. The IHX and pump are attached closely to each other and to the reactor vessel. In the first concept (Figure 4.39) the return pipe to the reactor is eliminated since the lower tubesheet is built into the reactor. Also included is a concentric pipe-pump concept. Since the fluid in the pipe-pump area is usually at 1300°F, and the inner pipe is not welded to the vessel wall, differential thermal expansion is not expected to be a problem. The large outer pipe is more capable of withstanding external forces and moments than a single, smaller pipe. Approximately 15 rows of 3/4 in. tubes run 360 degrees around the 20 ft diameter reactor vessel, and four pumps feed these tubes. The advantages of the hockey-stick-shaped tubes are the low thermal stresses during operation. The design concept also has the advantage of cooling the vessel walls. As with other cylindrical tubesheet concepts, this annulus of tubes presents inspection and repair problems compared to remote maintenance of a flat-plate tubesheet. Explosive tube plugging is a possibility for repair, but removal of the tube bundle is out of the question.

In the second concept of the coupled pump and IHX, the pipe that leads from the pump to the IHX is eliminated. Since some primary salt inventory is saved, it would be possible to elongate the return and outlet pipes attached to the reactor to give them more flexibility and lower stresses from external forces and moments. As in the previously discussed concept, this unit has a cylindrical tubesheet, and this causes tube inspection and tube repair problems. (See Figure 4.40.)

Figure 4.41 shows the third concept of a close-coupled pump with a standard hockey-stick IHX. A J-tube bundle IHX would also be applicable. The problems foreseen with this unit are not unique; with such short pipes the differential thermal expansion during thermal transients is expected to cause high nozzle and pipe loads.

These concepts generally do offer the advantage of a smaller reactor-pump-IHX containment area. This means less shielding and, perhaps, reduced plant capital cost. Further study may show that the savings of primary salt inventory, of Hastelloy, and of plant cost are not sufficiently great to warrant designing a close-coupled IHX and pump.

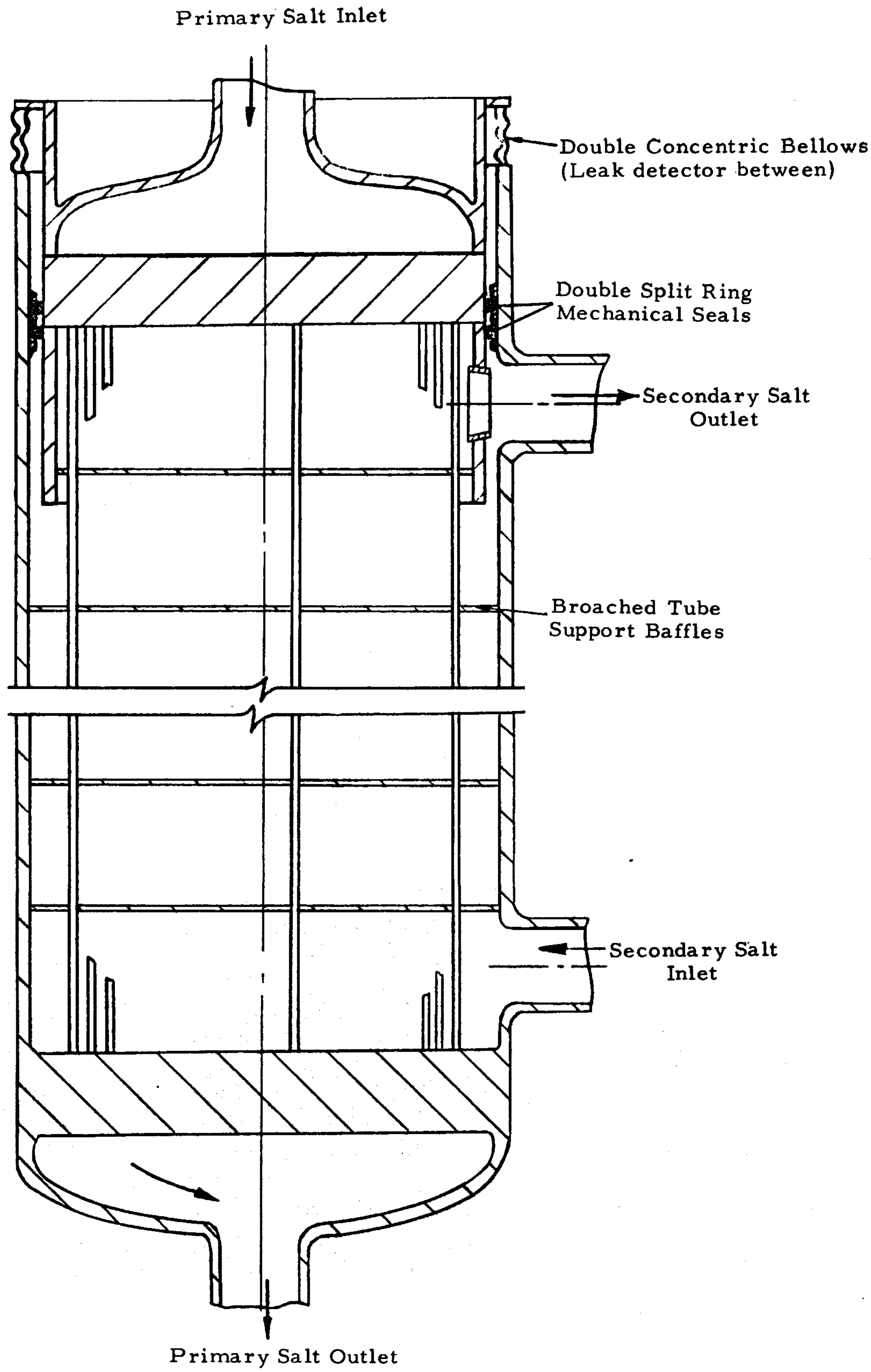


Figure 4.38: Fixed Straight-Tube Bundle with Bellows.

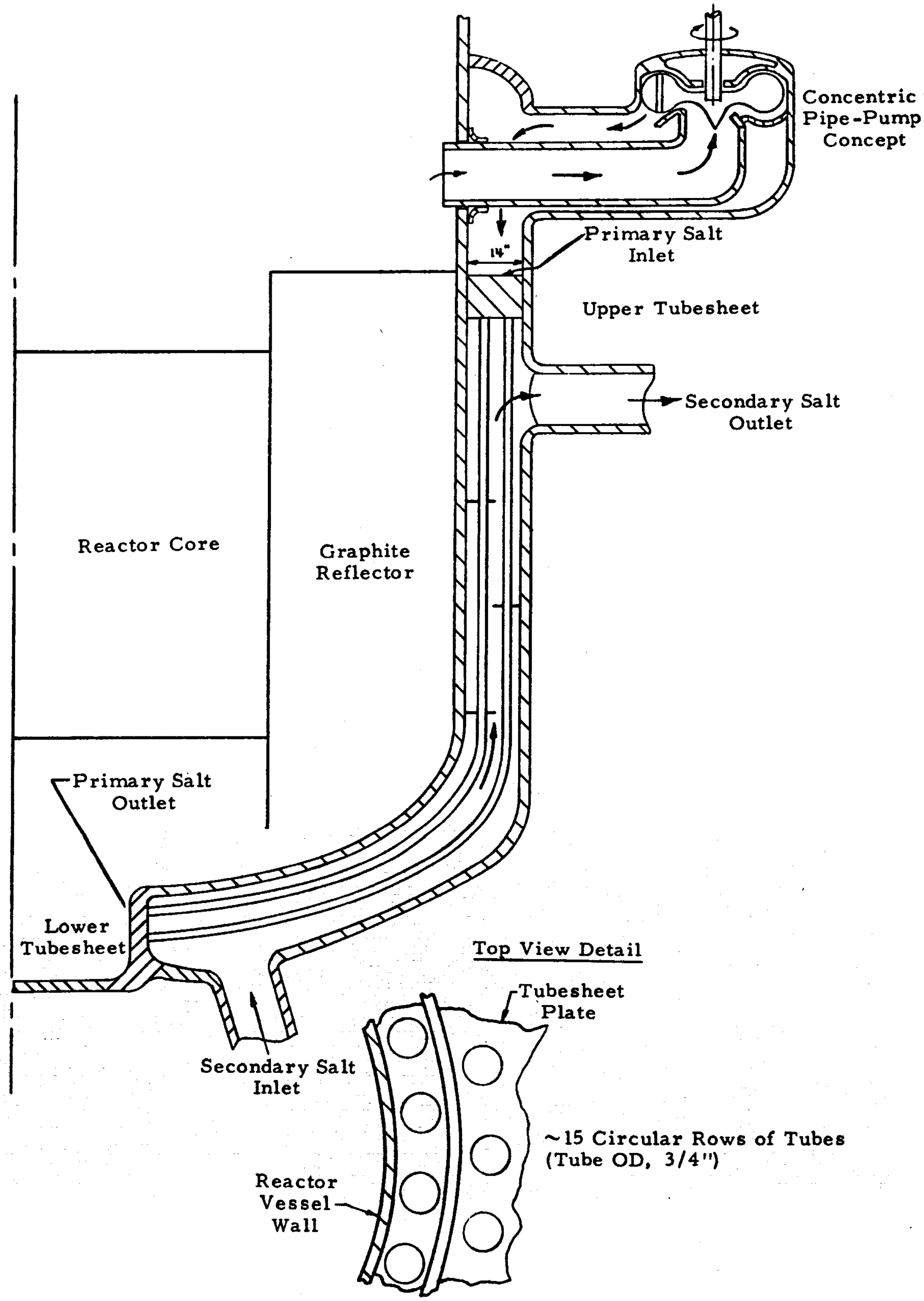


Figure 4.39: Close-Coupled Pump and IHX, Annulus of Tubes Surrounding Reactor.

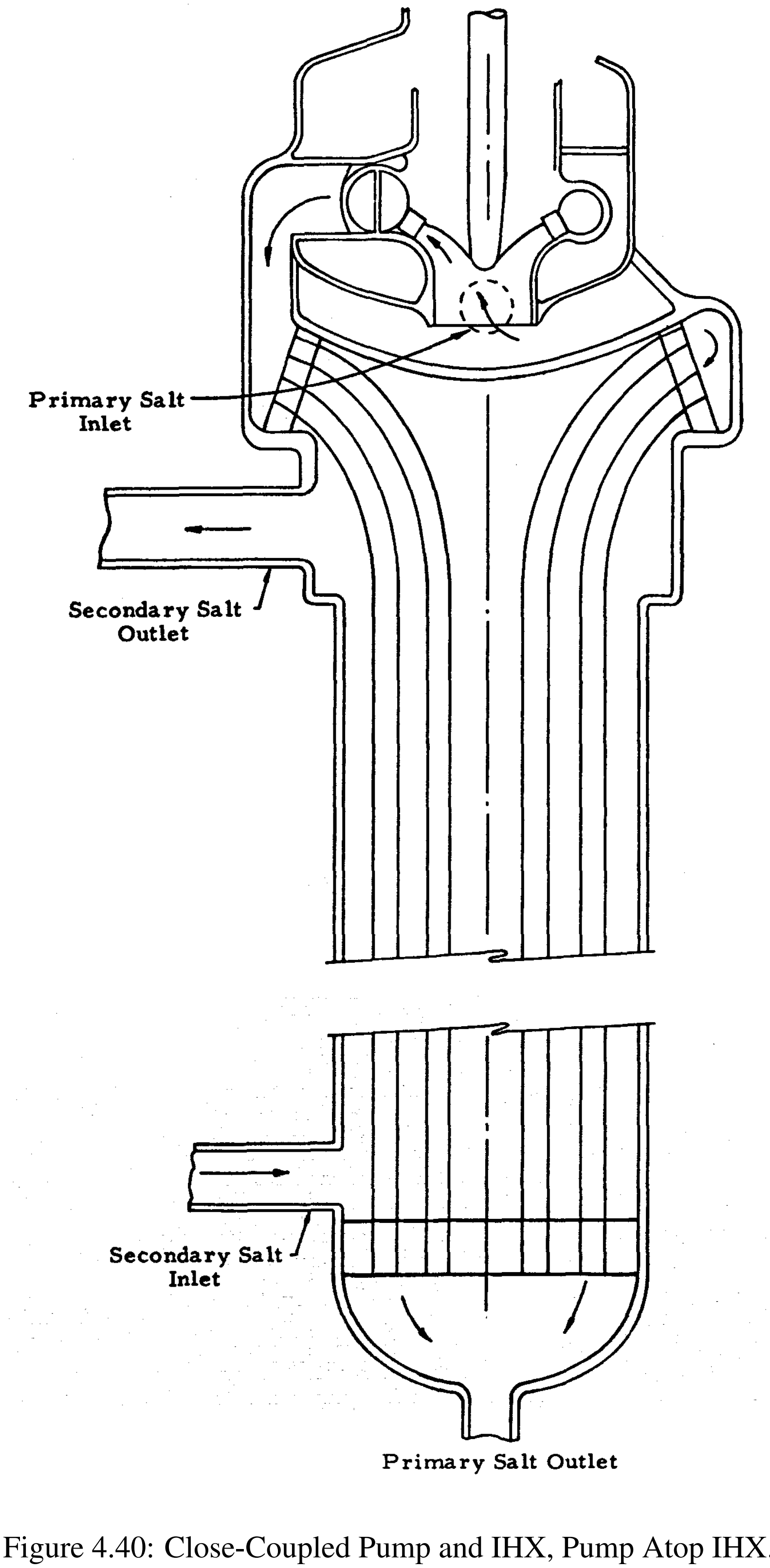


Figure 4.40: Close-Coupled Pump and IHX, Pump Atop IHX.

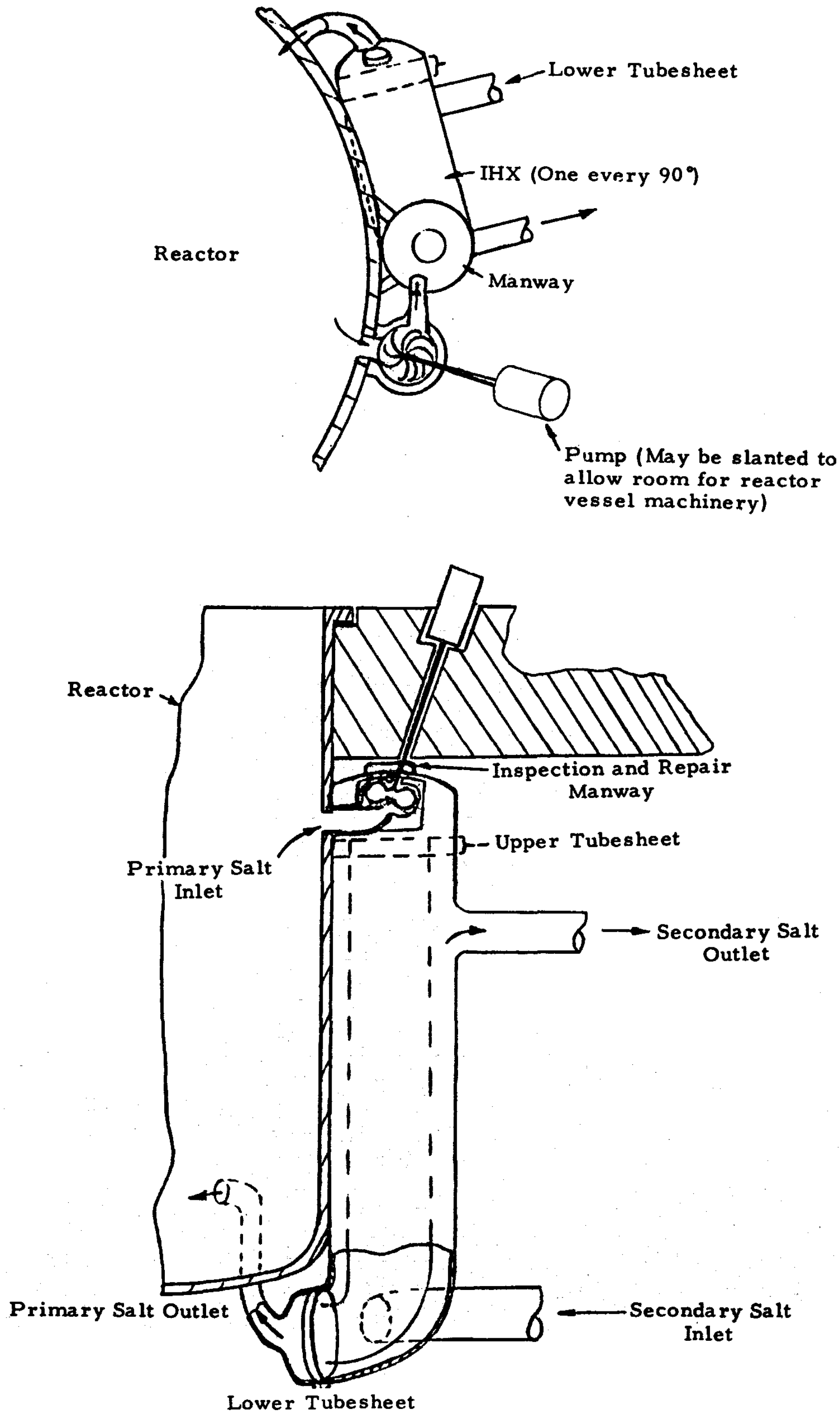


Figure 4.41: Close-Coupled Pump and IHX, IHX and Pump Mounted on Reactor.

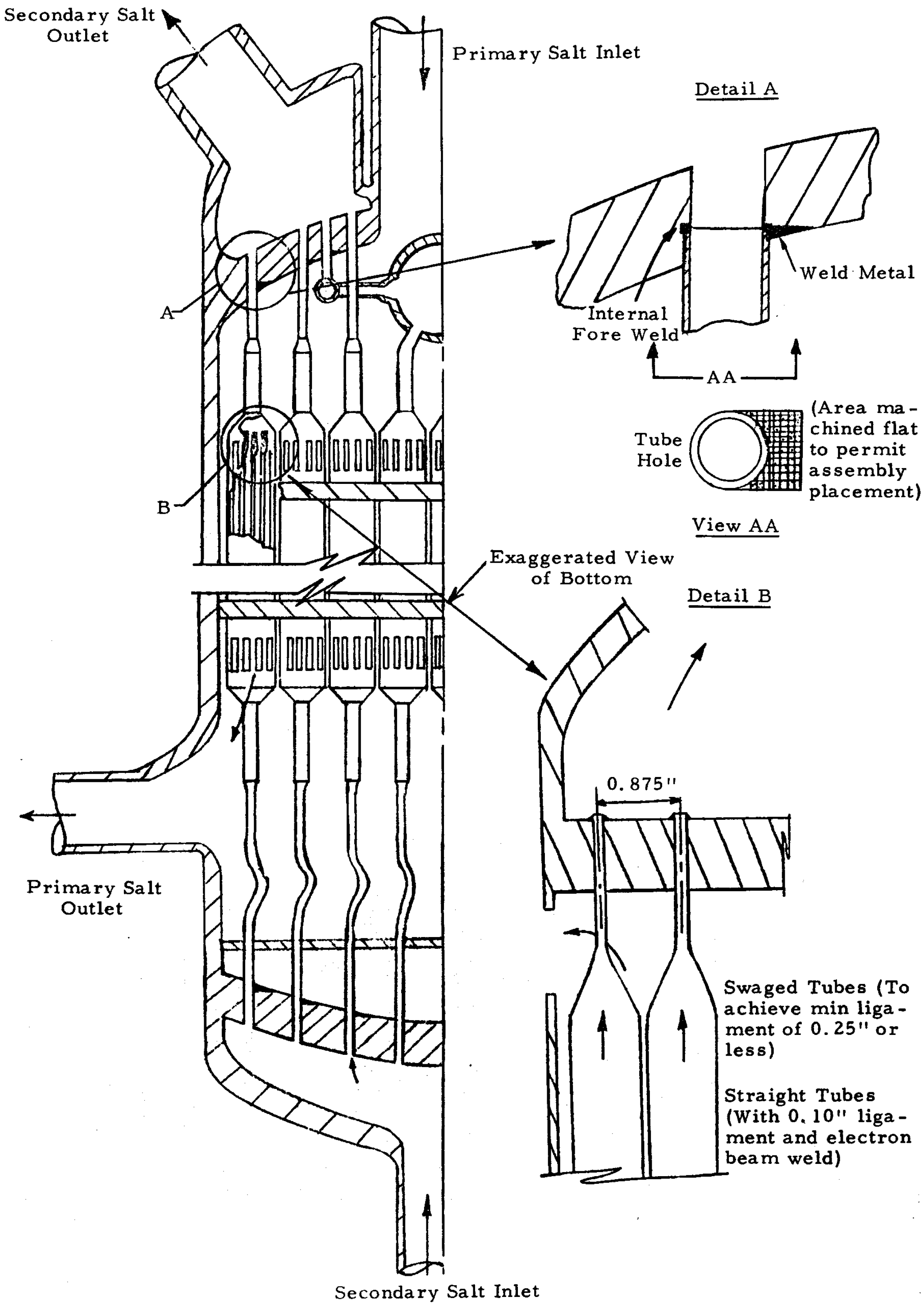


Figure 4.42: Modular Tube Assembly IHX.

Figure 4.42 depicts an advanced IHX concept with either primary salt on the tube side, utilizing very small-diameter tubes, or primary salt on the shell side, using a closely packed tube-bundle arrangement. Figure 4.43 portrays the design which utilizes primary salt on the shell side. The unit consists of a number of fuel-element-type modular assemblies of tubes which are attached to tubesheets. These tubes can be small in diameter yet be manufactured easily in the boiler shops. The modules can be fabricated in a separate shop, and the assembly operation can be performed by welding the modules in place; the concept of curved tubesheets and the specially machined weld-access preparation would be used. The small size of the tubesheets in each module permits small holes to be drilled without drill drift which affects the design of the tube-to-tubesheets. The small size of the modules makes it easy to perfrom the tubing operation by inserting the tubesheet on the tube rather than sliding the tube through two tubesheets—an operation that would eliminate the possibility of using swaged tubes. Using swaged tubes in this design permits very close packing of the tubes; this close packing decreases the primary system volume. An exaggerated view of the bottom of the unit, which shows sine-wave-type flexibility members, indicates the provision for differential thermal expansion in the design. Another reason for using this design would be its relatively easy manufacturing of small-diameter tubes to obtain a compact design with primary salt on the tube side. In this design the primary salt would flow down inside the tubes and out the bottom, and the secondary salt would enter in the sine-wave section and flow on the outside of the modules and out the top of the unit.

As stated previously, the major advantage of this design over other units, which use very small diameter tubes or use closely packed tube bundles, is that it will minimize the amount of delicate shop labor which is normally performed in the manufacturer's heavy vessel shop. A design with small-diameter tubes can be achieved by coordinating the technology used in fuel element manufacture and the technology used in the boiler shop. Further evaluation, of course, would have to be conducted to determine whether this design were a competitive design to the selected reference concepts.

For an advanced concept that could be applied to any IHX, a short study was conducted on the use of tubes with an elliptical cross section; circular and elliptical tubes were compared on the basis of equal hydraulic diameter, which is equivalent to comparing tube sections with equal pressure drop. The elliptical tube (for each calculation, the ellipse was taken with a major/minor axis ratio of 2) gave 25 percent more surface per unit mass flow in the tubes. In other words, the elliptical section-tube bundle could have about 25 percent less heat transfer area that the circular tube unit. One disadvantage with the elliptical tube is that the wall thickness is about 20 percent greater than that of the circular tube; however, most of the resistance to heat transfer is in the film coefficient, so that the increased wall thickness has little effect. The elliptical tubes would reduce the volume of the IHX but could introduce severe manufacturing problems.

4.3.4 Conclusions and Recommendations

The fixed sine-wave tube bundle and the ORNL designs are recommended as first-choice nonremovable and removable primary heat exchangers, respectively. Further study of the manufacturing problems will be required before a selection can be made among the concepts that contain the primary salt in the shell side of the unit. These concepts do have several attractive maintenance and inspection features.

It should be concluded also that four criteria were selected as the bases of evaluation for all units. The question of bundle removability was kept separate, and the four criteria considered, in order of their importance, were primary inventory, tube thermal stress, manufacturing, and maintenance.

Some conclusion should be made about the desirability of a removable unit versus a nonremovable unit. Current trends in the LMFBR program provide for tube-bundle removal because of certain factors, which are as follows:

1) The technology in sodium component design is not advanced to the stage where highly reliable units are possible.

2) It is undesirable to cut primary pipes because the reactor is not drained for maintenance, and because remote maintenance techniques would be required when they are not currently contemplated.

3) The component designer has quite a large amount of leeway in design arrangement because there are no requirements for minimizing primary inventory.

There appear to be incentives to utilize a fixed (nonremovable) tube-bundle design as a recommended reference concept. The incentives are related to system effects; therefore, the component designer can only consider these in a limited sense. These considerations are as follows:

1) It will probably be necessary to use remote maintenance and repair equipment on other components besides the IHX.

2) Pump maintenance will require the cutting of primary pressure boundaries.

3) The reactor can be drained down to the level of the bottom of the IHX (as a minimum).

4) The removal of a tube bundle requires the same type of handling as the removal of a complete unit because of fission product plate-out.

5) Remote equipment for cutting, welding, and inspecting is being developed by ORNL, and the use of this equipment has been somewhat successful.

Even though fixed tube bundles are being considered, more advanced remote tools may not alter the trend toward tube-bundle removal. Often units have exhibited vibration or maldistribution problems that could not be predicted analytically in advance. In these cases removal of the bundle for addition of baffle plates or flow guides becomes a very desirable feature. In concepts where removability can be achieved without any increase in the minimum primary salt inventory, this feature becomes even more attractive.

At this stage of development of the MSBR, it seems prudent to aim development toward the fixed-bundle design. By planning to conduct sufficient testing on flow, stress, and vibration in prototypes and/or models to ensure a trouble-free design, and at the same time by taking advantage of design development of heat-exchange equipment and analytical techniques over the next 10 years, B&W recommends the fixed-tube-bundle, sine-wave-tube IHX as the preferred approach.

4.4 Other Components

4.4.1 Coolant Salt Pump

The MSBR design study coolant salt pump is located in the cold leg of the salt system for five reasons:

1) To minimize the thermal duty (maximum temperature and rate of change of temperature) experienced by the pump.

2) To provide the maximum possible suction head for the pump.

3) To maximize the coolant salt pump pressure delivered to the intermediate heat exchanger. Since it has been established as a design condition that the tubes of the intermediate heat exchanger shall be held in compression by the coolant salt to control leakage and corrosion, location of the pump at this position reduces the cover gas pressure required during operation by about 200 psi.

4) To provide head for the purification of the coolant salt by filtration.

5) To allow storage and injection of makeup salt at the system cold-leg temperature.

The coolant salt pumps are of a standard centrifugal design utilizing salt-lubricated bearings and either a salt-controlled leakage or gas injection seal. Both salt seals and bearings are believed attainable in the time period of the study. Bearings are actually believed attainable today. If problems occur in salt seals, the presence of small quantities of noble gas (from a gas injection seal) in the salt should not deter the heat-exchanger performance. The gas would come off in the surge space provided in the rupture disk housings.

A coolant salt cover gas system provides the following functions:

1) Pressurizes the rupture disk housing gas space and the drain tank gas space.

2) Provides the appropriate concentration of BF3 in argon to stabilize the coolant salt concentration.

3) Transports tritium leaving the coolant salt to an off-gas system.

The coolant salt pumps are driven by fixed-speed pump motors located in the steam generator cell. Since all leakage is out of the coolant salt system into the primary, the major activity in the coolant salt will be 24Na formed in the intermediate heat exchangers. This activity is reported by ORNL to be 65 curies. When distributed through the volume of the coolant salt, the resulting concentration of 24Na is about 8 millicuries/ft3.

The radiation level in the steam generator cell should thus be between one and 100 mR/hr and be relatively accessible. The coolant salt pump will deliver a flow of 15.50 Mlb/hr at a pressure head of 350 psig.

4.4.2 Rupture Disk

A multipurpose rupture disk housing is located at the high point of the loop adjacent on the steam generator inlet. This housing provides the following functions:

1) To hold a rupture disk that will blow out in the event a tube break in the steam generator overpressures the system.

2) To provide a pressurization gas space for regulation of the coolant salt static pressure head by the cover gas supply system.

3) To accommodate thermal expansion of the coolant salt by gravity overflow through a line to coolant salt storage tank.

4) To provide a cover gas-coolant salt interface at the highest coolant salt temperature in the system to promote diffusion of tritium into the cover gas and absorption of BF3 into the salt.

The detailed design of the disk, its housing and the steam generator cell will be a Task II effort.

4.4.3 Coolant Storage Tank

A coolant storage tank will be provided for each coolant salt loop. The tank will provide:

1) Storage space for 2500 ft3 of coolant salt when drained from the loop.

2) A chemical treatment volume as the filtered salt is returned to the tank through the cover gas (argon and BF3) by means of a sparging ring.

3) A sump for makeup salt being returned to the loop.

4) A receiver for coolant salt additions to the system.

The tank will be built of Hastelloy-N material and be positioned in the steam generator cell to allow gravity drainage from all coolant salt components.

4.4.4 Coolant Salt Filter

A coolant salt filter is provided on a small side stream from the pump discharge. This filter will provide the following functions:

1) At plant start-up, remove particulate material left from construction.
2) After each maintenance, remove particulates resulting from maintenance operations and the ingress of moisture and air reacting with the system materials.
3) During normal operation, remove particulate materials resulting from wear and leakage.

Flow to the filter is controlled by a freeze valve. The flow rate is 1 percent of the system flow rate.

4.4.5 De-Mister

A de-mister consisting of loosely packed Hastelloy-N wool in a 4 in. pipe section is provided on the cover gas vent line. The de-mister prevents mist created by the sparging flow from being carried into the off-gas system.

4.4.6 Coolant Salt Melt Tank

A 25 cu ft Hastelloy-N tank is located on the roof of the steam generator cell. Additions to the system are made by filling the tank with dry frozen salt and heating the tank with flame heaters. The salt will flow by gravity to the drain tank after passing through a plumber's "U" seal which forms a freeze valve. The freeze valve is also flame-heated.

Chapter 5

Steam Power System

5.1 General

The steam power system proposed for use in the MSBR plant concept is designed to:

1) convert the thermal energy released in the multiple MSBR salt coolant loops into electrical power at a high plant efficiency,

2) to operate as the reactor heat absorption sink under all modes of reactor plant operation,

3) to permit the hot and cold start-up of the MSBR plant independent of outside steam supply, and

4) to convert reclaimed plant decay and chemical processing heat energy to electrical power output.

This system consists of a supercritical pressure steam cycle with once-through steam generation, condensing turbine generators, steam reheating and regenerative feedwater heating. Reactor heat is converted into turbine-generator motive steam at the four steam generators and four reheaters serving each of the four MSBR coolant loops. Plant decay and chemical processing reaction heat is introduced into the steam power system in the form of low-pressure steam for feedwater heating utilization. When operating with 3515 psia and 1000°F main throttle steam, with 1000°F reheat steam, and exhausting to 1.5 in Hg abs., the system delivers 1000 MWe nominal at an overall net plant efficiency of 43.0 percent.

Except for the steam generator and reheater components, special feedwater and reheat steam preheating components, and a supercritical auxiliary boiler, which are unique to the system, the system utilizes designs and equipment normally associated with conventional turbine-generator type power stations. Design of the proposed steam system concept should not limit the feasibility of the molten salt reactor plant concept since the constituent components are either conventional in present-day power station applications, or are within the extension of present technology. With the exception of the steam generator and reheat components, which are located in high temperature steam cells, the components of the steam power system are all located within the combined Feedwater and Turbine-Generator Building or in the area adjacent to it

5.2 Special Design Considerations

Integrating the proposed steam power system into the molten salt reactor plant concept requires providing special means to avoid thermal shock and possible coolant salt freezing problems in the system steam generator and steam reheating component's resulting from the relatively high operating and liquidus temperatures of the sodium fluorborate coolant salt.

In order to safely accommodate in the system steam generator and steam reheat components an average coolant salt operating temperature of about 1000°F, an estimated maximum allowable tube wall thermal gradient limit of about 300°F, and a coolant salt freezing temperature of 835°F, the feedwater and cold reheat steam flows to these components have to be preheated to inlet temperatures much higher than normally used in conventional steam power cycles. For the purpose of this study, the lower limit of the inlet feedwater and cold reheat steam temperatures to these components have been selected to be about 710°F and 650°F, respectively. These temperature limitations reflect the evaluation of the effects of reduction in plant cycle efficiency with increase in feedwater and cold reheat inlet temperatures, of turbine extraction limitations, of additional power and/or equipment size penalties for increased pressure drop and flow rates, and plant cycle start-up preheating requirements.

Various methods for providing the necessary feedwater and reheat steam preheating were reviewed, including indirect heating with either supercritical-pressure steam or coolant-salt utilizing high-pressure exchangers, and combination indirect heating and direct mixing with subcritical-pressure steam utilizing vapor compression in a modified Loeffler cycle. A rather unconventional approach, initially suggested in ORNL study report No. 4541, in which supercritical-pressure steam is utilized in series for the indirect preheating of reheat steam and the final stage feedwater heating by direct mixing, was evaluated as the more efficient and convenient method, and, as such, is selected for application in this study. This method dictates the use of supercritical steam in special reheat-preheater exchangers and feedwater mixing chambers, and special feedwater supply pressure booster pumps, the design of which, although not conventional, are feasible and within forseeable technological development.

5.3 Selection of Steam Cycle Conditions

In addition to providing a simple and effective means of achieving the necessary feedwater and reheat steam preheating, the use of a supercritical-pressure steam cycle as proposed, inherently offers a higher thermal efficiency than obtainable from a subcritical-pressure cycle, and as such, offers considerably lower electric power production cost, less fuel processing, less fission product accumulation, and less heat rejection to the environment. Also, the use of a supercritical-pressure steam results in less mass flow and provides superior steam-side heat transfer coefficients, such that, the cycle equipment capital cost is estimated to be not more, and very likely less than, the cost of a subcritical-pressure system for similar service.

Reheat and nonreheat cycle steam systems were compared, and on the basis of a thermodynamic gain in heat rate and a gain in effective turbine efficiency a reheat steam cycle is recommended for specific application to the MSBR plant concept. Improvement in turbine efficiency obtainable in the reheat cycle, due to the reduction in moisture losses, results in the substantial reduction in size of the turbine and steam generator units as well as the salt, main steam, feedwater and condensate piping, cycle heaters and pumps, and cooling water flow required for the rated duty. In addition, a 3600 rpm tandem-compound type turbine could be effectively utilized in the cycle.

These advantages are not offset in the nonreheat cycle by the elimination of the need for reheater and reheat preheater units, reheater steam and salt piping, and their respective controls. This is due primarily to the major disadvantage of the nonreheat cycle, in that the turbine required would have to be of a more costly cross-compound type, with an 1800 rpm low-pressure section, because of the relatively high moisture content of the steam in the turbine low-pressure section (18 percent moisture leaving the last stage blades). Also an expensive moisture separator is required between the turbine intermediate-pressure and low-pressure sections to minimize moisture and to increase turbine efficiency. In addition, the turbine building would have to be increased in size to accommodate the larger nonreheat cycle layout.

While neither a reheat or nonreheat 1000 MWe turbine-generator unit utilizing superheated high-pressure steam has, to date, been placed into operation, units of this size are presently under design and/or are within the capability of current technological knowledge. Based on evaluation of data provided by a leading turbine manufacturer of units of these types, the differential in estimated cost in favor of a reheat, tandem-compound, single-shaft turbine generator over the nonreheat, cross-compound type should be in excess of 4.5 million dollars. In addition to greater cost, the nonreheat cycle requires approximately 273 additional BTU/kW-hr estimated on a net plant basis or 254 additional BTU/kW-hr on a net turbine cycle basis. These values are differentially taken from the reference base value of 7528 BTU/kW-hr (ASME-62-WA-209).

In order to effectively and economically produce the required power to safely maintain the essential support systems and equipment in operation upon loss of the main turbine generator or a failure of the power grid, a dual admission, condensing type, standby turbine-generator unit is introduced as part of the concept cycle. This unit is capable of operating on either high-pressure main steam or intermediate-pressure turbine crossover steam, and is designed to operate in parallel with the main turbine to export power.

In the event of a main turbine trip and loss of plant load, an ample supply of high-pressure steam is available, from either decay afterheat removal in the cycle steam generator or from the auxiliary start-up boiler, to assure adequate standby power generation to maintain reactor operation at a banked, minimum load condition for any desired operating period. Sufficient afterheat disposal steam is generated to operate the standby turbine for at least several hours. For longer periods of power loss, the auxiliary start-up boiler can be fired up and placed on the line to supply the necessary standby motive steam.

The standby turbine-generator unit is required to always be available and kept at temperature. However, for the majority of the time the steam power cycle functions normally off the mains with no need for independent standby power. At such times, the standby turbine generator is efficiently operated at full load on main turbine crossover steam to contribute, together with the main turbine, to the total plant power output. When needed for standby protection, the unit is automatically switched over to high-pressure steam operation, and by means of load rejection devices, all nonessential electrical loads are dumped. In this manner, the unit is always available for standby operation and at the same time capable of economically generating a proportionate share of the total plant output, making full and effective use of the unit.

5.4 Design Criteria and Assumptions

The proposed steam power cycle is designed to comply with the following criteria basic to the MSBR plant concept:

a) The MSBR plant is intended for baseload operation.

b) With 3515 psia, 1000°F throttle steam and 1000°F reheat steam exhausting to 1.5 in. Hg abs., the rated net plant electrical output is to be 1000 MWe.

c) The power cycle is to be capable of providing means for reactor heat rejection under all modes of plant operation.

d) The plant is to be capable of hot start-up and indefinite banked load operation, as well as cold start-up operation independent of outside steam supply.

e) Due to use in the plant concept of the secondary coolant-salt as an intermediary heat transport fluid, steam generated by the cycle is not radioactive and the steam power cycle portion of the plant may be designed to conventional codes.

f) Adequate river water is to be available for cycle circulating cooling water application.

The following design assumptions, which are typical and representative of present practices and procedures employed in the design of conventional steam power stations, are utilized in the development of the proposed cycle:

a) Turbine-generator units are of a six-flow tandem-compound type rated on the basis of fully-loaded low-pressure steam end operation with valves wide open.

b) Eight stages of feedwater heating utilizing main turbine extraction steam are used to provide feedwater at about 548°F. Seven stages are of the closed type arranged in two independent, parallel, half-capacity circuits which share a single full-capacity deaerating heater stage in common. The heater units are of conventional shell-and-U-tube type design selected on a basis of:

1) Five percent pressure drop allowance from turbine extraction stage to heater inlet.

2) A 0°F feedwater terminal difference in the three top extraction heater stages (with the incorporation of desuperheating sections) and a 5°F feedwater terminal difference in the remaining closed type heaters.

3) All of the closed type heater drains cascade back to the next lower pressure heater stage with a 15°F drain cooler approach, except for the bottom low-pressure closed heater stage which cascades back to the condenser with a 10°F drain cooler approach.

c) Boiler feed and boiler feed booster pump units are to be tandem driven from a dual admission, steam drive turbine utilizing low-pressure, crossover extraction steam, or high-pressure main steam, while exhausting to the main condenser. The deaerator feed boiler feed booster pump is driven off the front end of the boiler feed pump turbine driver through an appropriate speed reducer. The boiler feed pump tie-in is located between the third and fourth feedwater heater stages from the top in order to keep the temperature of the feedwater at the suction to the pumps at a low level of about 350°F. A combined feedwater pressure rise of 125 percent above main throttle steam pressure is to be produced by the boiler feed and boiler feed booster pumps.

d) The main condenser is a three cell, single pressure, surface type designed to limit cooling water temperature rise to 15°F and flow velocity to about 7 ft per sec with less than a 20 ft head loss.

e) The maximum allowable total reheat steam pressure drop from turbine extraction to intercept valve is limited to 10 percent of the turbine extraction pressure, i.e., about 60 psi.

f) Course control of steam generator and reheater outlet steam temperature is achieved by adjustment in heat input (i.e., coolant salt pumping rate). Fine control is attained by desuperheating by means of feedwater injection (attemperation).

g) Cycle feedwater makeup is to be provided by the demineralization of raw water, and full flow capacity polishing demineralization is used to maintain feedwater condensate purity in the cycle.

In order to further define performance, or to simplify the steam power system operation and start-up analysis, the following developed design assumptions are included:

a) Component steam generator and reheat-preheater tubeside high-pressure steam pressure drops are limited to 300 psi and 100 psi, respectively, in order to minimize pressure-booster pump power consumption and its detrimental effect on plant cycle efficiency.

b) Minimum allowable flow rate through the cycle turbogenerator and steam generator components are to be limited to 5 and 10 percent of the rated flow rate, respectively. At lower flow rates there are problems in insuring stable operation.

c) A separate variable speed, motor-driven type pressure-booster pump is to provide final stage feedwater pressurization at each steam generator unit.

d) A separate temperature controlled, supercritical steam/feedwater mixing chamber and a feedwater pressure boosting pump are to provide the required feedwater flow at each of the four component steam generator units.

e) A separate supercritical steam heated reheat-preheater unit is to serve each of the four steam reheat units to provide the required cold reheat steam preheating.

f) Plant auxiliary power consumption is evaluated at about 36.7 MWe, and total stray heat loss from the reactor system is estimated at 25 MWt.

g) Electrical power load required during standby or emergency operation is evaluated at less than 18 MWe.

h) A supercritical pressure auxiliary steam pressure system consisting of a light oil-fired, 750,000 pound per hr, once-through type steam generator, with motor-driven forced draft fan and auxiliary boiler feed pump, start-up flash tank and fuel oil supply system, is selected as the most effective means to provide the required start-up feedwater and preheating design conditions as well as means for indefinite standby plant operation.

5.5 Description of MSBR Steam Power System

5.5.1 Normal Operation

Basic data for full load conditions in the conceptual design steam system are summarized in Table 5.1, and a simplified flowsheet is shown in Figure 5.1. Superheated steam leaves the four once-through type steam generators at about 3600 psia and 1000°F at a maximum total rate of about 10.5 Mlb/hr. Coolant salt at 1150°F is supplied to the steam generator units at a controlled rate to hold the steam outlet temperature to within a few degrees of 1000°F. Individual outlet steam attemperators, or desuperheaters, supplied with 700°F feedwater, assists in holding the outlet steam temperature to within tolerances.

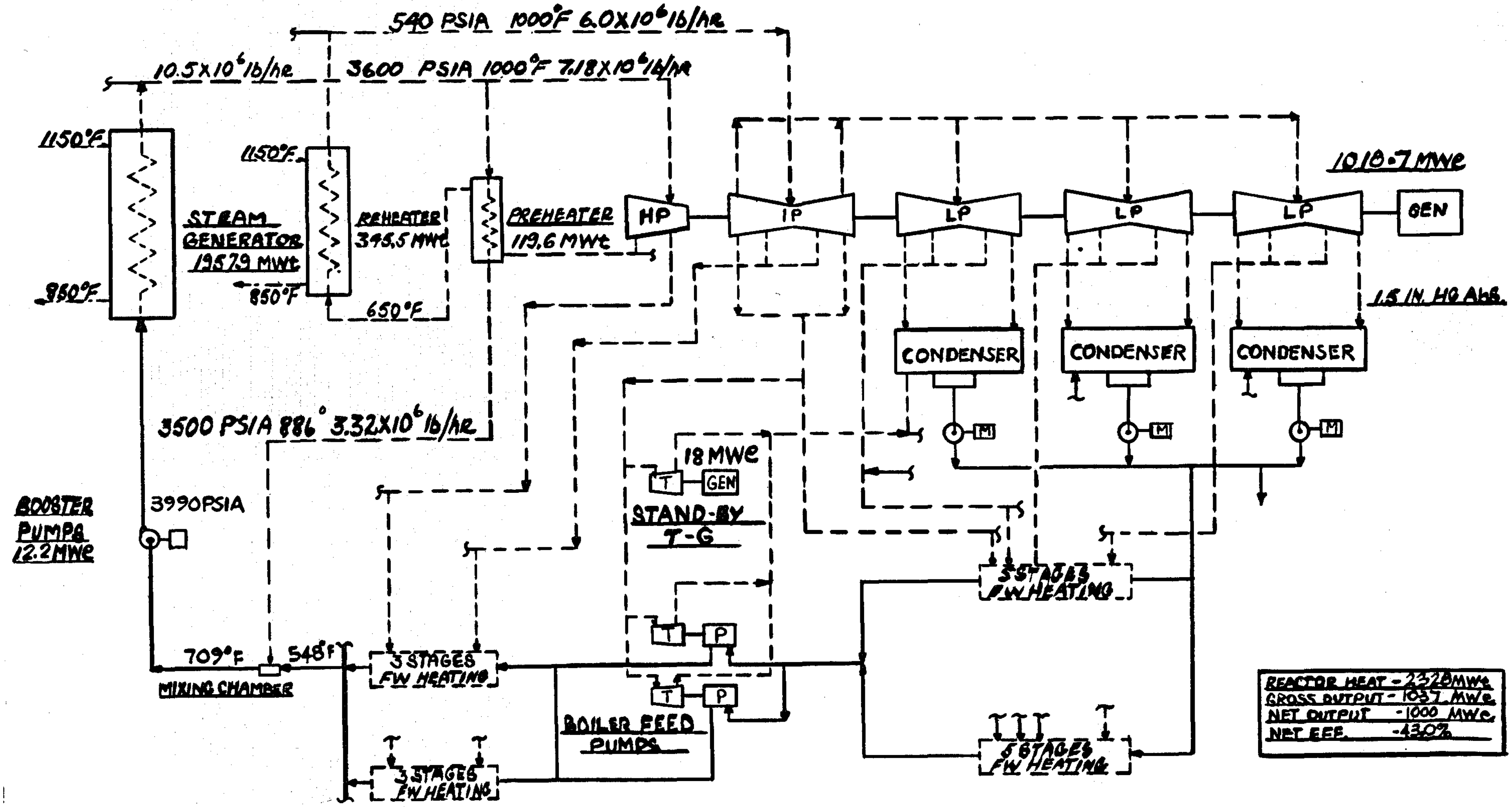


Figure 5.1: Simplified MSBR Steam System Flowsheet.

About 3.28 Mlb/hr of the steam leaving the steam generators is diverted for cold reheat steam preheating and the last stage of feedwater heating; the remainder enters the 3600 rpm high-pressure turbine throttle valve at 3500 psia and 1000°F. Part of the high-pressure turbine steam is extracted at a pressure suitable for the final stage of the regenerative feedwater heating. The remainder of the steam in the high-pressure turbine expands to about 600 psia and 549°F before exhausting into cold reheat steam mains leading to the reheat steam preheater. A portion of this exhaust steam is also used for feedwater heating in the No. 2 stage heaters.

The minimum temperature for the steam entering the reheaters is required to be 650°F. The 549°F high-pressure turbine exhaust steam is therefore preheated, or tempered, in the shell side of surface type heat exchangers using prime steam at 3600 psia and 1000°F in the tubes. The high-pressure steam leaves the tubes at about 3500 psia and 860°F and is used for preheating the feedwater, as described below. The preheated "cold" reheat steam, now at 650°F, then enters the four reheaters, which are supplied with coolant salt at 1150°F at a controlled rate to provide 1000°F steam at the exit. The reheated steam is supplied to the stop valves of the intermediate-pressure turbine section at about 540 psia and 1000°F.

Part of the intermediate-pressure turbine steam is extracted to heat the No. 3 stage feedwater heaters of the cycle. The remainder of the steam is exhausted as crossover steam to the low-pressure turbine, the tandem driven boiler feed pump and boiler feed booster pump turbine driver, and to the standby turbine generator. Although these drive turbines normally operate on crossover steam extracted from the intermediate-pressure turbine sections, they can also accept 3500 psia main steam during start-up or other times when extraction steam is not available from the intermediate-pressure turbine. Steam for the No. 4 stage feedwater heaters is also taken from the intermediate-pressure turbine exhaust.

Each of the six low-pressure turbine cylinders has three extraction points for the remaining stages of regenerative feedwater heating. About 4 Mlb/hr is finally exhausted from the low-pressure turbines into three surface condensers operating at about 1.5 in. Hg abs. Hot well pumps circulate the 92°F condensate through full-flow demineralizers for the condensate polishing necessary to obtain the high purity water required in a once-through steam generation. The feedwater flow then splits into two parallel paths for successive stages of regenerative feedwater heating and deaeration. Booster pumps taking suction from the cycle deaerator circulate the water through feedwater heater stage No. 4 and to the two main boiler feed pumps. Each is driven in tandem with the boiler feed booster pump by a steam turbine with a brake horsepower capacity of 19,400 hp.

The feedwater, now at a pressure in excess of 3600 psia, flows through the three top regenerative heaters and leaves at 3500 psia and 548°F. Each of the 3.59 Mlb/hr parallel-flow streams then enters a mixing chamber where the steam, at approximately 3500 psia and 860°F, from the tube side of the reheat steam preheater is mixed directly with it. The resulting mixture, actually compressed water at about 3475 psia and 700°F, then enters the boiler feedwater pressure-booster pumps. They are also shown as motor-driven pumps, but optimization studies would be likely to indicate an advantage for steam-turbine drives for some of the units. The feedwater, now at about 3990 psia and 710°F, is returned to the steam generator at a rate adjusted to the plant load by controlling the pumping rate.

5.5.2 Standby Operation

Upon main turbine tripout, or during prolonged steam cycle operation in a banked load condition, a standby cycle operation exists in which the main turbine is down, the standby turbine-generator unit is operating to provide necessary power to maintain the vital plant standby functions, and a minimum feedwater flow is maintained through the cycle steam generator units thus permitting the reactor to be shut down (standby operation with the reactor at low power is also possible without these additional steps).

In this mode, the steam system is initially operated on steam generated from reactor decay heat until the warmed-up auxiliary steam generator unit can be brought on stream in the following manner:

a) A cycle is established in which main steam is utilized to drive the condensing type standby turbine-generator and boiler feed pump drive turbine units to maintain feedwater flow from the main condenser, through the main steam heated feedwater mixing chambers, to the steam generator units.

b) Main steam output not utilized for standby turbine-generator and boiler feed pump operation, for feedwater heating in the mixing chambers, or for maintaining the condenier vacuum seal, is desuperheated and rejected to the main condenser, or if necessary may be rejected to the atmosphere.

c) Sufficient feedwater flow is then diverted through the auxiliary steam generator to permit commencement of burner firing with the consequent raising of boiler outlet pressure and temperature.

d) When the auxiliary steam generator outlet conditions match the main steam conditions the output flow replaces the main steam outlet flow from the reheat-preheaters to the mixing chambers serving cycle steam generator units. As the steam output from the cycle steam generator units decays with the decay in reactor afterheat generation, output steam from the auxiliary steam generator is phased in to make up the desired standby operation steam demand. This is achieved by restricting feedwater flow to the cycle steam generators, by means of the in-line feedwater flow control valves serving these generators, while proportionately increasing feedwater flow through the auxiliary steam generator to maintain a constant total standby steam output.

e) When the restricted feedwater flow can no longer be effectively converted to main steam in the cycle steam generator units, the full feedwater flow is diverted through the auxiliary steam generator, the cycle pressure booster pumps are bypassed, main steam flow bled through the reheat-preheaters is diverted to the main condenser, and sufficient output main steam flow is diverted, via the mixing chambers, through the cycle steam generators and reheat-preheater units to keep them at about 1000°F, in order to minimized any thermal stress problems. Thus, the auxiliary steam generator output is on stream and the cycle steam generator capability is kept in a banked condition ready for hot restart operation.

f) Adequate steam flow may be bypassed through the cold-reheat circuit and the extraction side of the regenerative feedwater heaters and exhausted to the main condenser to maintain them sufficiently warm to preclude thermal stress problems on turbine restart.

5.5.3 Start-up and Shutdown Operations

The steam power system start-up cycle depicted in simplified form on Figure 5.2 is intended to supply the necessary system warm-up, deaeration and motive steam for MSBR cold start-up, hot "black" restart and shutdown operations. This cycle is somewhat different from cycles of similar function in conventional fossil-fired, supercritical-pressure steam stations in that it utilizes a supercritical-pressure type auxiliary steam circuit to provide the required start-up steam. In order to avoid coolant-salt freezing and excessive thermal gradient problems in the system steam generator and reheater units during start-up, temperatures and pressures in these units should be maintained as close as possible to full load operating conditions before any steam loading is attempted in these units. The supercritical-pressure auxiliary steam circuit achieves this aim by providing a means to attain at these units a 1000°F inlet fluid temperature, at pressure, during zero power operation, in addition to providing a means to control the minimum inlet feedwater temperature to the steam generator units to about 700°F, and the minimum inlet cold reheat steam temperature to about 650°F, during steam loading in the power range.

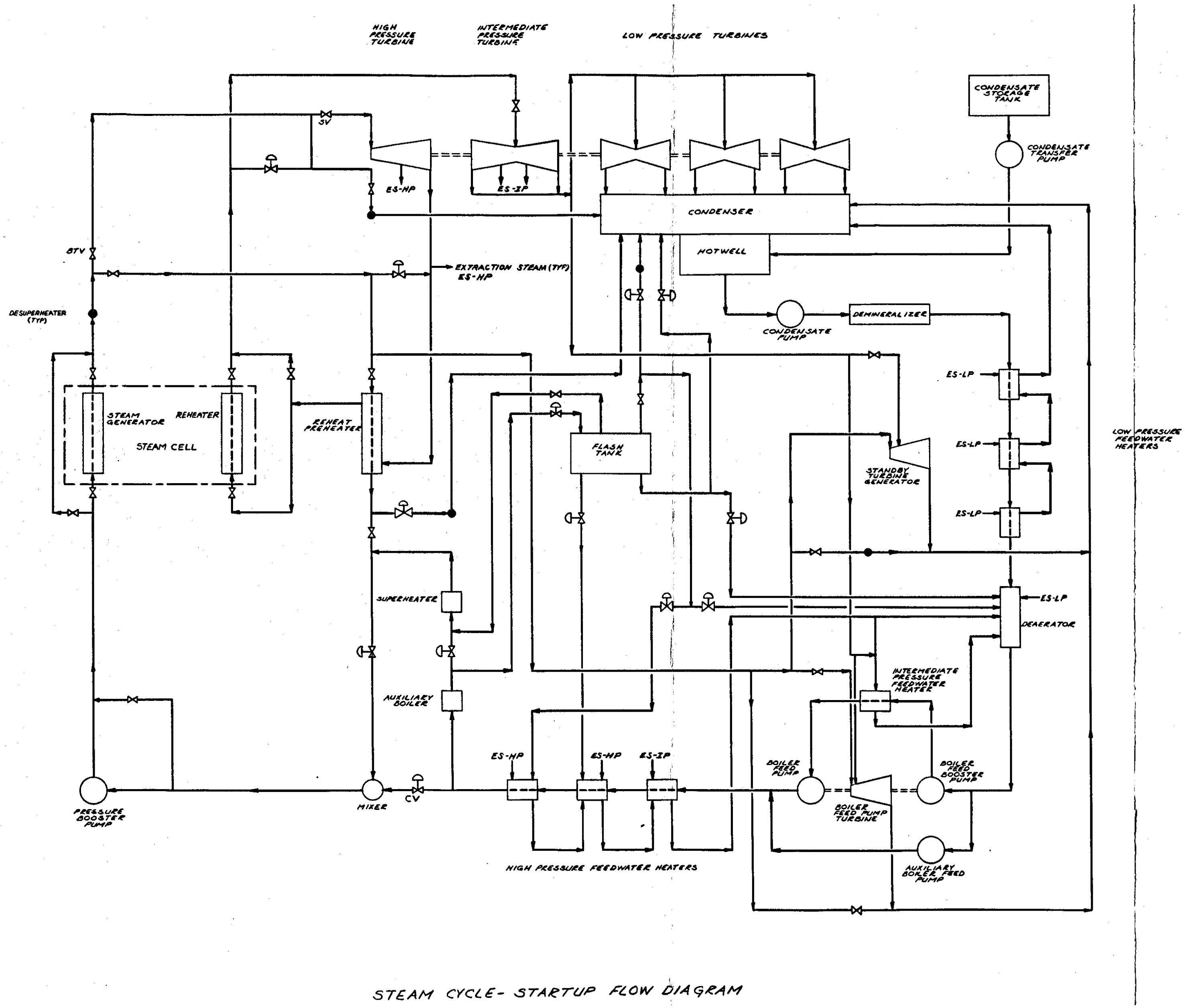


Figure 5.2: Steam Cycle - Startup Flow Diagram.

This start-up cycle basically consists of an independently fired auxiliary super-critical steam generator unit, with its associated start-up flash tank, pressure letdown and regulation valves, auxiliary boiler feed pumps, FD fan and support systems, connected to operate in parallel with the steam system and capable of supplying system start-up steam to approximately 10 percent of rated plant load.

5.5.4 Cold or Initial Start-up Operation

The basic steps in the cold start-up operation of the MSBR plant are listed below in which it is assumed all systems are in a cold and empty condition and no outside source of start-up steam is available.

Salt Systems

Primary and secondary cells are electrically heated and helium circulated in salt systems by primary and secondary pumps. When at temperatures above liquidus, fill salt loops and circulate isothermally at flow rates required for zero power mode operation.

Make reactor critical at zero power. Power controlled by automatic neutron flux level controller to maintain fuel salt temperature at 1050°F. Increase reactor power. Increase reactor ΔT. Increase coolant salt flow rate to match feedwater heat extraction. Increase reactor power.

At about 8 percent of full load, reactor is placed in a temperature control mode in which reactor outlet temperature set point is adjusted for subsequent load-following control.

Steam Power Systems

Charge feedwater circuit from condensate storage. Cycle feedwater through demineralizers for initial cleanup. Fire auxiliary steam generator to preheat and deaerate feedwater.

Warm-up steam lines, turbines and feedwater heaters while bypassing steam generator and reheater units. Roll and synchronize turbine-generator units and roll BFP turbine drive. Load turbine-generator units to minimum loads.

Fire auxiliary steam geneator to rated output conditions. Preheat steam generator and reheater units. Load BFP turbine and phase out motor driven auxiliary BFP. Reduce feedwater temperature by tempering with auxiliary steam to about 700°F. Place pressure booster pumps in operation and increase reactor output to maintain output steam conditions and minimum load. Divert main steam flow (from reheat preheater) to phase out auxiliary steam generator output utilized for feedwater heating. Take auxiliary steam generator off the line.

System is self-supporting at about 8 percent load. At about 20 percent load steam temperature control is activated in which reactor outlet temperature is regulated as a function of load while the steam temperature controller holds steam temperature at 1000°F.

On a cold or initial start-up, makeup feedwater is introduced into the steam cycle at the main condenser hot well from the site condensate storage facilities via condensate transfer pumps. Hot well condensate is pumped in sequence through polishing demineralizers, the lower 3 stages of regenerative feedwater heaters to the cycle deaerating heater, where level is maintained and from which all feedwater flow originates.

A pre-firing feedwater flushing cycle is then established in which feedwater is circulated in series, by means of the auxiliary boiler feed pumps, through the intermediate and high-pressure regenerative feedwater heaters, the auxiliary steam generator, the start-up flash tank, the main condenser hot well, the polishing demineralizers, the low-pressure feedwater heater units and back to the deaerator supplying the suction to the pump units. In this manner, a minimum feedwater cycle flow of about 10 to 30 percent of the rated auxiliary steam generator full flow is continually circulated until the concentration of solids has been reduced by demineralization to acceptable limits to provide initial feedwater cycle cleanup.

At this point in the start-up cycle operation, the superheating section of the auxiliary steam generator is bypassed so that the feedwater flows through the unfired furnace section of the steam generator, through a pressure-regulating boiler extraction valve station to a start-up flash tank. Pressure in the steam generator furnace section is controlled by regulation of the set pressure of the boiler extraction valve station. Feedwater level is set and maintained in the flash tank by means of a level control valve discharging to the main condenser.

Upon completion of the preliminary cycle cleanup operation, an initial firing (black start) cycle, which functions to bootstrap the feedwater temperature to suitable auxiliary steam generator inlet and deaerator pegging values, is established as follows:

a) While maintaining a minimum feedwater circulation rate, the flushing cycle is modified so that the makeup flow to the deaerator is blocked off and the flash tank drain flow is diverted from the condenser to the deaerator, thereby, temporarily isolating the condenser from the cycle.

b) The deaerator pressure is then set so that the corresponding saturation temperature is commensurate with the minimum allowable feedwater inlet temperature recommended for the auxiliary steam generator (20 psia). At this point, cooling water circulation through the condenser is established, and the mechanical-atomizing start-up burner, or burners, is fired at a low load operation sufficient to heat up the cycling feedwater to the deaerator saturation temperature (~260°F).

c) When proper inlet feedwater temperature is achieved, the condenser is reconnected to the cycle by directing the flash tank drain flow to both the condenser and the deaerator and readmitting condensate flow to the deaerator from the condenser, thus completing the feedwater cycle. Control valves in the drain lines to the condenser and deaerator function to control flash tank level and maintain deaerator set pressure respectively.

d) The firing rate is then increased, while simultaneously increasing the boiler extraction valve station set pressure, to raise the steam generator outlet heat content to permit the controlled increase in pressure and temperature at the flash tank. Any flash tank steam generated is admitted to the deaerator.

With the auxiliary steam generator pressure set at about 3550 psi and with increasing flash tank pressure and temperature, deaerator pegging and steam cycle warm-up operations can be initiated as follows:

a) Flash tank drain flow is diverted from the deaerator to the extraction inlet side of the No. 2 stage high-pressure feedwater heaters while the flash tank level control flow is maintained to the condenser. The deaerator is then pegged at about 30 psi with the drain flow from the No.2 stage heaters.

b) Steam generated within the flash tank flows through the auxiliary steam generator superheater section, cooling the superheater, and is utilized to warm up the steam lines, turbine units and the tubeside of the cycle reheat-preheater units, and to seal the turbines to permit establishment of condenser vacuum. Warm-up steam, however, is bypassed about the cycle steam cells.

c) As the flash tank pressure increases, steam is also admitted to the extraction side of the No. 1 high-pressure feedwater heater units completing feedwater heater unit warm-up.

d) Any excess flash tank steam is dumped to the condenser.

e) At this time, the coolant salt circulating system should be up to operating temperature.

By increasing the feedwater flow rate to about 10 percent of turbine full load flow and increasing the firing rate to raise the flash tank to its set point pressure of about 1000 psi, the warmed-up main and standby turbine-generator units may be rolled, synchronized and initially loaded to at least 5 percent of their respective full load flow rating with flash tank steam. Also, at this time, the boiler feed pump drive turbine may likewise be rolled and loaded sufficiently to permit establishment of a minimum boiler feed pump recirculation flow about the deaerator.

Turbine extraction steam generated is used for extraction heater and deaerator heating replacing flash tank steam as it becomes available. Reheat temperature to main turbine is established and controlled by bleeding warm-up steam through the tubeside of the reheat-preheater unit sufficient to heat the cold reheat extraction steam flow until the reheater units can be brought on line.

Auxiliary steam generator unit outlet flow is brought to rated output conditions of pressure and temperature by directly pressurizing the superheat section above 1000°F from the boiler section and increasing the firing rate to provide load at 1000°F rated output. The flash tank is automatically isolated from the start-up cycle by the boiler extraction valve station upon increase in steam pressure above the 1000 psi set pressure.

When the auxiliary steam system is at rated operating conditions, the standby steam generator is brought up to full load operation and the boiler feed pump drive turbine units are brought up to matching load with the motor driven auxiliary boiler feed pump output and feedwater circulation is transferred to the drive turbines while the motor driven units are taken off the line.

Bypasses around the steam cells can now be closed and 1000°F preheating steam admitted to the system steam generator units. Inlet steam flow temperature to the reheater units can be regulated to about 1000°F by indirect heating with auxiliary steam diverted through the system reheat-preheater units serving the reheaters.

With the steam cycle at pressure and temperature and operating at low load, the reactor is made critical. Some feedwater flow at the auxiliary boiler inlet is diverted to the mixing chambers, by means of the in-line control valve, to gradually reduce the steam temperature to steam generator units to about 710°F. Simultaneously, the reactor load is increased to maintain steam generator output temperature at 1000°F, and the boiler firing rate is reduced to match the resulting reduction in flow through the boiler. The reactor flux controller set point is regulated to keep salt temperatures at desired levels as the reactor power is matched to the load.

By diverting main steam flow from the tubeside outlet of the reheater-preheater units to the system mixing chambers, while simultaneously phasing out auxiliary steam generator output flow to the chambers, reactor generated main steam is substituted for auxiliary steam generator output and the auxiliary boiler can be taken off the line. Auxiliary steam generator loading and firing must be controlled during the phasing out of the steam flow to the mixing chambers until switchover is completed and the auxiliary generator can be shut down. Required steam cycle heat input is not completely supplied from reactor output.

With pressure booster pumps in operation, auxiliary steam generator unit, off the line and cycle at pressure and temperature, reactor is matched up with subsequent increase in steam loading to full load operation and the system is self-supporting. When the load reaches about 8 percent of full load flow, the reactor can be switched from flux control mode to a reactor outlet temperature control mode upon matching the set point temperature with the existing outlet temperature. Steam temperature control is initiated at about 20 percent load operation in which the reactor power control is regulated as a function of load while the main steam temperature controller holds outlet steam temperature at 1000°F.

Extraction steam from the main turbine will maintain the cycle feedwater temperatures and the flash tank is kept warm through interconnection with the deaerator.

5.5.5 Hot Restart Operation

Hot restart operation, as applied to the MSBR plant, consists of the ability to place the steam power system on the line from a standby or banked operating condition without the assistance of any outside power. Prerequisites for commencement of the hot restart operation are, therefore, a steam system operating at standby power level with either auxiliary steam at standby operating conditions, or the reactor steam at standby power conditions and all components of the steam warmed-up ready to accept loading. All necessary cycle feedwater, condensate and cooling water circulation is provided by power from the cycle steam or standby turbine-generator unit.

If the plant is on standby turbine-generator power, then the main turbine should be carefully pre-heated with system steam regulated to match the turbine bowl temperature limitation set by the turbine manufactuier (about 950°F). Turbine stop valve and main steam line drain valves should be opened for warm-up. Output from the auxiliary steam generator is sufficient to permit rolling, synchronization and loading of the main turbine for minimum load operation. At this time, steam flow from the auxiliary steam generator is replaced by steam from the cycle steam generator in the same manner during a normal cold start-up operation. Reactor power is matched to steam loading as the turbine is brought up to full power. The auxiliary steam generator can be kept warm and placed on standby service until stable operation is achieved.

5.5.6 Normal Shutdown Operation

Essentially, the normal shutdown operation of the MSBR steam system consists of reversing the start-up operation sequences until flow to the main turbine has been reduced to zero, at which time the system is placed on standby operation and held until afterheat rejection has decayed to a very low value, then the reverse start-up operations are continued until steam generation is terminated and the system is allowed to cool.

As the main turbine flow is gradually reduced to zero, the reactor control is transferred from output temperature and load control to flux control and the power output reduced to zero. The standby turbine generator and boiler feed pump drive turbines are switched to main steam operation on the reduction in extraction steam pressure and output accompanying the reduction in main turbine loading. Generated steam not required for reduced load, standby turbine and boiler feed pump drive turbine operation is rejected to the condenser.

Operating the system in the standby mode, with the auxiliary steam generator unit, standby turbine generator and boiler feed pump operating, assures maintenance of minimum system power requirements and continued feedwater circulation for afterheat removal and maintenance of desired salt temperature profiles. While operating on standby the reactor is shut down and the main turbine is allowed to cool gradually by controlled admission of steam through the turbine seals and warm-up system.

The auxiliary steam generator is taken off the line in a manner reverse to start-up. As output steam pressure is reduced to 1000 psi steam is generated in the flash tank and steam temperature and burner firing rate are controlled to maintain acceptable furnace gas temperatures as load is reduced. The auxiliary steam generator may be allowed to cool or intermittently fired and maintained warm ready for restart operation. Likewise, sufficient enthalpy is maintained stored in the flash tank by bleed steam to keep it in a standby condition to preclude the need for feedwater temperature bootstrapping operations of a cold start-up.

5.5.7 Waste Heat Rejection and Recovery System Operation

In this plant concept, decay heat rejected from the fuel salt drain tank cooling system and the heat from the chemical processing cooling system are intended to be rejected to a low-pressure steam circuit which, in turn, may either reject this combined heat directly to the atmosphere during emergency-dump operation, or into the plant steam power cycle, as low-pressure steam, during normal plant operation.

Radiant boiler units convert the heat from the individual NaK cooling circuits into low-pressure steam. About 18 MWt of fuel salt decay heat and about 7 MWt of chemical processing waste heat is converted continuously during normal plant operation. During emergency-dump operation the combined waste heat converted will increase to as much. as 53 MWt.

The schematic flow diagram, shown on Figure 5.3, illustrates this closed circuit, low-pressure, waste heat rejection and recovery system which is connected in parallel to both a water filled basin open to the atmosphere, and the steam plant feedwater cycle.

During the normal plant operation, the low-pressure steam generated from the waste heat is directly introduced into the plant steam system deaerating feedwater heater stage, thereby, reducing the amount of turbine extraction steam needed for regenerative feedwater heating. Level controllers serving the radiant boiler units regulate condensate return flow from the plant condenser to maintain the required circuit flow. In this mode of operation, the pressure of the waste heat steam generated is determined by the operating pressure of the deaerating heater, changing as the extraction pressure changes with plant load. At this time, a pressure activated control valve, set to open at pressures exceeding the deaerator pressure at rated load, isolates the waste heat steam circuit from its emergency-dump circuit.

During emergency-dump operation, the steam cycle extraction steam demand cannot absorb the increased waste heatsteam release, thus, causing the circuit pressure to rapidly increase. When the control valve set pressure is exceeded, the valve automatically opens to divert waste heat steam flow to the emergency-dump circuit where it is condensed, stored and the condensate pumped back to maintain proper circuit flow. Thus, the waste heat is rejected to the atmosphere by the evaporation of the basin water used to condense the steam in the coils within the dump basin. Although cooling water regulated by level is normally used to provide the dump basin makeup requirements, the 30 ft x 20 ft x 15 ft dump basin proposed contains enough stored water to provide evaporative cooling for at least three hours at rated emergency-dump heat input with no makeup. Any auxiliary makeup source, such as a river feed fire hose, capable of at least 150 gpm flow is sufficient to provide for indefinite operation of the dump basin, in the event of loss of normal makeup, under these conditions.

5.6 Performance

Figure 5.4 indicates the schematic heat balance, with major flow, turbine characteristics and cycle heat rate values, for the proposed reheat type steam power cycle at rate load operation. A gross cycle generating capacity of about 1036.7 MWe (proportioned on the basis of 1018.7 MWe from the main turbine-generator unit and about 18 MWe from the standby turbine-generator unit) provides the desired 1000 MWe net plant power output and about 36.7 MWe for estimated plant auxiliary

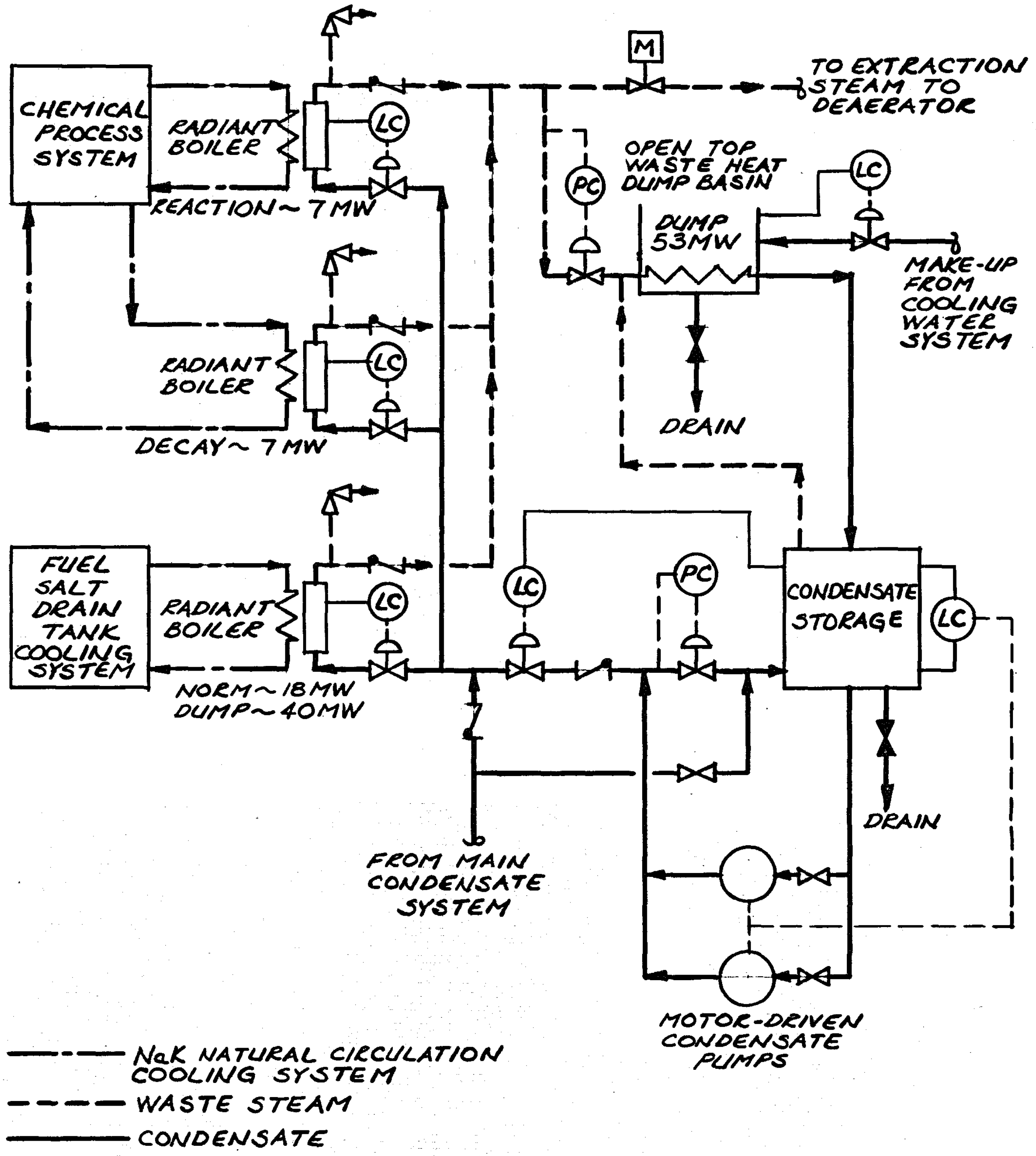


Figure 5.3: Waste Heat Rejection and Recovery System Flow Diagram.

electric loads. About 12.2 MWe of this auxiliary plant load is estimated to be required to drive the special pressure booster pump units feeding the cycle steam generators. The equivalent thermal energy input needed from the plant reactor to deliver this output, with the cycle as shown, is about 2303.4 MWt which results in a net cycle thermal efficiency of 43.4 percent and a net cycle heat rate of 7677 BTU/kW-hr.

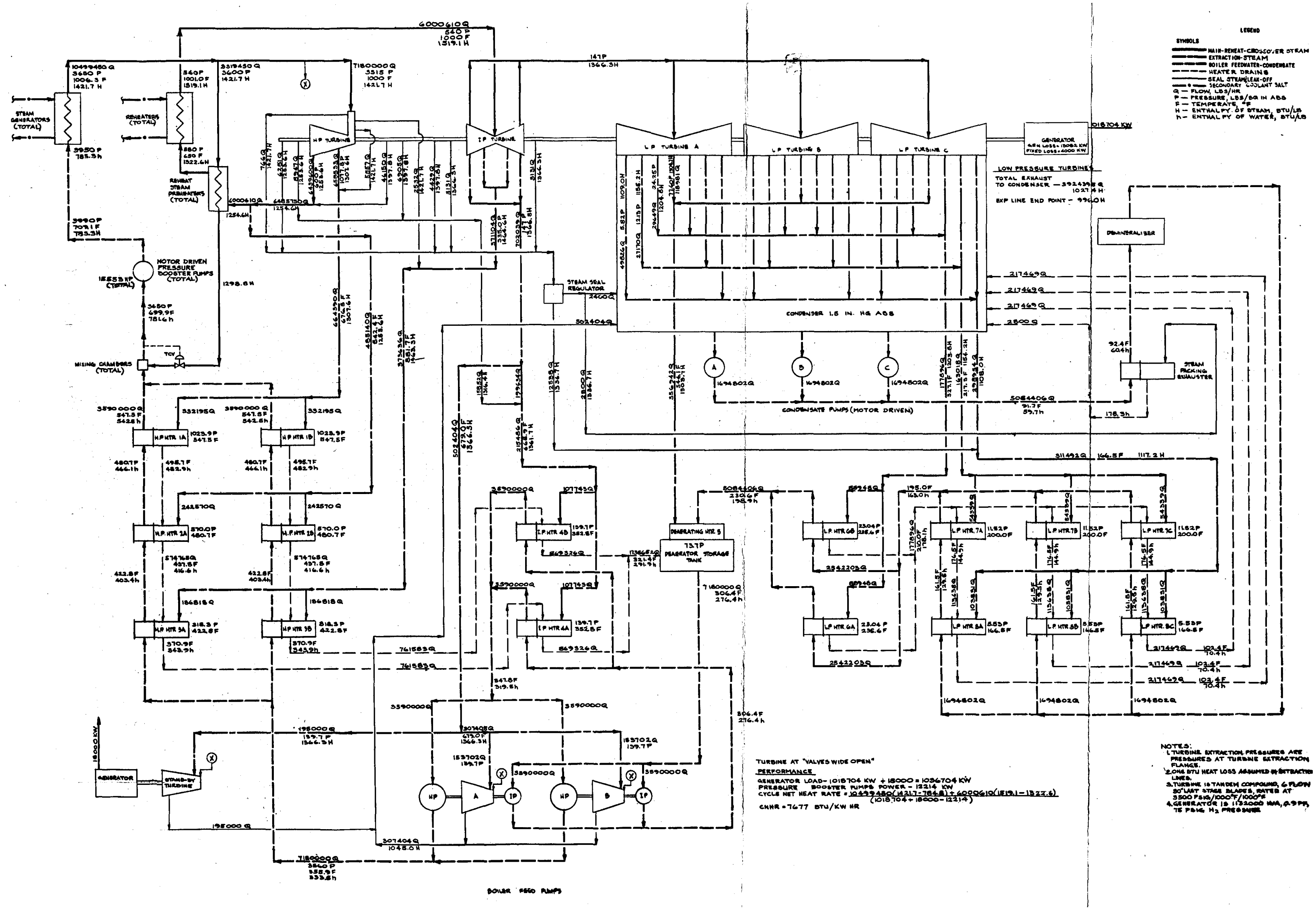


Figure 5.4: Steam Cycle - Heat Balance.

Although dependent on the nature of the final plant design, the estimated heat losses from the reactor plant, exclusive of decay and chemical processing heat, should be in the order of 1 percent of the total reactor output. For the purpose of this study, this reactor plant heat loss is estimated at about 25 MWt, thereby, making the total required energy output of the reactor about 2328 MWt. This results in an overall net station thermal efficiency of about 43 percent and a station net heat rate of 7866 BTU/kW-hr for the proposed cycle without the utilization of any plant decay or chemical processing heat which has not been included as part of the thermal capacity of the reactor.

Among the factors influencing the steam power cycle performance, the introduction of reclaimed reactor decay heat and chemical processing heat into the cycle represents the most attractive means for material improvement of cycle capability. The schematic heat, balance, shown in Figure 5.5, indicates the effects to the proposed cycle obtainable from the method suggested in this concept for the introduction of the 18 MWt reactor decay heat and the 7 MWt chemical processing heat into the cycle as an additional source of low-pressure feedwater heating steam.

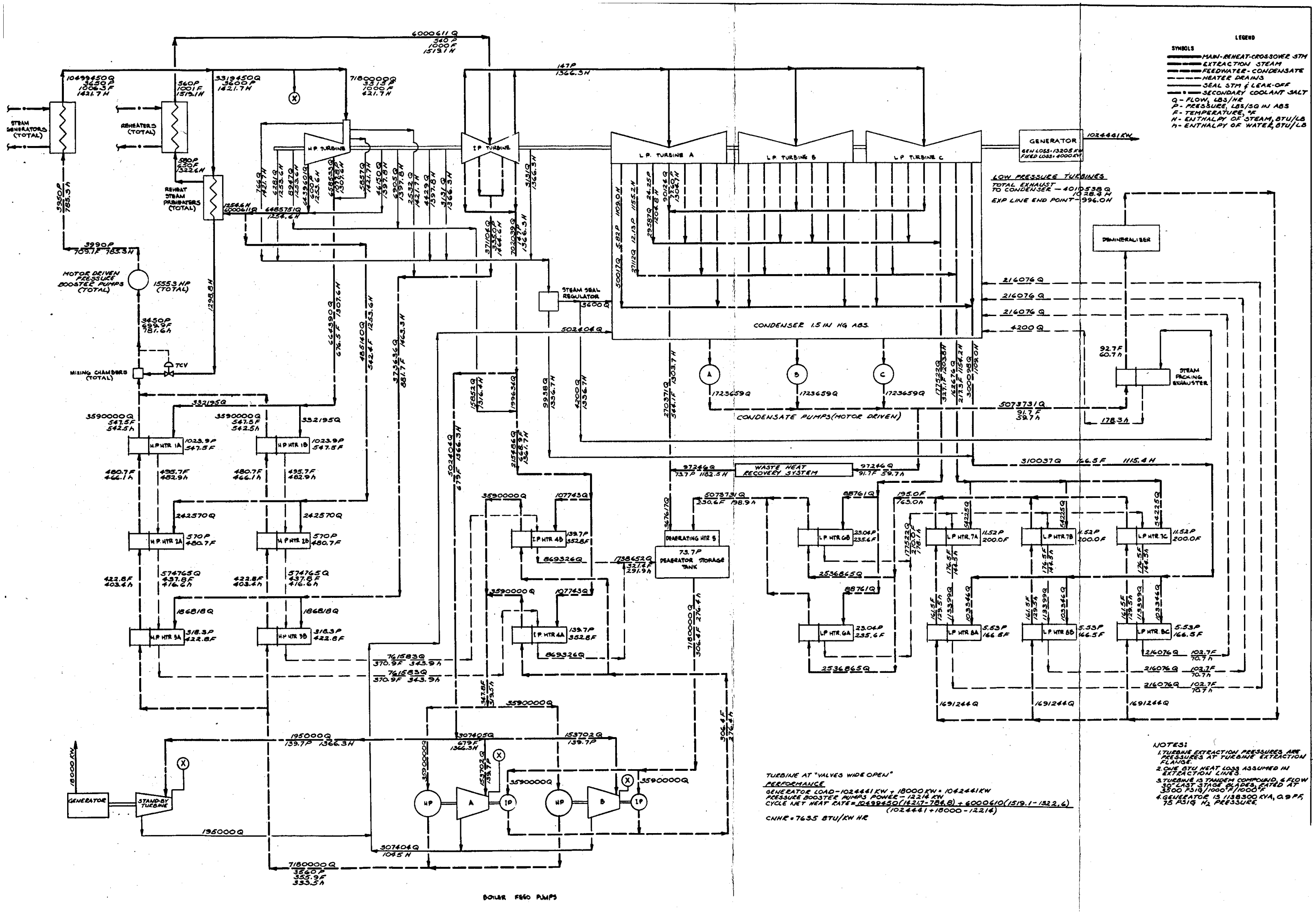


Figure 5.5: Enhanced Steam Cycle - Heat Balance.

As shown, this method of cycle enhancement is capable of increasing the obtainable gross generating capacity to 1042.4 BTU/kW-hr without any additional increase in reactor power output. Assuming the allowances for auxiliary plant load and reactor plant heat load remain unchanged, the net plant power output capability is increased to 1005.7 MWe by this enhancement. This results in a net cycle thermal efficiency of 43.7 percent and a net cycle heat rate of 7635 BTU/kW-hr, or an overall station thermal efficiency of 43.2 percent and a station heat rate of 7625 BTU/kW-hr.

Further improvement of the cycle performance can be expected from optimization of the special pressure booster pump power requirements and the utilization of steam turbine drivers in lieu of variable speed motor drivers used in this study for these units. Less significant changes to cycle performance as indicated are possible from trade-off studies to determine the most economical feedwater cycle type and number of feedwater stages, arrangement of boiler feedwater pumping, and in cycle, extraction steam line pressure drop allowance, etc.

Chapter 6

Buildings and Structures

6.1 Site

The site defined by contract for the design study is the Atomic Energy Commission's standard hypothetical site as specified in NUS-531 - Appendix A, dated January 1969. This site is a level grass covered riverbank 15 ft above mean river level. The soil has a depth of 8 ft and is underlaid by a limestone strata 30 ft thick. The limestone has a bearing capability of 18 kp/ft2. The river has adequate cooling water to maintain 1.5 in. Hg back pressure in the condenser. A plot plan of the site is presented in Figure 6.1. The site is served by a single source of transmission which is subject to occasional outages.. This requires the site to have redundant emergency power sources. Tornado frequency is specified so as to require Class I structural design. The seismic criteria for the Task I design effort was specified as 0.07g horizontal ground acceleration. The site imposes no special environmental criteria on the plant design other than the normal licensing requirements.

6.2 Buildings

Each of the principal buildings is situated on a common concrete mat with sufficient space between the walls of each building to allow for seismic displacement without disruptive contact between the buildings. With this arrangement relative displacement between the buildings would not threaten the integrity of the reactor building or of the steam generator building.

Plan and elevation layout drawings of the reactor building are shown in Figure 6.2- 6.7. Plan and elevation layout drawings of the steam generator building and turbine building are shown in Figure 6.8- 6.13.

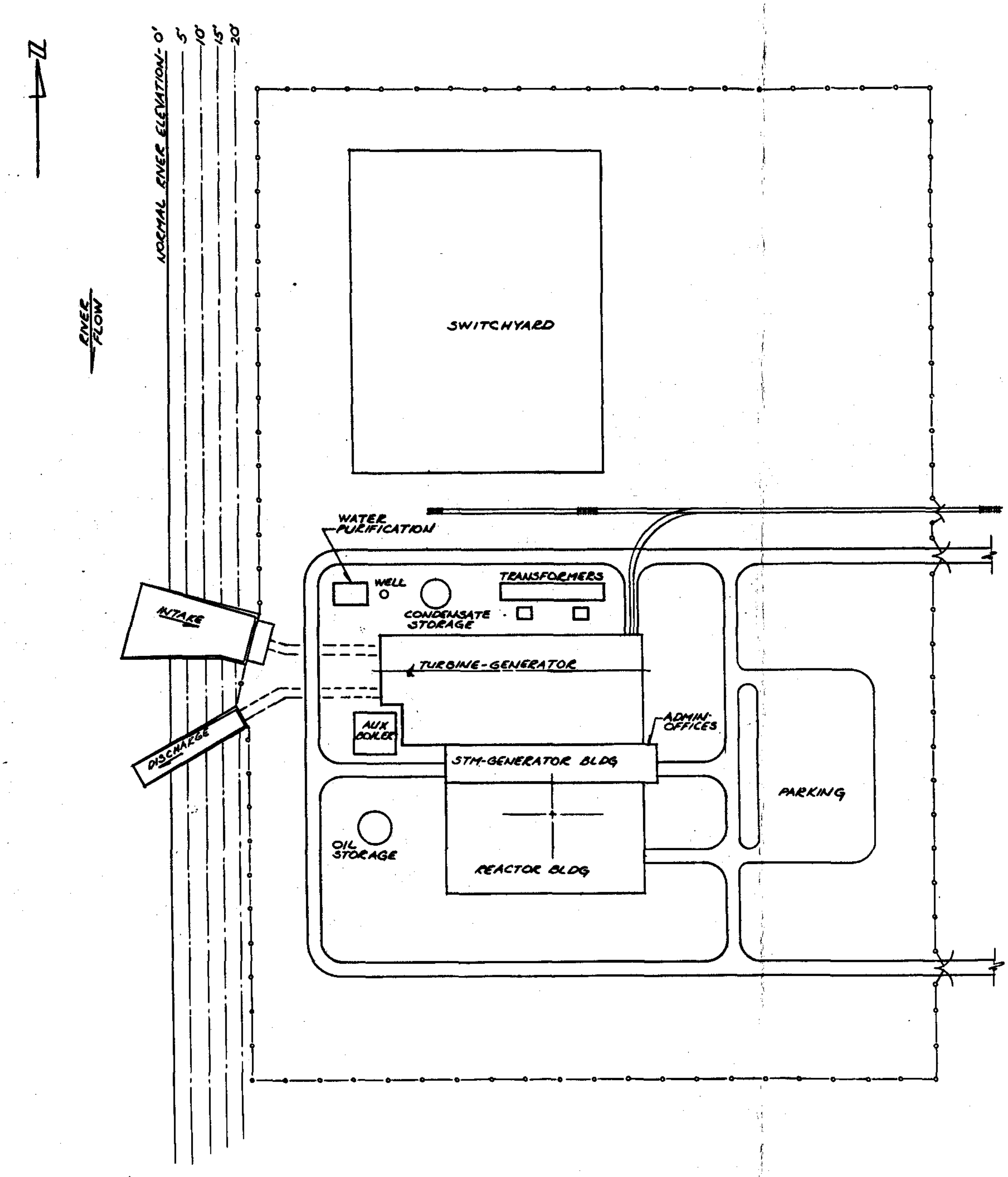


Figure 6.1: Site Plot Plan.

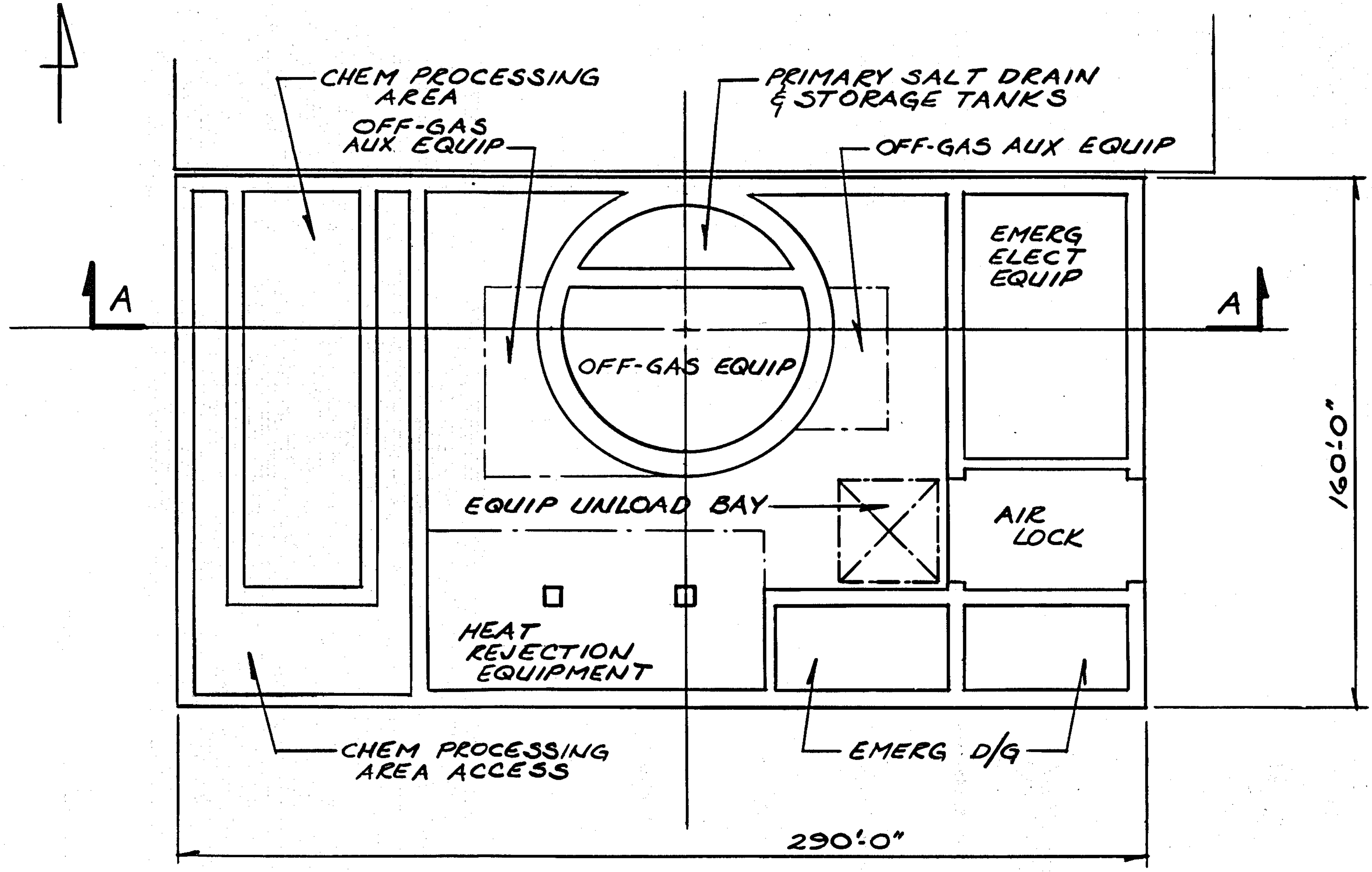


Figure 6.2: Reactor Building Grade Elevation 100'-0".

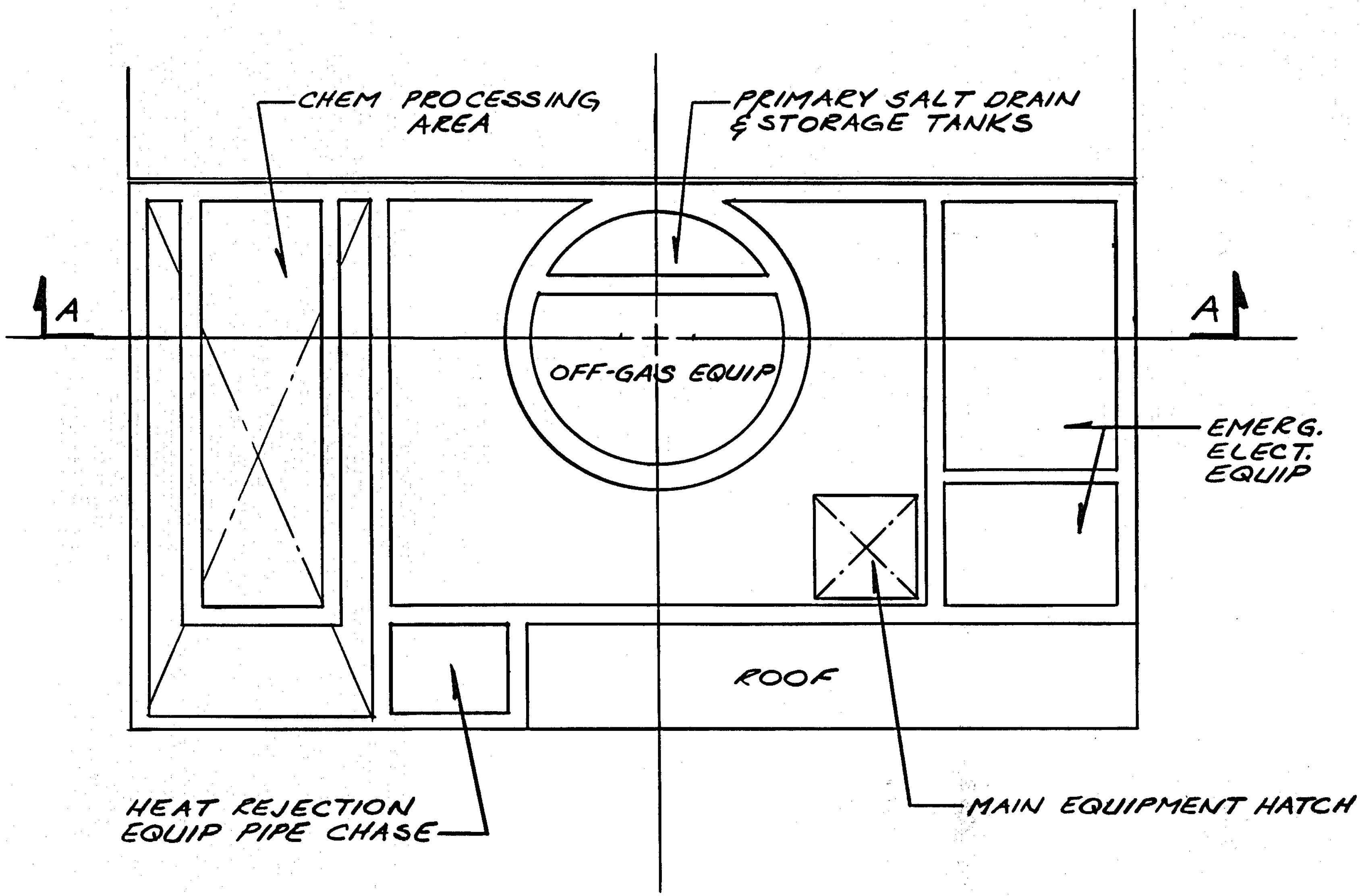


Figure 6.3: Reactor Building Grade Elevation 125’-0".

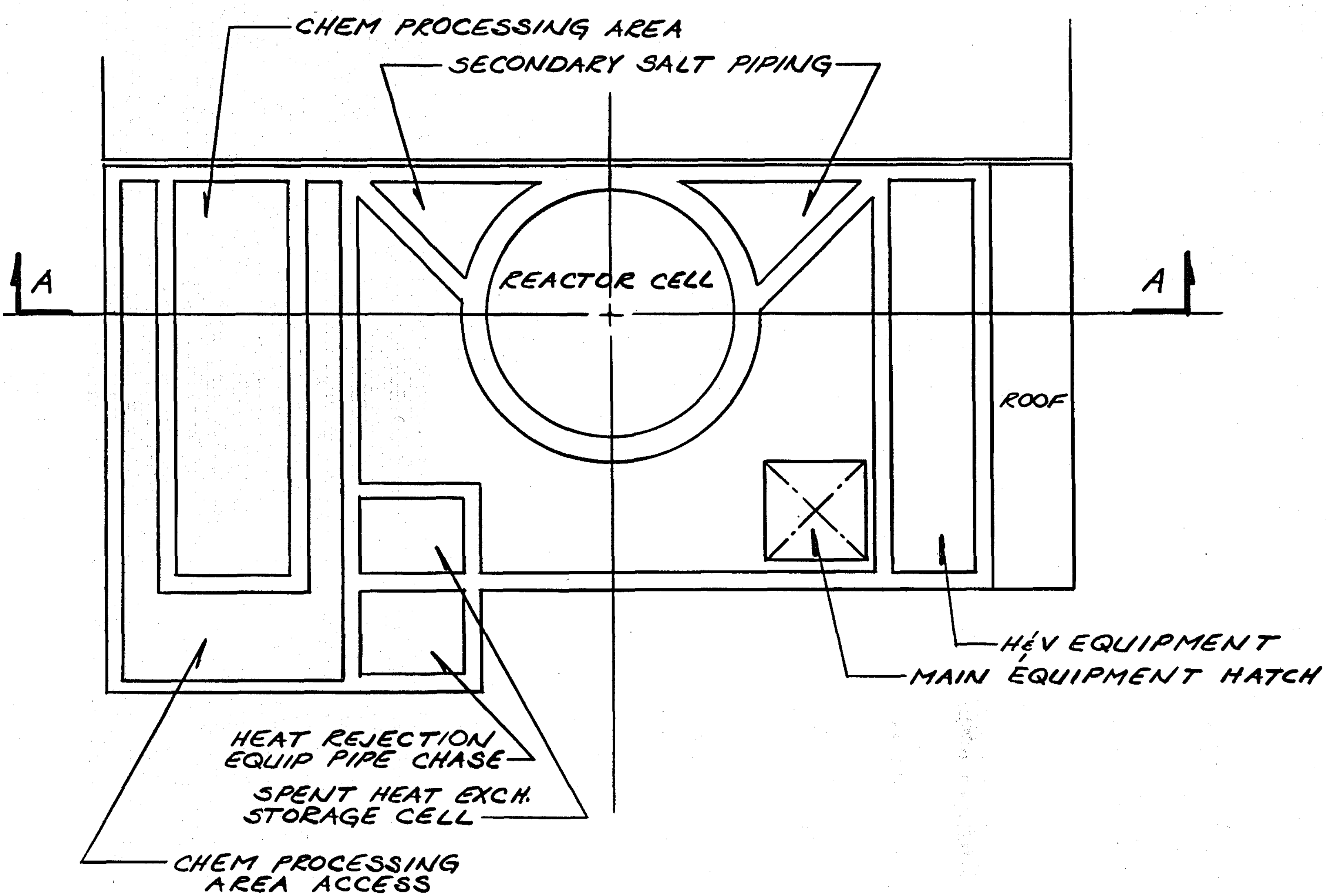


Figure 6.4: Reactor Building Grade Elevation 150’-0".

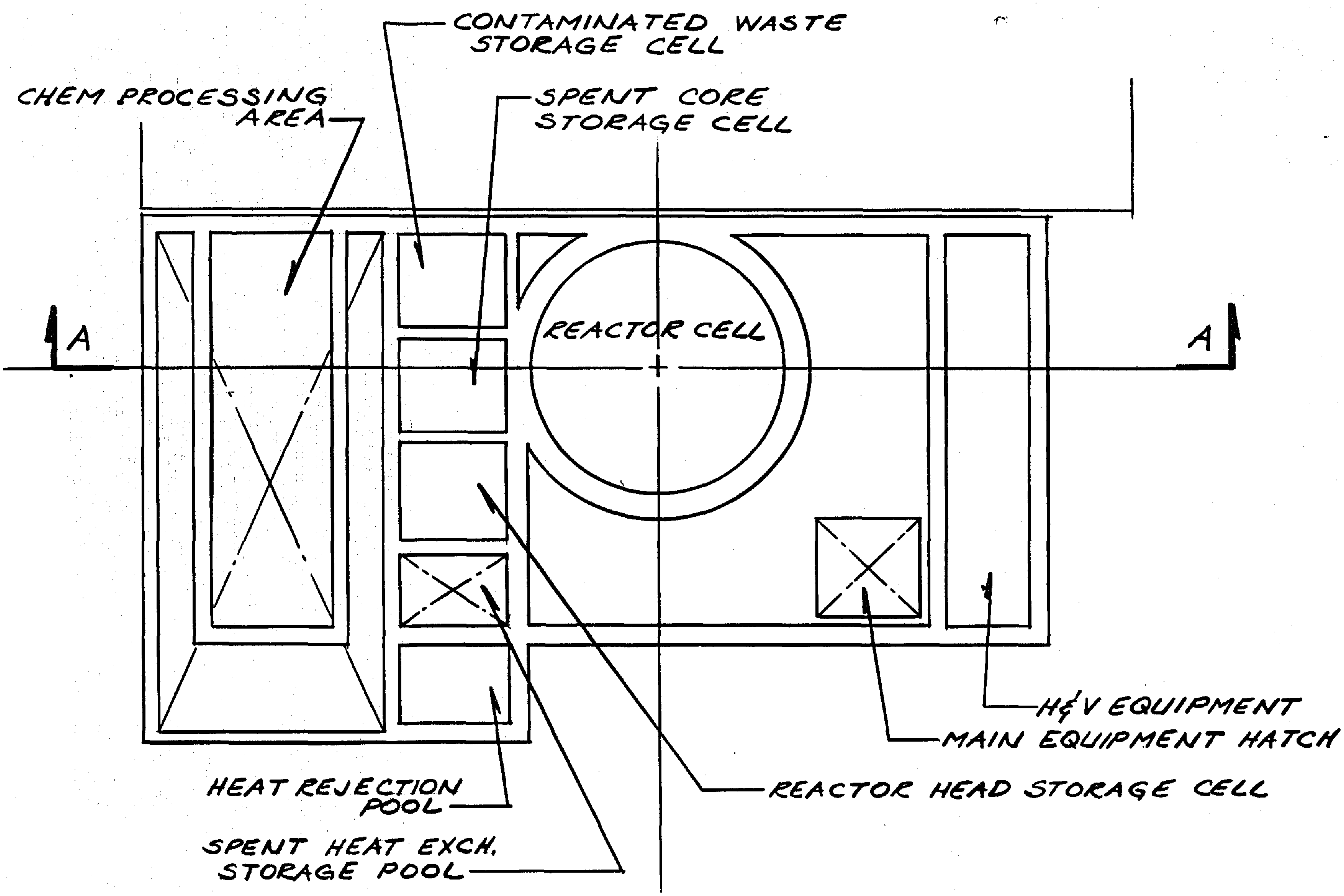


Figure 6.5: Reactor Building Grade Elevation 175’-0".

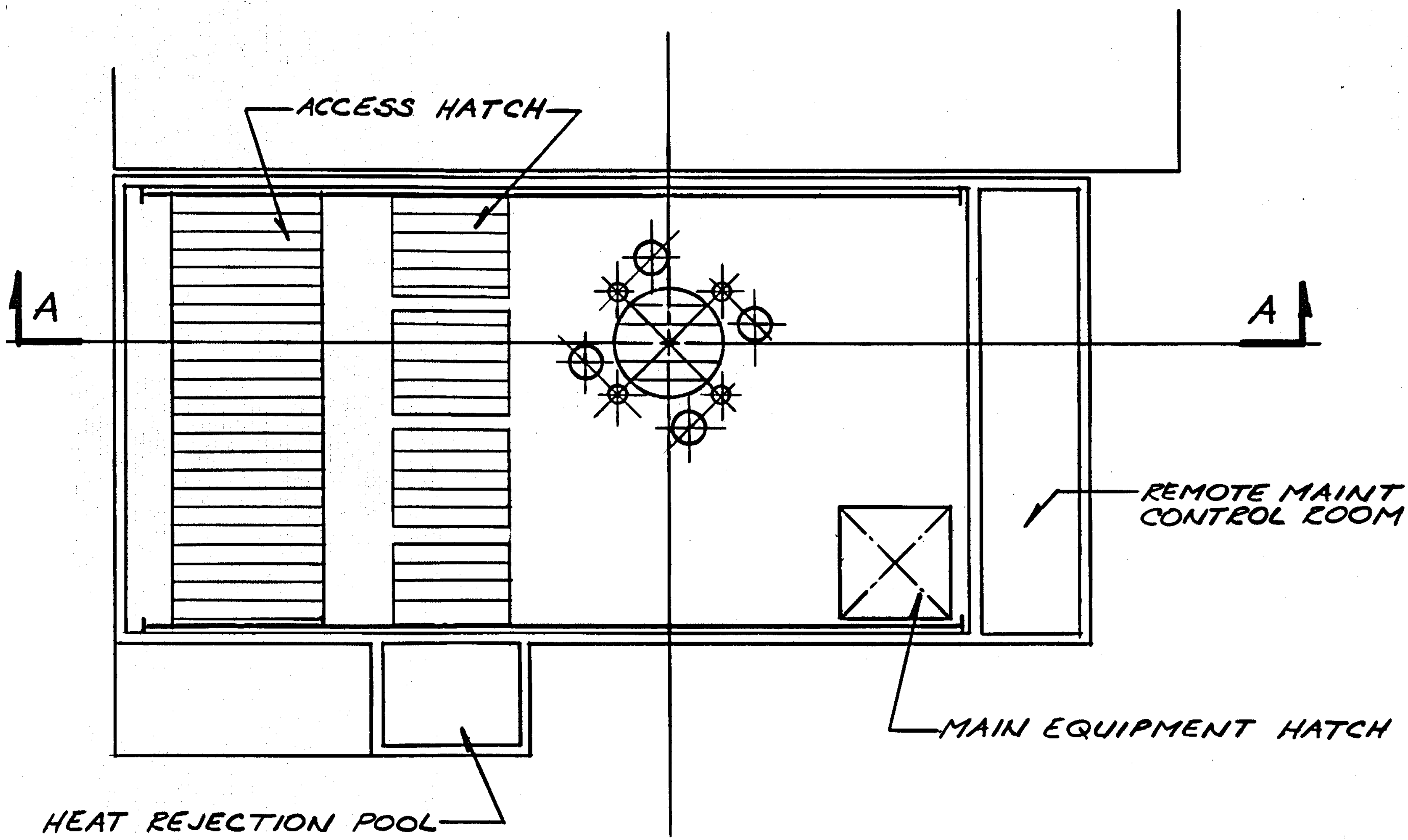


Figure 6.6: Reactor Building Grade Elevation 200’-0".

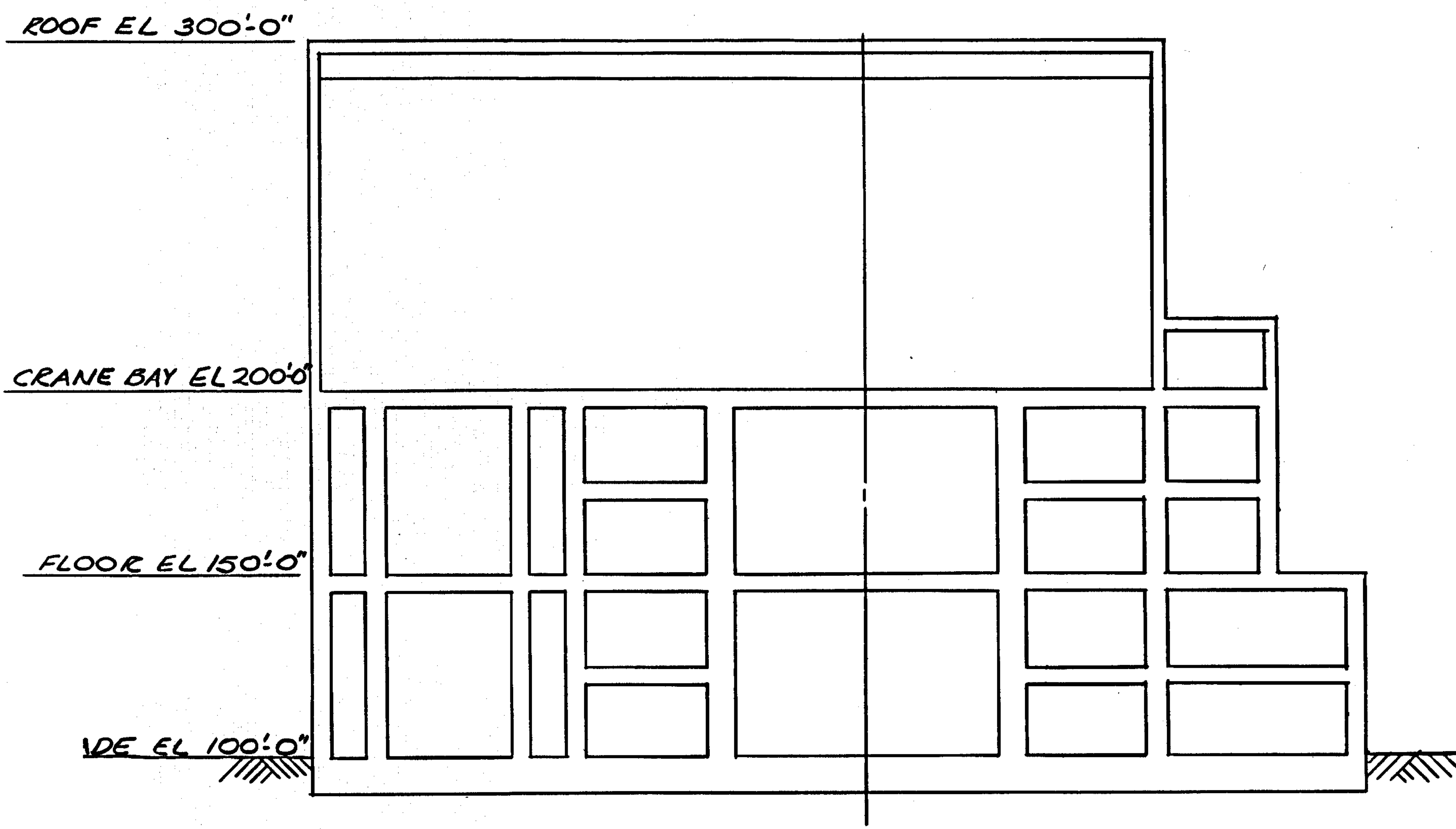


Figure 6.7: Reactor Building - Section "A-A".

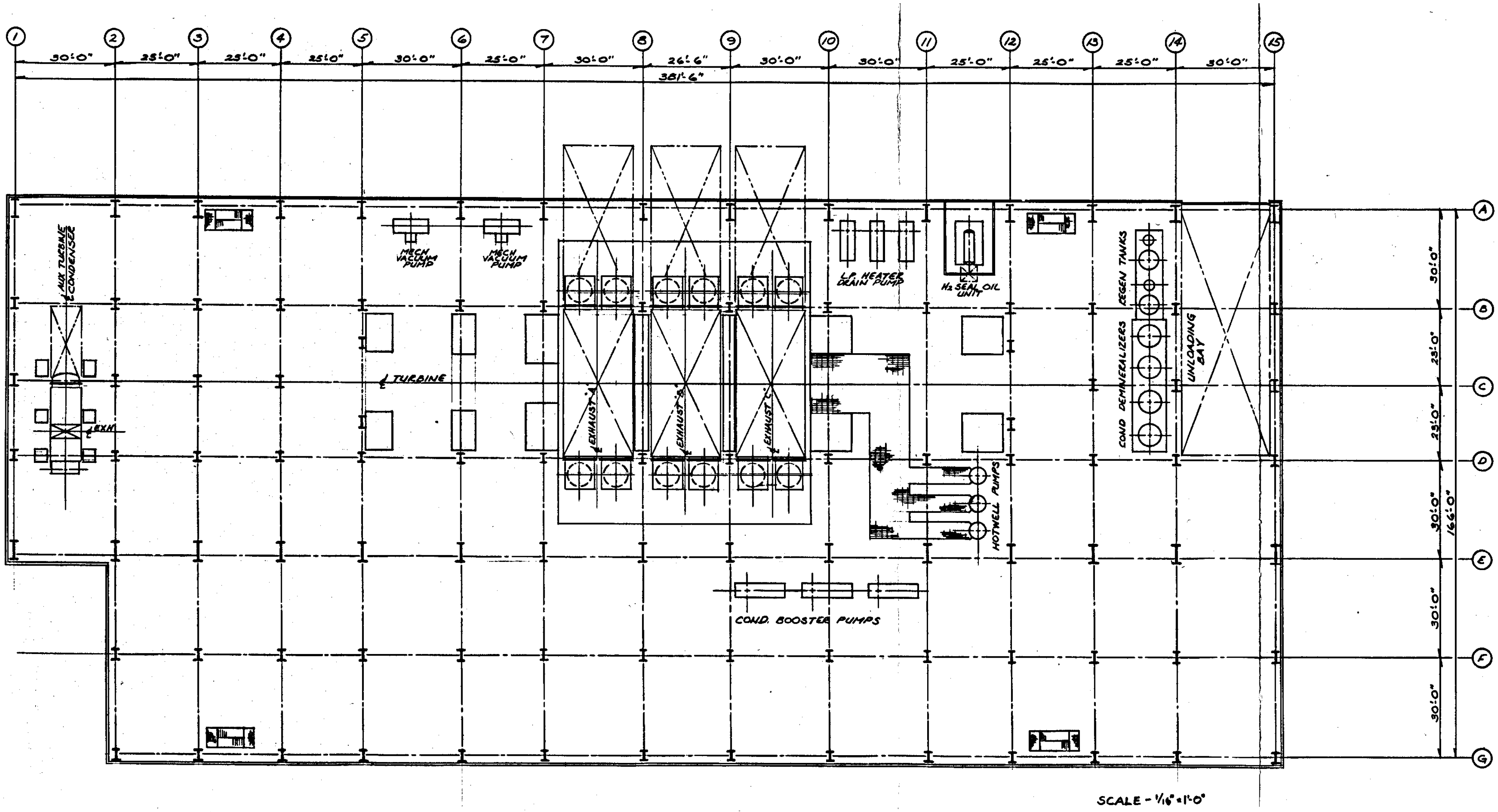


Figure 6.8: General Arrangement - Turbine Building Ground Floor Plan.

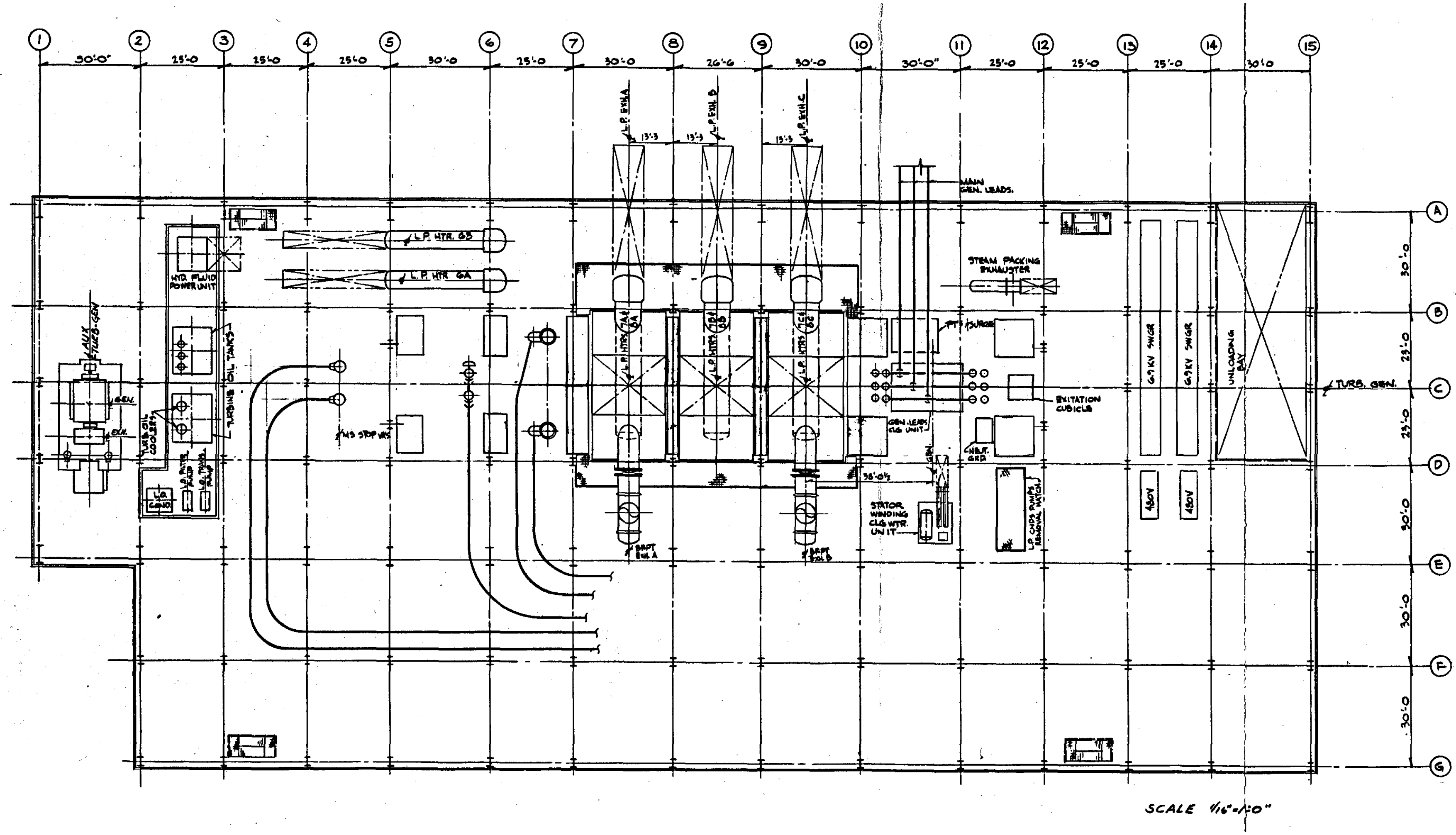


Figure 6.9: General Arrangement - Turbine Building Mezzanine Floor Plan.

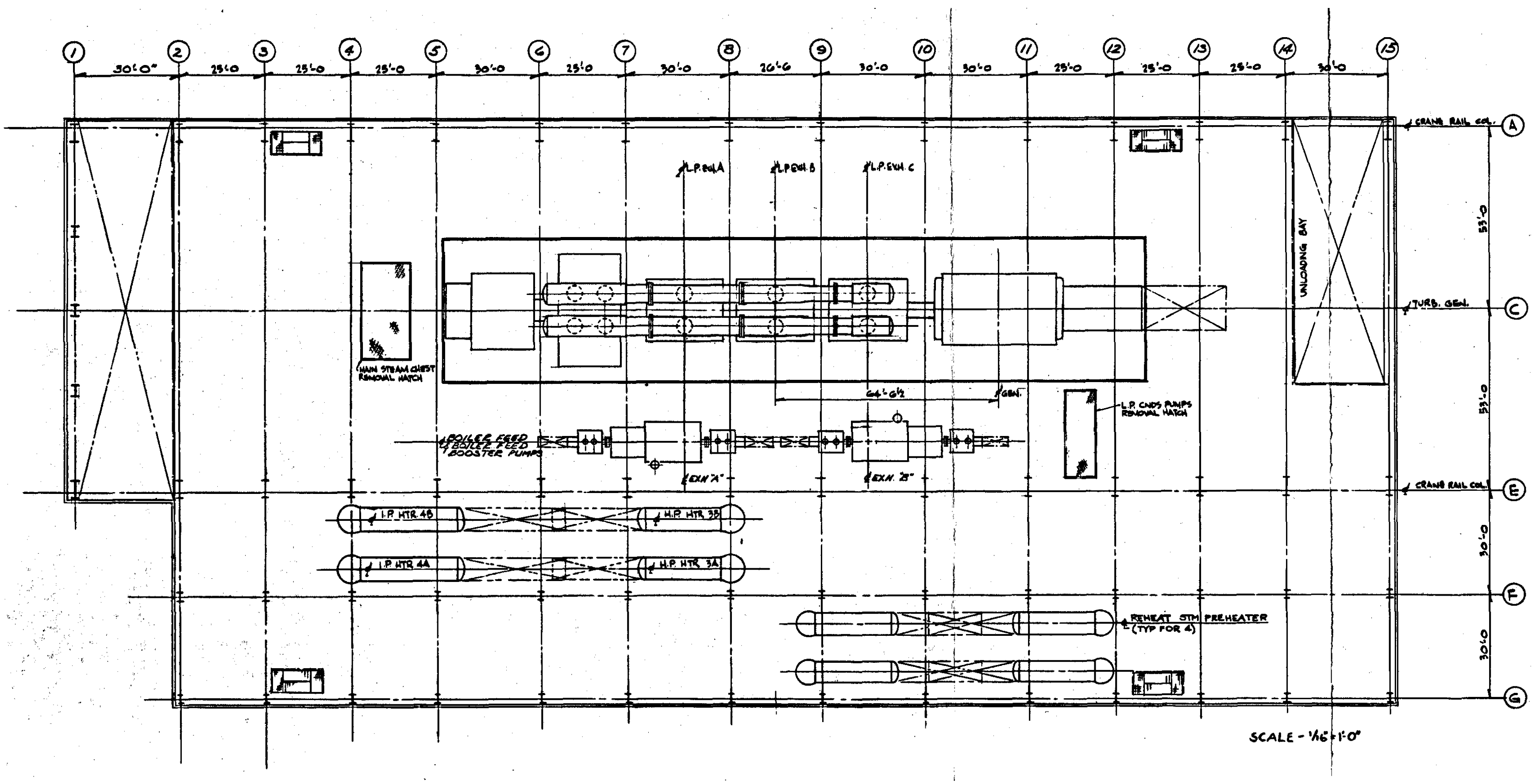


Figure 6.10: General Arrangement - Turbine Building Operating Floor Plan.

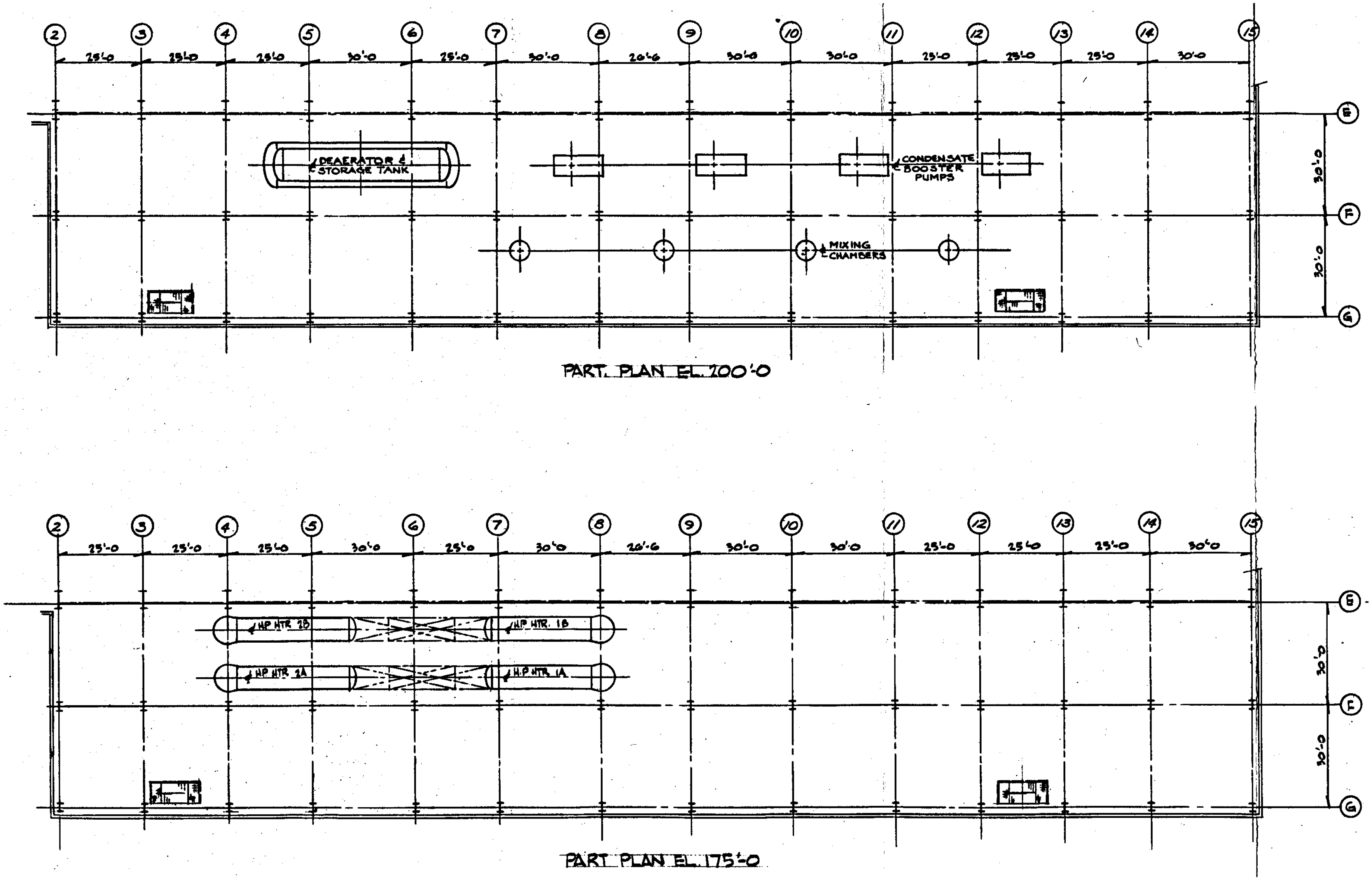


Figure 6.11: General Arrangement - Turbine Building Partial Plans.

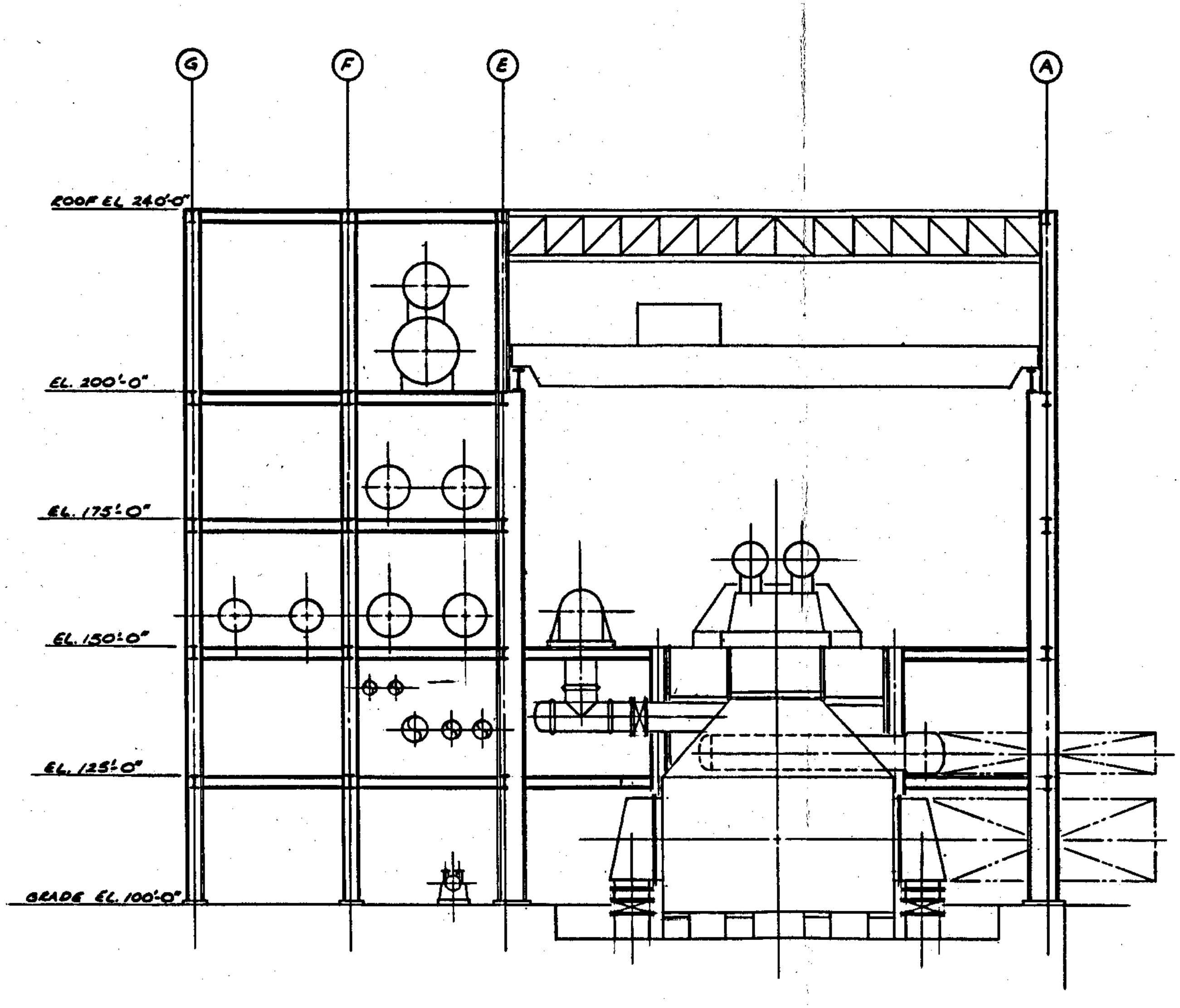


Figure 6.12: General Arrangement - Turbine Building Sections.

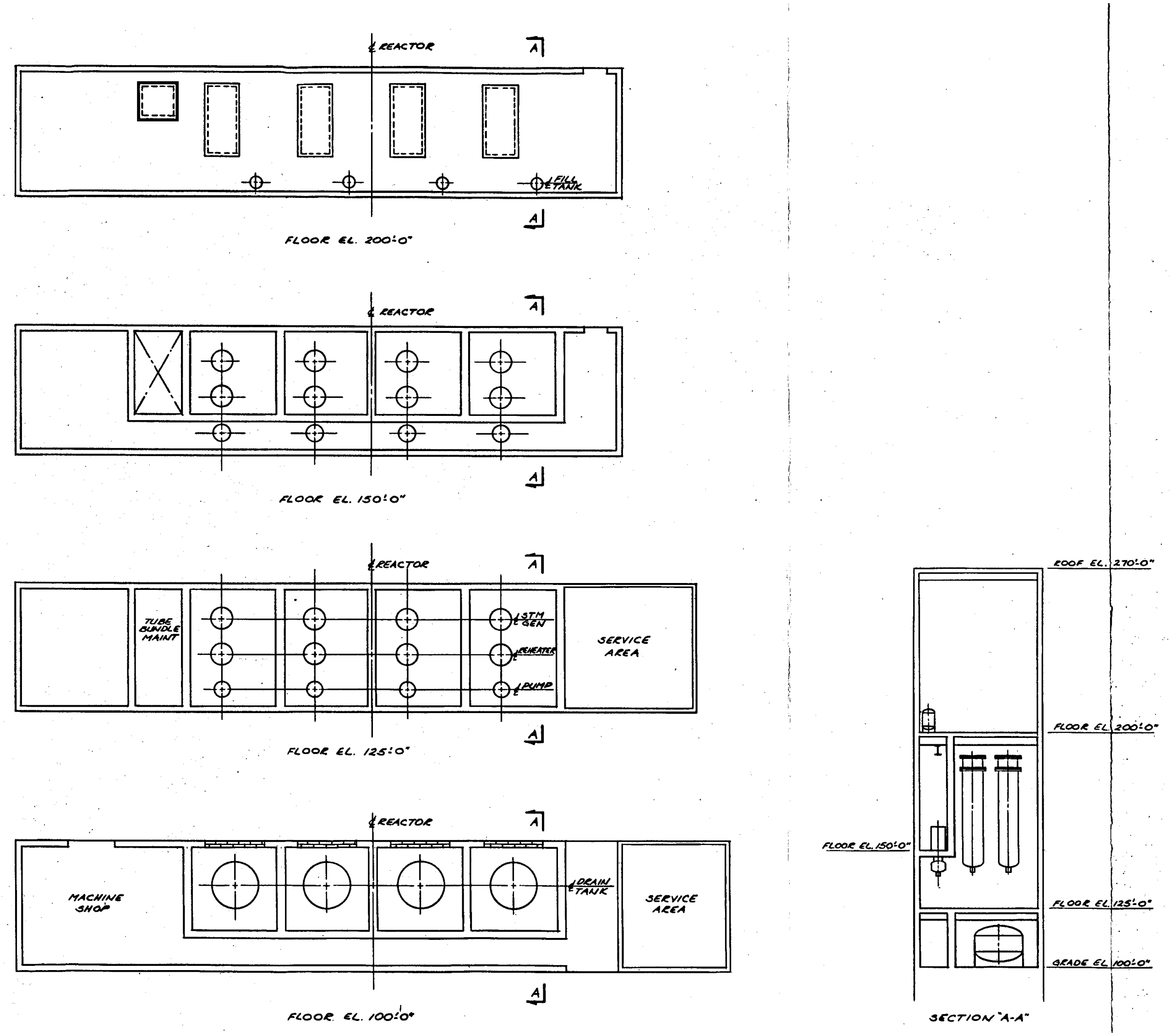


Figure 6.13: General Arrangement - Steam Generator Building.

6.2.1 Classification of Structures, Systems, and Equipment

a) Definitions of Seismic Classification

Class I

Class I structures, systems, and equipment are those whose failure could cause uncontrolled release of significant amounts of radioactivity, or those essential for safe shutdown and immediate or long term operation following a design basis accident. When a system as a whole is referred to as Class I, portions not associated with the vital function of the system are to be designated as Class II.

Class II

Class II structures, systems, and equipment are those whose failure would not result in the release of significant amounts of radioactivity and would not prevent reactor shutdown, but could interrupt power generation. These structures are to be designed to conform to the requirements of the Uniform Building Code - Zone II. Class H structures, systems, and equipment shall not degrade the integrity of those designated Class I.

Class III

Those structures and components which are not related to reactor operation or containment. Earthquake is not considered in the design of these structures.

b) Seismic Classification of Structures, Systems, and Equipment

1) Class I Structures

a. Primary containment

1. Reactor vessel support
2. Primary heat exchanger support
3. Primary pump support
4. Reactor cell liner support
5. Horizontal support structure

b. Reactor building

1. Reactor building mat
2. Reactor building enclosure
3. Reactor building crane, crane runway, and crane support structure

c. Main control room and cable vault

d. Battery room

e. Primary salt drain tank cell

f. Off-gas cell

g. Freeze valve cell

h. Control rod storage cell

i. Emergency generator room

j. Off-gas heat-rejection cell

k. Air lock

l. Off-gas auxiliary equipment cell

2) Class II Structures

The following are considered to be Class II seismic structures:

a. Turbine building

b. Turbine generator pedestal

c. Turbine building crane supports

d. Steam generator building

e. Off-gas ventilation stack

f. Circulating water intake structure

g. Coolant salt drain tank cell

h. Auxiliary equipment cell

i. Waste storage cell

j. Spent core cell

k. Spent heat exchanger cell

l. Hot cell and work areas

m. Chemical processing cell

n. Components storage and assembly areas

o. Radwaste building

3) Class III Structures

Buildings containing conventional facilities.

c) Class I Systems and Equipment

The following are considered to be Class I seismic systems and equipment:

1) Reactor Fuel Salt System

   a. Reactor vessel and internals

   b. Primary fuel salt pump

   c. Primary fuel salt heat exchanger

   d. Freeze valves

   e. Control rod and drive system

   f. Control rod drive housing

   g. All piping connections from the reactor vessel up to and including the first isolation valves

2) Fuel Salt Drain and Off-gas Holdup System

   a. Fuel salt drain tank and piping

   b. Fuel salt storage tank and piping

   c. Off-gas heat reject system equipment

   d. Off-gas and jet pump piping

3) Fuel Salt Transfer Pump

4) Standby Electrical Power Systems

   a. Station battery system

   b. Standby motor generator system

   c. Heating, ventilating, air conditioning and lighting in reactor building control room

5) Instrumentation and Control Systems

   a. In-core instrumentation

   b. Control rod drive system instrumentation and control

6) Emergency Fuel Storage Tank and Piping

7) Gas Treatment System

8) Chemical Processing System

Class II Seismic Equipment and Piping Systems

The following are considered to be Class II seismic equipment and piping systems:

a) Turbine-generator system

b) Main condenser and circulating water systems

c) Turbine building cranes

d) Secondary salt circulating system

e) Condensate storage and transfer system

f) Station auxiliary power buses

g) Electrical controls and instrumentation (for above systems)

h) Radwaste system

i) Turbine system moisture spearators

j) Condensate demineralizer system

k) Station service water system

l) Compressed air system

m) Steam generators and reheaters

n) Coolant salt rupture discs

o) All other piping and equipment

Class III Seismic Equipment and Piping Systems

a) Conventional equipment, tanks and piping, other than I and II classes.

d) All Class I structures are designed against the possibility of an on site tornado occurence.

6.2.2 Reactor Building

The reactor building is the secondary containment structure for the reactor primary system containment cell, the fuel salt processing system cell, the hot storage cells, the coolant salt drain and storage tank cell and the off-gas system cell during normal operation. In addition, the reactor building forms the primary containment for all cells opened during maintenance operations.

The reactor building is a single integrated reinforced concrete multi-story structure as shown in Figure 6.7. The reactor is housed in a cylindrical furnace cell within the reactor building. Plan views at the five major levels are shown in Figure 6.2- 6.6. The building is approximately 290 ft long by 160 ft wide and 200 ft high. The top of the foundation mat is set at grade 100 ft in order to avoid negative numbers and is equivalent to elevation zero. This is compatible with the site conditions having the top of the limestone formation about 8 ft below grade.

The first level (elevation 100 ft) of the reactor building contains cells and space for chemical processing drain tanks, heat rejection equipment, off-gas handling, fuel salt drain and storage tanks, emergency diesel equipment as shown in Figure 6.2. Each respective elevation, as shown in Figure 6.3- 6.6 provides the space and cells for the remaining equipment necessary for nuclear plant operation. The structure also supports a traveling bridge crane which would span the length of the building to service the reactor and other equipment within the building.

During normal operation the building is maintained at slightly below atmospheric pressure by a controlled ventilation system discharging through filters and up a stack. The primary purpose of the reactor building normal ventilation system is to limit exposure of personnel to airborne contaminants and to maintain appropriate temperature conditions for operating personnel and equipment.

The reactor building normal ventilation system shall:

a) Migrate air from clean accessible areas to areas of progressively higher contamination or potential contamination.

b) Remove the normal heat losses from all equipment and piping in the reactor building during station operation.

c) Filter outside air to limit the introduction of airborne particulate matter to the station.

d) Exhaust potentially contaminated leakage to the stack through the ventilation air treatment system.

The reactor building normal ventilation system consists of a supply and exhaust side. The supply side includes in the direction of air flow, outside louvers, dampers, filters, heating coils, and two supply fans each sized for the full system capacity. The exhaust side consists of two exhaust fans, each having full system capacity exiting through two treatment systems.

The main supply and exhaust ducts penetrate the reactor building, through two butterfly isolating valves in series, which are automatically closed by a primary containment isolation signal. The valves in the main supply duct are powered from different buses. This is also true of the valves in the main exhaust duct. All isolating valves fail closed.

Supply air will be distributed by means of a duct system to provide equipment cooling in various areas within the building as required. Air will be routed from clean areas to areas with progressively greater contamination potential. Gravity dampers are provided at key points in the duct distribution system to prevent backflow of air from contaminated to clean exhaust duct branches. All exhaust air will be routed through a return duct system where the exhaust fans direct the exhaust air to the treatment system which monitors the air and 1) discharges to the stack if of acceptable quality or 2) processes the air through filters, charcoal adsorbers, and if necessary, through delay tanks if contaminated.

Operating personnel would have access to the major portion of the reactor building at all times except during certain phases of maintenance operations. During these periods remotely-controlled equipment can be viewed through shielded windows in the remote maintenance control room wall at the crane bay level (elevation 200 ft).

The design criteria for the reactor building is as follows:

1) Withstand a 300 mph wind; a storm caused 3 psi negative pressure differential, and a 2500 lb missile, 15 in. in diameter, traveling at a speed of 150 mph.

2) Design basis earthquake equipment to 0.07g horizontal base acceleration.

Reactor Cell

The reactor cell is the containment for the primary system. It is about 72 ft ID by 50 ft deep. The design criteria for the reactor cell is as follows:

1) The atmosphere of the reactor cell is to be operated at about 13 psia and at 1000°F $\pm$ 25°F and a He atmosphere.

2) The cell is designed for 50 psig. This pressure would be supplied from He storage tanks to assist in fan coolers transferring the heat of deposited fission products from the drained reactor and primary system.

3) All wall construction must provide both thermal insulation and gamma shielding to protect concrete structures from rising above 150°F.

4) Previous projects which had base accelerations of approximately 0.07g and foundations in rock were therefore examined to obtain what could be considered reasonable values. Building periods of around 0.4 sec and maximum accelerations (at 80 ft from foundation) were about 0.10g. Therefore, if the supported equipment within the cell is considered to have a period of 0.05 seconds (rigid), the acceleration at 80 ft from the foundation for the reactor piping system was taken as 0.25g and for the structure as 0.5g.

5) Prevent the escape of radioactivity both during normal operation and during accidents.

The reactor heat exchangers and salt piping are all supported from the cell floor whereas the fuel salt pump is mounted in the roof of the reactor cell and is supported on a set of springs to allow for the relative thermal expansion of the other components. The primary heat exchangers are mounted on roller bearings to allow for horizontal thermal growth but are restrained by a three tier girder arrangement as shown in Figure 6.14 for seismic protection of the piping, reactor vessel and heat exchangers. Calculations indicate that the restraining members, if made of a metal closely matching the piping system thermal expansion coefficient, will be of the wide flange type with maximum sizes as follows:

a. Horizontal tiers-flanges — 2 in. x 18 in. wide, web — 1 in. x 36 in. deep

b. Vertical support-flanges — 7/8 in. x 14-5/8 in. wide, web — 9/16 in. x 12-5/8 in. deep

Supporting the reactor vessel and heat exchangers from the reactor cell floor thus facilitates construction as the primary containment may be erected first and post-weld heat treated followed by

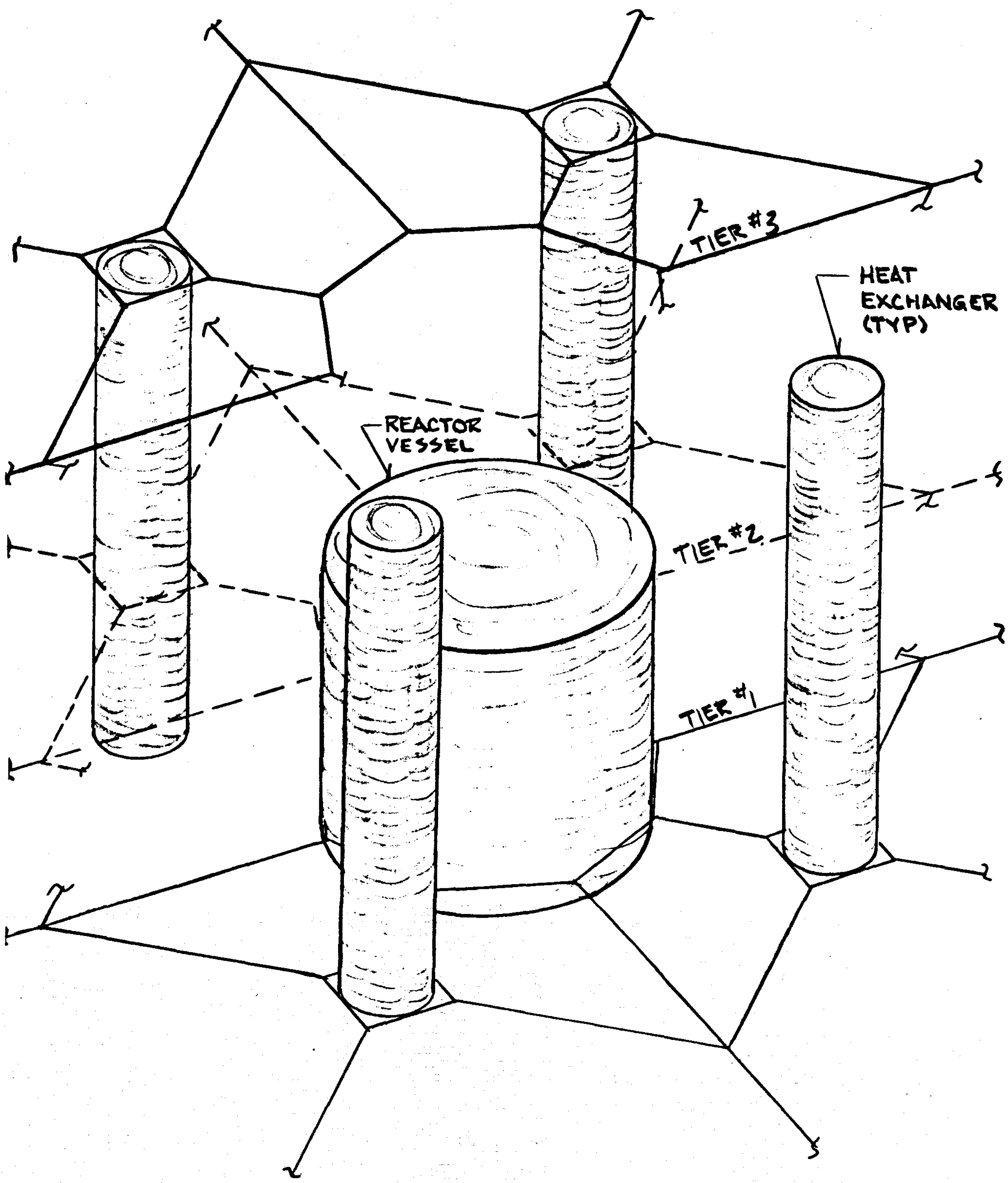


Figure 6.14: Three Tier Horizontal Equipment Restraint General Scheme.

a placement of concrete for reactor cell shield walls. Sizes of primary containment vessel plates need not be restricted as in the top support method.

Of the materials investigated for use as structural members within the reactor cell, two possibilities exist: type 304 stainless steel (ASTM 240) and Inconel 625 (ASTM B443). Due to the environment within the reactor cell, the following were considered in the selection of the above materials:

1) Tensile strength, yield point, allowable stresses and their decrease in values at elevated temperatures
2) Creep and creep rupture
3) Fatigue
4) NDT temperature, precipitation hardening and temper brittleness
5) Corrosion (carbide instability and oxidation of scaling)
6) Radiation effects on above items

Based on the more critical items above, it is anticipated the Inconel 625 will be ultimately selected as it appears the more favorable of the two materials. Allowable stress or stress intensity, with reference to the ASME Code Section VIII (Case 1409-1) is 26 000 psi (Grade 1 at 1100°F) and, with reference to ASME Code Section III (Case 1422) is 31 000 psi (Grade 1 at 800°F) versus 8700 psi and 14 800 psi, respectively for type 304 stainless steel. With respect to creep rupture, Inconel 625 again offers the best possibility. The below information was obtained from manufacturer's literature:

Stress for Rupture	Inconel 625 (psi)	Type 304 (psi)
(1,000 hrs)	94,000 (approx)	24,500
(10,000 hr)	86,000 (approx)	18,300

Reactor Cell Future Study

1) Insulated pipe and equipment simplifies support, and wall construction complicates maintenance.

Chapter 7

Chemical Processing

7.1 General

Task I objective for CONOCO was to develop a spatial layout for the Conceptual Design Flow Sheet of the chemical processing system supplied by ORNL.

The information provided by Oak Ridge National Laboratory contained the basic chemistry for their Conceptual Design Flow Sheet and the major values for processing component sizes and stream flow rates.

An engineering analysis by CONOCO of the ORNL concept was performed to define the pumps, valves, auxiliary vessels, and other processing components which would be necessary to complete the chemical processing system. From these engineering flow sheets the spatial layout was developed.

Section 7.2 gives the design basis from which the process flow and spatial layout were developed. Section 7.3 gives a process description of the chemical processing plant. Detailed process flow drawings described by the process description and layout drawings which define the spatial requirements are given in Section 7.4.

7.2 Design Basis

The basis for the process flow and spatial layout of the chemical processing plant was developed both from meetings with the personnel of Ebasco Services Incorporated, and Oak Ridge National Laboratory, and from reports issued by ORNL.

Radioactivity in the chemical processing cell, which is at a very high level, requires all operations and maintenance be performed remotely; human access to the cell is not possible after initial activation. Design is for a 30-year nominal life expectancy for the reactor. The cell enclosure is heated to the average process temperature. All process components are capable of being drained by freeze valves to tanks equipped with a fail-safe cooling system. The cell area is preferably rectangular in shape with process components located along the walls of the cell.

More specific premises upon which the process flow diagram and spatial layout were developed are contained in the following lists. The first three state premises concerning (1) process flow, (2) safety, and (3) plant maintenance.

7.2.1 Process Flow Bases

a) The process is continuous except for the periodic operation of the rare earth salt fluorinator and UF_6 product removal system.

b) The reactor salt to be processed is pumped continuously to the chemical processing cell.

c) The control functions necessary to the steady state operation of the process are indicated. Instrumentation is assumed to be available and sufficient space is allowed for it.

d) All process lines in periodic use are capable of being blown with argon which is at process temperature.

e) Overflow as a method of moving fluids is preferable to employing pumps.

f) Positive displacement pumps are equipped with necessary instrumentation to function as flow metering devices. Gas pressurization as an alternate to pumping will be investigated.

g) It is possible to use control valves on all process streams.

h) Certain process components are at sufficiently low levels of radioactivity that they may be located in a hot cell adjoining the main processing cell.

i) A means of sensing a salt-bismuth interface is available.

j) The makeup of beryllium and thorium salts to reactor salt is not done in line because of the possibility of solids plugging lines.

k) The purified bismuth streams leaving the two hydrofluorinators for recycle back to the extractors may be mixed together in a common holding tank.

l) Block valves which are used solely for isolating a component during maintenance will not require extensions through the cell wall. They will be opened and closed by the manipulators used for maintenance.

m) Hastelloy-N, nickel, graphite, and molybdenum are assumed to be the only feasible materials of construction.

7.2.2 Safety Bases

a) A drain tank system is provided to drain all process vessels and lines except those containing aqueous solutions.

b) Aqueous solutions are disposed of by direct routing to the Radwaste disposal system.

c) Direct mixing of H_2 and F_2 is not to be permitted. It is assumed that H_2 and F_2 mixtures react violently.

d) The drain tank system provides a number of tanks to prevent H_2 and F_2 from contacting and to maintain segregation of the three bismuth phases which contain 0.2, 5.0, and 50.0 mole percent lithium concentrations.

e) A floor drain system drains spills to either a temporary drain tank or a flush salt drain tank.

f) Freeze valves provide a fail-safe system for draining all vessels and for bypassing pumps and control valves.

g) All process lines are sloped to be self draining during shutdown.

h) The drain tanks are sized on the basis of double the volume of the vessels which each must drain.

i) Each drain tank is vented to the vessels it drains through an argon manifold. Argon is added and relieved as necessary to compensate for changes in liquid holdup.

j) The fluids in the drain tanks can be pumped back to the processing system.

k) All pumps, valves, and vessels are cooled with the NaK operational coolant system.

l) The NaK operational coolant system operates with its own pumps and surge tank.

m) The drain tanks are equipped with a fail-safe NaK coolant system for emergency use in addition to the operational cooling system.

n) All off-gases are disposed of through a central off-gas disposal system.

o) The process cell atmosphere is argon gas under slight negative pressure. Argon seals are provided for controlled in-leakage.

7.2.3 Plant Maintenance Bases

a) A rectangular shape for the chemical processing cell is compatible with the building concept which has 40 ft bay widths.

b) Normal equipment entry into the process cell is through a baffled air-lock.

c) All electric motors for process pumps are located outside processing cell.

d) All maintenance and component replacement is done remotely. Overhead cranes and wall manipulators are provided for normal remote handling. Removable concrete ceiling slabs allow the reactor crane access to the processing cell interior if necessary.

e) Viewing of the process cell is by view windows, mirrors, and television cameras.

f) All wall penetrations for pump and valve drive extensions are below knee and above head heights.

g) Vessels are located in a plane parallel to the wall. A minimum wall clearance of 1 ft and a minimum clearance between vessels of 3 ft is required.

h) Pumps are located at a minimum of 3 ft for horizontal spacing and 2 ft for vertical spacing.

i) All lines have 6 in. clearance.

j) All process components are capable of being blocked off by valves and removed.

k) The location of each component in the cell, is indexed for positioning of replacements. However, final alignment of components and operation of tools is by some visual means.

l) The flanging of lines in a leakproof manner is assumed.

7.3 Fuel Salt Chemical Processing

7.3.1 Purpose of Chemical Processing

The chemical processing system has the objectives of isolating protactinium-233 (^{233}Pa) from regions of high neutron flux during its decay to uranium-233 (^{233}U), and removing fission and corrosion products from the reactor salt. The processing system functions to achieve these objectives by first removing uranium from the reactor salt by fluorination. The protactinium is then removed from the reactor salt to a bismuth stream by reductive extraction. The bismuth stream in turn transfers the Pa into a decay salt phase where the Pa is held for its decay to uranium. The major portion of the uranium removed from the decay salt by fluorination is returned to the reactor salt while a small portion is routed to a product uranium hexafluoride container.

After Pa removal, the rare earth fission products are reductively extracted from the reactor salt into a bismuth phase. The rare earths are then transferred sequentially into a LiF phase, into another bismuth phase, and finally into a salt phase rich in rare earths. The rare earth salt is batch fluorinated periodically, to remove any remaining uranium before being transferred to storage tanks to await disposal.

The reactor salt begins reconstitution by purging a small amount of salt periodically to be batch fluorinated with the rare earth salt, and adjusting the salt composition with beryllium and thorium salts. Uranium hexafluoride from the fluorinators is returned to the reactor salt by hydrogen reduction. The salt is lastly treated with a hydrogen and hydrogen fluoride gas mixture and filtered to remove corrosion products.

7.3.2 Process Description

The chemical processing plant is divided functionally into seven sections (Sections 100 through 700). Section 100 is the reactor salt system which has the function of removing the Pa and rare earths for further processing and preparing the reactor salt for its return to the reactor drain tank. A small stream of 0.88 gpm reactor fuel salt is taken continuously from the reactor drain tank to be contacted with fluorine in the reactor salt fluorinator R-101 where about 95 percent of the uranium salt (UF_4) is removed as gaseous UF_6. The salt is then contacted with hydrogen in the UF_5 reduction reactor R-111 to reduce the remaining nonvolatile uranium fluoride salts back to the UF_4 salt.

The salt continues to the protactinium (Pa) extractor W-102 where the remaining uranium and protactinium salts undergo reductive extraction into a liquid bismuth phase containing 0.2 mole percent lithium. The salt and bismuth are contacted counter-currently at a metal-to-salt volume ratio of about 0.125. The lithium and thorium dissolved in the bismuth reduce the uranium and protactinium salts to the metals which are soluble in the bismuth phase. The bismuth stream containing the Pa continues to Section 300 for further chemical processing.

The fuel salt continues to the rare earth (RE) extractor W-103 for removal of the di- and trivalent rare earth salts from the salt phase by reductive extraction into a bismuth phase containing 0.2 mole percent lithium. The bismuth and salt are counter-currently contacted at a metal-to-salt volume ratio of 14. With the lithium acting as the reducing agent, the rare earth salts are reduced to the rare earth metals which are soluble in the bismuth. The major portion (12.39 gpm) of the bismuth stream leaving the RE extractor goes to Section 200 to the rare earth transfer (RET) extractor W-201 while a small portion (0.11 gpm) is the source of bismuth for Pa extractor W-102.

Reconstitution of the fuel salt for its return to the reactor system begins in a small settler D-109. Periodically a small volume of salt is transferred to a mix tank where thorium and beryllium salts are added to adjust the composition of the reactor salt. After mixing, the salt is transferred back to the settler. A reactor salt purge of about 0.16 cu ft/day is taken daily from the settler to the rare earth salt hold-up tank D-402 in Section 400.

From the settler, reactor salt flow to the UF_6 reduction reactor R-104. A gaseous UF_6 and F_2 mixture from the fluorinators reacts with a recycle salt stream containing UF_4 to form subfluoride salts such as nonvolatile UF_5. The salt is then contacted with hydrogen to reduce the subfluoride salts to UF_4.

Lastly, the salt is counter-currently contacted with a gaseous mixture of hydrogen and hydrogen fluoride in nickel gauze packed reactors R-106 and R-107 to achieve a desired UF_4/UF_3 ratio and to remove impurities such as bismuth, nickel, and iron fluorides. The salt passes through porous metal filter F-108 on its return to the reactor drain tank.

Section 200 is the transfer salt system. Its function is first to transfer the rare earths extracted into the bismuth stream in Section 100 into a lithium chloride salt phase and second to transfer the di- and tri-valent rare earths into separate bismuth phases. The bismuth stream from RE extractor W-103 carries rare earths to RET extractor W-201. The bismuth is counter-currently contacted with a 33.4 gpm stream of lithium chloride (LiCl) salt. The rare earth fission products are converted from dissolved metals in the bismuth phase to their rare earth chlorides which are soluble in the LiCl stream. The bismuth is recycled to Section 100 to RE extractor W-103 after a makeup mixture of 0.2 mole percent lithium in bismuth is added to replace the bismuth routed to W-102.

The rare earths are removed from the LiCl stream in two stages. The trivalent rare earths are transferred from the LiCl phase to a bismuth stream (8.28 gpm) containing 5 mole percent lithium by counter-current contact in RE3 extractor W-203. The bismuth is circulated from the RE3 heat exchanger X-205 to provide sufficient heat removal. Daily a portion of the bismuth is drawn off to Section 400 to the RE3 hydrofluorinator R-401 for bismuth purification. An equal volume of recycled bismuth at 5 mole percent lithium concentration is made up daily from a Bi-Li holding tank. The major portion of the LiCl is returned to the RET extractor W-201 while 0.69 gpm (about 2 percent of the total LiCl flow) is routed to RE2 extractor W-202. The divalent rare earths are removed from the LiCl phase by counter-current contact with a 0.23 gpm bismuth stream containing 50 mole percent lithium. For heat removal purposes, the bismuth is circulated through RE2 heat exchanger X-204. Daily a portion of the bismuth is transferred to the Pa-RE2 hydrofluorinator R-301 in Section 300 for bismuth purification, and an equal volume of recycled bismuth at the 50 mole percent lithium concentration is replaced from a Bi-Li holding tank.

Section 300 is the Pa salt system which accepts Pa and divalent rare earths from Sections 100 and 200, respectively. The bismuth streams from the Pa extractor in Section 100 and RE2 extractor in Section 200 flow to the Pa-RE2 hydrofluorinator R-301 where the bismuth is contacted with the Pa salt and hydrogen fluoride gas. The protactinium, uranium, and rare earths dissolved in the bismuth are oxidized by the hydrogen fluoride to their salt forms which are soluble in the Pa salt. The purified bismuth is recycled to the RE and RE2 extractors in Sections 100 and 200, respectively, after proper lithium addition in holding tanks. The Pa salt flows to the Pa salt fluorinator R-302 at 0.68 gpm where it is contacted with fluorine. Ninety mole percent of the uranium in the salt is removed by fluorination to gaseous UF_6 which is normally returned to R-104 for reconstitution of the reactor fuel salt. The Pa salt is then contacted with hydrogen in Pa salt reduction reactor R-305 to reduce remaining nonvolatile uranium fluoride salts to UF_4. The salt stream then continuous to Pa decay tank D-303 where a salt volume of 130 cu ft is held to allow the protactinium sufficient time to decay to the uranium salt UF_4. The uranium resulting from the decay of Pa is removed in the Pa salt fluorinator as the Pa salt is recycled back to R-302. A daily purge of about 0.1 cu ft of Pa salt is sent to Section 400 to the rare earth salt holdup tank D-402.

Section 400 is the rare earth salt system which serves to concentrate rare earths into a salt phase for storage and to recover any remaining uranium before salt disposal. The reactor salt purge from D-109 and the Pa salt purge from D-303 are held in the RE salt hold-up tank D-402. This salt is continuously circulated to the RE3 hydrofluorinator R-401 to counter-currently contact the bismuth from the RE3 extractor of Section 200 and hydrogen fluoride gas. The trivalent rare earth metals dissolved in the bismuth are oxidized by the hydrogen fluoride to their salt forms which are soluble in the RE salt. The purified bismuth is recycled to RE3 extractor W-203 after lithium addition in a holding tank. Every 68 days 18.5 cu ft of RE salt is transferred to the RE salt batch fluorinator R-403. The salt is contacted with fluorine for a day to remove remaining traces of UF_4 as gaseous UF_6. After fluorination, the salt is discarded to RE salt drain tanks of Section 500 for storage.

In Section 600 the uranium production is achieved by routing, for about an hour daily, the gaseous UF_6-F_2 stream from the Pa salt fluorinator R-302 to a NaF bed W-602. This gas mixture normally returns the UF_6 to the fuel salt in reactor R-104. The NaF bed absorbs about 133 grams/day of UF_6 which is later desorbed and condensed in a UF_6 product cylinder for shipment.

Section 500 contains a system of tanks equipped with a fail-safe cooling system. These drain tanks provide liquid storage and emergency cooling capabilities for periods of abnormal process operation.

Section 600 is basically the gas recovery system which provides the HF, F_2, and H_2 gases for the process. The hydrogen and hydrogen fluoride gas mixtures from the process return to Section 600 where the HF is condensed and converted to H_2 and F_2 in an electrolytic cell. The hydrogen gas is scrubbed with a KOH solution and dried for recycle to the process.

Section 700 contains those process components, which because they are at a lower level of radioactivity, can be located in a hot cell adjacent to the main processing cell.

7.4 Index of Drawings

Title	Figure Number	CONOCO I.D. Number
Block Flow Diagram	7.1	SD1700-42-1-C
Process Flow Diagrams:		
Reactor Salt System (Sec. 100)	7.2	SD1700-42-2-D
Transfer Salt System (Sec. 200)	7.3	SD1700-42-3-D
Pa Salt System (Sec. 300)		
RE Salt System (Sec. 400)	7.4	SD1700-42-4-D
Drain Tank System (Sec. 500)	7.5	SD1700-42-5-D
Gas Recovery System (Sec. 600)	7.6	SD1700-42-6-D
Auxiliary Processing Area (Sec. 700)	7.7	SD1700-42-7-D
Plans and Sections	7.8	SD1700-54-1-D
Process Equipment Arrangement	7.9	SD1700-54-2-D
Sec. 100 Isometric and Miscellaneous Details	7.10	SD1700-54-3-D

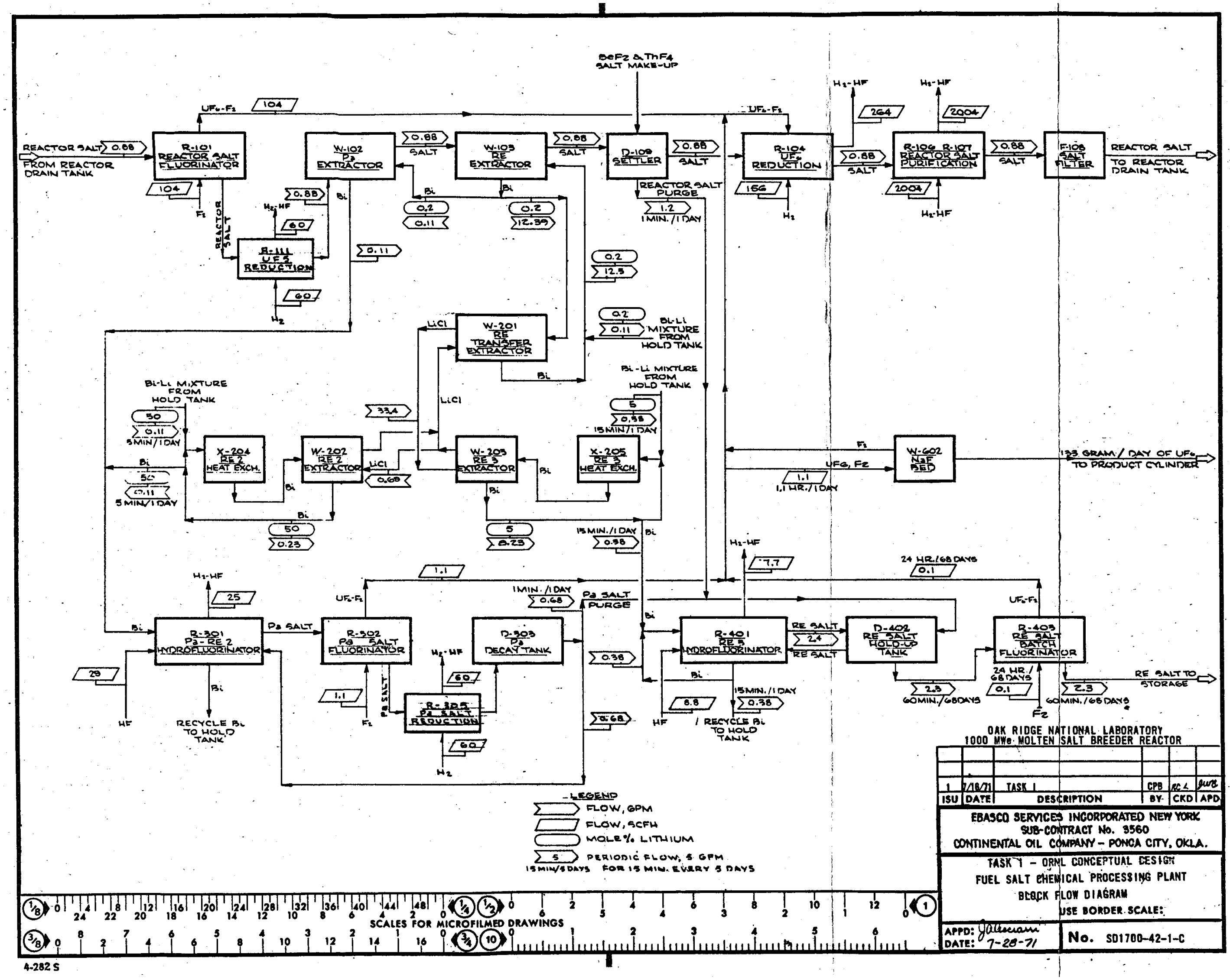


Figure 7.1: Fuel Salt Chemical Processing Plant Block Flow Diagram.

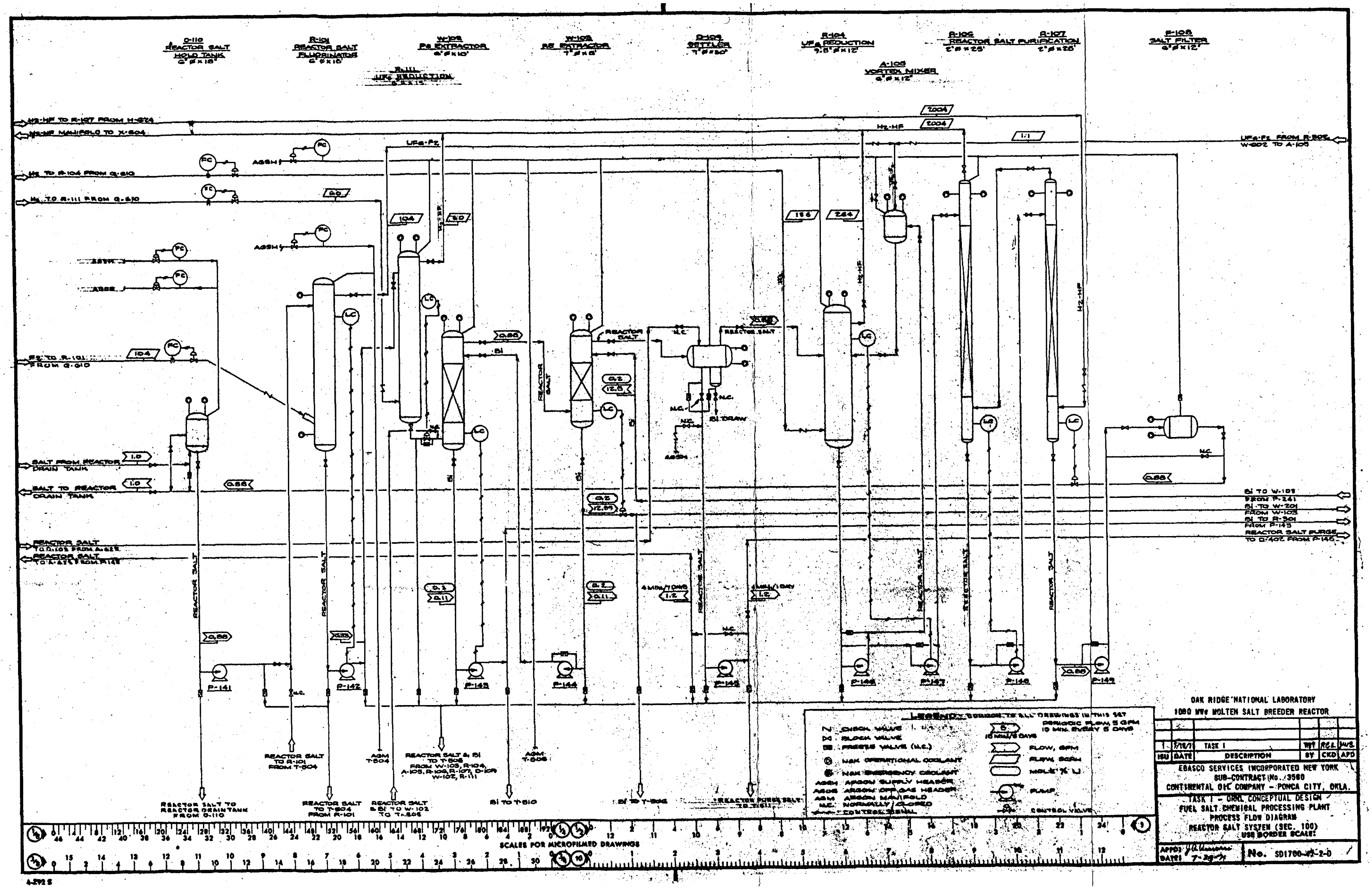


Figure 7.2: Process Flow Diagram Reactor Salt System (Sec 100).

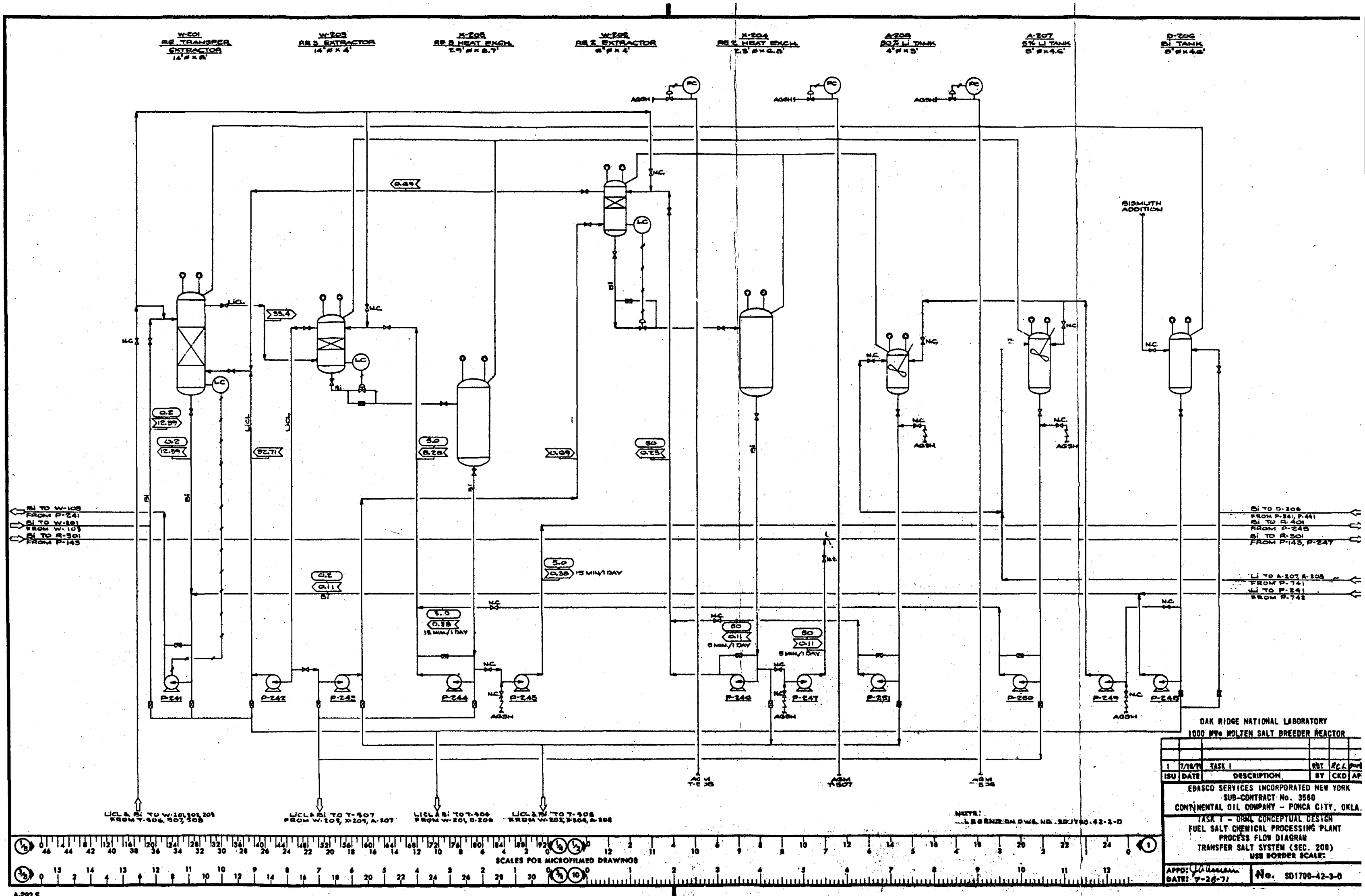


Figure 7.3: Process Flow Diagram Transfer Salt System (Sec 200).

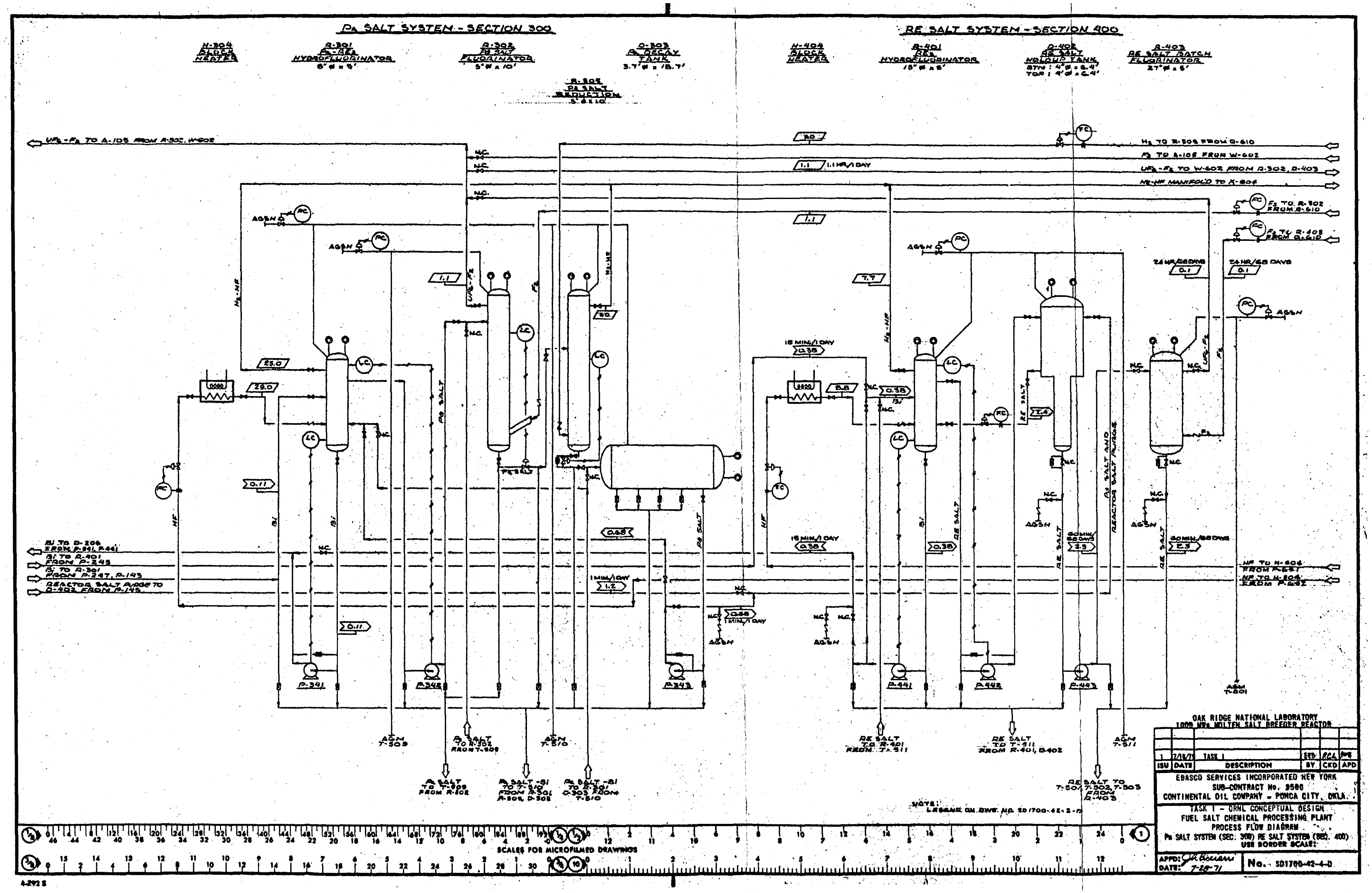


Figure 7.4: Process Flow Diagram Pa Salt System (Sec 300), Re Salt System (Sec 400).

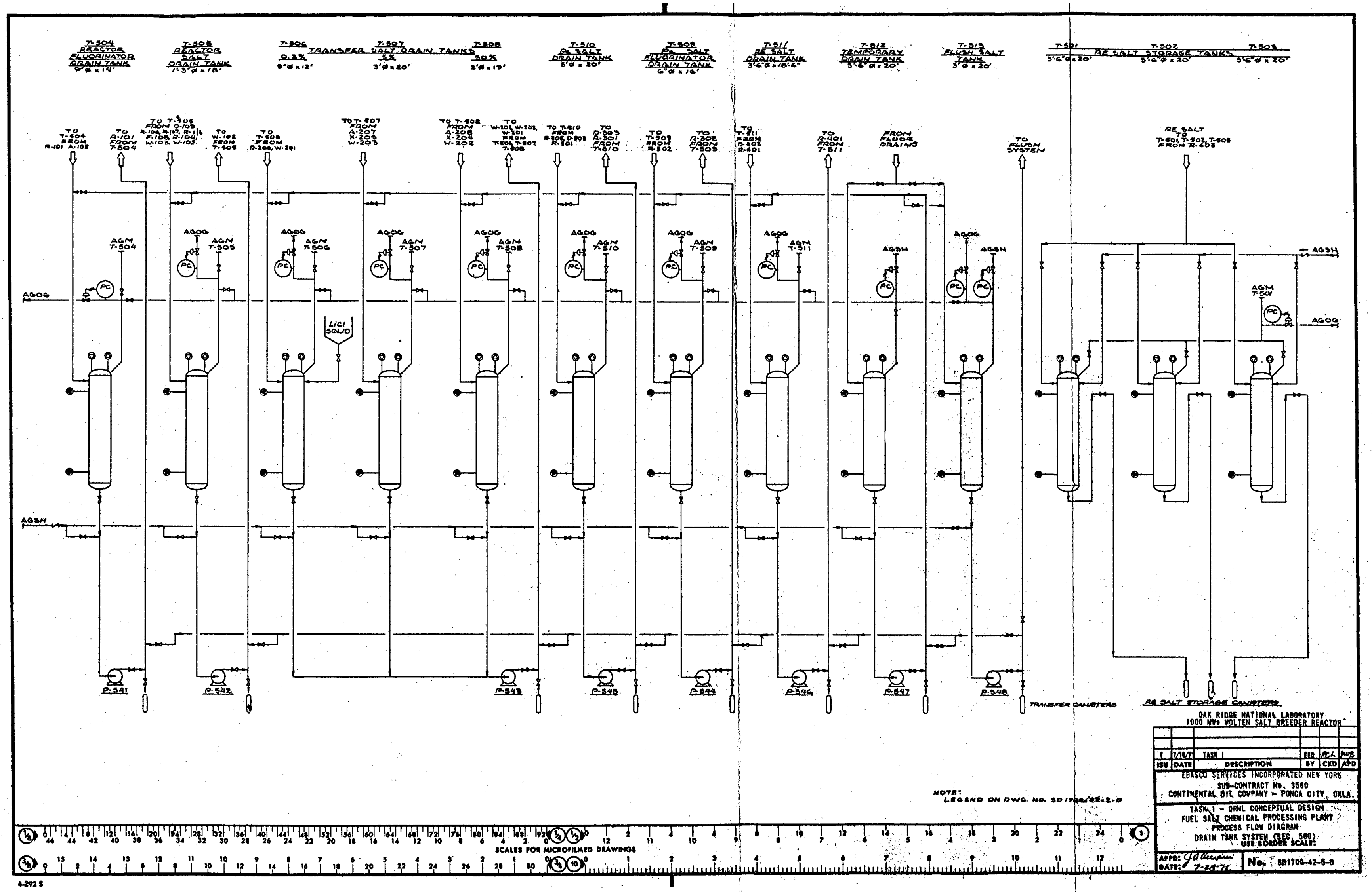


Figure 7.5: Process Flow Diagram Drain Tank System (Sec 500).

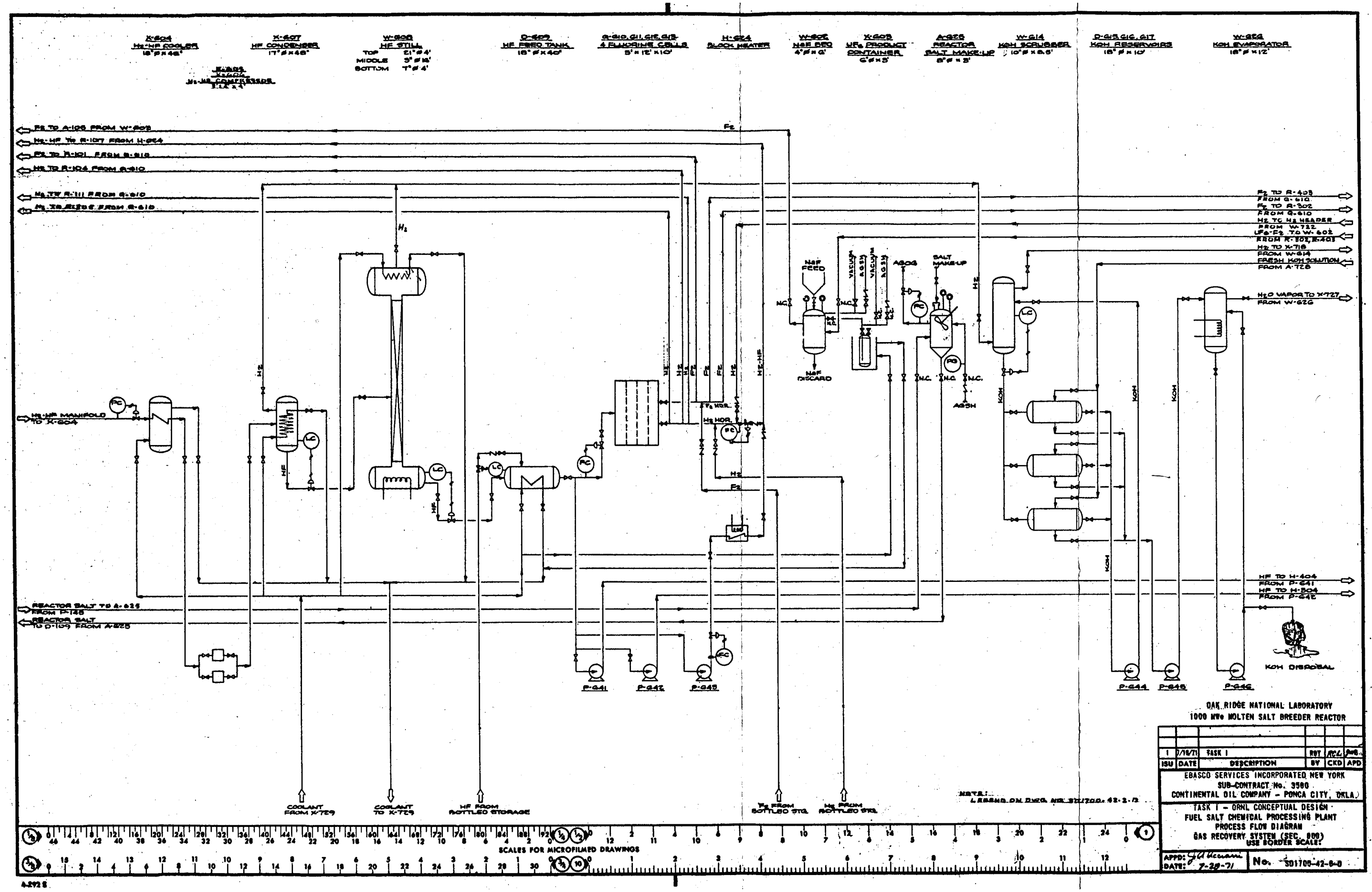
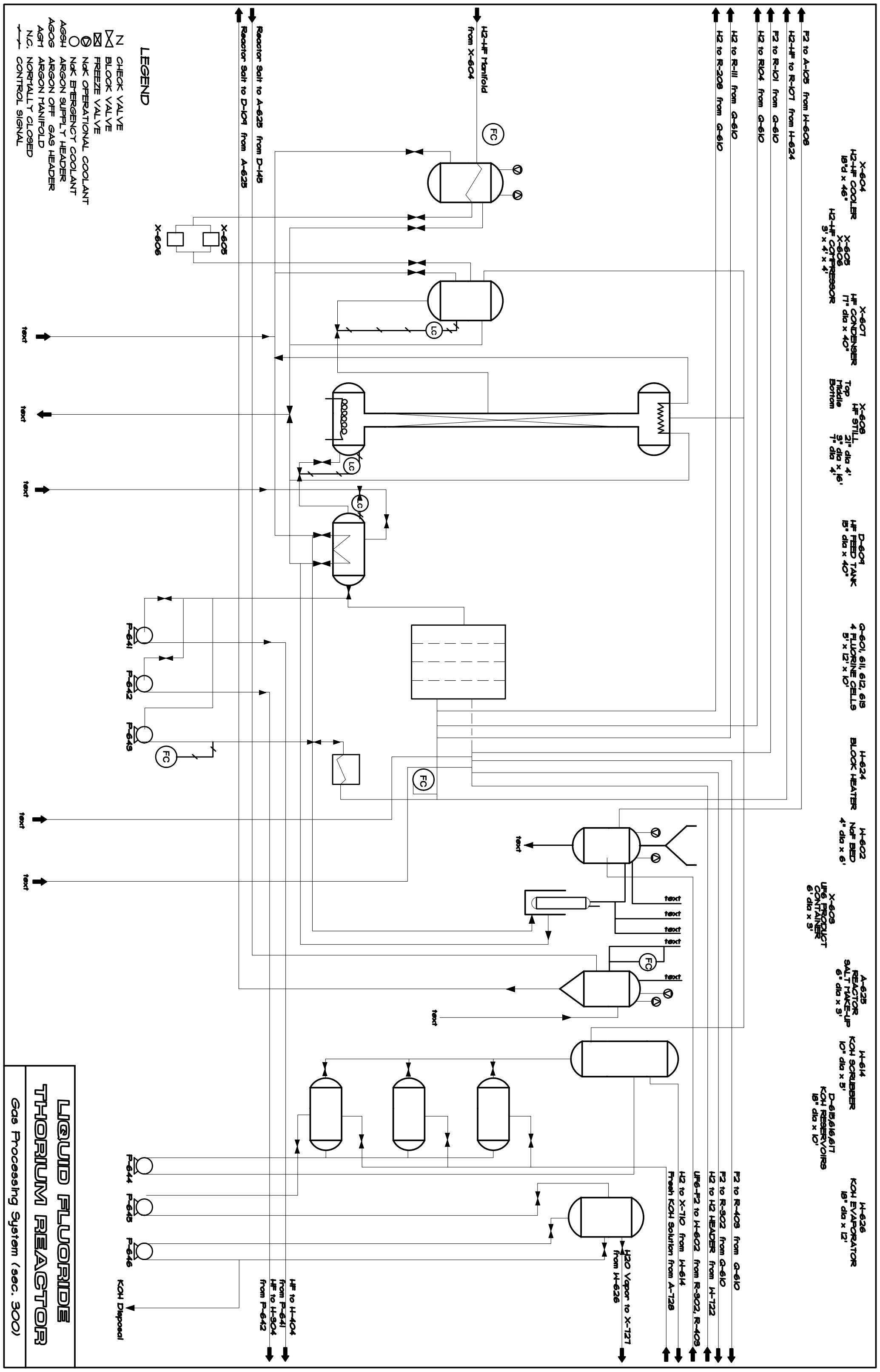


Figure 7.6: Process Flow Diagram Gas Recovery System (Sec 600).


LIQUID FLUORIDE
THORIUM REACTOR
Gas Processing System (sec. 300)
LEGEND
CHECK VALVE
BLOCK VALVE
FREEZE VALVE
NaK OPERATIONAL COOLANT
NaK EMERGENCY COOLANT
AGSH ARGON SUPPLY HEADER
AGOG ARGON OFF GAS HEADER
AGM ARGON MANIFOLD
N.C. NORMALLY CLOSED
CONTROL SIGNAL
F2 to A-105 from H-605
H2-HF to R-107 from H-624
F2 to R-101 from G-610
H2 to R104 from G-610
H2 to R-111 from G-610
H2 to R-208 from G-610
H2-HF Manifold from X-604
Reactor Salt to A-625 from D-115
Reactor Salt to D-104 from A-625
X-604 H2-HF COOLER
X-605 X-606 H2-HF COMPRESSOR
X-607 HF CONDENSER
X-608 HF STILL
D-607 HF FEED TANK
G-601, 611, 612, 613 4 FLUORINE CELLS
H-624 BLOCK HEATER
H-602 NaF BED
X-603 UF6 PRODUCT CONTAINER
A-625 REACTOR SALT MAKE-UP
H-614 KOH SCRUBBER
D-615,616,617 KOH RESERVOIRS
H-626 KOH EVAPORATOR
P-641
P-642
P-643
P-644
P-645
P-646
FC
LC
F2 to R-403 from G-610
F2 to R-302 from G-610
H2 to H2 HEADER from H-722
UF6-F2 to H-602 from R-302, R-403
H2 to X-710 from H-614
Fresh KOH Solution from A-728
H2O Vapor to X-727 from H-626
HF to H-404 from P-641
HF to H-304 from P-642
KOH Disposal

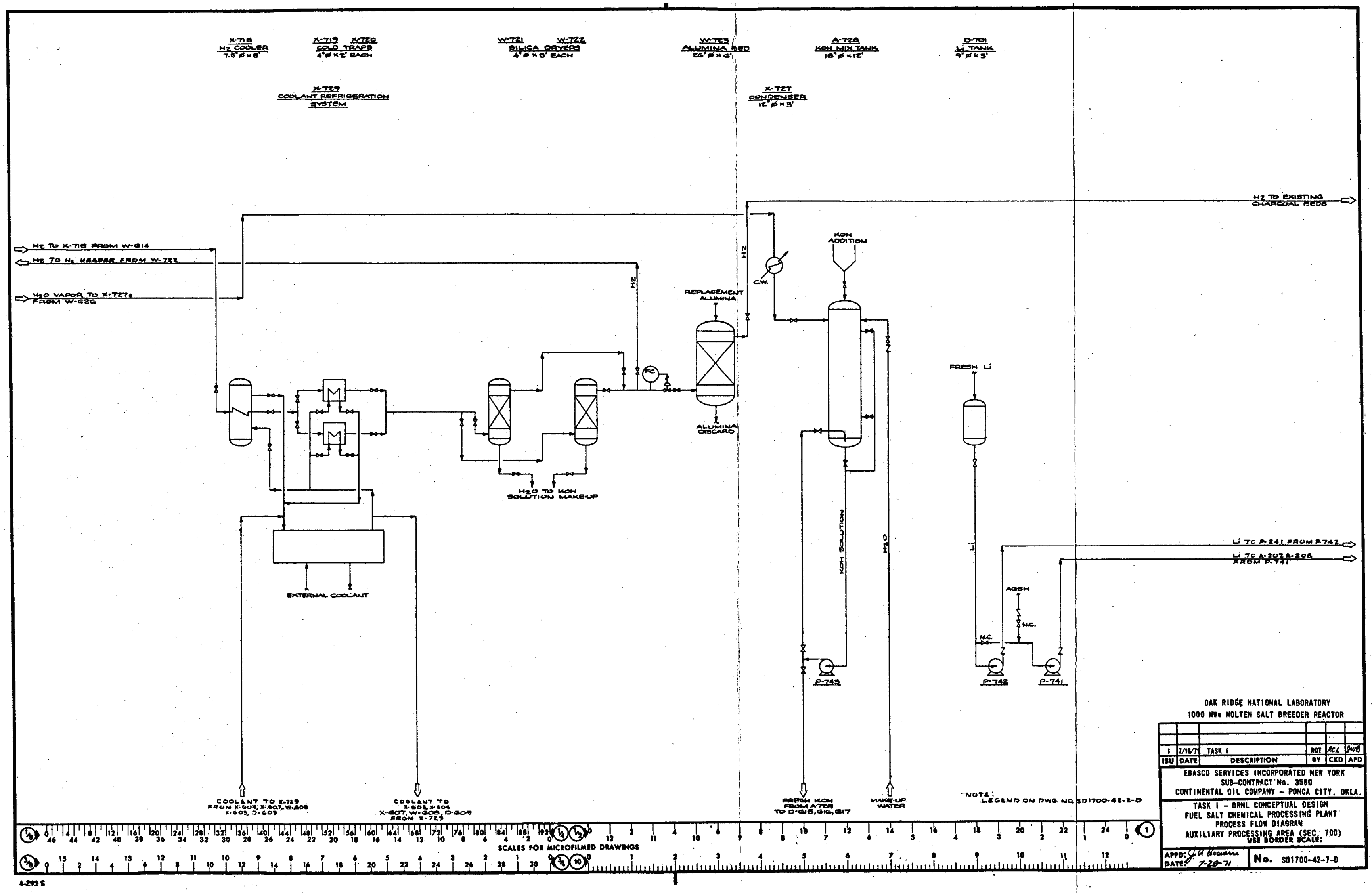


Figure 7.7: Process Flow Diagram Aux Processing Area (Sec 700).

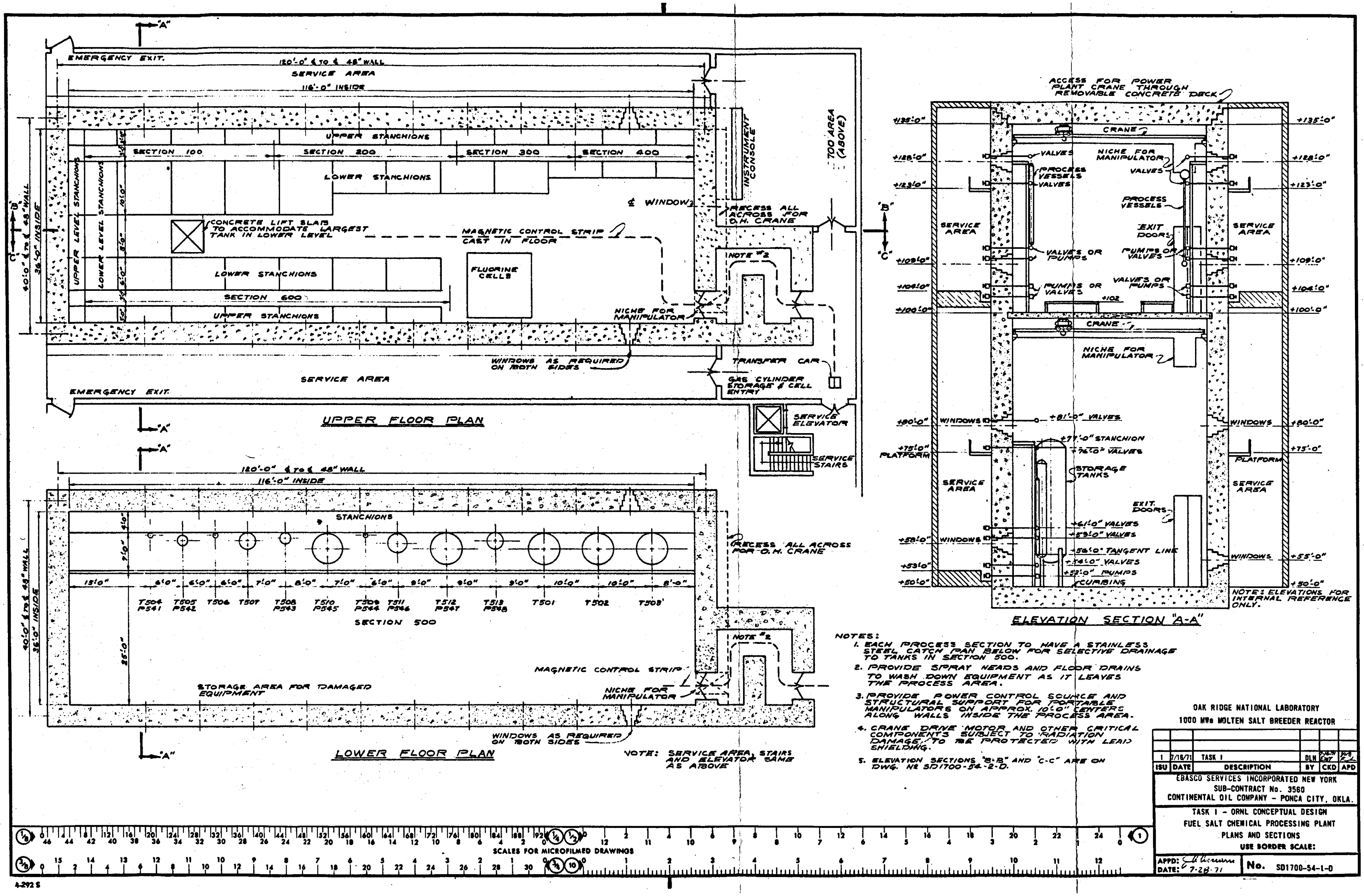


Figure 7.8: Plans and Sections.

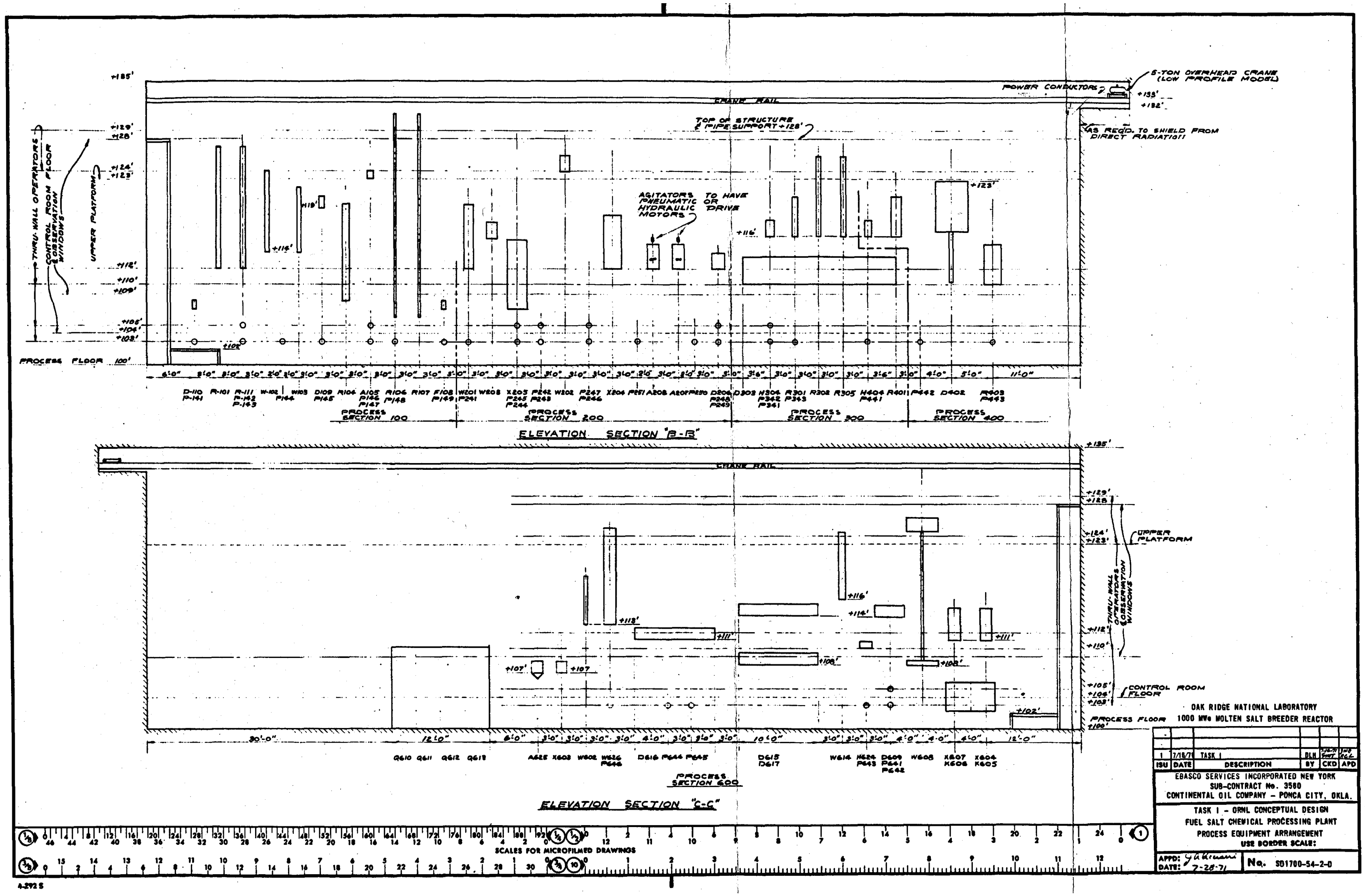


Figure 7.9: Process Equipment Arrangement.

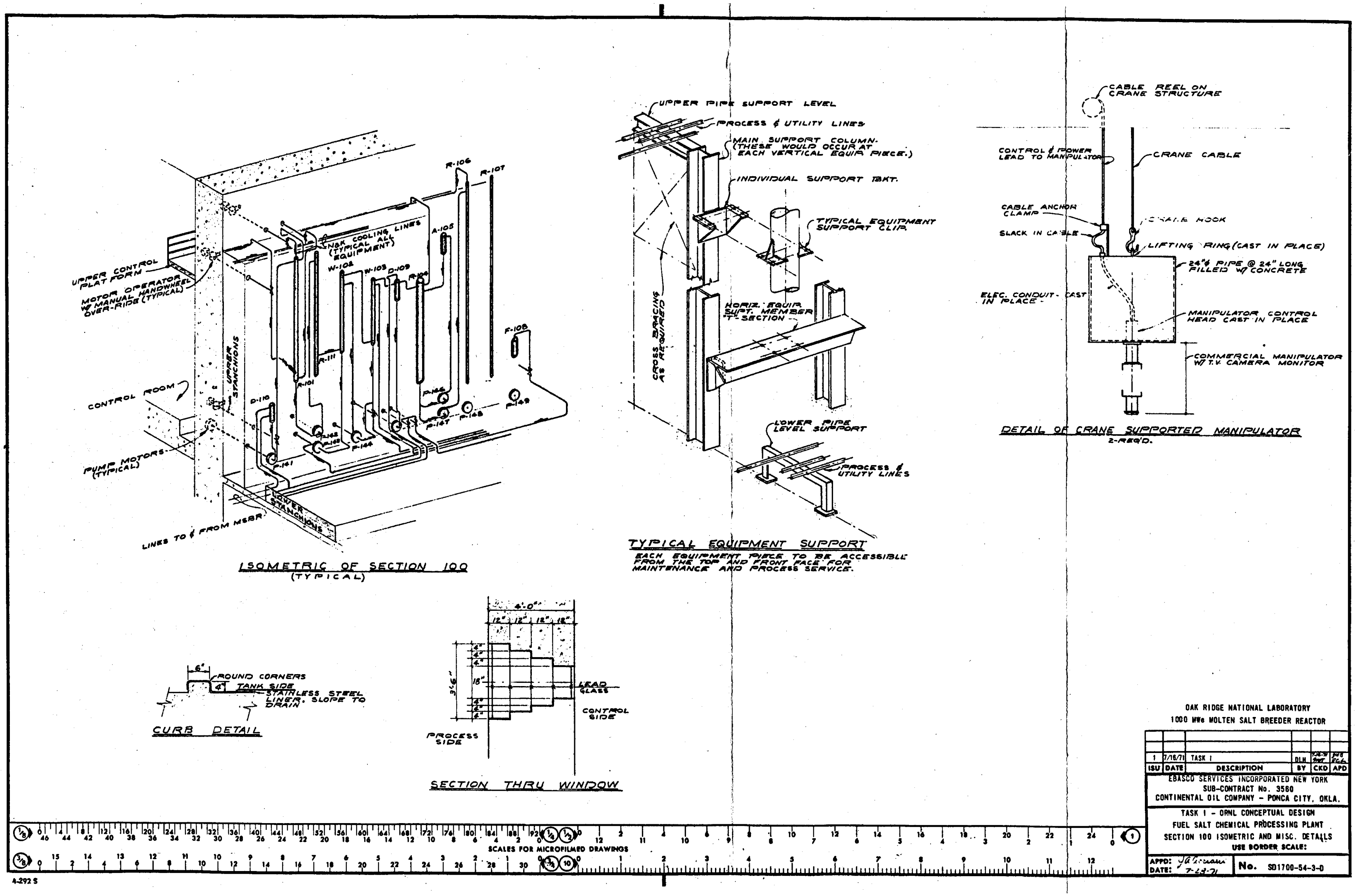


Figure 7.10: Section 100 Isometric and Misc. Details.

Chapter 8

Reactor Off-Gas System

8.1 System Requirements

The reactor off-gas system must perform the following functions:

a) maintain sufficiently low ^{135}Xe concentration in the fuel salt,

b) remove tritium from the purge gas,

c) accommodate other forms of contamination such as noble metals, particulate, or hydrocarbons which might be picked up by the purge gas,

d) have 100 percent availability,

e) provide positive confinement of all radiation under all normal modes of operation.

8.2 Basic Assumptions

The following assumptions provide the bases for the off-gas system design:

a) At least 90 percent of the xenon-135 produced enters the off-gas system. This amounts to 6.2 micro-moles/second.

b) Virtually 100 percent of all longer-lived fission gases enter the off-gas system.

c) Virtually 100 percent of the tritium (nominally 2400 curies/day) produced enters the off-gas system.

d) The purge gas emerging from the primary system will also contain short-lived fission gases in the amount determined by ORNL (ORNL-4541).

e) The carrier gas may be either helium or argon. Helium is arbitrarily selected as the tentative choice for the Ebasco reference concept.

f) The rate of decay heat production in the purge gas as it emerges from the fuel salt is about 0.1 MW/ft3(STP).

8.3 Reactor Interface System

The off-gas system is interfaced to the primary system by some method or device which injects clean purge gas and removes contaminated purge gas from the salt. It is assumed that R&D efforts and ORNL will lead to such a satisfactory system or device.

8.4 Purge-Gas Cleanup System

Two methods of cleaning the purge gas have been proposed. One method involves physical holdup of the purge gas for ^{135}Xe decay. The other involves actual cleanup of the purge gas by passing it through charcoal beds. Each system has specific advantages and disadvantages. The following sections present the merits of each system and provide the bases for choosing a reference concept.

8.4.1 Physical Holdup System

The purge gas emerging from the fuel salt flows directly to a holdup tank. The volume of the holdup tank, together with the volume flow rate of purge gas, determine the holdup time. After the desired holdup interval, the gas in the tank is recycled directly to the bubble generator. In this system, the concentration of fission gases will build up until the decay rate of fission gases in this gas reservoir equals the rate at which the fission gases are transferred to the gas reservoir. If the volume of this reservoir is small, the steady-state fission-gas concentration will be high. In this case, the ^{135}Xe back pressure can impede ^{135}Xe transfer from the fuel salt to the bubbles. If the gas volume is made sufficiently large, however, the steady-state ^{135}Xe concentration will be low enough that the back pressure will not impede ^{135}Xe transfer. Figure 8.1 shows the poison fraction and other parameters as a function of the purge-gas volume. It can be seen that a low volume gives a high fractional back pressure, and therefore, a poison fraction approaching 5 percent, i.e., the case for which there is virtually no ^{135}Xe removal. For a very high reservoir volume, the back pressure approaches zero, and the ^{135}Xe transfer is limited by the bubble-surface area. The desired poison fraction can be attained by the proper choice of purge-gas volume, together with the proper choice of surface area. If the core graphite is sealed, a purge-gas volume in the range of 4000 cu ft, together with a bubble-surface area of about 15,000 ft2 is adaquate. This surface area is slightly larger than assumed by ORNL, but is sufficient to compensate for the much higher back pressure than assumed by ORNL. These values are based on the assumptions that the graphite would be sealed and the bubbles fairly large (0.02 in. diameter). It is now believed that the bubbles will be smaller than the surface area roughly ten times larger. The back pressure will therefore become more significant and perhaps justify going to a somewhat larger purge-gas reservoir. If the gas reservoir were pressurized, then the same level of dilution could be achieved with a smaller volume. Two atmospheres of pressure would permit the reservoir volume to be roughly the same as that of the drain tank. If the drain tank were used, its cooling system would be available to remove fission-gas decay heat. Conversely, the fission-gas decay heat could be used to continually drive the drain-tank cooling system and thereby demonstrate its operability and availability. The decay heat of fission gases alone has been estimated to be about 10 MW. An additional 10 MW might be generated from decay of noble metals which accumulate in the drain tank.

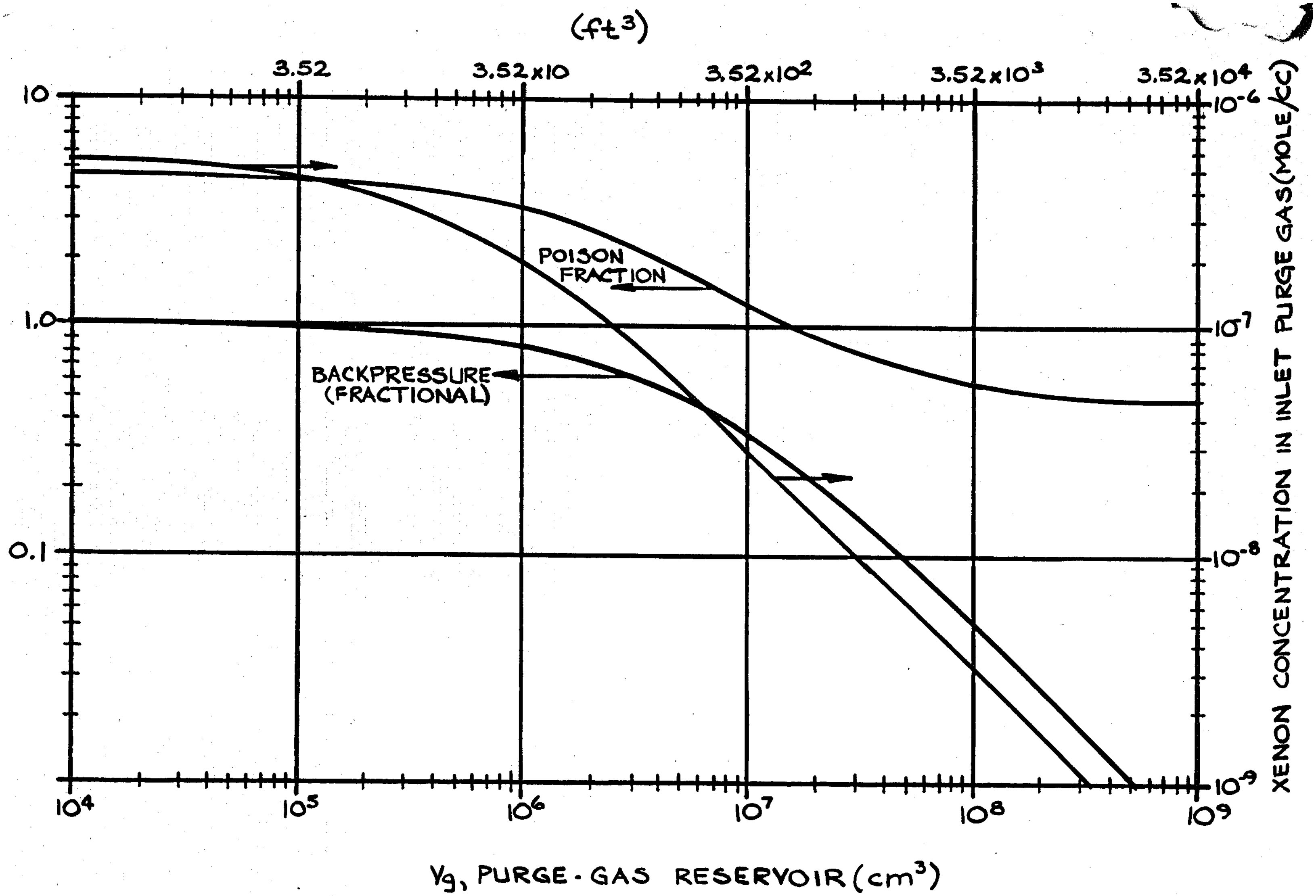


Figure 8.1: Poison Fraction and Related Quantities.

The holdup tank will contain about 600 megacuries of fission gases. This might present licensing difficulties, especially if the tank is pressurized. Alternately, several smaller tanks could be employed to reduce the likelihood of a large fission-gas release, to insure 100 percent availability, and to permit on-line maintenance and repair.

It is recognized that some purge-gas cleanup via charcoal filtration will probably be required. This will certainly be true if there is a continuous flow of clean purge gas into the contaminated purge gas, as, for example, the flow through the purge glands in the primary pumps. In this case, the supply of clean purge gas must be replenished by cleaning and recycling the contaminated gas. If, on the other hand, purge glands are not required, then it seems reasonable to have two separate gaseous systems: One system containing only highly contaminated gas and not diluted by clean gas. This would include purge gas and cover gas only. The second system would include the reactor cell atmosphere and a clean circulation system which purges cell penetrations and discharges into the cell. The two gas systems would have to be kept separate by gas-tight seals. The primary pumps would therefore require gas-tight oil seals in order to keep the two gas streams separate. Although the reactor-cell atmosphere normally would not be contaminated, it is anticipated that it might inadvertently become contaminated because of a seal failure. Thus, a charcoal filtration system should be available in the event that it is needed.

8.4.2 Charcoal Bed System

This section presents a design description of a concept proposed by ORNL. The performance criteria assumed in this study are the same as those assumed by ORNL. It is understood that these criteria are largely arbitrary and subject to change.

Assumptions

1) Fission gas production rates are based on a reactor power of 2329 MWt and a fuel of ^{233}U.
2) The carrier gas is helium, with a total flow to the off-gas system of 11 scfm. This total is the combination of flows from each of the four pump loops, consisting of 2.25 scfm from each of the gas separators and 0.5 scfm of purge gas for each of the pump shafts. Net flow of fission products and materials other than helium is at most 0.1 percent, or 0.01 scfm.
3) The atom flow rates of Kr and Xe into the off-gas system are based on calculated atom flow rates at the gas separator discharge, with appropriate corrections for a 2-hr residence time in the fuel-salt drain tank. All solids which are gas-borne at the outlet of the drain tank (including noble metals, salt mist, and solid daughters of the noble gases) will be removed by a filter before the gas stream enters the off-gas system. The total yield of tritium (^{3}H) from all mechanisms will be 2400 curies/day, and all tritium will remain in the off-gas stream, that is, for the purpose of studying the off-gas system, the rate of diffusion of tritium through vessels and pipe walls is assumed to be zero.

4) The gas will enter the off-gas system at 10 to 15 psig. Nine scfm will be returned to the bubble generators at 5 psig. Two scfm will be returned to the purge-gas header at 45 psig.

5) Shielding will be provided for attenuation of penetrating radiation to permissible levels. Instrumentation will warn of excessive leakage of gas or penetrating radiation.

6) The target reliability of the system is 100 percent; that is, spare units will be provided, and the maintainability of units will be such that predictable failures in the off-gas system will not result in shutdown of the reactor or loss of the contaminants to the environment.

The flow of gas in the off-gas system can be represented by two recycle loops, a 47-hr Xe holdup loop, and a long delay (~90-day) Xe holdup loop, as shown in Figure 8.2. These holdup times do not include the 2-hr residence time of the off-gas stream in flowing through the fuel-salt drain tank. The 47-hr loop circulates through the bubble generator and gas separator to strip the ^{135}Xe from the fuel salt. The long-delay loop carries the balance of the gas flow in the fuel system. The two loops are joined together at the salt entrainment separator and flow concurrently through the primary drain tank and the 47-hr holdup system.

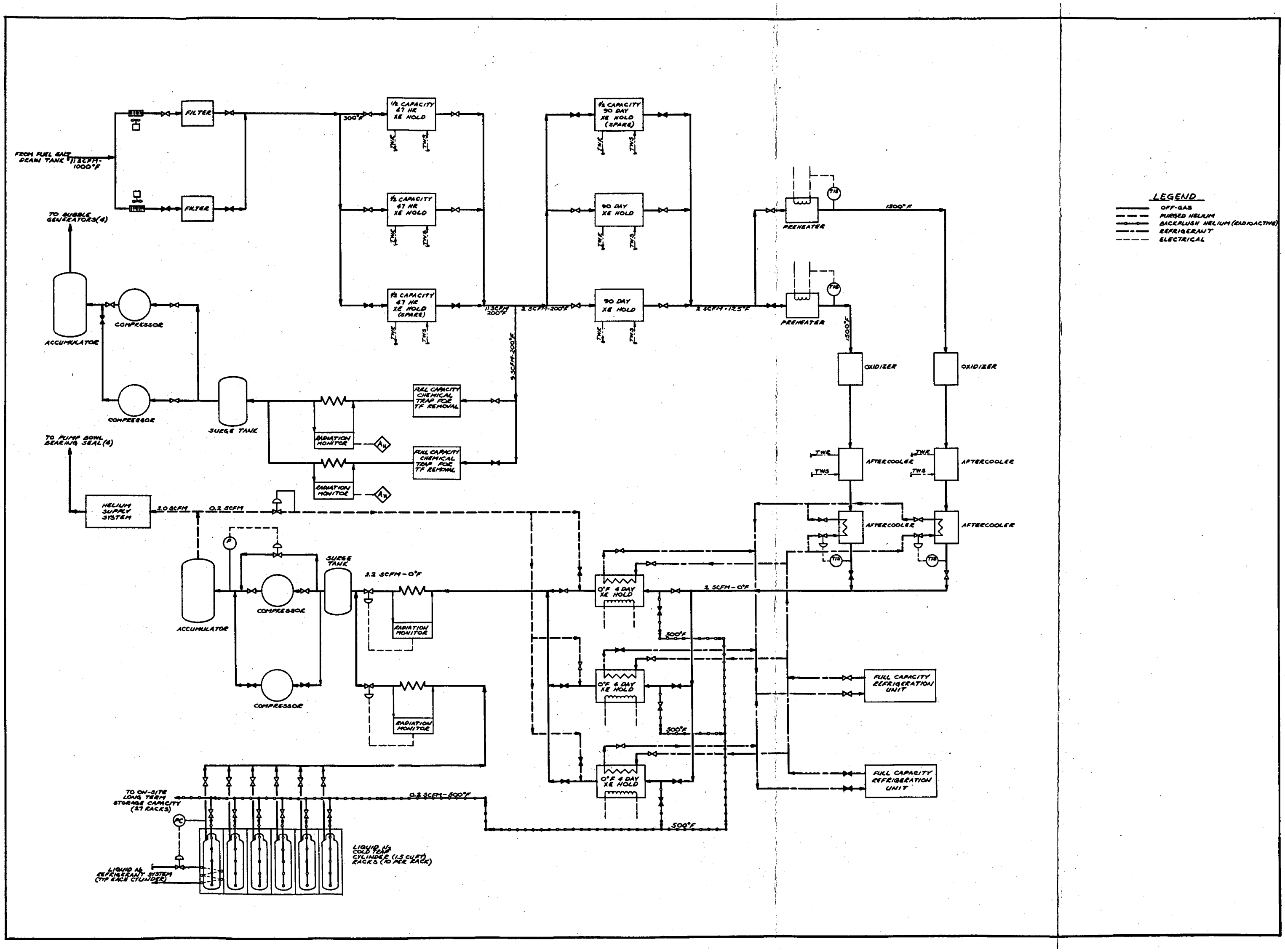


Figure 8.2: Off-Gas System - Flow Diagram.

The concurrent stream enters the primary off-gas system from the fuel-salt drain tank and is cooled by means of a radiator or forced convention type air coolers. The purpose of cooling the gas would be to increase the effectiveness of the 47-hr holdup system. The drain tank will probably serve as an efficient collector of particulates in the gas, but if it proves necessary, a particle trap, or filter, can be added, as shown in Figure 8.2. At this point, the gas will have been stripped of non-gaseous components (noble metals, salt mist, and non-gaseous daughters of the noble gases), so that the primary contaminants are Kr, Xe, and ^{3}H. About 2 hr will have elapsed since the gas first left the fuel salt system. The gas first passes through the 47-hr Xe holdup system to provide a residence time for xenon molecules sufficient to permit the ^{135}Xe to decay to about 3 percent of the inlet amount. The 47-hr holdup system will utilize charcoal for the dynamic adsorption and holdup of krypton and xenon. The decay heat will be transferred to a forced circulating water system within the shells of heat exchanger units.

At the outlet of the 47-hr system, the gas stream is divided into the two recycle loops. In the 47-hr recycle loop, 9 scfm, or about 80 percent of the total flow, passes in succession through a chemical trap and alarm sytem, a surge tank, a compressor, and an accumulator. From the accumulator the gas is metered to the bubble generators at the four circulating pumps. In the second recycle gas stream, 2 scfm, or 20 percent of the total flow, passes first through the long-delay Xe holdup system where the residence time for krypton and xenon are sufficiently long to allow all radioisotopes except the 10-yr ^{85}Kr to decay to insignificant levels. The gas then passes throuel a gas cleanup system which reduces the level of any remaining contaminants (^{85}Kr, H, stable isotopes of Kr and Xe, water, hydrocarbons, etc.) to an acceptable level, then through a surge tank, a compressor, and an accumulator, and finally is returned to the primary system.

Design Criteria for the Gas Cleanup System:

1) Carrier gas is helium at a flow of 2 scfm and an inlet pressure of 20 psia. The design pressure drop is 4 psi.

2) The level of each contaminant in the effluent gas is not more than 1 percent of the value at inlet.

3) The gas contains some ^{131m}Xe, which is negligible from a mass flow standpoint, but which must be considered in the design of shielding and the heat dissipation system.

4) The stable noble gases, as well as essentially 100 percent of the ^{85}Kr, and ^{3}H, will be carried into the gas cleanup system at a rate equal to the rate of production in the reactor (assuming that no tritium is lost to other parts of the reactor system by diffusion through pipe and vessel walls).

5) The tritium oxidizer preheater and aftercooler have heat loads of 3 kW and designed for negligible pressure drop.

6) The tritium oxidizer is 2 in. ID by 3 ft long, is packed with 13 lb of copper oxide, and operates at 1500°F. The tritium flow is 0.036 cu ft/day with an allowable Δp of 2 psi. The CuO consumption at breakthrough if 60 percent and the operating life of a unit is to be 1000 days. Development work will be needed to confirm the efficiency and pressure drop estimates, how-

ever.

7) Each adsorber is made up of 16 pieces of charcoal-packed 8 in. pipe, with 1-1/2 in. interconnections. The total length of 8 in. pipe is 288 ft, arranged in two branches to provide a Δp of 2 psi. The pipes are closely stacked inside a 3 ft to 4 ft diameter pipe with a heated or cooled fluid circulated in the interstitial spaces, to provide an average on-stream operating temperature of 0°F and a temperature of 500°F when on the regeneration cycle. Using an adsorption coefficient of 4.8 cu ft/lb the total charcoal requirement is about 3000 lb. The operating cycle is 8 days-4 days on stream and 4 days regenerating.

8) Nitrogen storage bottles, similar to a 1.5 cu ft high-pressure gas cylinder, each container would be kept on line for 12 cycles, or 48 days. About 30 lb of xenon, 6 lb of krypton, and 0.1 lb of tritiated water would be accumulated in each bottle. Each freshly filled bottle would contain about 240 curies of ^{85}Kr, equivalent to a decay energy of about 0.4 watts per bottle. The bottle pressure after equilibrating to room temperature would be 1000 psi. About 230 bottles would be filled during the 30-year life of an MSBR station. After each container was filled, it would be transferred to long-term storage, where, after a period of about 100 years, the ^{3}H and ^{85}Kr would decay sufficiently for the contents to be released or sold without radiological protection.

9) The gas-cleanup compressor has a capacity of about 2 scfm helium with an inlet pressure of 14.7 psia and an outlet pressure of 60 psia. A major requirement for the compressor is to provide positive sealing for the pumped fluid so that the highly purified gas is not recontaminated.

10) The Off-Gas System pipe design will be such as to minimize the effects of solids accumulations, such as at valve seats, pipe bends, etc., where fission product decay heating would tend to cause hot spots.

11) Wherever necessary, valves will be provided with welded bellows for positive stem sealing. Positive-sealed end connections, either buffered O-rings or butt welds, are used. Where necessary, provisions are made for remote maintainence of valving.

12) Gas system piping and components are provided with a controlled-circulation ambient air system, which assures prompt detection of gas leaks, and the channeling of such leaks to an absolute filter system.

Description of Gas Cleanup System

Upon entering the gas cleanup system, as shown in Figure 8.2, the off-gas first passes through a preheater, which raises the gas temperature to 1500°F. It then passes through an oxidizer, which converts the tritium to $^{3}H_2O$, and then through a water cooled aftercooler and a refrigerant cooled aftercooler reducing the gas temperature in two steps from 1500°F to 100°F and then to 0°F. The function of the aftercoolers is to reduce the heat load on the ensuing components. The off-gas then passes through a charcoal-packed adsorber which is maintained at 0°F. The $^{3}H_2O$ and the kryptons and xenons are retained on the charcoal while the carrier gas passes through the bed. After leaving the refrigerated adsorber, the carrier gas is recompressed and recycled to the reactor purge system. In normal operation, two adsorbers are alternated on a fixed cycle. A regeneration process is used to transfer the adsorbed gases in the off-stream unit to a receiver cylinder for permanent storage.

The helium gas used for regeneration is taken from the He purge header, and preheated, if considered necessary. During regeneration, the gas flow is about 10 percent of normal on-stream flow and moves through the adsorber unit in the opposite direction. After leaving the heated adsorber bed, the regenerating gas, now laden with ^3H_2O, krypton and xenon, passes through a storage bottle maintained at a liquid nitrogen temperature of -325°F by use of an external liquid nitrogen refrigeration system. The water, krypton and xenon are trapped in the bottle and the purified effluent returned to the main carrier-gas stream. A compressor is used to return the effluent of the gas cleanup system to the purge-gas cycle.

Design Criteria for the 47-Hour Xenon Holdup System

1) The residence time for xenon is 47 hours. This time is exclusive of the volume holdup in the primary system drain tank and other vessels and ducts. A 47-hour delay time permits 97 percent of the 9.14 hr ^{135}Xe to decay.

2) The estimated heat load is 2.14 MW, 42 percent of which is due to daughter-product decay. The design capacity of the heat removal system is 125 percent of calculated, or 2.7 MW.

3) The 47-hr Xe recycle system is to be designed to operate on the available pressure drop, so a compressor probably will not be required. However, if one is needed, the flow will be 9 scfm and the compression ratio will be fairly low, about 1.4 to 1. Positive sealing will be essential to prevent outleakage of the highly radioactive gas. Other requirements will be radiation resistance and remote maintainability.

4) A dynamic adsorption system is used for delay of the xenon. The adsorbent is activiated charcoal with transfer of the decay heat to forced circulating water. The design temperature of the charcoal pipe wall is dependent upon the average circulating water temperature in the respective decay heat generating regions, The average temperature of the charcoal in regions I and II is 285°F and in region III is 254°F.

5) The assumed charcoal properties are:

Bulk density	30 lb/cu ft
Thermal conductivity	0.03 BTU/hr sq ft.°F/ft
Size range	6 to 14 Tyler Sieve Series (1/8 to 3/64 in.)

6) The average decay heat distribution is obtained by selecting three regions at holdup time intervals of 17, 14 and 16 hours, resulting in respective decay heat rates of 1.1, 0.7 and 0.4 kW/min.

7) The efficiency of the bed is assumed to decrease with time due to accumulation of solid daughters. Thirty percent spare capacity is provided and provision is made for replacement of modules by remote maintenance techniques.

8) Carrier gas flow is 11 scfm and the overall pressure drop is less than 5 psi. An estimate of the size of the charcoal bed is obtained by using the empirical relationship developed by Browning and Bolta.[1]

$$t_h = \frac{km}{f} \tag{8.1}$$

where:

t_h = holdup time
m = mass of charcoal
f = volume flow rate of carrier gas at local conditions and k is a proportionality factor which is known as the adsorption coefficient and which varies with the carrier gas composition, the absorbent, the adsorbate and the temperature. For typical commercial charcoals, Ackley and Browning[2] have determined the following relationship between k and temperature from xenon at temperatures between 32°F and 140°F.

$$\text{k (Xe)} = 3.2 \times 10^4 \exp \frac{5880}{\text{T°R}} \text{ cu ft/lb} \tag{8.2}$$

Equation 1 indicates that the holdup time increases directly with k. However, an increase in holdup time increases the heat generation which results in an increase in charcoal temperature and decrease in k in accordance with Equation 2. An increase in temperature causes an increase in f (local flow rate), which results in a decrease in holdup time. For any given section of the bed, k and t_h will seek equilibrium values which are a balance between these opposing forces.

For the purpose of this conceptual design, the assumption was made that Equation 2 is valid up to 300°F. Equation 2 indicates that this temperature would be equivalent to an adsorption coefficient of 0.87 cu ft/lb for Regions I and II and 1.2 cu ft/lb for Region III. For a holdup time of 48 hr and a flow of 11 scfm, Equation 1 indicates that the required mass of charcoal would be 12 900 lb for Region I, 10 700 lb for Region II and 8680 lb for Region III.

Description of 47-Hr Xenon Charcoal Beds

Using the dynamic adsorption method of calculating charcoal requirements, the average adsorption coefficient is affected by 1) Inside wall temperature, 2) Pipe diameter, 3) Heat flux, 4) Gas flow. For a given heat flux, the inside wall temperature can be changed by altering the coolant water temperature. Because the heat flux is low, the fluid film temperature drop is negligible, and since the metal thermal resistance is low, the inside wall temperature can be assumed to be approximately equal to the coolant water temperature in that region. Straight tubes filled with charcoal inside a

[1] W.E. Browning and C.C. Bolta, Measurement and Analysis of the Holdup of Gas Mixtures by Charcoal adsorption Traps, ORNL-2116, July 1956.

[2] R.D. Ackley and W.E. Browning Jr., Equilibirum Adsorption of Kr and Xe on Activated Carbon and Linde Molecular Sieves, ORNL internal correspondence CF-61-2-32 (February 14, 1961).

baffled-delay type exchanger would thus transfer its decay heat to water in the shell side. This water would then be cooled by a conventional cooling water system as shown on Figure 8.3. The baffled-delay type exchanger would be similar to standard shell and tube type heat exchangers except that the tubes would be packed with charcoal. One 47-hr charcoal bed would consist of two half-capacity vessels in series. The first vessel would be designated as Region I, whereas Regions II and III would be common to the second vessel. These regions were aribitrarily chosen in order to obtain better estimates of average sectional heat decay rates.

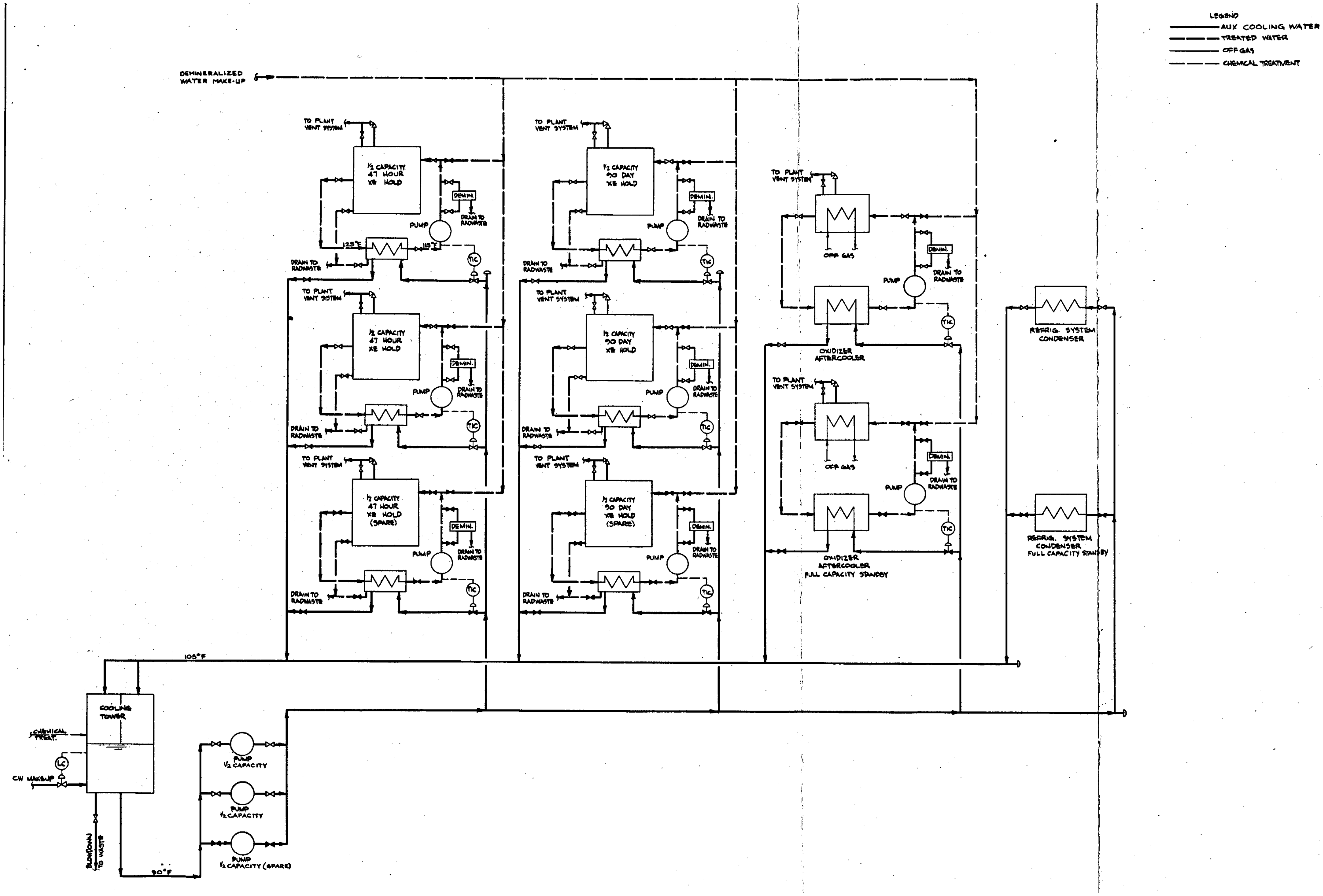


Figure 8.3: Aux Cooling Water System - Flow Diagram.

Because of the higher heat load in Region I, the first vessel would consist of 1-1/2 in. tubes, whereas the second vessel (Regions II and III) would consist of 2 in. tubes. It should be noted that, within limits, the average charcoal temperature can be adjusted by the pipe diameter and heat removal capability. Due to the complex interaction of variables, however, the optimum system would not necessarily be the one with the smallest mass of charcoal.

The size of the first vessel is estimated to be 22 ft long by 8 ft in diameter with approximately 1000 tubes. The mass of charcoal is estimated to be 8400 lb and the length of tubing is 19,000 ft of 1-1/2 in. size. The second vessel is estimated to be 21 ft long by 8 ft in diameter with approximately 1000 tubes. The mass of charcoal is estimated to be 12,700 lb and the length of tubing is 17,000 ft of 2 in. size.

There would be six vessels, each pair capable of handling half capacity with one pair of vessels acting as a standby in the event of operational problems. The vessels are oriented vertically.

A minimum of two containment barriers are provided to guard against leakage of the radioactive fission gas into areas which would be hazardous to personnel. The cooling water heat exchanger capacity is 2.7 kW which is 30 percent over the maximum estimated heat load.

The 47-hr charcoal holdup interval was based on a 97 percent attenuation of xenon-135 concentration of the outlet purge gas relative to the inlet. This criterion was arbitrarily chosen. It appears that this delay, time could be substantially reduced without affecting the poison fraction of the reactor. Calculations have been performed that indicate a 16 hour holdup would provide adequate decay to allow satisfactory stripping of the xenon from the fuel salt by bubble circulation. In Task II a trade-off study will be performed to determine the optimum decay period for both the bubble reinjection decay period and the clean reuse decay period.

Design Criteria for the 90-Day Long-Delay Charcoal Beds

1) Holdup time for xenon is 90 days.

2) The heat load is 0.25 MW. The average heat load is 2 W/min holdup time.

3) The physical properties of the charcoal are the same as those noted in the description of the 47-hr xenon holdup system, Section 3-C.

4) The gas flow rate is 2 scfm at an inlet pressure of 5 psig. The design Δp is 5 psi.

5) The gas composition is 99.9 percent helium, with trace quantities of contaminants. Since noble gas daughters will be deposited on the charcoal during operation, there will be gradual reduction on the effectiveness of the charcoal. About 30 percent spare capacity is provided to offset this loss in effectiveness.

6) The heat will be transferred to cooling water. The average temperature of the charcoal is 125°F.

Description of 90-Day Long-Delay Charcoal Beds

At the outlet of the 47-hr xenon holdup system, the off-gas flow is split into two streams, as shown in Figure 8.2. One stream of 9 cfm is returned to the primary system by the way of the bubble generator and the other stream, of 2 cfm, is fed to the long-delay charcoal beds. The assumed design residence time is somewhat arbitrary since whatever load is not handled by the long-delay bed must be dissipated by the gas cleanup system. The incentive, however, is to handle as much load as possible with the long-delay bed, since its construction and operation is more simple than that of the gas cleanup system.

The size of the long-delay bed was estimated using a method similar to that used for the 47-hr xenon holdup charcoal bed. However, in this case, it is contemplated to use helically wound tubing within or external to the charcoal packed vessel. Each long-delay bed would consist of two vessels in series and oriented vertically. There would be six vessels, each pair capable of handling half capacity with one pair of vessels acting as a standby in the event of operational problems.

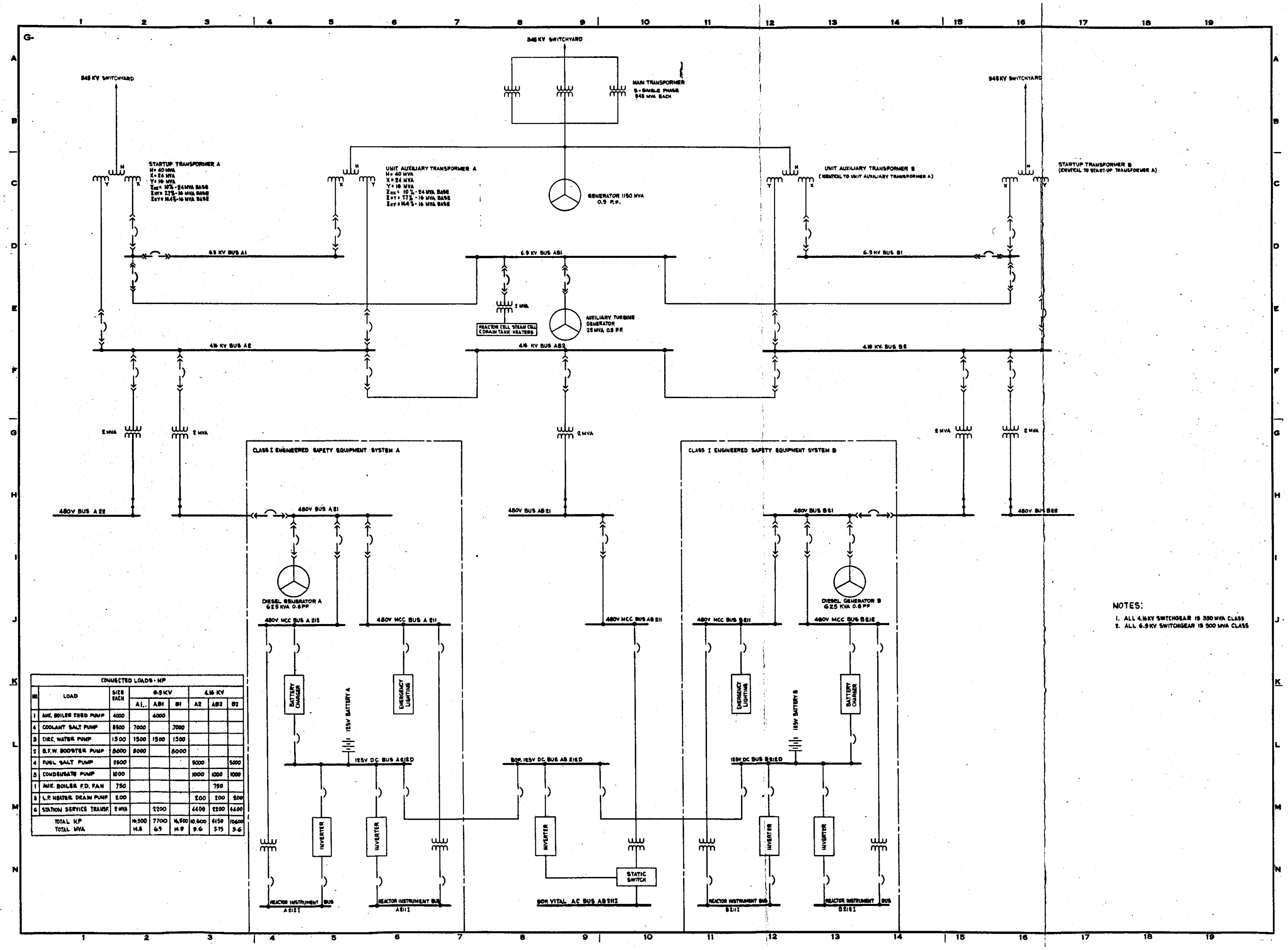
The size of each vessel is estimated to be 22 ft long by 5 ft in diameter. The mass of charcoal in each vessel is estimated to be 17,800 lb. The average charcoal temperature is 125°F with the decay heat transferred to the cooling water system. Thirty percent spare capacity is provided and any unit may be isolated from the rest of the system.

Chapter 9

General Description of Electrical Distribution System

The on-site electrical distribution system is designed to provide reliable sources of electrical power during all modes of operation and shutdown conditions. Electrical equipment related to normal operation of the power station will be connected to their respective normal electrical buses and the auxiliaries required for safety-features operation will be supplied from redundant emergency buses. This engineered safety loading is small and consists mainly of reactor instrumentation, emergency lighting and control-room air conditioning.

In the event that all off-site power is lost coincident with a nuclear accident, two redundant diesel generator sets will provide on-site emergency power to the engineered safety-feature auxiliaries. Reactor instrumentation will be powered from redundant 120V AC instrument buses which in turn will be supplied by inverters from the station batteries.



CONNECTED LOADS - HP								
NO.	LOAD	SIZE EACH	6.9 KV			4.16 KV		
			A1	AB1	B1	A2	AB2	B2
1	AUX. BOILER FEED PUMP	4000		4000				
4	COOLANT SALT PUMP	3500	7000		7000			
3	CIRC. WATER PUMP	1500	1500	1500	1500			
2	B.F.W. BOOSTER PUMP	8000	8000		8000			
4	FUEL SALT PUMP	2500				5000		5000
3	CONDENSATE PUMP	1000				1000	1000	1000
1	AUX. BOILER F.D. FAN	750					750	
3	L.P. HEATER DRAIN PUMP	200				200	200	200
6	STATION SERVICE TRANSF.	2 MVA		2200		4400	2200	4400
	TOTAL HP		16,500	7700	16,500	10,600	4150	10600
	TOTAL MVA		14.8	6.9	14.8	9.6	3.75	9.6

Figure 9.1: One Line Diagram.

The electrical distribution system is shown on Figure 9.1 and consists mainly of two half-capacity start-up transformers which provide start-up power and full-capacity standby service in the event of auxiliary transformer nonavailability. An auxiliary turbine generator provides power to hold the reactor in a standby condition. Under normal operation, the electrical power it generates can be exported via one of the start-up transformers. Switchgear interrupting capability precludes the simultaneous use of both transformers for this purpose.

In the event of a loss of off-site power, it is planned that the auxiliary turbine generator will be rapidly isolated from the electrical transmission system for any occurrance which would result in a trip of the main station turbine generator. At the same time other selected loads would be dropped, leaving only those which are necessary for shutdown and which are within the capability of the auxiliary turbine generator.

Chapter 10

Fuel-Salt Drain System

10.1 Purpose

The principal purpose of the fuel-salt drain system is to provide a place where the salt can be safely contained and cooled under any accidental or intentional situation. Without impairing the above-mentioned principal function of the drain system, the drain tank can be conveniently used for other purposes, such as a holdup volume for off-gases to allow about a 2-hr decay time before the gasses are processed. This arrangement would reduce the heat load on the off-gas system while at the same time providing assurance that the cooling system is operable and could accomodate a major drain. The internal surfaces of the drain tank, particularly the cooled ones, may also act as sites for deposition of noble metals in the off-gas and will possibly eliminate the need for a particle trap in the off-system. The drain tank also serves usefully as a surge volume to which salt can be continuously overflowed from the primary pump bowl as well as transferring fuel salt to the chemical processing facility for processing independent of reactor operation.

10.2 Design Objectives of Fuel-Salt Drain Tank System

The fuel-salt drain system is to be designed to handle:

a) A maximum heat load of about 40 MWt, if about 7 minutes is allowed for fuel drainage to take place.

b) A maximum fission gas inventory of about 250 megacuries

c) An overflow fuel salt rate to drain tank of 90 gpm at a temperature of approximately 1200°F.

d) An off-gas flow rate of approximately 11 scfm

e) A maximum transient heat release of about 53 MWt which would occur after a sudden salt drain.

f) About 18 MWt during full-load operation.

10.3 Design Objectives of Drain Tank Cooling System

a) It must be able to keep the maximum drain tank temperature well within the safe operating range even under the worst condition of transient heat loads.

b) The system must be reliable, with a minimum of reliance on the electric power supply or operator initiated actions.

c) A double barrier be provided between the tank coolant and the fuel salt so that leakage of the coolant into the salt would be highly improbable.

d) The cooling system should impose a minimal risk for freezing of either the fuel or the cooling system coolant.

10.4 Design Criteria of Drain Tank

a) Vessel to be designed for 40 psig, a wall temperature of 1300°F and meet ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, Class A requirements.

b) Fuel-salt storage volume of approximately 2500 cu. ft.

c) Internals and supports designed for a seismic disturbance equivalent to 0.007g to 0.07g horizontal ground acceleration.

10.5 General Description of Drain Tank with NaK-Cooled Heat Disposal System

In addition to eliminating a relatively tall natural-draft stack, as probably would be required for a salt-cooled primary drain tank system, it is believed a NaK-cooled system would provide a more efficient, dependable and freeze-free emergency cooling system, as well as offering the following improvements.

NaK can be heated to relatively high temperatures and can experience significant radiation fluences without problems of dissociation or high vapor pressure. Its density and viscosity variations with temperature are favorable for natural circulation in the system, thus no auxiliary power or action by the plant operators is required to initiate and maintain circulation. Use of NaK, and placing primary emphasis on radiant heat transfer (which varies as the fourth power of the absolute temperature), accommodates the wide ranges of temperature and heat loads which may be encountered between the normal off-gas heating load and the maximum transient after a sudden salt drain. The NaK is compatible with Croloy or stainless steel and does not require the more expensive Hastelloy-N used in the salt systems. Since NaK can have a eutectic temperature such that it will be liquid at room temperature, no preheating of the NaK circuits prior to filling is required. The only major disadvantage of NaK is its poor compatibility if mixed with the fuel salt. However, the probability of this happening can be greatly reduced by providing an isolation barrier of a suitable salt between the fuel salt and the NaK. For this study sodium fluoraborate was selected as the isolation barrier salt, and a 47 percent NaK composition with a eutectic temperature of about 60°F was selected as the coolant salt.

A simplified flowsheet for the NaK-cooled drain system is shown in Figure 10.1. The drain tank is located below and to the side of the primary circuit such that the fuel salt will drain by gravity whenever the freeze-plug type valve is opened. A small circulation of fuel salt is normally maintained in the drain line between the reactor and the freeze valve to prevent overheating due to stagnant salt.

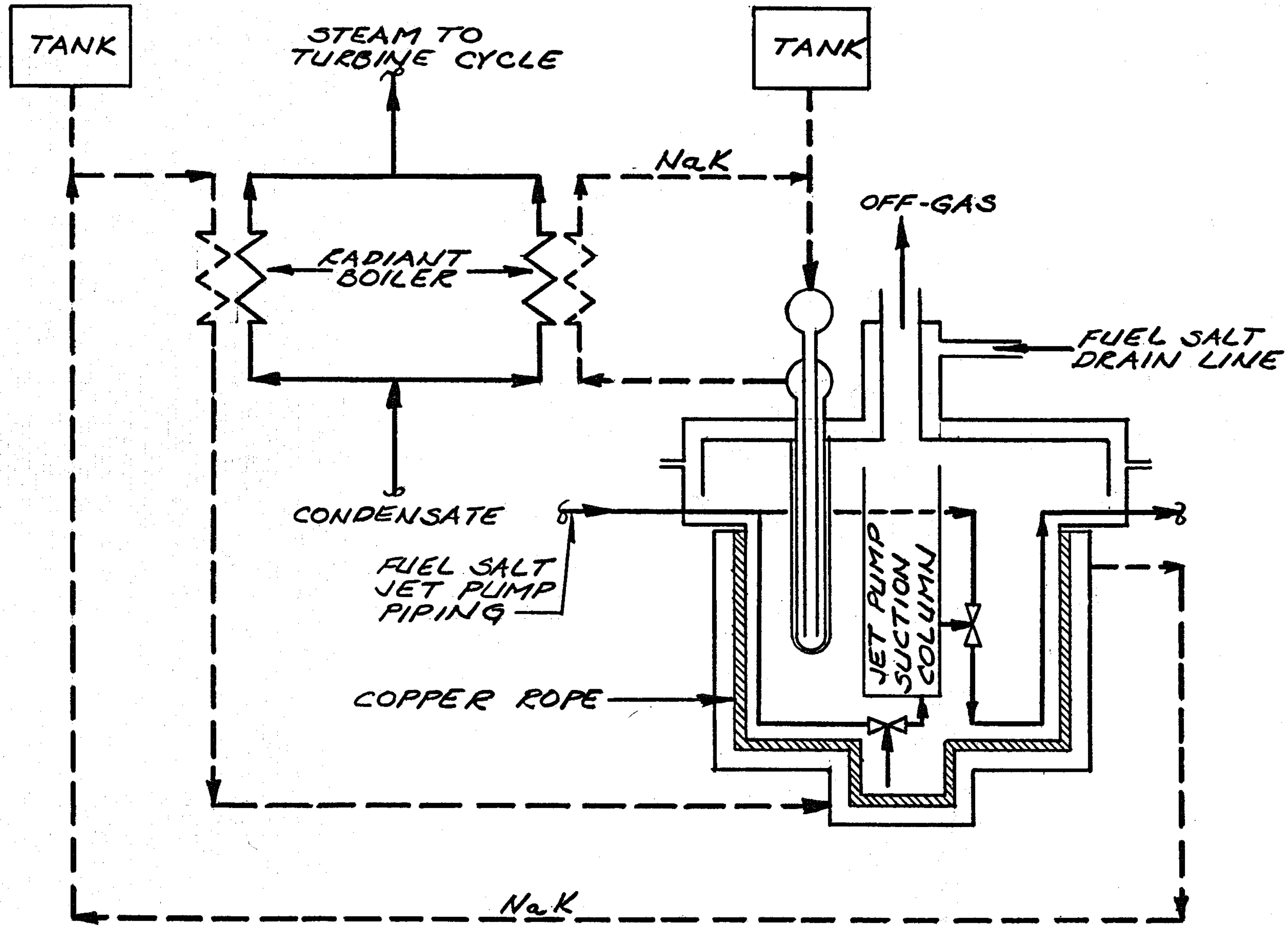


Figure 10.1: Drain Tank Cooling System Schematic.

During normal operation of the reactor a quantity of fuel salt overflows from each circulating pump bowl. The gases stripped from the fuel salt at the gas separator, laden with highly radioactive fission-product gases and particulates, are combined with the overflow salt from the pump bowls in the batch tank (see Section 3.3) before flowing to the drain tank. The overflow gas-salt mixture, which reaches the drain tank at an estimated temperature of about 1100°F, enters the top of the drain tank and is first directed beneath the top head by means of a liner positioned under the head and then downward into the body of the tank.

The drain tank design is for a vessel about 15 ft in diameter by 20 ft high with the bottom head containing a plenum for salt-actuated jet pump operation. The jet pumps transfer fuel salt back into the primary system and to the chemical processing facility. The drain tank is constructed of Hastelloy-N with approximately 1000 Croloy or stainless steel open-ended bayonet tubes and mating thimbles extending vertically downward into the tank, as shown in Figure 10.1 and 10.2. Each thimble in turn is inserted in another thimble, made of Hastelloy-N. The two thimbles are separated from each other by an annulus filled with the barrier salt. The eutectic NaK circulates down the bayonet tube and up inside its mating thimble by natural convection to radiant boilers. A reservoir about 60 ft above the drain tank provides the thermal driving head for natural circulation.

The drain tank is surrounded by essentially two open-topped stainless steel vessels. The inner vessel is filled with tightly packed copper rope, the purpose of which is threefold: to minimize the salt volume which can occupy the annulus in the event a major fuel-salt leak develops in the drain tank, to provide a good conductor of heat to the outer vessel and to act as a gamma shield. The outer vessel provides the necessary cooling by acting as a channel for circulation of the NaK. It is assumed that about 60 percent of the heat generated within the drain tank is in the form of gamma rays, much of which will be absorbed by the vessel walls, the copper rope and by the thimbles and thus be directly transferred into the NaK. Most of the generated heat will then be removed by the cooling thimbles. Heat is transferred from the outer thimble wall to the NaK cooling system by conduction through the salt isolation barrier. Since the outer thimbles and NaK cooling components are not in physical contact, either in the drain tank or in the radiant boilers, a leak in any system is unlikely to contaminate another.

The NaK cooling system is arranged with several autonomous circulating loops so that failure of one circuit would not cause a severe loss of cooling capacity and necessitate an immediate shutdown of the plant. Twelve separate loops, 8 of them tied into the thimble arrangement and 4 of them connected into the bottom plenum of the outer tank, were assumed in this study. An electromagnetic pump (acting as a brake) is installed in each of the NaK circuits to retard or stop the NaK natural circulation as necessary to protect against freezing of the fuel salt in the drain tank. This arrangement is particularly advantageous during start-up or partial load operation.

The radiant boilers have NaK heated inner surfaces separated from the water-cooled outer surfaces by an inert-gas- filled space. Heat exchange is entirely by radiation. Heat rejected from the NaK coolant loops is converted to low-pressure steam for either reclamation by introduction into the steam power system or for rejection to the environment by means of a water-cooled condensing basin as discussed in greater detail in Section 5.5.7.

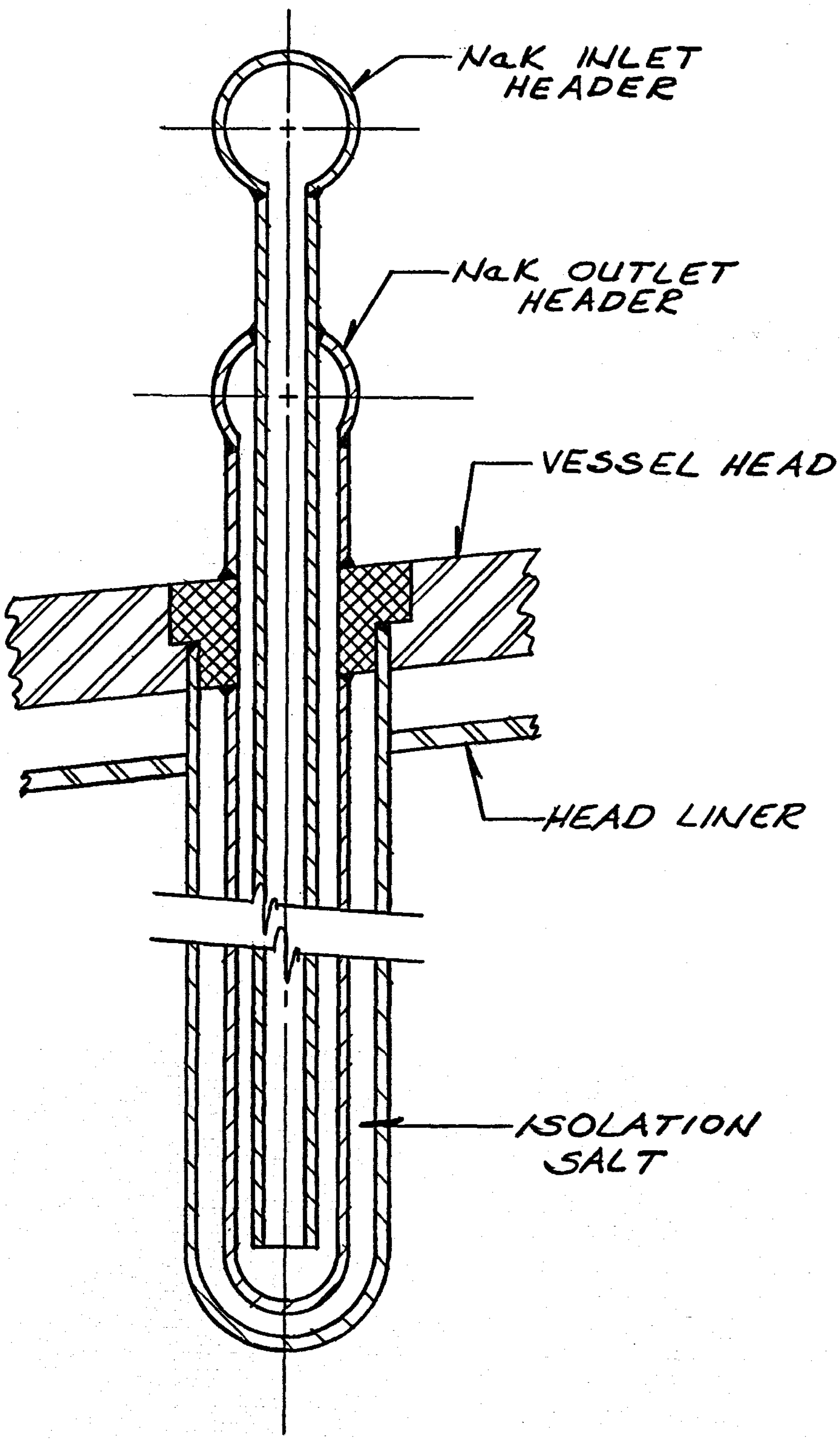


Figure 10.2: Drain Tank Thimble Detail.

Appendix A

Stress Analysis

A.1 Background Investigation

The investigations which were undertaken in determining the final layout of the primary salt piping evolved as follows; basically, the system had to meet two criteria:

1) For functional and economic reasons, the overall length of primary piping had to be maintained at a predetermined minimum.

2) Stresses had to be within limits prescribed by the applicable Codes and Code cases for all anticipated conditions.

Two layouts were considered initially; one with the reactor, pump and exchanger in line and the other with the major equipment forming an approximate right triangle in plan (the one finally selected).

Initial computer runs were made for the normal, shutdown, scram and one-pump-out cases with the triangular layout using wall thicknesses of 1-1/2 in., 1-1/4 in. and 3/8 in. for the outlet piping from the reactor, pump and exchanger, respectively. These wall thicknesses were based on examination of previous studies.

At this time two horizontal trusses or lateral equipment supports against earthquake were contemplated. The material for the trusses was to be such that they would exhibit the same coefficient of thermal expansion as the piping. Despite this factor which tended to minimize thermal constraints, the piping reactions on the truss were in excess of 165,000 lb. Furthermore, excessive stresses existed in the piping.

The wall thicknesses were reevaluated by calculating them for pressure and allowing for the existence of thermal, seismic and other stresses. This resulted in 1/2 in. for the reactor outlet, 1 in. for the pump outlet and 3/4 in. for the exchanger outlet. This latter increase (3/8 in. to 3/4 in.) was an attempt to "balance" the system which appeared overstiff in the upper piping section.

Using these new thicknesses the normal, shutdown, scram and one-pump-out cases were run for both the in-line and triangular layout. Reactions on the truss were reduced to 65,000 lb with a lesser degree of overstress in the piping. The in-line layout showed higher (145,0001b) reactions on the truss and further piping overstress.

Analysis of these calculations made at this time indicated a preference for the triangular layout and an "overstiff" upper piping section for both layouts. Additionally, it was becoming apparent that the system was sensitive to minor changes in temperature and geometry.

The elevation of the reactor relative to the pump and exchanger was varied, while keeping the overall volume of piping constant, and the four cases plus seismic effects reevaulated for both the in-line and triangular cases. A position for the reactor was found which minimized reactions of piping on the trusses and generally gave the best stress results.

At this point, the triangular layout appeared to be superior from a reaction and stress viewpoint as well as having a minimum overall layout configuration. Further studies were therefore directed toward the triangular layout.

All of the above studies neglected the growth of the reactor and exchanger supports. With the reactor now optimally positioned, studies were made to evaluate these factors with the reactor cell at temperatures from 950°F to 1100°F since holding the cell temperature to an exact temperature of 1000°F would be virtually impossible—and the system was so sensitive to these minor differences.

This additional, seeming minor constraint, on the "tight" layout produced high stresses for certain cases and lead to the following conclusions:

1) The one-pump-out case would be taken care of by the use of pump spares and bypass lines within the secondary salt coolant loop.

2) The reactor cell temperature would be limited to 1000°F minimum.

With these restrictions, the piping system was within tolerable stress limits for all combinations of loading conditions. However, the bottom of the reactor support was 1 ft 9 in. above the bottom of the exchanger support. This complicated the layout of the cooling plenum beneath the reactor cell.

When the system was rerun for the applicable cases with the reactor support at the same level as the exchanger support, resultant stresses were in excess of the allowable for some cases.

The pipe from the pump to the reactor was increased by approximately 7 in., and length of the pipe from the exchanger to the reactor was increased accordingly. At the same time, it was thought desirable to use a side outlet from the exchanger to facilitate drainage of the primary system rather than use the bottom elbow outlet as originally designed. The vertical distance between the inlet and outlet on the reactor was also revised to 14 ft 9 in from the 13 ft 9 in originally used in the calculations to account for a change in reactor head design. Appropriate vertical piping length changes were made to suit.

All cases were rerun to ascertain the effect of these changes. The results indicated that the stresses were reduced to within allowable values of the code. The feasibility of the system, from the viewpoint of the primary salt loop, was thus amply demonstrated.

It should be noted, however, that the system remains quite sensitive to minor changes and that more than ordinary care must be taken with fabrication and erection. In addition, very minor modifications which on an ordinary plant might be undertaken without extensive study must be investigated carefully in order not to induce an unanticipated overstress.

A.2 Basic Assumptions

a) Limit fuel salt volume to minimum permitted by reasonable pressure drop and required piping flexibility.

b) Reactor vessel wall temperature to be maintained at 1050°F.

c) Ambient temperature of reactor cell is 1000°F minimum, with possible temperature variations of 1100°F.

d) The piping system design pressures are as follows (Figure A.1):

Points 4-7	75 psig
Points 9-10	250 psig
Points 15-20	125 psig

e) Equipment restraint members have the same coefficient of expansion as that of the piping.

f) Localized stress raisers such as welds and changes in wall thickness at vessel connections have not been taken into account.

g) Design basis earthquake equivalent to accelerations of 0.25g horizontally, and 0.16g vertically acting on the piping.

h) The piping contains thermal sleeves to minimize thermal transients.

A.3 General Description

The piping was analyzed for three operating conditions, namely: a) normal operation, b) shutdown, and c) scram. The temperature conditions associated with these modes of operation are all shown in the figure. The loss of a secondary coolant salt pump was analyzed as a separate case. The results of this latter analysis indicated that piping stresses considerably above allowable would be introduced into the primary piping system. Therefore, for this study it is assumed that provisions will be available whereby the loss of a secondary coolant salt pump will not affect primary piping stresses. Such provisions could consist of pump spares and bypass lines within the secondary coolant salt loop.

The piping was analyzed for conformance to equations 9 and 10 of Section 1-705 of the Code for Pressure Piping, ANSI B31.7, Nuclear Power Piping. The analysis was made on an elastic basis using the basic stresses for Hastelloy-N per Code Case 1315-3 dated April 25, 1968 and the appropriate paragraphs of Code Case 1331-4 dated August 15, 1967.

Using the equipment and piping temperatures shown on Figure A.1, the analyses were made for a reactor cell temperature of 1000°F with possible variations in cell temperature of plus or minus 50°F.

All points of the piping system meet the requirements for primary membrane stress intensity (equation 9). Under operating conditions the system meets the requirements for primary plus secondary stress intensity range (equation 10) with the reactor cell at 1000°F except for a slight overstress of 346 psi at point 10.

With the reactor cell at 950°F, an overstress of 1000-2000 psi exists at points 5, 9, 901 and 10 under the operating condition. Additional analyses under "Simplified Elastic-Plastic Discontinuity Analysis," may show these points to be within tolerable limits without major piping changes. However, rather than subject the system to further analyses at this time, the reactor cell environment will be limited in this report to a minimum temperature of 1000°F.

One fact revealed by the piping analyses performed to date is the piping system sensitivity to small displacements. For example, small changes (one to two ft) in the relative heights of supporting skirts under the reactor and the primary heat exchanger cause sizable changes in the piping stresses. Similarly, fluctuations in reactor cell temperatures which effect component skirt and horizontal restraint expansions act in a like manner. Satisfactory conformance to stress intensity limits indicated herein are therefore predicated upon the assumed shell temperatures of the major equipment. Final designs must be checked against actual shell temperatures obtained by a more detailed study.

Further, the deflection sensitivity of the system dictates that care must be taken with fabrication procedures to minimize "locked in" erection stresses.

A thermal stress analysis was conducted to investigate the maximum thermal transients that might occur in the primary piping system. The transient thermal stress conditions were, of necessity, established for the worst possible cases — e.g., (1) assuming that the changing of conditions between operating (1300°F) and scram (1050°F) from points 9 to 10 was almost instantaneous (approximately 10 sec); (2) assuming that upon the loss of a secondary coolant pump, the reactor fuel salt inlet temperature would rise from 1050°F to 1300°F in approximately 30 sec.

The computer program employed, uses input information describing the geometry, thermal properties, and boundary conditions for an orthotropic or axisymmetric body and calculates the transient temperature field resulting from specified heat sources on a finite element formulation basis. The temperature distributions so generated were used to calculate the absolute ΔT temperature stresses in accordance with equation 10.

For Case 1, an additional thermal stress of approximately 23,000 psi occurred after 14 sec between points 9 and 10. For Case 2, a maximum thermal stress of approximately 20,000 psi occurred after 31 sec between points 15 and 20. Stresses of these magnitudes, superimposed on the other stresses to be considered for equation 10, would far exceed the allowable values for primary plus secondary stress intensity range of 3 Sm. Based on these conclusions, a thermal sleeve design was selected and analyzed using the same program as previously described, but with a few modifications. The thermal stress analysis was performed assuming a bypass flow of 30 gpm contained within a 1/2 in. annulus, the annulus being formed by the I.D. of the primary piping and the O.D. of a 3/8 in. thick thermal sleeve. To keep the fuel-salt inventory constant, the thermal sleeve sizes were limited from 16 to 21 in. in diameter. The 30.5 gpm fuel salt bypass flow was not considered as adding to the fuel-salt inventory.

With the thermal sleeves incorporated in the piping system, the maximum temperature difference across any pipe wall caused by any transient is less than 3 degrees and results in negligible stresses.

A summary of the pressure containing pipe sizes, allowing for thermal sleeves, are as follows:

Points	Sizes	Wall Thickness
4-7	23 in. O.D.	1/2 in.
9-10	19 in. O.D.	1 in.
15-20	18-1/2 in. O.D.	3/4 in.

The above pipe wall thicknesses have been shown to be adequate for the conditions and analyses cited above, but may be modified by future optimization studies.

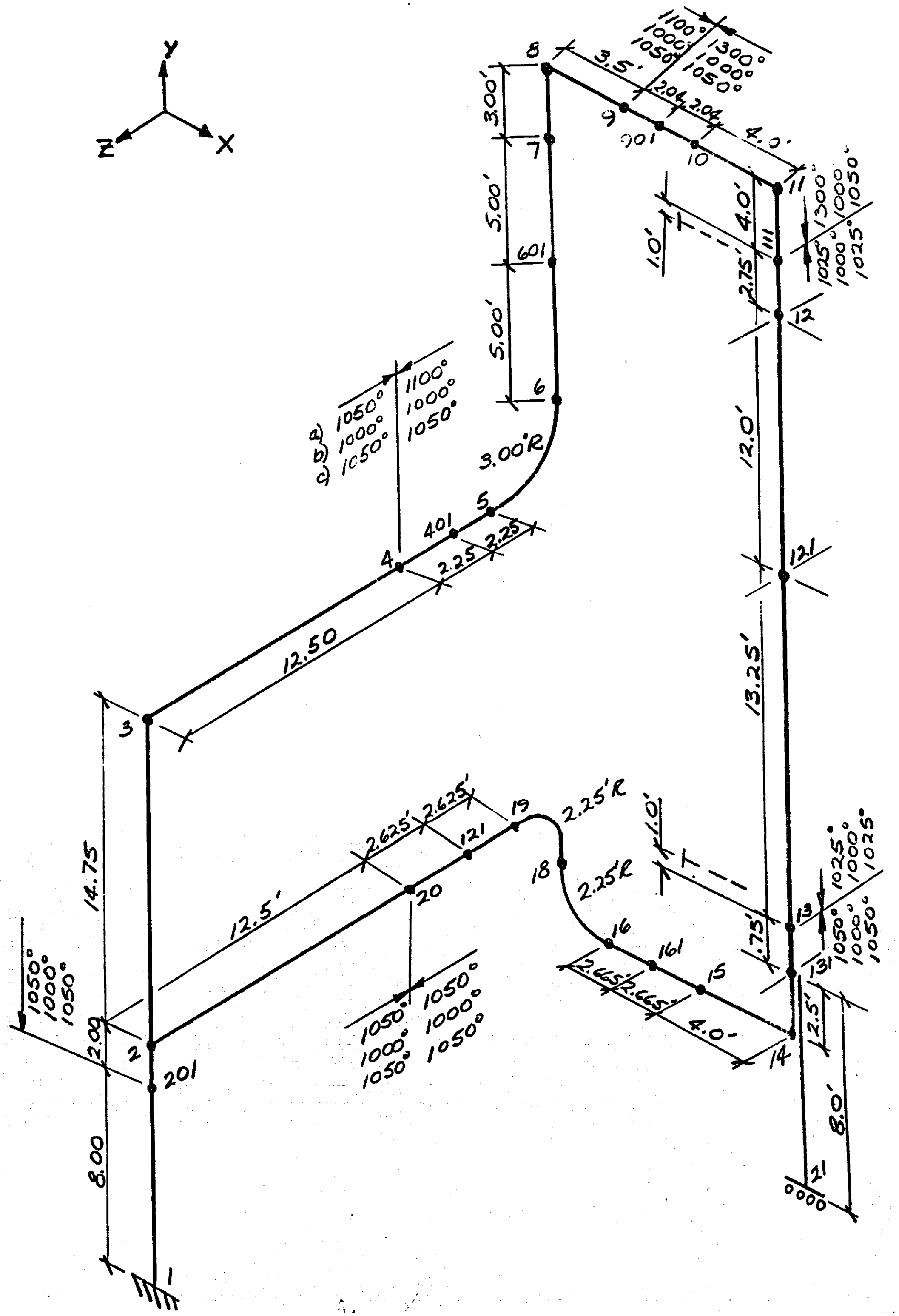


Figure A.1: Primary Piping Isometric.

Appendix B

Number of Steam Generators and Reheaters

A study was conducted to assist the system designer to determine the optimum number of steam generators and repeaters in the system. For the steam generator, 4, 8, and 16 units were sized, the pressure-drop ranges were calculated, and the amount of material was estimated (see Figure B.1). A similar study was performed on the reheater, but 1, 2, 4 and 8 units were considered.

The results of these studies can best be understood if a few basic points are explained. Components are generally designed to utilize the maximum pressure drop allowed by the system's designer, and this design tends to ensure a most compact unit. On this basis, whether the design is for 4 units or 16 units, the allowable pressure drop will be the same. If the tube size and tube spacing are the same, the use of the same pressure drop will ensure identical fluid velocities in all units. Accordingly, if 4 units are designed to the same pressure drop, tube size, and tube spacing as 16 units, the lengths will be equal; the difference will lie in the diameters. Another way of interpreting this is to imagine 4 units, each with 1000 tubes. These would obviously be equivalent to 16 units, each with 250 tubes of identical size and arrangement. Although the optimum selection may be slightly different for each number of units, the studies showed that this analysis gives an accurate picture of the problem. The conclusions are as follows:

1) The steam generator can be built and shipped in 4, 8, or 16 units. The amount of material required is not dependent on the number of units and is approximately constant. The major disadvantage of 4 units is that the tubesheets tend to be very thick (–25 inches). The major disadvantage of 16 units is that the total manufacturing costs will be slightly greater due to extra numbers of components. Also the larger number of units (8 or 16) will cause extra piping problems.

2) The reheater can be built and shipped in 1, 2, 4, or 8 units; the same conclusions that apply to the steam generator apply to the reheater.

3) After discussions with Ebasco on the results of the study, it was mutually agreed that it became

basically a choice for the system designer, and consequently 4 units were chosen for the steam generator and the reheater.

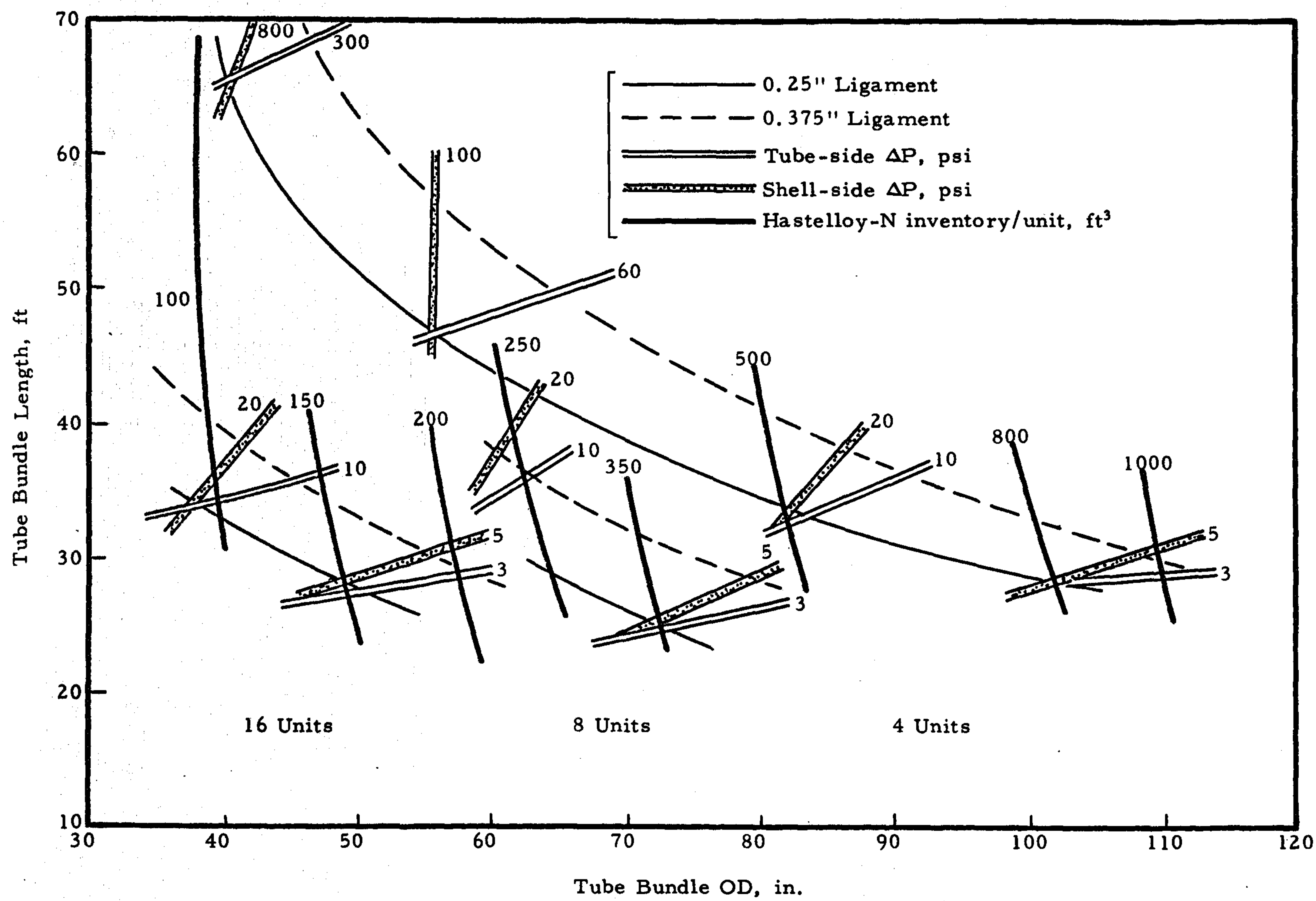


Figure B.1: Number of Steam Generators, Parametric Curves.

Appendix C

Flow Stability of Supercritical Fluid in Steam Generators

One of the major problems in steam generators is the flow instability of the heated fluid. There are two major types—dynamic instability and flow reversal. Dynamic instability is characterized by rapid pressure and velocity fluctuations and can take place over a wide range of loads. Flow reversal may occur at very low loads and result in some tubes having fluid flowing opposite to the desired direction.

The dynamic stability characteristics of a steam generator can only be evaluated by using highly complex analysis; however, a preliminary analysis can be made by assuming quasi-steady conditions. The static stability curve is shown by the example in Figure C.1 and is drawn for a given load condition (say, 20 percent load, for example). Upon receipt of operating temperature conditions, the steam generator designer can then calculate the required mass flow to satisfy load and temperature conditions; this is indicated by M_A in the figure. The next step is to find out how the P varies with small changes in mass flow. If the gradient is positive (such as C to D) in the region of M_A, the load, flow, and temperature conditions will most probably be stable. If the gradient is negative (such as B to C), the dynamic instability will probably occur during operation.

Rough calculations of the MSBR units have been made on the static stability aspects. It was assumed that the mass flow was proportional to load or, in other words, a constant ΔT across the unit. The results indicated that the minimum flow would be in the range of 10 to 15 percent to avoid dynamic instability.

Flow reversal generally occurs at very low loads and is caused by the density change in the up-leg of a loop, as shown in Figure **??**. The cold leg and hot leg may be inside the steam generator shell, or the cold leg may be part of the piping system outside the unit. The heated up-leg creates a change in density such that the outlet pressure (P_3) is higher than the inlet pressure (P_1). A situation is created where it is possible for some tubes to have flow in one direction while adjacent tubes have flow in the opposite direction. The solution is to increase the mass flow until the friction a equals or exceeds the pressure gain caused by the density change.

The initial calculations were based on the same assumptions as previously stated; that is, the ΔT across the unit was constant at all loads. It was found that to avoid flow reversal, the minimum mass flow would probably be in the range of 10 to 15 percent of full load.

To avoid these large mass flow requirements the graph in Figure **??** was prepared. This graph shows that flow reversal can be avoided with mass flows of 5 percent if the inlet temperature is increased to 750°F. The reason is that the flow reversal problem is aggravated by high density changes. Since a large portion of the density change occurs between 700 and 750°F (34 lb/ft3 to 12 lb/ft3), the solution is made easier by increasing the inlet temperature to 750°F.

As it has not been checked, it can be stated with a high degree of assurance that, the conclusion of Figure **??** can be applied to the dynamic stability analysis; in other words, the unit will by dynamically stable with smaller mass flows if the temperature range is decreased.

The conclusions are as follows:

1) With an inlet temperature of 700°F and an outlet temperature of 1000°F, the steam generator will need minimum tube-side mass flows of 10 to 15 percent of full flow to ensure freedom from flow reversal and dynamic instability problems.

2) If the inlet temperature is raised to 750°F, the steam generator may be free from flow reversal and stability problems at mass flows as low as 5 percent of full flow.

3) The relatively large variation in outlet temperature does not affect the minimum mass flow requirement for a stable operation because the density variation near the outlet conditions is negligible compared to the inlet conditions.

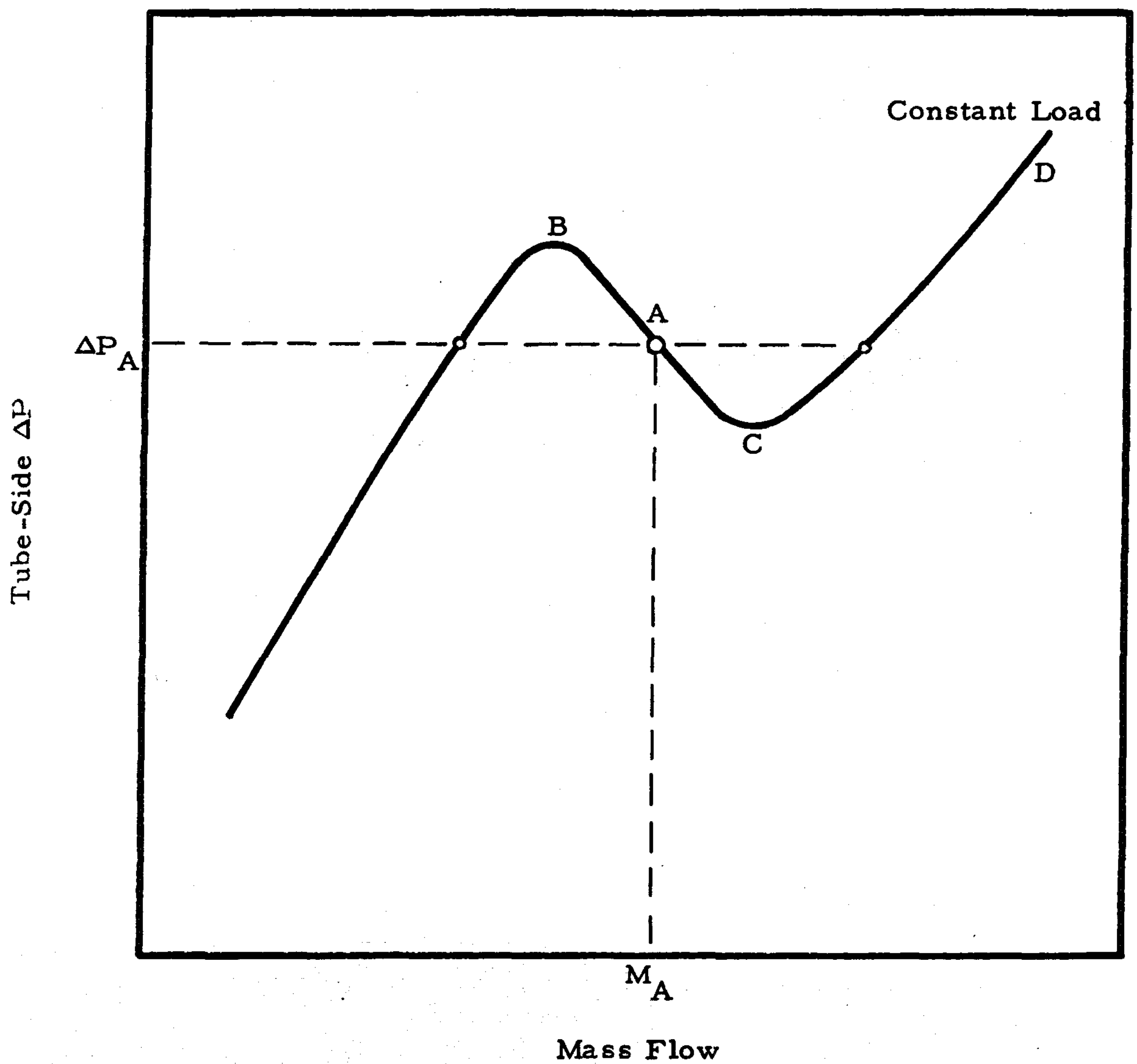


Figure C.1: Characteristic Flow Instability.
If the unit is operated at mass flow corresponding to point A, instability occurs. Any perturbation has the potential to cause oscillation from region B to region C.
Note the multi-valued mass flow of ΔP A This is a characteristic of flow which has a high density change.

Appendix D

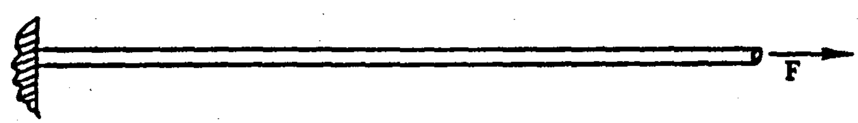
Stress in Tubes Due to Differential Thermal Expansion

1) General

This appendix gives a comparison of the thermal stresses in the types of tubes which are considered for the IHX. These types are straight, sine-wave-bend, hockey-stick, C, and hockey-stick/sine tubes.

All tubes are stressed due to the difference in the average temperatures of tubes and shell. It is assumed that the axial temperature variations in tubes and shell are linear and that the tubes are forced to expand the same amount as the shell would expand with no constraints. In other words, the tubes' stiffness is negligible compared to that of the shell. All types of tubes have the same cross-sectional dimensions.

2) Straight Tube



The formula for the axial force acting in a straight tube is

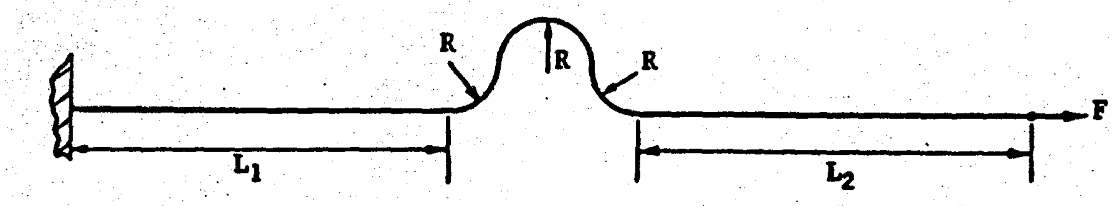
$$F = AE\alpha\Delta T$$

$$\sigma = F/A = E\alpha\Delta T = (25 \times 10^6)(10 \times 10^{-6})(87) = 21,750 \text{ psi}$$

where E = elastic modulus, α = coefficient of thermal expansion, ΔT = temperature difference between shell and tubes, σ = stress, A = cross-sectional area of the tube.

The value of 1.5 Sm at 1300°F is 5250 psi. This value indicates that a straight tube will be thermally overstressed.

3) Sine-Wave-Bend Tubes

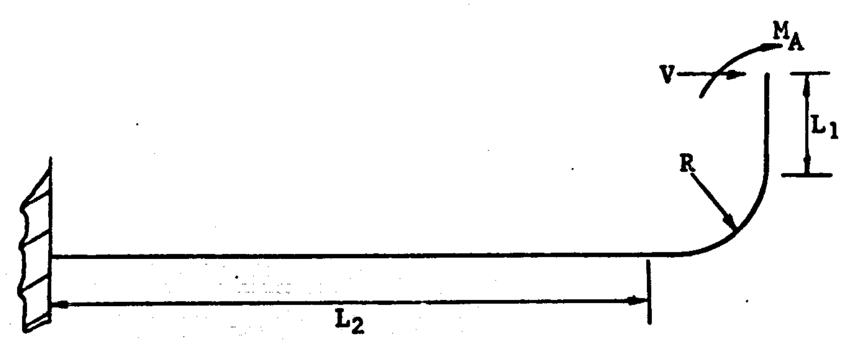


$$F = \frac{E\alpha\Delta T}{\dfrac{L_1 + L_2}{A} + \pi R\left(\dfrac{1+2\nu}{A} + \dfrac{3R^2}{I}\right)}$$

where E = elastic modulus, α = coefficient of thermal expansion, ΔT = temperature difference between shell and tubes, A = cross-sectional area of the tube, I = moment of inertia of the tube, ν = Poisson's raio, L_1, L_2, R = defined in the picture, $L_1 + L_2 + 4R$ = 300 inches.

The maximum bending stress in the tube occurs at the midpoint of the sine wave. For this reason the wave will be located in as cool a region as possible. The maximum bending stress and the P/A stress for different values of R are shown in Figure D.1. By comparing these stresses to 1.5 Sm at 1100°F, it is shown that R should be greater than 4 inches.

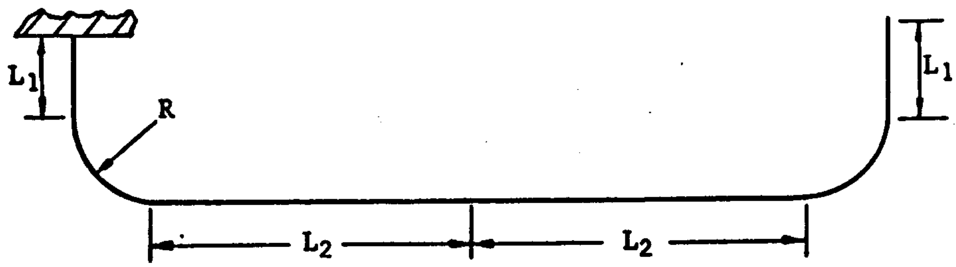
4) Hockey-Stick Tubes



The assumptions are that the forced tube displacement is the same as the free thermal expansion of the shell at point A, and that there is no rotation at A. A force V and moment M are developed at A to satisfy these assumptions. The maximum stress occurs at point A. Graphs of stress versus R, L_1, and L_2 are shown in Figure D.2, D.3, and D.4. The tube dimensions and T are the same as for the sine-wave tube. The stress for a tube with a different OD is obtained by multiplying the stress shown on the graphs by the ratio of the new to the old OD.

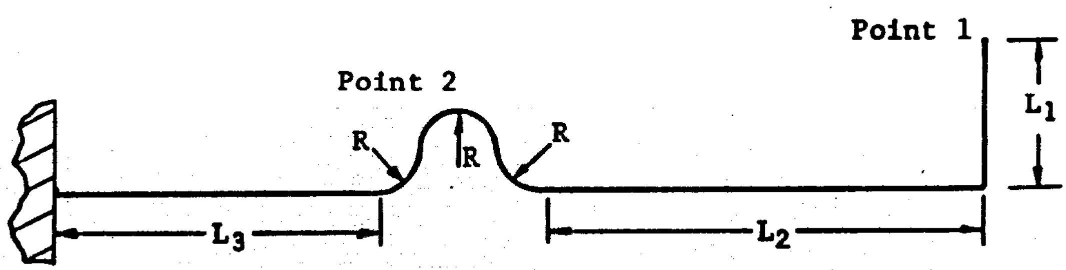
The graphs show that the maximum bending stress varies significantly with R and L_1 and is practically independent of L_1 and R should be greather than 25 in. to have stresses below 1.5 Sm at steady state. This leaves an additional 1.5 Sm for stresses that occur during transients since the total stress intensity range allowable for thermal secondary type stresses is 3 Sm.

5) C Tube



A C tube can be thought of as being made up of two hockey-stick tubes. Each end will be forced to displace about one-half as much as the end of the hockey-stick tube. Figure D.4 shows that the stress in the hockey-stick tube is practically independent of L_2; therefore, the maximum stress developed in the C tube will be about one-half of that developed in the hockey-stick tube. This means that the sum of L_1 and R should be greater than 12 in. to have maximum stresses below 1.5 Sm.

6) Hockey-Stick/Sine-Wave-Bend Tube



The model for the combination hockey-stick/sine-wave-bend tube is shown below; the distance $L_2 + L_3 + 4R$ is 300 inches.

The stress at points 1 and 2 is shown in Figure D.5 and D.6. It is evident that the stress at point 2 is greater for a given combination of R and L_1. Figure D.6 shows that the sum of L_1 and R should be greater than 14 in to have stresses less than 1.5

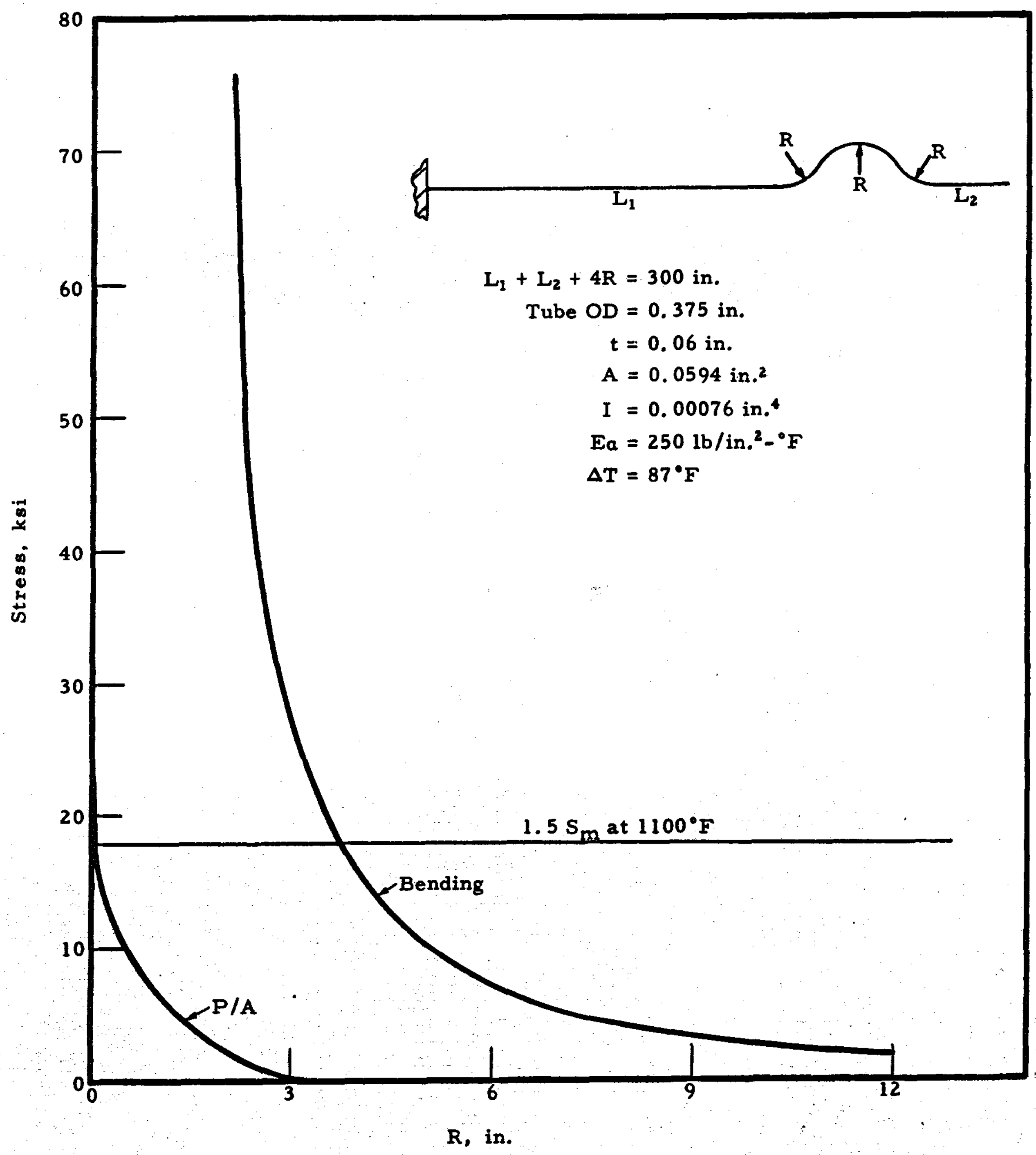


Figure D.1: Stresses in Sine-Wave Tubes vs. Radius of Curvature.

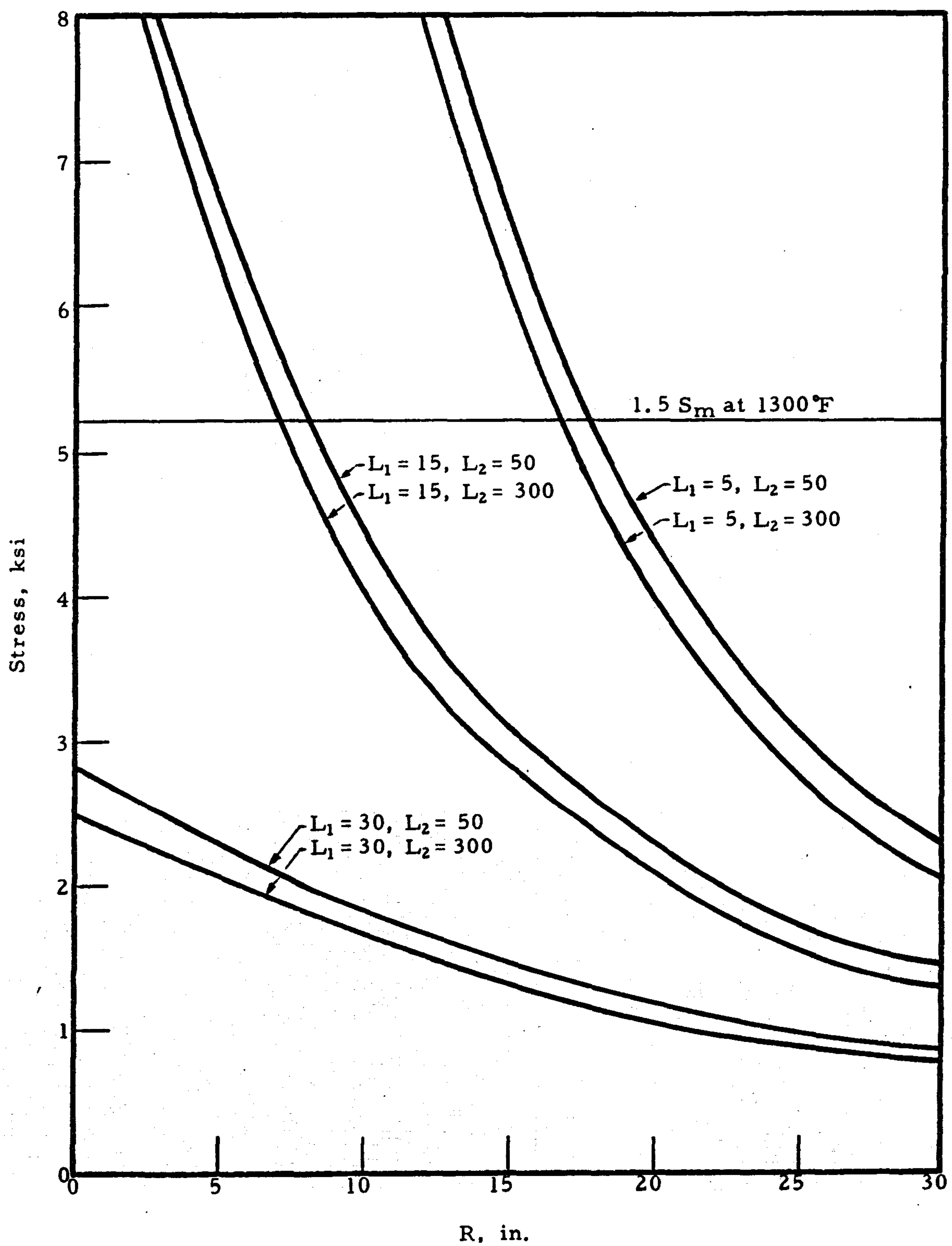


Figure D.2: Stress in Hockey-Stick Tube vs. Radius of Curvature.

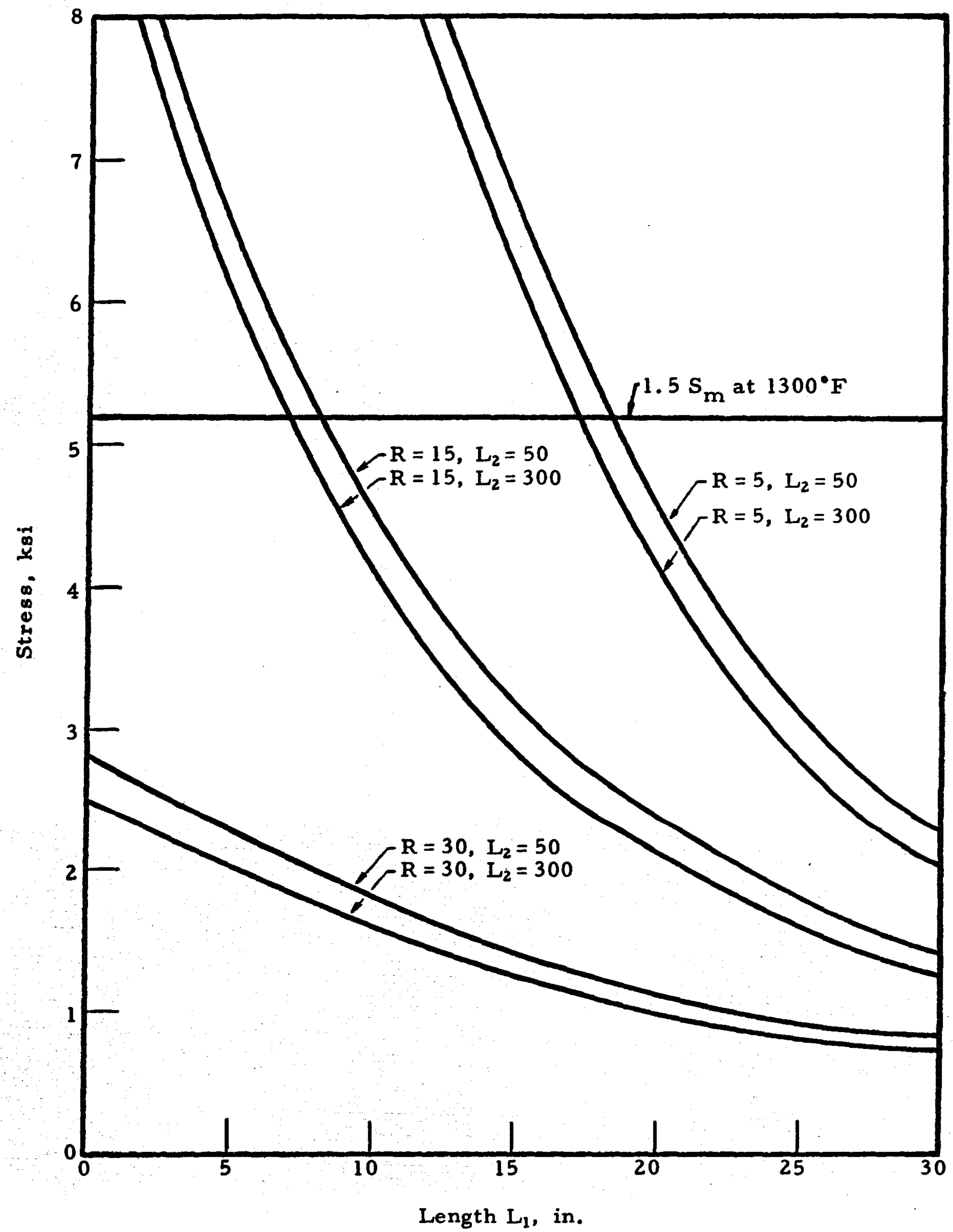


Figure D.3: Stress in Hockey-Stick Tube vs. Length L_1.

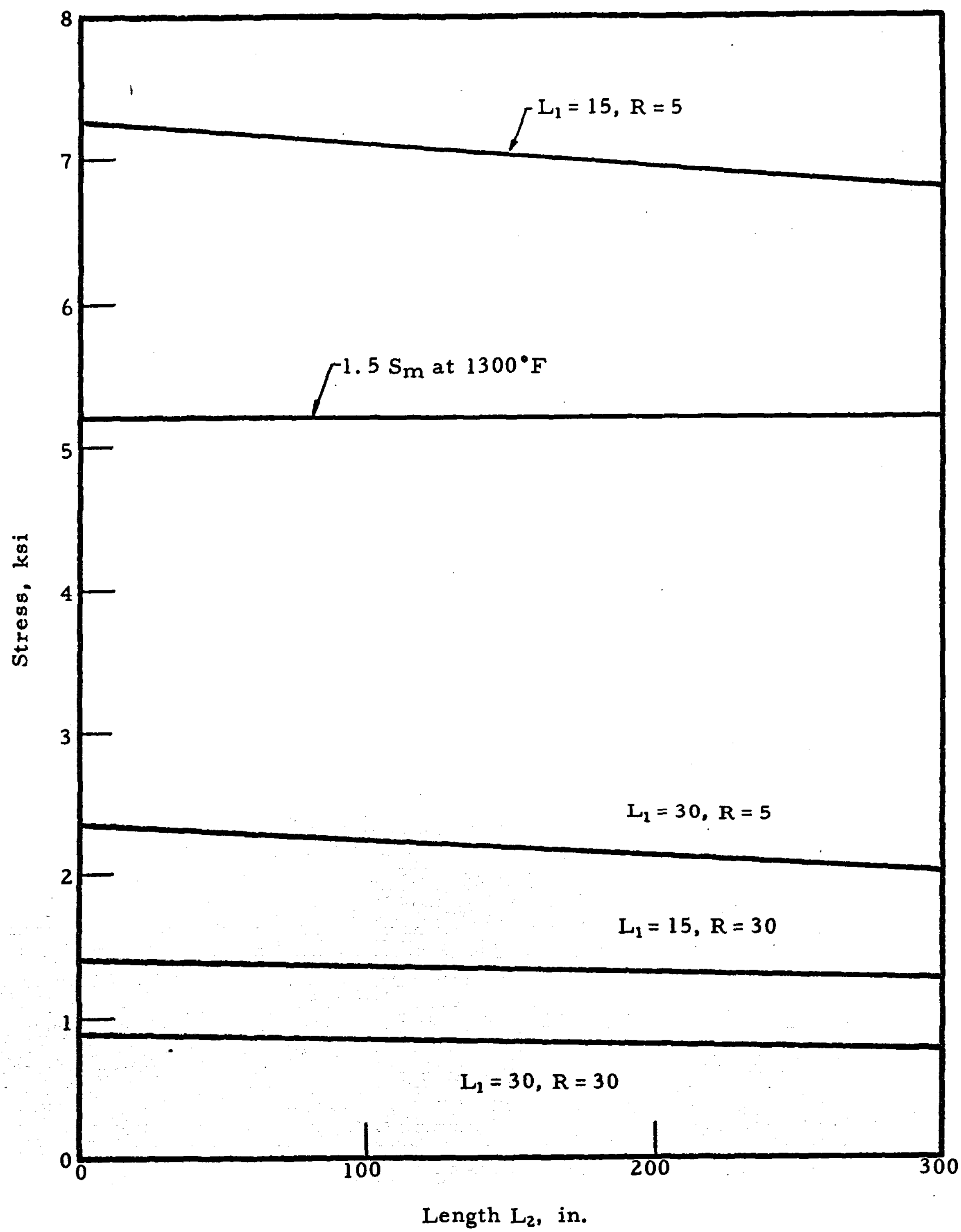


Figure D.4: Stress in Hockey-Stick Tube vs. Length L_2.

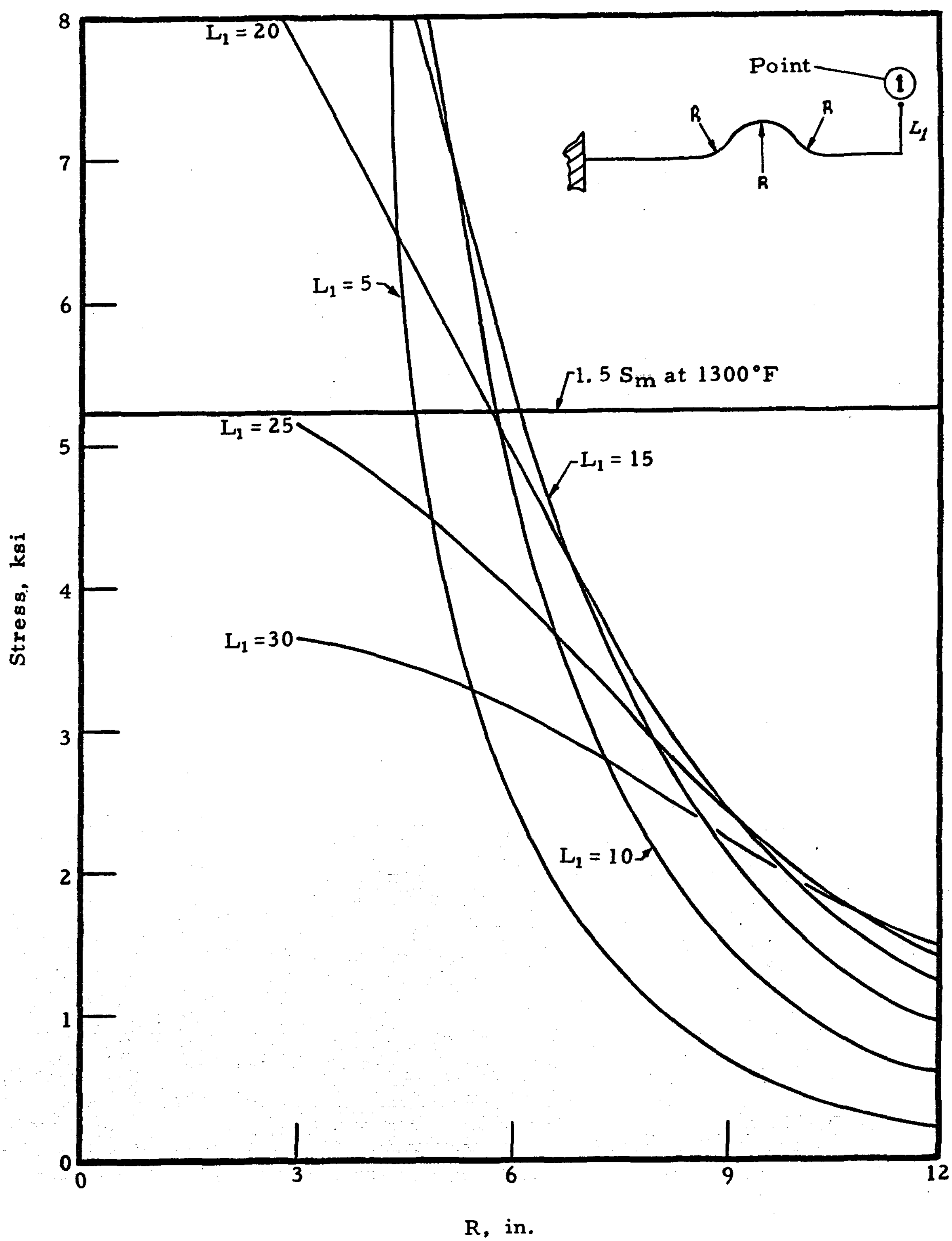


Figure D.5: Stress in Hockey-Stick, Sine-Wave Tube vs. Radius of Curvature, Point 1.

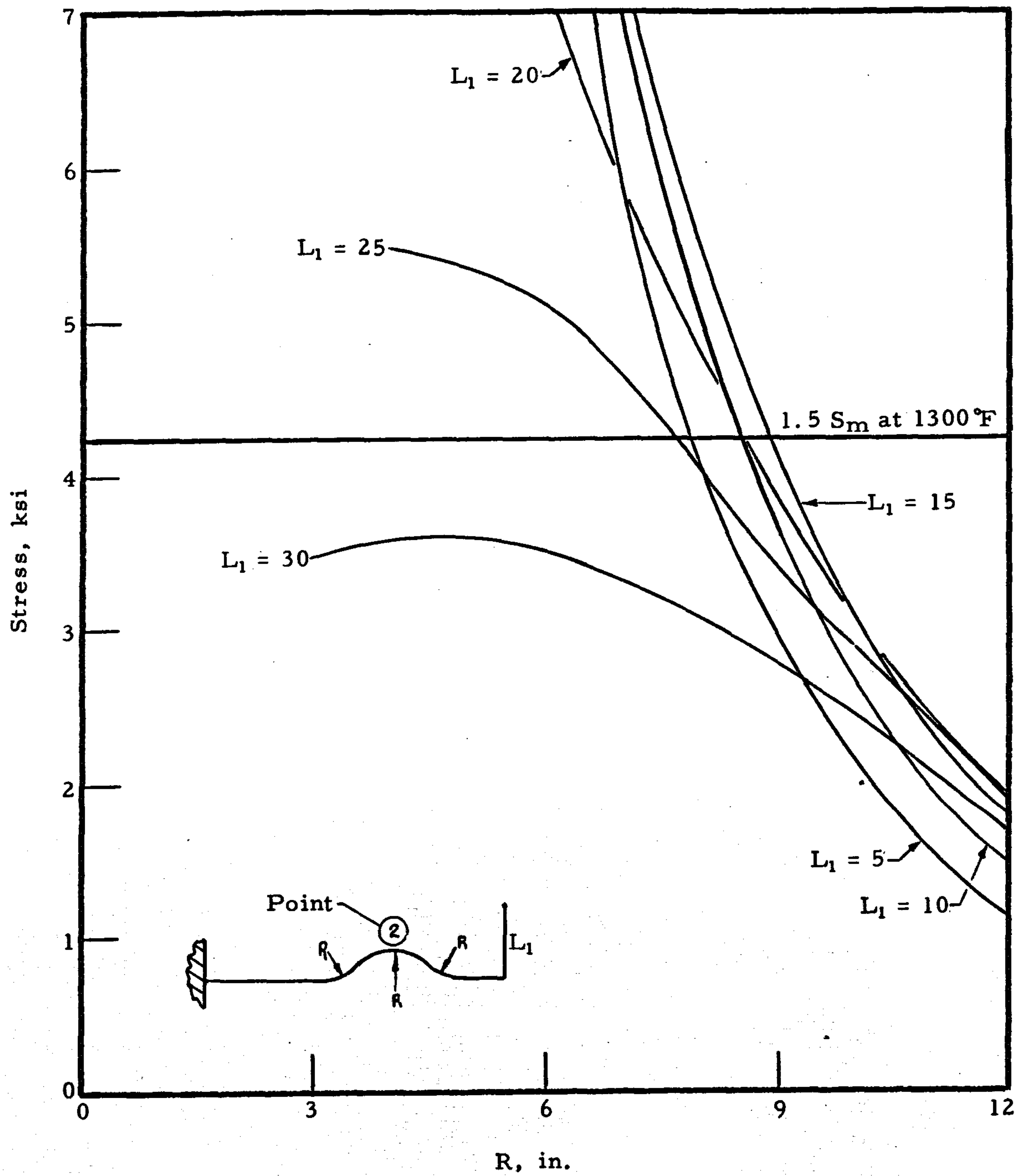


Figure D.6: Stress in Hockey-Stick, Sine-Wave Tube vs. Radius of Curvature, Point 2.

Appendix E

Relevant Sections from ORNL Molten-Salt Reactor Program Reports

E.1 ORNL-4622: Industrial Study of 1000-MWe Molten-Salt Breeder Reactor

M. I. Lundin
J. R. McWherter

Preparations are nearly complete for issuance of a request for proposals for an industrial study of a 1000-MWe MSBR. Internal reviews and approvals of the request for proposal package have been obtained, and the package has been submitted to AEC-RDT for review. A preliminary expression of interest has been received from some 26 industrial firms.

The study, to begin in FY 1971, will consist of several sequential tasks. The first task is the development of a concept for a 1000-MWe reactor plant which will have a chemical processing plant based on information furnished by Oak Ridge National Laboratory. The second task is the evaluation of the effects of various parameters on the power production cost. Other tasks include the study of a modified plant concept having a chemical processing plant proposed by the contractor and the recommendations by the contractor for the molten-salt program research and development effort.

E.2 ORNL-4676: Industrial Study of 1000-MWe Molten-Salt Breeder Reactor

M. I. Lundin
J. R. McWherter

Proposals were solicited from a number of industrial firms to perform design studies of a 1000-MWe molten-salt breeder reactor. An evaluation team visited each group that submitted a proposal. The Ebasco Services group, consisting of Ebasco, Conoco, Babcock and Wilcox, Cabot, Union Carbide, and Byron-Jackson companies, was selected as the one that could most nearly meet our objectives. A subcontract is being negotiated with them.

They will initially develop their concept of a molten-salt breeder reactor plant. Using this concept as a base, trade-off and parametric studies of the nuclear steam supply system, the energy conversion system, and the fuel processing system will be made. After incorporation of the results of these studies in the reference concept, they will estimate the plant capital and fuel-cycle costs. A review of the research and development program will be made. An independent assessment of chemical processing and a safety review of the proposed plant will be conducted. Technical liaison is being furnished by ORNL. All the work will be reported.

E.3 ORNL-4728: MSBR Industrial Design Study

M. I. Lundin
J. R. McWherter

The subcontract[1] negotiated between ORNL and the Ebasco Services group, consisting of Ebasco, Babcock and Wilcox, Byron Jackson, Cabot, Conoco, and Union Carbide Companies, was signed. This subcontract covers the conduct of an industrial design study of a 1000-MWe MSBR plant.

Task I, the selection of a reference conceptual design, is essentially complete. Two progress reports were submitted.

The overall core size is the same as that in the ORNL reference concept, but a slab-type graphite element is proposed. Several of these slabs are held at the top and bottom in a hexagonal array as shown in Figs. 1.10 and 1.11. This assembly is sufficiently small to be handled and replaced as a unit. One hundred fifty-seven such arrays are required for the core. A 2-ft-thick radial graphite reflector surrounds the core. A boron-containing thermal shield separates the reflector from the pressure vessel to reduce the radiation damage resulting from transmutation of boron in the Hastelloy N.

[1] MSR Program Semiannual Progress Report Feb. 28, 1971, ORNL-4676, p. 36.

E.4 ORNL-4782: MSBR Industrial Design Study

M. I. Lundin

The Ebasco MSBR reference concept was completed during this period.

The reactor consists of a 2-in.-thick cylindrical vessel (nominally 22 ft OD x 20 ft high) supported from the bottom. Salt enters through four inlet nozzles at the bottom at 1050°F and exits through four outlet nozzles at the top. The vessel contains 415 tons of graphite which defines salt flow channels in the following regions: core, axial and radial blankets, inlet and outlet salt plena, and axial and radial neutron reflectors. Each region has a specific salt fraction chosen to produce the desired nuclear characteristics of that region. Based on an evaluation of the ORNL reference concept, it was decided to retain the physics characteristic of that concept for task I. This was done by preserving the salt composition, the region salt fractions, and region dimensions specified in the ORNL reference concept (case CC-120).

The Ebasco concept does, however, have two minor variations in the graphite region dimensions:

1. The salt annulus between the radial blanket and the reflector was eliminated. This annulus, whose function was to provide clearance between permanent and replaceable graphite for unit core replacement, is no longer required. In the Ebasco concept, individual graphite assemblies will be replaced on a four-year schedule. Only those assemblies which cannot survive another four-year exposure interval are replaced.
2. The inlet and outlet plena have been extended into the axial blanket regions for improved flow distribution.

Neither of these changes will make an appreciable effect on the nuclear performance of the reactor.

The core and blanket moderator bundles consist of ribbed graphite plates arranged into hexagonal assemblies 15.6 in. across flats.

Fuel salt flows from the reactor into four parallel circuits, each with a salt-circulation pump in the hot leg and an intermediate heat exchanger (IHX) where the heat is transferred to the secondary salt.

The IHX is a vertical sine-wave-bent tube design with a nonremovable tube bundle. Fuel salt enters the top plenum, flows downward through about 7000 tubes (3/8 in. OD), and exits at the bottom. The coolant salt enters at the bottom, flows up in a mixed countercurrent flow, and exits at the top.

The secondary system also has four parallel circuits, each containing one IHX, steam generator, steam reheater, and circulation pump. The pump is in the cold leg to pressurize the IHX sufficiently to force any leakage to be directed into the primary system. The steam generator concept is a supercritical, once-through, helical-coil tube design. Supercritical fluid enters at the top, flows down through annular rows of unheated downcomer tubes, turns, and flows up through the heat transfer zone. The steam flows through 815 tube coils, countercurrent to the coolant salt. The steam reheater is identical to the steam generator except for its size.

The steam system basically consists of a supercritical steam cycle using a tandem-compound turbine generator with reheat and feedwater heaters. Except for the steam generators, reheaters, supercritical feedwater pumps, and preheater-reheater, the system utilizes conventional power plant technology and designs. The feedwater temperature is 700°F to prevent freezing of coolant salt in the steam generator. This high feedwater temperature causes the steam system to differ from a completely conventional supercritical steam system.

The chemical process plant permits the reactor to operate as a breeder by removing ^{233}Pa and certain soluble parasitic neutron absorbers from the fuel salt. It also reconstitutes the salt and returns it to the primary system. The plant flowsheet was developed and supplied by ORNL.

This conceptual design reflects CONOCO's experience and judgment regarding need and location of pumps, valves, surge volumes, drain systems, safety, control system, and spatial layout.

The chemical processing cell is heated to prevent salt freezing. It is of a modular design for replacement of equipment by remote techniques. The upper level contains process equipment; the lower level contains drains and storage tanks. The cooling system uses NaK and is independent of other cooling systems in the reactor building.

The reactor off-gas system removes fission gases, particularly ^{135}Xe and tritium, from the fuel salt. A purge gas (helium) throughput of (nominally) 10 scfm, together with efficient bubble separation from a 10% salt sidestream, will keep the salt void fraction in the core to about 1% (about 0.6% volume-weighed loop average). Based on these conditions, it is speculated that a poison fraction of 0.5% (0.005 neutron absorbed in ^{135}Xe per absorption in fissile isotopes) can be achieved with unsealed graphite.

Reduction of the ^{135}Xe concentration in the purge gas is accomplished primarily by decay during holdup in the drain tank, and the gas is recycled directly to the bubble injector. It is anticipated that some gas cleanup via charcoal adsorption will be required. These charcoal beds consist of coils of charcoal-filled piping submerged in cylindrical water tanks. Xenon is removed from the helium by dynamic sorption. The decay heat (about 2 MW) is removed by forced circulation of water through coolers. The off-gas system will also remove krypton, tritium, and volatile hydrocarbons from the purge gas.

The fuel tank-drain tank system is intended to provide a safe place to store the salt at any time under all conceivable circumstances. It also provides holdup and cooling for the purge gas during normal operation. This tank is located below the reactor cell to permit drainage by gravity. The salt (or purge gas) is cooled by about 1000 bayonet tubes inside thimbles mounted into the tank head. The coolant is NaK, which circulates by natural convection through many redundant external cooling circuits. Fission gas decay heat (about 18 MW) is transferred to the main steam system when in operation. Otherwise, it is transferred to a closed-cycle, boiling-water, heat rejection system.

The reactor cell, chemical plant, off-gas system, drain tank cell, graphite handling equipment, and emergency power generators are located in the reactor building, a rectangular class 1 structure. Seismic supports are provided for the reactor and intermediate heat exchangers in a horizontal plane by a three-tier support structure of Inconel beams. Structural support is provided at the bottom of the reactor and heat exchangers. The reactor building provides containment against release of radioactivity.

E.5 ORNL-4832: MSBR Industrial Design Study

M. I. Lundin

The Task I Report[2] was delivered to ORNL early in the report period. The report described a conceptual design for a 1000-MWe Molten-Salt Breeder Reactor plant, the salient features of which were described in the last semiannual report.[3] Task II was started and is currently in progress. This effort is defined as follows:

1. demonstrate the feasibility of the conceptual design presented in Task I;
2. prepare CSDDs for the design;
3. conduct trade-off and parametric studies to optimize the design;
4. perform the benchmark physics calculations;
5. prepare a cost estimate for the recommended design.

Work is active in each of these five areas.

In the design engineering area, where practicality of the design must be demonstrated, work is currently in progress on the drain tank, the transient analysis of the plant, handling of the reactor vessel top head and of the reflector graphite, structural support of the primary system, and design of a reactor cell structural support system. The present arrangement is shown in Fig. 1.3. The primary system concept has been studied, and a decision has been made to utilize a lined system to protect the plant from thermal shock. This cooled liner also increases the design allowable strength of the reactor vessel metal to 11,000 psi.

Babcock & Wilcox completed a transient analysis of the intermediate heat exchanger. The scram transient specified resulted in a drop in fuel salt inlet temperature of 250°F in two stages: 200°F in 10 sec followed by a 50°F drop in 4 sec. The recommended Task I design withstood this transient. This design utilizes a sine wave tube with a nonremovable tubesheet but having access to the tubesheet for tube plugging.

[2] 1000-MWe Molten Salt Breeder Reactor Conceptual Design Study — Final Report — Task I, Ebasco Services, Inc.

[3] MSR Program Semiannual Progress Report Feb. 29, 1972, ORNL-4782.

E.5.1 Drain Tank

A parametric study is under way to identify and quantify the important design parameters and to permit the selection of a credible design. The design base is a hypothetical steady-state condition in which the fuel salt produces 50 MW while at a uniform temperature of 1300°F. A tentative design based on this condition will be examined under normal steady-state operation and under transients. These conditions may provide bases for refining the design.

The conceptual design uses bayonet tubes mounted in the tank head for cooling the tank contents. For simplicity, reliability, and ease of maintenance, it is desirable to have as few thimbles as practical. To achieve this, the present Ebasco concept utilizes duplex tube thimbles having a mechanical bond. In this concept heat transfer is limited primarily by the salt film coefficient rather than by radiation as in the ORNL concept. Thus, a greater heat flux and far fewer thimbles are possible in the Ebasco concept.

Heat transfer characteristics of fuel salt to cooling thimbles were examined. Figure 1.5 shows the calculated film coefficient as a function of salt and metal temperature based on a free convection model.[4] It can be seen that a coefficient of about 200 BTU/(hr-ft2-°F) can be expected if the cooling surfaces are maintained at 1100°F and if the salt is allowed to reach 1300°F. Being cooler than the bulk, fuel salt near the thimble surface flows downward, becoming turbulent very near the free surface. Because the flow is turbulent over almost the entire thimble length, the film coefficient is constant over the entire length. The thickness of the turbulent boundary layer was calculated based on a vertical flat plate model.[5] The results suggest that it may exceed 1 in. near the bottom of a long thimble. The same model was used to estimate the temperature distribution and the velocity distribution in the boundary layer. It can be seen that most of the temperature rise occurs within half the boundary layer thickness, and the salt velocity can approach several feet per second. These results indicate that natural convection provides a great deal of mixing, so a uniform bulk salt temperature can be assumed for design purposes.

Based on the film coefficient shown in Fig. 1.5, the heat transfer surface required to accommodate 50 MW was determined. If the cooling surfaces operate at 1100°F (average) and the bulk salt is permitted to approach 1300°F, it can be seen that an area of 4200 ft2 is required. The heat transfer area together with the fixed heat load determine the heat flux and, hence, the temperature drop between the thimble surface and the flowing NaK. Figure 1.11 shows calculated NaK temperatures for various choices of salt and salt temperatures. For 1300°F salt and 1100°F metal the NaK temperature would have to be about 960°F. This result is based on bare thimbles, that is, no fins. However, if vertical fins are put on the thimbles so that the number of thimbles can be reduced, the heat flux will be greater and the NaK temperature will have to be substantially lower.

Because heat transfer through the thimbles is limited primarily by the salt film coefficient, fins greatly improve the heat-removal capacity of a thimble. A standard analyis of fin efficiency shows

[4] M. Jacob, Heat Transfer, Vol. I, Wiley, 1949, p. 529.

[5] E. R. G. Eckert and T. W. Jackson, NACA Technical Notes 2207, October 1950.

that a fin height of 1/2 in. is near optimum. Calculations show that 1/8-in.-thick fins spaced 1/2 in. apart increase the effective heat-removal capacity of a thimble by approximately 50%.

The temperature rise of the NaK as it flows down and up a thimble was determined by constructing two heat balances on a thimble segment; one for downflow and one for upflow. Based on a uniform heaf flux Q, the solutions to the two differential equations are presented and plotted in Fig. 1.12. It can be seen that an approximately uniform thimble temperature can be achieved by choosing design parameters such that the NaK temperature at the bottom is equal to that of the upflow at the top (i.e., at the salt-free surface). Equal temperatures (top and bottom) can be achieved by choosing design parameters such that

$$2\pi r_D U_D N H \Delta T / Q = 2$$

where ΔT is the overall NaK temperature rise (see nomenclature). Assuming that the heat-rejection system does not impose additional constraints on the drain tank design, the problem is now reduced to one of choosing design parameters which satisfy all the previous relationships. It is desired, however, for the number of thimbles N and their height H to be as low as practical and for ΔT to be as large as practical. The latter is to provide adequate driving head with a minimum elevation differential between heat source and sink. Thus, in the previous equation the down-comer perimeter $2\pi r_D$ and the overall heat transfer coefficient U_D between NaK upflow and downflow are the only parameters which can be arbitrarily adjusted to satisfy the equality. Current efforts are directed toward determining the practical limits of each parameter so that a definitive choice can be made.

E.5.2 Physics Calculations

As part of the industrial MSBR design study, Ebasco is performing physics calculations to provide an independent check on the breeding ratio reported[6] by ORNL for the reference design. To attain a high level of independence Ebasco is using different sources of data, different computer codes, and different mathematical treatment from that used by ORNL. Specifically, Ebasco is using ENDF/B data, whereas ORNL used the XSDRN library; Ebasco is using the 3D Monte Carlo code ESP for generating multigroup cross sections, whereas ORNL used the 1D-Sn code XSDRN; fmally, Ebasco is using the multigroup, multidimensional neutron diffusion code CITATION, whereas ORNL used ROD, a multigroup neutron diffusion code which synthesizes two dimensions via a buckling iteration procedure. This system of codes offers not only a high level of independence but rigorous mathematical treatment as well. Because it does not have the capability to search for the equilibrium salt composition, it is inappropriate for routine MSR calculations. But for a reactor having a well-defined composition, this limitation does not constitute a handicap.

[6]Conceptual Design Study of a Single Fluid Molten-Salt Breeder Reactor, ORNL-4541 (June 1971).

E.5.3 Chemical Processing

Continental Oil Company has developed engineered process diagrams and mechanical arrangement drawings for the chemical processing flowsheet supplied by ORNL. This work represents the reduction of the ORNL flowsheet to a practical engineering design in which attention has been given to the details of cooling, heating, shielding, construction, remote maintenance, safety, recovery from accidents and process interruptions, and replacement of major process components. A systematic attempt has been made to include all components and controls that would allow for normal operation. Further, an attempt has been made to utilize the materials that have the lowest possible cost for the service conditions. Specifically, graphite, nickel, and Hastelloy have been utilized in addition to molybdenum.

The process, equipment, and plant layout are described in the Task 1 report for the flowsheet shown in Fig. 1.13. Continental Oil Company is presently preparing conceptual systems design descriptions for the plant.