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ENGINEERING AND CONSTRUCTING THE HALLAM NUCLEAR POWER FACILITY REACTOR STRUCTURE

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REACTOR TECHNOLOGY
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ENGINEERING AND CONSTRUCTING THE HALLAM NUCLEAR POWER FACILITY REACTOR STRUCTURE

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The personal and technical contributions by those many individuals who contributed to analysis, design, field engineering, and quality control of the HNPF reactor structure are hereby acknowledged. Without their personal interest and knowledge, a project of this magnitude could not have been successful.

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ABSTRACT

The Hallam Nuclear Power Facility reactor structure, including the cavity liner, is described, and the design philosophy and special design requirements which were developed during the preliminary and final engineering phases of the project are explained.

The structure was designed for 600°F inlet and 1000°F outlet operating sodium temperatures and fabricated of austenitic and ferritic stainless steels. Support for the reactor core components and adequate containment for biological safeguards were readily provided even though quite conservative design philosophy was used. The calculated operating characteristics, including heat generation, temperature distributions and stress levels for full-power operation, are summarized. Shop fabrication and field installation experiences are also briefly related.

Results of this project have established that the sodium graphite reactor permits practical and economical fabrication and field erection procedures; considerably higher operating design temperatures are believed possible without radical design changes. Also, larger reactor structures can be similarly constructed for higher capacity (300 to 1000 Mwe) nuclear power plants.

I. INTRODUCTION

The Hallam Nuclear Power Facility (HNPF) project is the first power plant version of the Sodium Graphite Reactor (SGR) concept which was authorized by the Atomic Energy Commission (AEC) and was constructed at Hallam, Nebraska. This project began in November 1957, with the basic objective of demonstrating the economic and technical practicability of the SGR concept for central station power.

Separate prime contracts were drawn between the AEC and Atomics International (AI), Bechtel Corporation, Peter Kiewit Sons', Inc., and Consumers Public Power District of Nebraska for the HNPF project. Atomics International, under its prime contract, was responsible for the preliminary and final design and installation of the reactor structure, with the exception of the cavity liner. The Bechtel Corporation was responsible for the final engineering of the cavity liner, which is an integral part of the building substructure. The Baldwin-Lima-Hamilton Corporation was awarded a subcontract by AI for the fabrication-installation of the reactor structure. The contract for the fabrication and installation of the cavity liner was awarded to the Henry Pratt Company by Peter Kiewit Sons', Inc., who was responsible for the facility construction supervision.

The HNPF reactor structure is, in many basic respects, similar to the Sodium Reactor Experiment (SRE), which was also designed and constructed by AI. The SRE reactor structure encloses a 6-ft active core compared to a 13.5-ft diameter active core for HNPF. The corresponding reactor vessel diameters are 11 and 19 ft.

The purpose of this report is to describe the HNPF reactor structure, including the cavity liner and biological shielding, indicating the engineering criteria which were developed during the preliminary and final design of these components. A companion report[3] describes the basic criteria for shielding and methods of calculation for the gamma, neutron flux, and heat generation distributions. This report describes the expected full power operating temperatures, and stress distributions within the reactor structure. In addition, a brief summary of the shop fabrication and field erection experiences is given.

II. DESIGN FEATURES

Certain general design criteria were established for the reactor structure. These are:

1) Provide firm support and allow accurate positioning of the core components in relation to the loading face shield.

2) Adequate shielding must be provided to protect the operating personnel from nuclear radiations from the reactor core.

3) The reactor vessel must, in addition to containing the core and the liquid sodium coolant, provide inlet and outlet plenums, so that the flow of sodium coolant can be properly directed through the core.

4) To contain the liquid sodium coolant in the event of a leak in the reactor vessel or the process pipes located between the reactor vessel and the primary blocking valves. The loss of sodium from the reactor vessel must be limited such that the outlet nozzles will be submerged and coolant flow through the core maintained.

5) An inert gas over the pool of liquid sodium must be maintained. The escape of either liquids or gases from the reactor cavity and the entrance of moisture into the reactor cavity must be prevented.

6) Thermal insulation and shielding must be provided to protect the concrete structure of the facility from excessive heat generation and temperatures from the high-temperature reactor components and to keep the heat losses from the reactor at a low level.

A structure having six major components was developed to fulfill these objectives. These components are indicated in Figures 1 and 2 as the loading face shield, reactor vessel and internals, outer vessel, thermal shields, support structure, and the cavity liner and diaphragm seal.

The top opening of the reactor cavity, called the upper cavity liner, supports the loading face shield. The top surface of the loading face shield is at the operating floor level. The reactor atmosphere is sealed from the operating area by a frozen metal seal* around the periphery of the loading face shield.

*Cerrobend† Alloy, consisting of lead, cadmium, tin, and bismuth, which freezes at 158°F

†Registered Trade Mark of Cerrode Pasco Corporation.

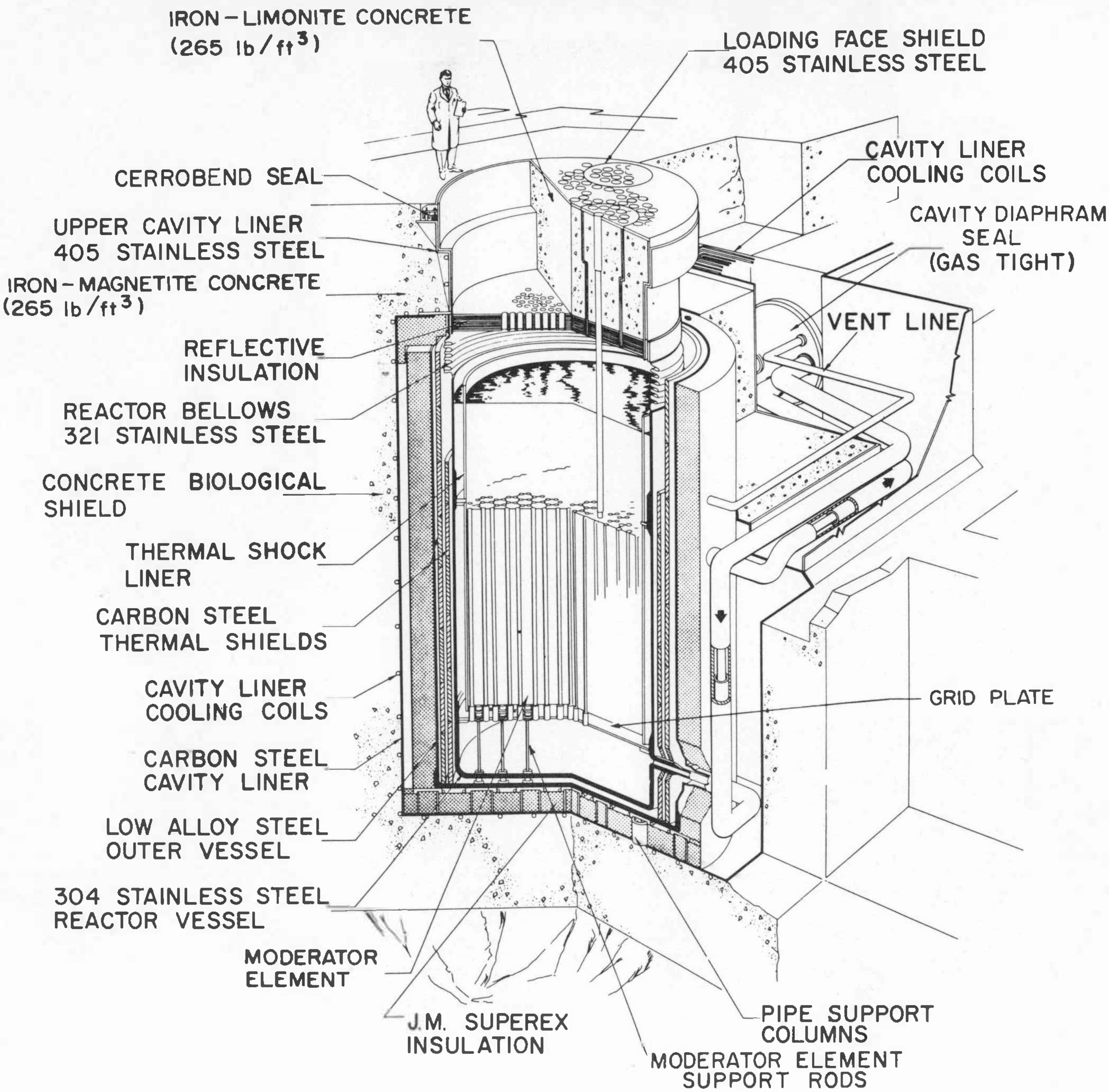

Figure 1. HNPF Reactor Structure

Immediately below the loading face shield is the reactor core which is immersed in liquid sodium. The core is supported in the reactor vessel and aligned to the loading face shield by a grid plate and adjustable core clamps. Thermal shields are located in the annulus between the reactor vessel and the outer vessel. The vessels and thermal shields are supported from the cavity foundation by the reactor support structure. The cavity itself is sealed by the cavity liner, which extends into the pipe galleries to the vicinity of the primary loop blocking valves.

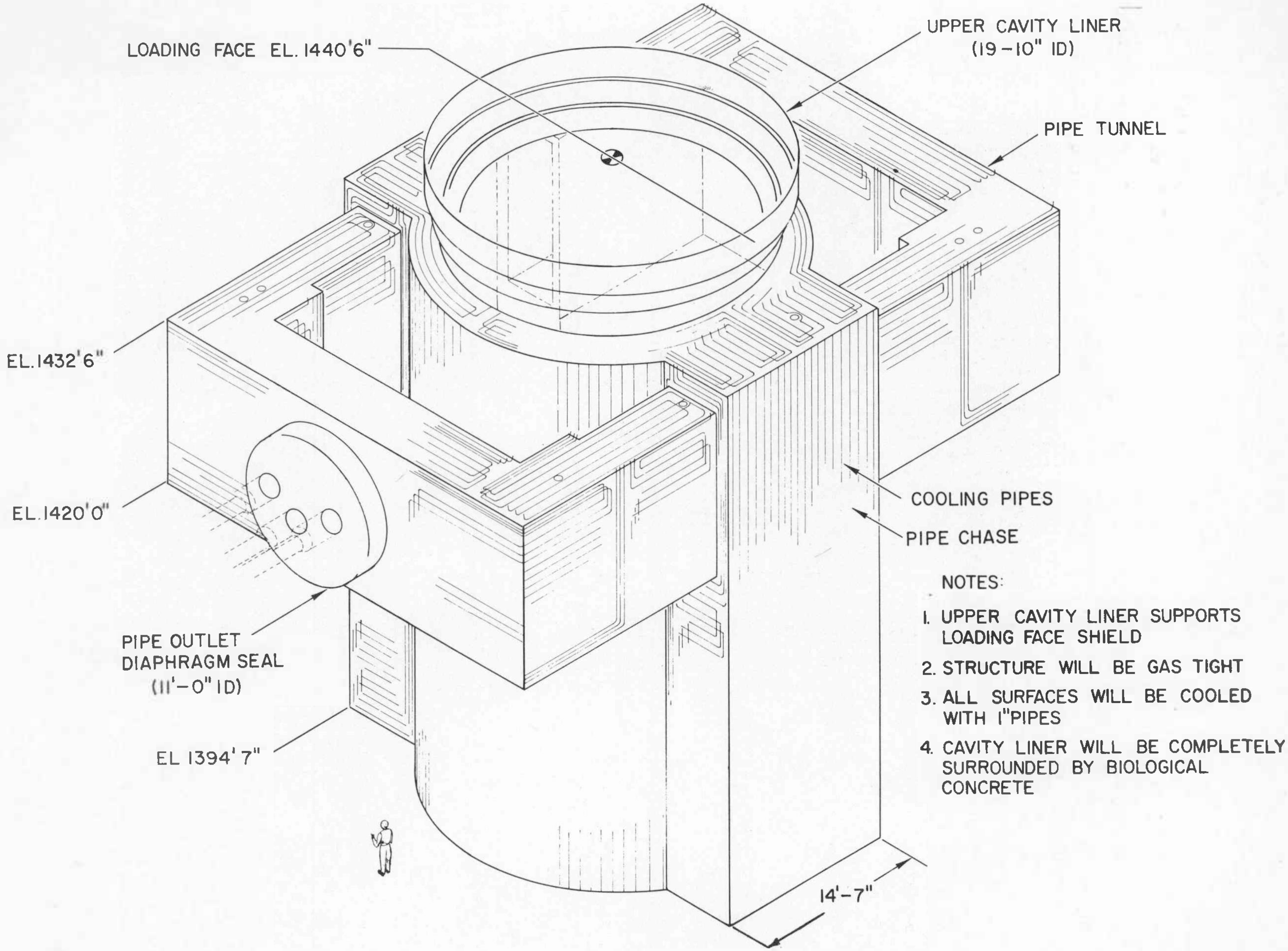

Figure 2. HNPF Cavity Liner

At this point the cavity diaphragm seal is provided to separate the atmospheres of the reactor cavity and the pipe galleries. Detailed design features of each of these components are described in following pages.

A. LOADING FACE SHIELD

The loading face shield provides the biological shielding and the radioactive gas barrier above the reactor core. The nominal dose rate of 0.75 mrem/hr at the loading face at full power was specified. In order to achieve the required shielding and yet maintain reasonable manufacturing tolerance, the loading face shield was designed as a single-stepped plug, 19 ft-2 in. in diameter above the step and 18 ft-6 in. in diameter below the step. This is the minimum diameter that will allow removal of the outermost moderator-reflector elements. The lower gap between the shield and cavity liner is located over the sodium pool and not directly over the gap between the reactor vessel and the thermal shield. The average gap between the shield and the liner is 1/4 in. with a maximum of 3/8 in. The overall depth of the loading face shield is 7 ft-3 in.

Reflective insulation consisting of 13 Type 304 stainless steel (SS) sheets is attached below the 1-in. bottom plate of the loading face shield to decrease the cooling load in the shield. The sheets are spaced 23/32 in. apart. The bottom sheet is 16 gage and the remaining are 22 gage. Thermal shielding is provided by the bottom plate of the loading face shield and a 1-1/2-in. layer of lead poured on top of the bottom plate. Imbedded in the lead are the shield cooling pipes which carry nitrogen coolant. Above the lead is 5 ft 10 in. of iron limonite concrete having a density of 265 $lb/ft.^3$ The iron (steel punchings) limonite concrete was selected because of its high density and the ability of limonite to retain 25% of its initial water content when elevated to relatively high temperature (500°F).

There are three large circular plugs in the loading face shield for removing moderator and reflector elements. The plugs are approximately 58 in. in diameter at the top and are single stepped. The average radial gap of these plugs is 3/32 in. There are also 208 penetrations for core elements, each of which is ~ 6 in. in diameter and single stepped with an average radial gap of 1/16 in.

B. REACTOR VESSEL AND INTERNALS

The reactor vessel, which is attached to the upper cavity by the bellows, contains the reactor core, its associated pool of liquid sodium, and the sodium vapor and helium gas atmosphere above the liquid pool. It also acts as the surge tank for the primary system. The vessel is 19 ft in diameter and 33 ft long, the bottom of the vessel is 2 in. thick, whereas the side wall thickness is as required to withstand the operating loads. Thermal shock liners, 1/4 in. Type 304 SS, are provided in the reactor vessel in the region of the core and at the inlet and outlet nozzles. The vessel was manufactured according to the best known practices, employing all practical means of non-destructive testing.

The reactor vessel wall thickness at the various elevations of the vessel was largely determined by the loads in that region. The wall of the reactor vessel is designed to resist an external pressure of 5 psi, in addition to the internal hydrostatic pressure. A short distance above the grid plate support ring the vessel wall is reduced from the two in. of the lower plenum to 3/4 in. The primary load on the reactor vessel wall at this point is the hydrostatic pressure of the sodium. The region around the outlet nozzle of the reactor vessel is 2 in. thick in order to resist the piping reactions. The wall thickness between the outlet nozzle and the bellows is 1 in. This is due to the requirement that the vessel wall be capable of resisting the 5-psi external pressure.

In order to maintain the alignment of the reactor vessel during an earthquake or in the event of a variation in the coefficients of friction on the bearing surfaces supporting the vessel, alignment keys are provided. The alignment keys for the reactor vessel are connected to the bottom head. These keys fit into keyways which are connected to the outer vessel and then into the reactor foundations.

Sodium from the intermediate heat exchangers enters the inlet plenum from three return lines (one per loop). A grid plate, which separates the inlet plenum from the vessel above, rests on a support ledge in the reactor vessel. The grid plate is also supported by the moderator element support rods. The support rods act as tension rods holding the grid plate and bottom together when at operating pressure.

The nozzles for the process tubes extend through the grid plate into the process channels to provide the flow paths from the inlet to the reactor core. The lower end of the fuel elements slip into the process tube nozzles. Piston rings on the fuel element provides the joint between each process tube nozzle and the fuel element.

A low-pressure sodium inlet line supplies moderator coolant to the plenum between the base of the moderator and reflector elements and the grid plate. This moderator coolant flows upward, past the moderator and reflector elements, and core elements, to the upper plenum where it mixes with the sodium flowing from the process channels. Three outlet nozzles (one per loop) are provided in the upper plenum of the reactor vessel.

The reactor vessel is served by three other lines. One line located in the upper portion of the vessel serves as a supply for the helium cover gas and a second in the same region serves as a vent. Another line, extending to the bottom of the vessel, is used for draining the vessel.

Since the reactor vessel is essentially a flat bottom-open top vessel supported at the bottom, allowances had to be provided for the differential thermal expansion between the reactor vessel and the cavity liner to which it is sealed. This connection was achieved through the use of a 20-ft diameter expansion joint welded to the reactor vessel and upper cavity liner.

Core clamps provide lateral support at the top of the moderator elements and maintain proper alignment of the moderator elements.

C. OUTER VESSEL AND THERMAL SHIELDS

The outer vessel surrounds the reactor vessel. The inlet and outlet sodium pipes penetrate the outer vessel but are not attached to it. The guard pipes surrounding these sodium pipes terminate at and are attached to the outer vessel, thereby providing a path into the outer vessel for any sodium which might leak from the sodium pipes serving the reactor vessel.

Mounted directly below the bottom of the outer vessel are heater elements which maintain reactor temperature during prolonged shutdowns and aid in preheating the reactor structure. The heaters are backed with a polished stainless steel sheet which acts as reflective insulation.

The outer vessel limits the drop in sodium level in the reactor core in the event of a major sodium leak. If there was a rupture so severe that sodium leakage could lower the sodium operating level uncontrollably, the outer vessel would retain this sodium and prevent the sodium level in the reactor vessel from dropping below the outlet nozzles. This is provided so that the reactor afterflow heat can always be carried away by natural thermal convection. Also, should a leak occur anywhere within the cavity, guard pipes are provided to carry the sodium to the bottom of the outer vessel.

Thermal shielding is provided in the annulus between the outer vessel and the reactor vessel. The purpose of this shielding is to protect the concrete structure surrounding the reactor cavity from damage due to heat generated by gamma rays and neutron radiation escaping from the reactor. No thermal shield is required at the bottom of the reactor. The concrete foundation below the reactor is adequately protected from overheating due to the attenuation of nuclear radiation by the shielding effects of the sodium in the lower plenum, the tank bottoms and steel support structures. Thermal insulation is placed inside and around the reactor support pedestals to reduce the transfer of heat to the concrete and reduce the heat loss from the reactor.

In addition to the thermal insulation around the outer vessel and the reactor support pedestals, thermal insulation is provided at the top of the cavity and elsewhere as needed to prevent the exposure of the concrete to high temperatures. The thermal shields and thermal insulation, in addition to protecting the concrete from overheating, decrease the heat losses from the reactor by causing heat to be transferred back into the reactor and sodium coolant.

D. SUPPORT STRUCTURE AND CORE HEATERS

A firm and level support structure is provided for both the outer vessel (which in turn supports the reactor core vessel) and the thermal shields. The support structure is in the form of many short pipes, standing on end as columns, with a combination shim and cap on the top. The cap was machined to provide any shimming required for leveling of the vessels and thermal shields. The pedestals are welded to the cavity liner.

The core preheaters are used to bring the reactor structure and moderator elements up to ~ 350°F before sodium is admitted to the core. The heaters are removed before the sodium is admitted. The heaters are enclosed by carbon steel tubes to prevent damage to the moderator elements during insertion and do not have shield plugs since it is not expected that they will be used after the initial heating. The reactor atmosphere is sealed by an expandable rubber seal at the top of the element. Since the preheaters cannot be handled by the fuel element handling cask, the reactor atmosphere will be opened momentarily during insertion and withdrawal.

E. CAVITY LINER AND REACTOR CAVITY DIAPHRAGM SEAL

The previously described reactor structure components are located within the reactor cavity, which is sealed at the top by the loading face shield. A steel cavity liner is provided which extends into the pipe tunnels to the gallery diaphragm seals. The cavity liner provides a gas-tight envelope between the reactor cavity and the outside concrete structures and shields. This envelope is needed to prevent moisture from entering the reactor cavity as well as to enclose the dry helium atmosphere surrounding the reactor structure and vessels. The liner also will contain any sodium vapors in the case of a leak from the reactor vessel or the primary piping, interior to the blocking valves; or, if the circumstance arises wherein a leak occurs in both the reactor vessel or primary piping and the outer vessel and/or guard piping, the cavity liner will become the final containment barrier for the sodium.

The cavity liner is cooled to remove heat generated in the concrete and cavity liner by the attenuation of the radiation from the reactor as well as sensible heat. Cooling is provided by water circulating through pipes on the exterior surface of the liner. The coolant piping is 1-in. -diameter, Schedule 80, ASTM-A-106 Grade A. The pipe is attached to the cavity liner with a maximum permissible local gap of 1/16 in. between the pipe and the liner. Three parallel coolant loops are provided, each capable of handling the entire cooling load.

The reactor cavity diaphragm seal is provided to maintain a gas-tight barrier between the pipe tunnel and the reactor cavity. In this capacity, the diaphragm seal must be capable of resisting an applied pressure of 10 psig and the piping reactions. The outlet sodium line during normal operation is at a temperature of approximately 945°F which causes large expansions of the piping. These expansions would cause large piping reaction loads in the diaphragm seal if it were not for the use of a bellows expansion joint connection at that point.

The bellows absorb these expansions without the resulting high stresses in the diaphragm seal. A bellows is not required on the inlet sodium line; consequently, the largest piping reactions to the diaphragm seal occur at this point.

The concrete biological shielding surrounding the cavity liner is ordinary concrete. In general, the thickness of the concrete is dictated primarily by shielding considerations. The temperature due to heat generation in the concrete because of the gamma ray and neutron attenuation will be minimized by cooling the concrete. The cooling system consists of pipes attached to the exterior of the cavity liner as described above. This cooling system will limit the maximum concrete temperature to ~ 175°F. The concrete is designed to take the mechanical loads due to the weight of the reactor structure and the thermal loads due to heat generation. The stress is limited to 1350 psi in the concrete and to 20,000 psi in the reinforcing steel, in accordance with the ACI building specifications.[4]

III. DESIGN CRITERIA AND EXPECTED OPERATING CHARACTERISTICS

Design of the components of the reactor structure emphasizes safety and practicality of fabrication. This is necessary, not only from the standpoint of minimizing hazardous conditions, but because of the measures required for repair after operation begins. It is of the utmost importance that integrity and usefulness of the reactor structure be maintained through all the anticipated and unanticipated (or casualty) operating conditions; the secondary consideration is economics. However, the most economical design was strived for. In all cases, consideration was given to feasibility and construction and erection procedures.

Components of the reactor structure normally in contact with liquid sodium are made of Type 304 SS to provide for corrosion resistance and strength. The exceptions are the loading face shield and core clamp assembly where a ferritic stainless steel was used because of its low coefficient of expansion. Type 405 was chosen because its welding characteristics are better than those of most other ferritic stainless steels. Those components not normally in contact with sodium are made of carbon or low alloy steel.

A conservative structural design was achieved by maintaining relatively low stress levels within the reactor structure components, as well as minimizing the degree and number of stress raisers. All the known pre-operational and operational loading conditions were incorporated into the analyses for the reactor structure and the latest design analysis methods were used to evaluate the effect of these loading conditions. Whenever a critical component was considered too complex for investigation by analytical methods, experimental programs were carried out.

The design criteria, which are incorporated in the ASME Boiler Code, Section VIII,[4] were used to evaluate safety of the reactor vessel for the mechanical loads. The stress intensity due to design values of internal pressure, mechanical forces, or their combinations (based on the average stresses across the thickness of any section and neglecting structural discontinuities and stress concentration) were limited by the allowable tensile values given by the ASME Boiler and Pressure Vessel Code.[4]

The peak stress intensity at any location due to design pressure, mechanical forces, pipe reactions, or their combinations, including the effects of any structural discontinuities (but not stress concentrations), were limited to one and one-half times the allowable tensile values as given by the ASME Boiler and Pressure Vessel Code.[4]

All thermal stresses were considered as transient conditions and treated in accordance with the method outlined in the "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components".[5] The AISC specifications[6] and the ACI Building Code[7] were used where applicable.

The thermal design criteria for the bottom support structure, the top and side shields, and the insulation were based on maintaining concrete temperatures generally below 175°F. The temperatures of other materials were kept below their practical use limit, and heat generated from absorption of gamma ray and neutron radiations was returned to the reactor to the maximum possible extent.

The shielding design was based on full-time occupancy of the reactor room during normal operation. The bulk shielding was designed to limit the radiation level at the loading face shield top surface to 0.75 mrem/hr for an imperforate shield. Where there are penetrations through the shielding and where gaps, ducts, or voids exist, the shielding is designed to limit the radiation level to the tolerance level (7.5 mrem/hr).

In addition to the above general requirements, additional criteria were established for safeguards and normal operation. These are further detailed below.

A. SAFEGUARDS REQUIREMENTS

Special design criteria, relative to the containment of reactor liquids and gases, were also established to assure safety to the operators and the general public. These are:

1) Reactor operation will be possible with minor sodium leakage from the core vessel or primary piping into the outer vessel and/or associated guard pipes.

2) Continued reactor operation will not be allowed in the event of a major sodium leak.

3) Reactor operation is possible with minor gas leakages from the core vessel, except for leakage through the top shield. The maximum permissible gas leakage rate from the loading face shield is 0.11 ft^3/day.[10]

4) Even though pressures in the reactor vessel greater than the cover gas pressure (2 ft-6 in. H_2O pressure) are not expected due to the inherent shutdown features of the SGR core, maximum pressure containing ability is provided.

5) The outer vessel and associated guard pipes, in the event of a sodium leak interior to the primary loop blocking valves, assure that natural convection flow through the core can be maintained.

6) Only small quantities of sodium vapor will be permitted in the reactor cavity, external to the reactor vessel and interior to the reactor cavity diaphragm seal. Materials in this space must be compatible with this small amount of sodium vapor.

7) Because of criterion 3 above, compatible atmospheres must be maintained in the reactor vessel and cavity liner.

8) All of the primary sodium on the reactor side of the blocking valves and within the blocking valves is provided with secondary liquid containment.

9) There are at least two maintainable barriers between the reactor gases and the atmosphere of the reactor room of the facility.

10) The maximum permissible leakage rate from the reactor cavity is 1.37 ft^3/day and the design leakage rate is 1 ft^3/day.[10]

The maximum permissible gas leakage rate (0.11 ft^3/day) from the loading face shield is based on assumed failure of one percent of the fuel rod cladding, one percent fission gas release from the fuel, two air changes per hour in the reactor building, and a permissible concentration of noble gas fission product activity in the building of not greater than 1×10^{-6} $\mu c/cc$.

The maximum permissible gas leakage rate (1.3 ft^3/day) from the reactor cavity is also based upon the above conditions but further assumes leakage through the other regions in the reactor structure and the dilution of the gaseous fission products by the cavity liner atmosphere.

B. NORMAL OPERATION, INCLUDING PREHEATING

Rotation of the loading face shield, if necessary, will only take place when the reactor is shut down and fuel elements removed. This should be an infrequent operation and will be required only for removal of moderator and/or reflector elements and possibly for reactor core repairs. This operation consists, essentially, of rotation of the shield to place one of the three large moderator removal plugs into position above the element to be removed. The shield rests on rollers which provides support at all times and permits rotation. Guide rollers are provided to maintain shield alignment and concentricity during rotation. To rotate the loading face shield, all core elements, i.e., fuel, dummy, source, temperature instrument, liquid level, control connecting rod connector assemblies, etc., must be removed. The holes in the shield through which these elements were inserted must be properly closed with plugs provided for this use. The cooling lines, electrical and instrumentation leads to the loading face shield must be disconnected as well as the ones to the moderator removal plug that is to be removed. The frozen metal seal at the periphery of the loading face shield must be melted by means of its electric heaters. The shield is then ready to rotate. It is estimated that it will require a force of about 6000 lb applied at the periphery to rotate the shield. The shield is rotated by attaching cables to two of the gusset stiffeners on the shield support ring. The cables are run through sheaves mounted tangent to the periphery of the shield and attached to the 15-ton hook on the building crane.

The initial core and structure preheating will be accomplished using in-core heaters and the reactor vessel bottom heaters. During reactor shutdown the reactor vessel bottom heaters will be used to maintain the reactor temperature. In order to minimize thermal distortion the temperature of the lower portion of the containment vessel shell must not exceed ±50°F relative to the average temperature of its head, and the lower portion of the reactor vessel shell must not exceed ±35°F relative to the average temperature of its head. The maximum ΔT between any two points in the head of either vessel must not exceed 85°F.

The initial preheating of the reactor will be done with the vessel dry prior to filling the system with sodium. Preheating will be done using the in-core heaters and one set (30 kw) of the bottom heaters. There is a total of 54 in-core heater units. Thirty-three heaters with a total heat output of 48.8 kw are arranged in two rings in the moderator can region. Six heater units with a total heat output of 29.3 kw are provided in the upper plenum. Fifteen heater units with a total heat output of 9.6 kw are provided in the inlet plenum. The 33 moderator can region heaters are turned on until the reactor vessel wall opposite the moderator can reaches 330°F (~ 200 hr); then the outermost row of moderator can heaters is turned off. The inner row of moderator can heaters is left on until the inner moderator cans reach approximately 350°F; then they are turned off. The six heaters in the upper plenum and the 15 heaters in the lower plenum are turned on and off as necessary to keep the reactor vessel wall temperature uniform over its length. The bottom heaters are turned on and off as necessary to keep the temperature differences in the lower regions of the containment vessel and the reactor vessel within the limits given above.

During prolonged shutdown of the reactor the temperature of the core and structure will be held at 350°F by use of the bottom heaters. The connected heat output of the bottom heaters is 60 kw with 12 spare heaters (30 kw).

Heat generation distributions for full power operation in the reactor structure components are shown in Figures 3 and 4. The basis for these results are given in reference 3. Heat generation distribution for the loading face shield is shown in Figure 5.

C. COMPONENT REQUIREMENTS

Based upon the above design criteria the expected operating conditions for each major component are given in the following sections.

1. Loading Face Shield

The nuclear radiation at the top surface of the loading face shield will be due to either direct penetration of the shield or streaming through the gaps surrounding the numerous plugs. Calculations have shown that the radiation level due to direct penetration (for gamma rays or neutrons) is much less than the design limit of 7.5 mrem/hr. The shield thickness is needed primarily to provide sufficient length for the attenuation of streaming through the various gaps.

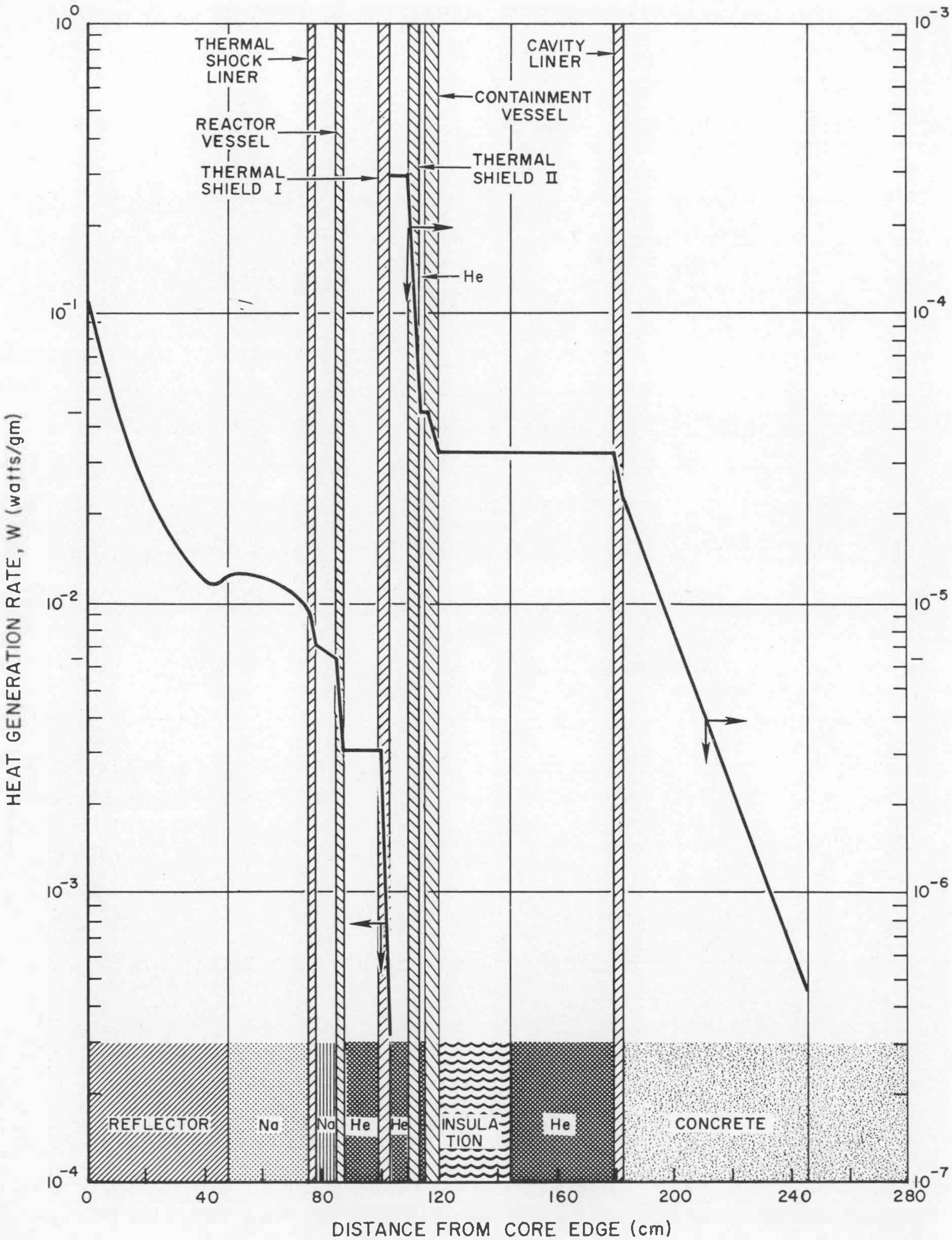

Figure 3. Overall Heat Generation Distribution in the HNPF Radial Shield

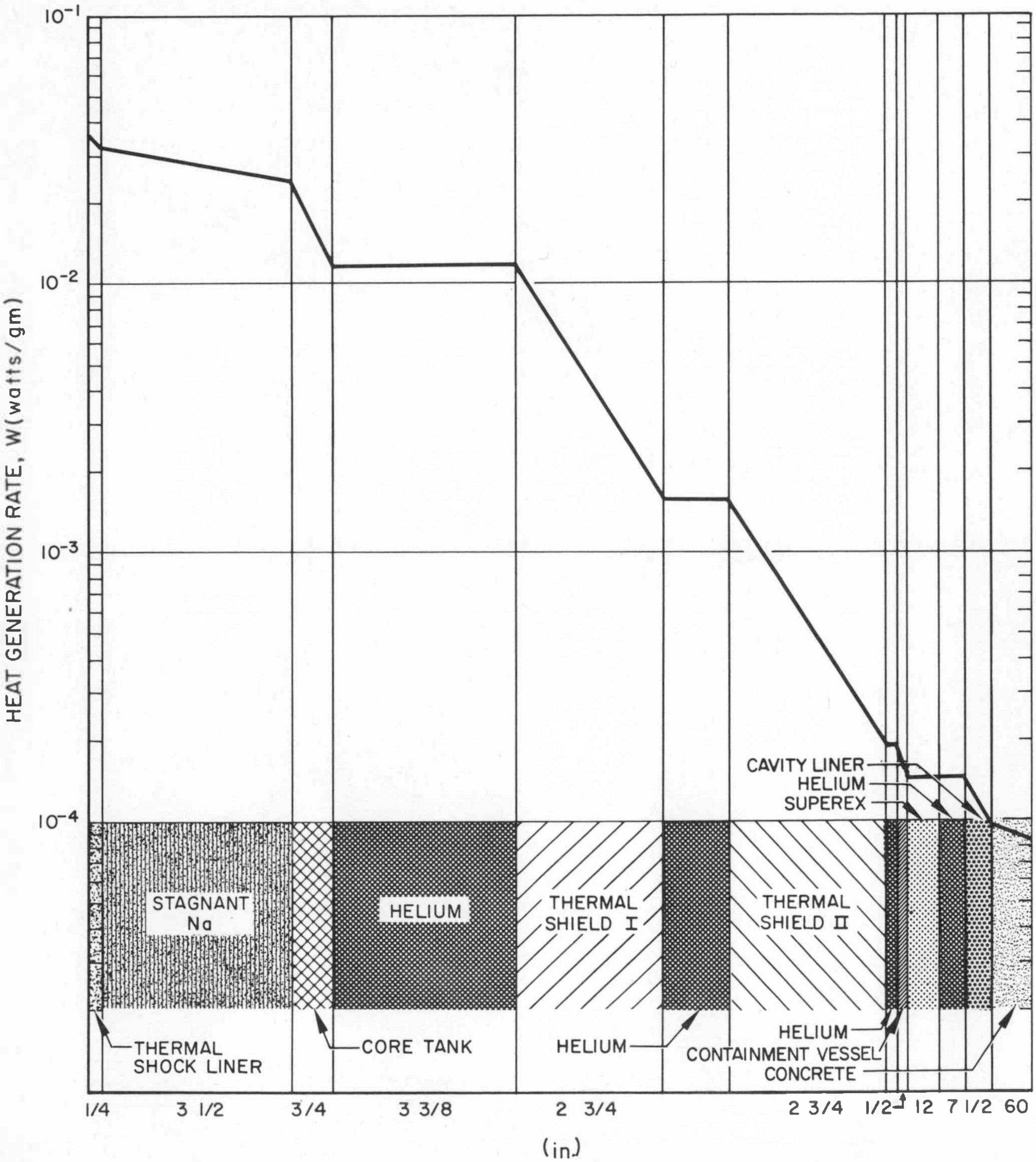

Figure 4. Internal Heat Generation Rates for 5-1/2-in. Thermal Shield

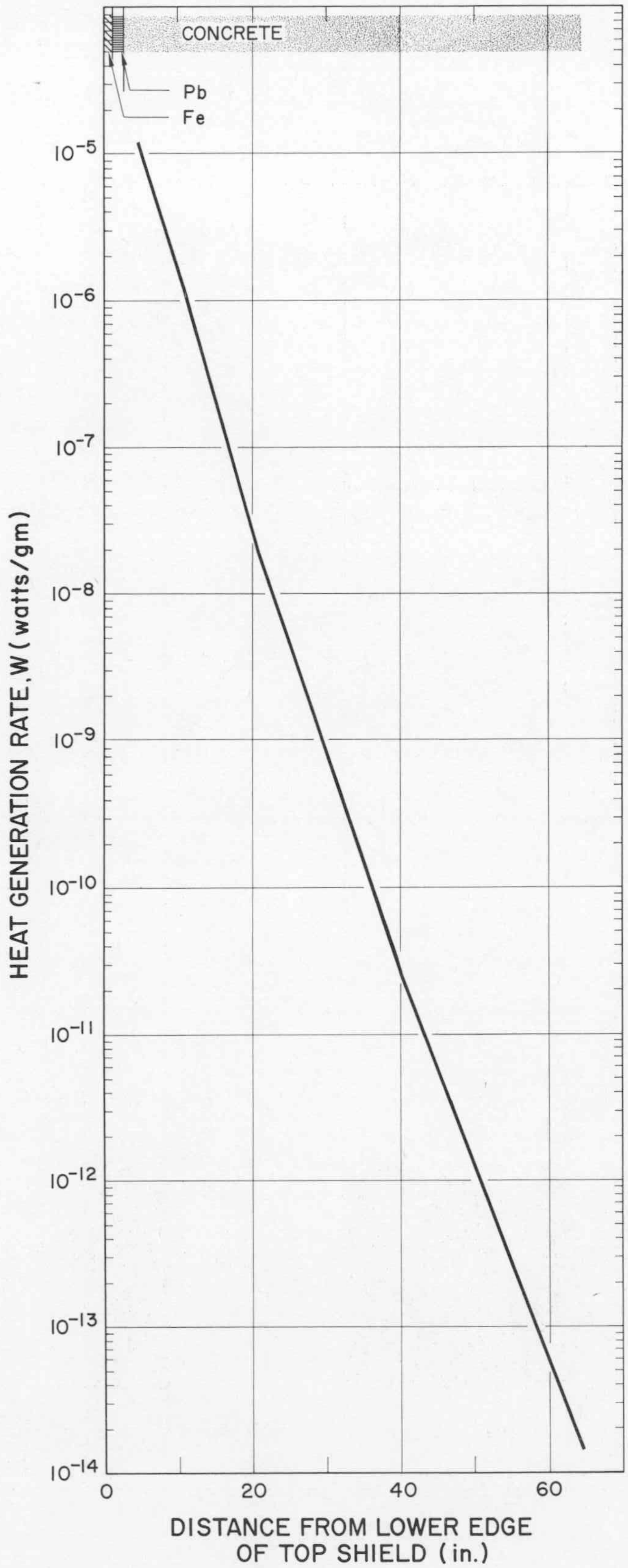

Figure 5. Gamma Ray Heat Generation in Loading Face Shield

Calculations indicate that radiation streaming through the gaps around the plugs is quite small (0.75 mrem/hr). These calculations were based on a single plug and assume that there will be a minor cumulative effect due to the large number of such plugs. The streaming dose rate through the 1/4-in. gap surrounding the loading face shield is about 2.0 mrem/hr, which is below the design limit of 7.5 mrem/hr. The induced activity of the nitrogen coolant is as follows: C^{14}-5.3 x 10^{-6} μc/cc-yr and A^{41}-8.1 x 10^{6} μc/cc based on 0.1% argon impurity. These activities present no problem either from a hazards standpoint or from increased radiation levels.

The basic structural loads carried by the loading face shield include the dead weight of the shield itself, the weight of the fuel and miscellaneous elements, the control rod carriage, and the dynamic loading imposed by the control rods. In addition, the loading face shield carries the impact loading of the fuel handling machine movable shield during the fuel handling sequence. The various combinations of these loadings impose large bending moments causing compression in the shield top surface and tension in the bottom. The shield structure is accordingly designed as a reinforced concrete plate.

The allowable reinforcing steel stress is 20,000 psi (intermediate grade steel based on the ACI Building Code[6] requirements for reinforced concrete).

The effect of the large moderator removal plug holes on the redistribution of the plate bending moments was determined by a series of tests on a one-tenth scale model aluminum plate.[11] The purpose of this model testing procedure was the accurate determination of the stress distribution around the complex geometry of the three large, eccentric, moderator-removal plug holes. The effect of the fuel plug holes on the concrete design allowable stresses was also determined experimentally. The maximum concrete compressive stress due to the downbending condition is 530 psi, while the maximum reinforcing steel tensile stress is 18,000 psi. The minimum factor of safety is 2.2 based on yielding of the reinforcing steel.

For a positive internal pressure of 50 psig, the maximum concrete compressive stress was calculated to be 820 psi and the reinforcing steel tensile stress is 37,300 psi.

The maximum vertical deflection of the shield for down-bending due to all the basic loads, including impact loading due to the fuel handling machine movable shield, was calculated to be 0.004 in. as a lower limit and 0.029 in. as an upper limit. The former value is applicable when the concrete is uncracked and the latter is applicable when the concrete is fully cracked.

The shield is supported by a side shell which frames into a ring girder, which in turn rests on a total of 40 rollers. The roller bearing allowable load was based on manufacturer's recommendations. The allowable bearing stress on the hardened rolling surface of the ring girder was based on conservative practice. Relatively low usage was anticipated which allowed higher than normal design stresses for the bearings. The maximum bending stress in the ring girder was calculated to be 19,600 psi (compression). The maximum static bearing stress on the rollers is 142,000 psi; the bearing stress when rolling is 98,000 psi. Since a hardened rolling surface is attached to the ring girder and the rollers are also hardened, no significant brinelling is expected.

The top and bottom plates of the shield are essentially non-structural components, inasmuch as they do not resist the primary bending movements in the shield. They do serve a secondary structural purpose in framing concentrated loads to their supports (the side shell, fuel plug liners, and the edge ring girder). The unsupported spans are such that no excessive stresses exist. The stainless steel bottom plate and side shell are essentially nonstructural items providing containment for the reactor atmosphere and protecting the concrete from sodium vapor. However, the side shell does act as a portion of the edge ring girder in transferring the shield dead weight and the live loads (i. e., handling equipment) to the roller bearings.

The shield is subject to thermal stresses from the following operational conditions: startup, shutdown, steady-state operation, defrosting, and loss of coolant. The effect of the thermal loads is to increase the compressive stress in the concrete portion of the shield to 390 psi and decrease the tension stress in the reinforcing steel. The maximum vertical deflection of the loading face shield could be increased by 0.125 in. if a 15-hr loss of coolant transient occurred. This is in addition to the deflection due to mechanical loads. Such a transient would cause a compressive stress of 35,000 psi in the bottom plate.

The cooling load for the loading face shield is 165,000 Btu/hr. This heat is removed by the loading face shield cooling system. Nitrogen at a pressure of 250 psi is used as a coolant. The nitrogen enters the loading face shield at 95°F and leaves the shield at 135°F. The total nitrogen flow rate is 16,500 lb/hr.

The maximum concrete temperature in the loading face shield during normal operation is 165°F in the main portion of the shield and 175°F in the plugs. The 1-1/2-in. layer of lead above the bottom plate reduces the heat generation in the concrete to a negligible amount with the result that the temperature profile through the concrete is linear from the lead-concrete interface to the top of the shield.

The maximum temperature at the bottom plate of the reflective insulation will be 928°F, based on a 1000°F sodium pool temperature. The temperature decreases to 165°F at the loading face shield bottom plate.

The junction of the bottom plate and side shield is subject to thermal cycling from transient temperature excursions which could result in thermal displacement stresses above the yield point. The analysis accordingly is governed by fatigue criteria set forth in the "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components."[5] The various items connected to the bottom plate, (side shell, side skirt, moderator plug liners, fuel plug liners, and reinforcing rods) were analyzed for fatigue damage since the cycling from thermal transients results in displacement stresses above the yield point. The critical items are the attachment of the skirt to the side shell. The fatigue damage factor for the former after 50 cycles of a 15-hr loss of coolant transient is 0.143 and is 0.013 for the latter after 100 cycles from shutdown to steady state.

Melting of sodium which may accumulate after long periods of time on the reflective insulation plate is accomplished by increasing the inlet temperature of the nitrogen coolant to 200°F inlet and adjusting the flow rate to obtain a temperature of not more than 240°F outlet. The maximum concrete temperature which will occur during melting of frozen sodium is 280°F. This would occur in a moderator plug. If all 13 of the insulation plates are frozen solid with sodium, ~ 11 hours will be required for melting. A hot liquid can be used, if necessary, in the upper cavity liner coolant pipes to melt any sodium which may become frozen in the annulus between the upper cavity liner and the loading face shield and the cavity liner. Based on a 10-hr heating time, a heat input of 275,000 Btu/hr will be required.

2. Reactor Vessel and Internals

The axial temperature distribution in the reactor vessel from the bottom to the top of the sodium pool is shown in Figure 6 for full power operation. The maximum estimated temperature gradient is 3.4°F per in. occurring approximately 10-1/2 ft from the bottom. The reactor vessel temperature at the horizontal mid-plane is 837°F. The average vessel temperature in the active fuel region is 819°F and the average vessel temperature from the bottom of the vessel to the bellows is 855°F. Figure 7 shows the axial temperature distribution between the top of the sodium pool and the top of the bellows skirt.

The maximum radial temperature gradients in the 3/4-in. -thick sections of the reactor vessel wall are of the order of 7 to 10°F. The maximum radial temperature differences in the 2-in. -thick sections of the vessel are ~ 14°F.

The most severe temperature transient that could occur would be a change in the coolant temperature equal to the coolant temperature rise through the reactor (338°F). For the reactor vessel inlet area the maximum change in fluid temperature would be from 607 to 945°F. Above the core the maximum change in fluid temperature would be from 945 to 607°F.

a. Lower Plenum

The lower 42 in. of the reactor vessel below the grid plate forms the inlet plenum where 30 psi sodium is returned from the intermediate heat exchanger to the reactor. This is the highest-pressure plenum of the vessel and has a wall thickness of 2 in. The inlet plenum is designed to resist the piping reactions and an internal pressure of 30 psi. The internal pressure is due to the hydrostatic pressure of the liquid sodium and the pump pressure. The weight of the entire reactor vessel, internals, and core must also be transferred through the bottom head. The normal operating temperature of the lower plenum is 607°F. However, the design temperature used in evaluating the integrity of the reactor vessel was chosen as 1050°F.

The mechanical stresses in the lower head of the reactor vessel due to the transfer of the weight (~ 761,000 lb) of the core and vessel are minimized by the large number of support pads attached to the bottom head. The support pads are aligned with the moderator can support rods, so that the weight of the moderator cans is directly transmitted to the pads without causing bending

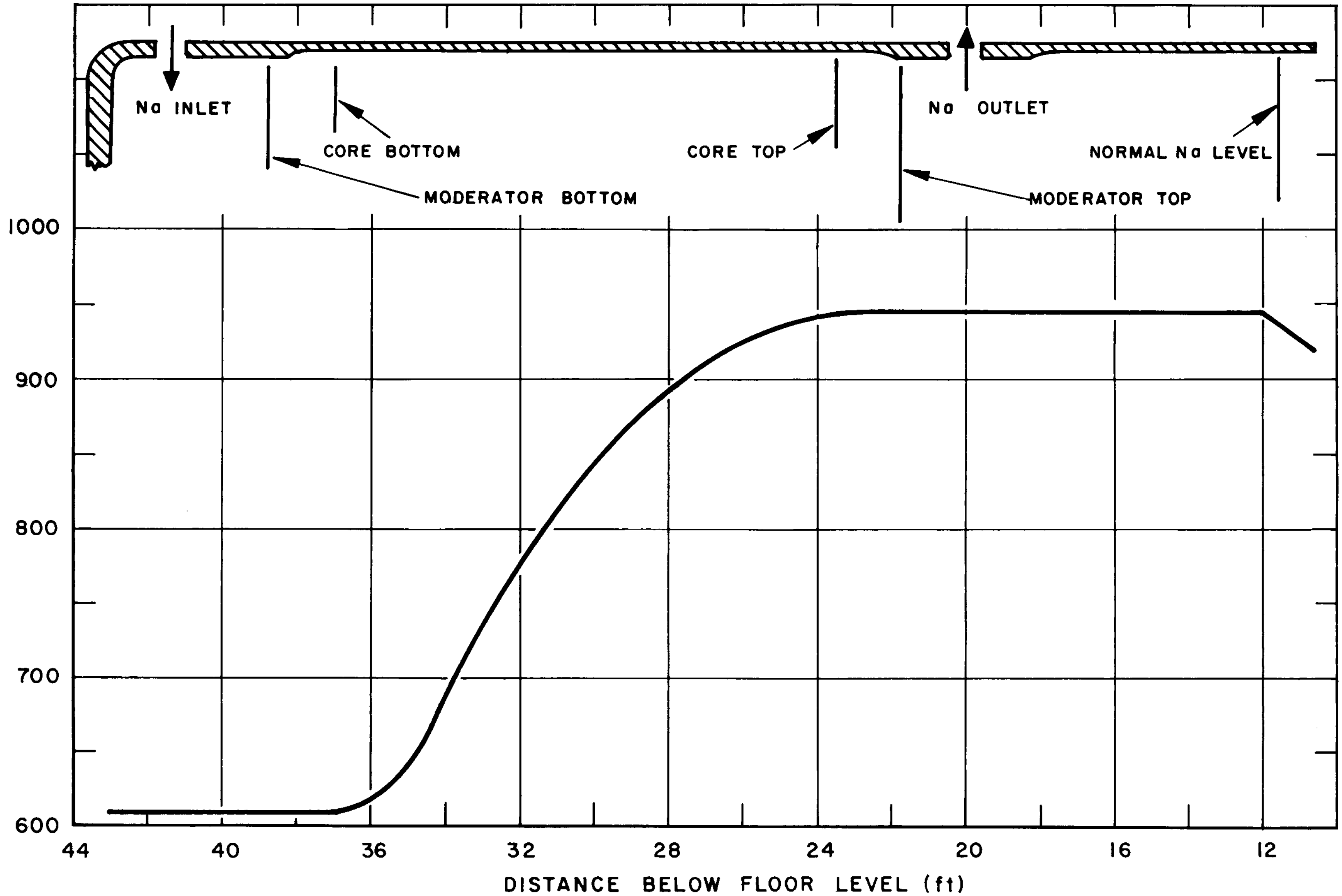

Figure 6. Axial Temperature Gradient in Reactor Vessel from Bottom to Upper Plenum

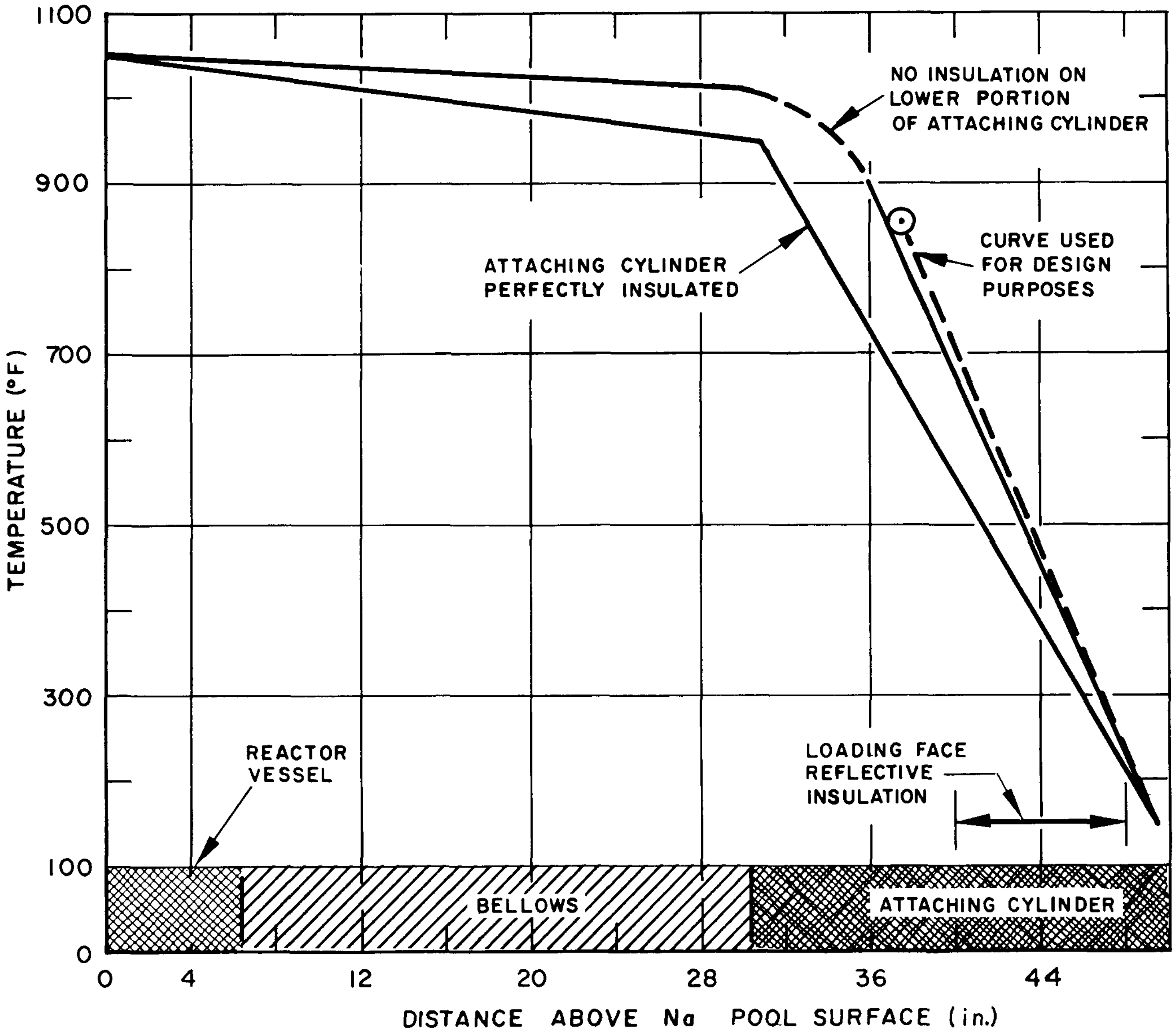

Figure 7. Reactor Vessel Axial Temperature Gradient in Upper Plenum Region

stresses in the lower head. The allowable bending stress, based on 1-1/2 times the membrane allowable given in the ASME Code, Section VIII,[4] at 1050°F is 12,750 psi.

The thermal stresses that result from normal power swings will not cause any significant change in the stress levels of this portion of the reactor vessel. The transient conditions, however have two significant effects. The first of these is the stress resulting from the temperature distribution through the plate thickness. The second effect is that of a redistribution of the load that the plate carries. This occurs because of the attempted upward bowing of the plate due to the temperature distribution through the plate thickness. In this case of thermal loading, a true measure of the structural capability of the plate is found by fatigue considerations. Because of the simple geometry of this element, stress and strain concentrations were not considered. The amount of damage and weakening of the plate due to these transient conditions is measured by a cumulative damage factor. The computations of this damage factor followed the procedures as given in the "Tentative Structural Design Basis for Reactor Pressure Vessels."[5] The stress due to the thermal transients has been found to be ~ 35,000 psi. When this thermal stress is added to the mechanical stress, the total stress equals 43,700 psi. Considering this combination of thermal and mechanical stress from the standpoint of fatigue failure, it was found that a rather substantial factor of safety exists, since the damage factor was computed as ~ 0.02. This compares with the allowable value of 0.8.[5] Therefore, the bottom head will function safely throughout the lifetime of the reactor.

b. Vessel Nozzles

The inlet nozzle region of the reactor structure is perhaps the most complex region from the stress analysis standpoint. The complexity of this area results from the close interaction between the reactor vessel shell, the inlet nozzles, and the grid plate support ring. This region must be capable of resisting the applied loads due to piping reactions and hydrostatic pressure and the thermally induced loads. The mechanical loads due to the piping reactions determined that the most highly stressed inlet nozzle is the southeast one. The stresses resulting from the piping reactions were found to be a maximum of 2,000 psi. The membrane stresses developed due to the hydrostatic and pressure heads acting in the lower plenum were found to be ~ 4,300 psi, after the application of an appropriate stress concentration factor.

Loads which result from internal pressure, seismic loading, and the piping reactions were at combined stress levels low enough to satisfy the requirements of the ASME Boiler Code.[4] The seismic loadings, which are applied to the reactor vessel wall in this region, were computed using the Uniform Building Code.[9]

Thermal stresses are developed by virtue of the requirement that the lower plenum of the reactor vessel be capable of resisting a step change from 607 to 945°F. Shortly after the introduction of the 945°F sodium into the lower plenum which could be caused by loss of power to the secondary pump impeller, the temperature of the reactor vessel wall on the lower plenum side increases while the shell above the grid plate remains at the same temperature. The existence of the temperature difference between these two segments of the reactor vessel wall results in stresses in the vessel wall. It has been found that these stresses are not as large as those existing at the point where the reactor vessel inlet nozzle attaches to the reactor vessel and, therefore, create no problem.

The thermal stresses developed in the nozzles due to the above transient condition induce relatively high stresses. The resulting thermal stresses were analyzed on the basis of fatigue considerations in accordance with the "Tentative Design Basis for Reactor Pressure Vessels."[5]

The total combined mechanical and thermal stress was found to be a maximum at the connection of the inlet nozzle to the reactor vessel wall. Thermal stress fatigue analysis has shown that the damage factor resulting from the sum of all the anticipated transient conditions is less than 0.1. This is less than the allowable value of 0.8, as given in Reference 5.

The outlet nozzle region of the reactor vessel must carry loads resulting from internal pressure and the piping reactions. With a ΔT of 14.5°F across the 2-in.-thick nozzle course, a design pressure of 10 psi, and piping reactions, the resulting combined stress is 5210 psi. The worst transient condition occurs during a scram when the pumps fail to shut off. This results in a ΔT of 123°F across the 2-in. wall and a maximum ΔT of 338°F between the nozzle and outlet pipe. The effective alternating stress is 24,350 psi, which will result in negligible fatigue damage.

c. Reactor Vessel Walls

The analysis of the reactor vessel wall between the upper cavity liner and the bellows consisted of (1) evaluating the thickness requirements to resist a 5 psi internal and external pressure, and (2) evaluating the capabilities of the section to resist the strains resulting from the thermal gradients during all phases of reactor operation.

The requirement that the vessel resist either an internal or external pressure of 5 psi dictated a wall thickness of 1/2 in. The existence of the 5-psi external pressure difference, coupled with the high temperatures existing in this region, made the 1/2-in. wall necessary for adequate buckling resistance.

The thermal stresses are highest near the end of the bellows skirt where it connects to the upper cavity liner. The region of maximum stress occurs where the Type 304 SS bellows skirt connects to the Type 405 SS cylinder of the upper cavity liner. These maximum stresses result from (1) the discontinuity due to the thermal gradients along the axis of the vessel, and (2) the difference in the coefficients of expansion of Type 405 and Type 304 SS. The "elastic" stress levels existing here are above the yield strength of the material. However, the nature of these stresses permits the evaluation of the structural capability on the basis of thermal stress fatigue. The analysis followed that given in Reference 3. It was found that the damage factor, which is a measure of the fatigue damage, is less than 0.1. This compares with the allowable value of 0.8.[5]

d. Reactor Vessel Bellows

The reactor vessel bellows provides for the expansion of the reactor vessel relative to the reactor cavity. The axial movement of the bellows is expected to be ~ 4 in. The normal operating conditions show that the internal gas pressure acting on the bellows will not be greater than 0.5 psi. However, the bellows is designed for an internal and external pressure of 5 psi.

The reactor vessel bellows is constructed of Type 321 SS, which has high corrosion resistance, considerable high temperature creep strength, and can be formed and welded easily. At installation this bellows was expanded cold so that the spring load on the bellows will reduce when the reactor is at normal operating temperature. This permits the bellows to operate at very low stress levels at operating conditions.

Design stresses due to combined effects of deflection and pressure was limited to twice the yield strength at any given temperature. The bending stress component due to pressure was limited to 1-1/2 times the stresses specified in the ASME Boiler and Pressure Vessel Code.[4] Based on information given in Reference 2 and considering a design temperature of 1050°F, bending stresses and membrane stresses in the bellows convolution due to pressure loading were limited to 19,600 psi and 13,100 psi respectively. In order to conform the above limitations, the bellows were subjected to an extension of 2.75 in. by cold springing.

To establish behavior of the bellows during operation, a series of test programs has been performed. The results of these tests were compared with theoretical values and were incorporated into the analysis and final selection of the reactor vessel bellows.

The external pressure is the primary factor in the selection of the thickness of the reactor vessel bellows skirt. However, consideration must be given to the thermal gradients in order to describe the capabilities of the section adequately. The temperature at the upper cavity liner is ~ 150°F, because of the cooling coils placed on the exterior of the cavity liner. The temperature of the wall slightly above the bellows is essentially the same as the sodium pool temperature of 945°F. The difference in temperature between these two points results in a rather large thermal gradient. Consequently, this portion of the reactor vessel is particularly sensitive to the temperature changes in the upper plenum. Any temperature change in the sodium pool results in the flexing of the joint connecting the reactor vessel and the cavity liner. Inasmuch as the mechanical stresses are low, the primary design considerations are with respect to the cumulative fatigue damage. This was evaluated in accordance with the "Tentative Structural Design Basis for Reactor Pressure Vessels."[5]

e. Grid Plate

The grid plate separates the lower plenum of the reactor vessel and the moderator coolant plenum. In this capacity, the plate is subjected to thermal, as well as mechanical, loads. The mechanical loads result from the pressure differential across the grid plate. The stresses resulting from this pressure drop are minimized by the large number of moderator element support rods connected to the grid plate, and result in bending stresses in the plate between the moderator element support rods of ~ 3000 psi. This includes a stress concentration factor of two to account for the numerous holes in the grid plate.

Stresses are also induced in the plate as a result of the relative expansion of the various components in the reactor vessel. The relative vertical expansion of the rods causes bending and distortion in the grid plate adjacent to the grid plate support ring. It is calculated that the maximum bending stress in the plate due to this action is ~ 27,500 psi.

The mechanical loads result from a pressure difference between the lower plenum and the upper plenum. The design pressure difference is ~ 17 psi. Because the grid plate and the moderator can support rods are clamped to each other, the span of the plate to resist this applied pressure is not very large. The allowable bending stress used at 1050°F was 12,750 psi, which is based on 1.5 times the ASME Code[4] allowable.

Thermal stresses could be included in the grid plate as a result of an assumed transient temperature change from 607 to 945°F occurring in the lower plenum. The thermal stresses developed in the plate from the thermal shocks are due to the gradient through the plate thickness and the bending stresses which are developed at the outer edge of the grid plate due to the expansion of the moderator element support rods. The bending stresses result from the moderator element support rods increasing in length at a rate greater than the walls of the reactor vessel. Since the grid plate is bolted to the reactor vessel wall through a grid plate support ring, the relative motion between the grid plate and its support ring results in bending stresses in the grid plate. It was found that the elastic stresses due to this thermal shock were in excess of the yield strength. Consequently, it was necessary to evaluate the effects of this thermal shock on the basis of thermal stress fatigue. The analysis has shown that the resulting damage factor for all the transient conditions anticipated is lower than 0.1. The allowable value as given in Reference 3 is 0.8. Consequently, the grid plate will maintain its structural integrity throughout the lifetime of the reactor.

The lower grid plate rests on and is bolted to the grid plate support ring such that sliding can occur. In order to minimize the leakage of sodium from the lower plenum into the reactor core, a flexible metal seal has been located at the junction. This seal must be capable of resisting the loads applied to it satisfactorily throughout the reactor life. These loads are: (1) an internal pressure of 15 psi due to the pressure differential across the grid plate, and (2) those loads induced by the expansion and contraction of the grid plate which may be as much as 1/4 in. The material used for the seal is Type 321 SS with a thickness of 0.050 in. The stresses resulting from the 15-psi pressure drop have been found to be ~ 6750 psi. The ASME Boiler Code allowable stress is 13,500 psi at 1000°F. Stresses resulting from a 1/4-in. displacement were computed on an elastic basis to be ~ 88,000 psi. The combination of the mechanical and displacement stresses was analyzed by the fatigue criteria of Reference 3, and it was found that the damage factor was ~ 0.004. This compares with the allowable value of 0.8.[5]

f. Moderator Element Support Rods

The moderator element support rods are subject to mechanical as well as thermal loads. The mechanical load which must be transmitted during normal reactor operation is the pressure differential across the grid plate and the weight of the moderator elements less the effect of buoyancy.

The sodium, as it enters the lower plenum, impinges on the support rods developing lateral forces. Further, this flow impingement promotes the formation of vortices, and their breaking away from the rods produces vibrations in the rod.

The thermal loads on the moderator element support rods result from the temperature changes occurring in the lower plenum. These temperature changes cause expansion and contraction of the various elements of the lower plenum. Thermal stresses are also developed in the rod by virtue of a change in the ambient temperature. The circumferential and radial stresses which result further complicate the stress distribution.

The rods which are subject to the most severe loading conditions are located directly in front of the inlet nozzles. The maximum mechanical stress in a rod at this location has been found to be ~ 15,000 psi. This stress level slightly exceeds the ASME Code allowable value of 13,850 psi at 950°F. However, it is felt that the amount of deviation is not significant. The maximum thermal stresses occur at this location as well, due to the direct impingement of hot sodium on these rods. Thermal fatigue analysis of the support rods showed that the damage factor due to the total of all accident transient conditions as well as normal operation is less than 0.01.

g. Core Clamp Assembly

The core clamp assembly positions and restrains the moderator element at their tops. There are six core clamps arranged in a hexagonal pattern around the moderator cans. The clamp assembly is mounted on a centering ring which is mounted on a support ledge located on the inside wall of the reactor vessel. Both the clamps and the ring are made from Type 405 SS to match the coefficient of expansion of the moderator element spacers. The clamps have a built-in locking device such that when the clamps are in the closed position the linkage is past dead center and force from the core cannot loosen the clamps. Locating pins and lugs are welded to the centering ring and support ledge to allow for radial expansion but prevent gross movement of the core with respect to the reactor vessel. A slight nominal radial clearance (0.035 in.) between the six clamping mechanisms and the adjacent moderator element spacers ensures that there will be no clamping loads under design operating conditions.

Under seismic loading, the ring will resist the horizontal thrust of the moderator elements. However, the ring will not resist the total seismic design loading (0.05 g, per the Uniform Building Code for Zone I) due to its flexibility. Under the full lateral seismic load (11,000 lb), the ring will transfer the load to the upper ledge of the reactor vessel. The maximum bending stress in the clamping bars from the seismic loading is estimated to be 18,000 psi, which is below the yield stress of 23,000 psi at 1000°F.

Under transient temperature conditions, the thermal growth of the clamping assemblies and the ring may lag the moderator element spacer thermal growth. The maximum temperature difference between the moderator element spacer and the ring occurs during preheating and is 100°F. The maximum combined stress in the ring is estimated as 31,000 psi, which is below the yield stress of 40,000 psi at the preheating temperatures (room temperature to 350°F). Temperature transients during reactor operation will not cause thermal stresses in the ring.

3. Thermal Shields and Thermal Insulation

Thermal shielding is required to attenuate the incident gamma and neutron radiation to prevent the concrete biological shielding from overheating. The thermal shielding is provided by two, 2-3/4-in.-thick concentric cylinders of carbon steel located in the annular space between the reactor vessel and the outer vessel. It is built up from segments which are attached to the outer vessel by "T" shaped brackets running the depth of the vessel longitudinally. These brackets keep the thermal shields in proper alignment while permitting radial expansion.

The thermal shields are non-structural items, their design being governed by shielding requirements. There are no significant thermal stresses within the shields themselves, since the temperature gradients are linear and the shields are not mechanically restrained. The amount of thermal bowing is relatively minor.

Block thermal insulation is mounted on the outside periphery of the outer tank and consists of 12 in. of Superex insulation. The thermal shields and block insulation reduce the energy loss from the reactor and the cavity cooling load.

The top of the cavity is insulated with similar Superex insulation wherever it is exposed to heat radiating from the thermal shields or from the core vessel. Also, there is block insulation between the outer vessel and the cavity liner, in and around the vessel support columns. This insulation is compatible with limited amounts of sodium vapor.

The radial and axial temperature distributions in the thermal shield and thermal insulation are shown in Figures 8 and 9. The maximum temperature is ~ 1030°F and occurs slightly below the top of the active core. There are no significant structural problems in the thermal shields and insulation.

4. Outer Vessel

The outer vessel surrounds the reactor vessel. The side walls of the vessel are 1/2 in. thick and are made of ASTM-A-387 Grade A for resistance to graphitization at high temperature. The bottom head area is ~ 1-1/2 in. thick, and since the temperatures in this area are low, ASTM-A-212 Grade B is used for ease of fabrication. The gap between the reactor vessel and containment vessel is ~ 10 in.; however, the thermal shields occupy 5-1/2 in. of this space. The guard pipes surrounding the primary piping are fabricated from ASTM-A-155 Class 2.

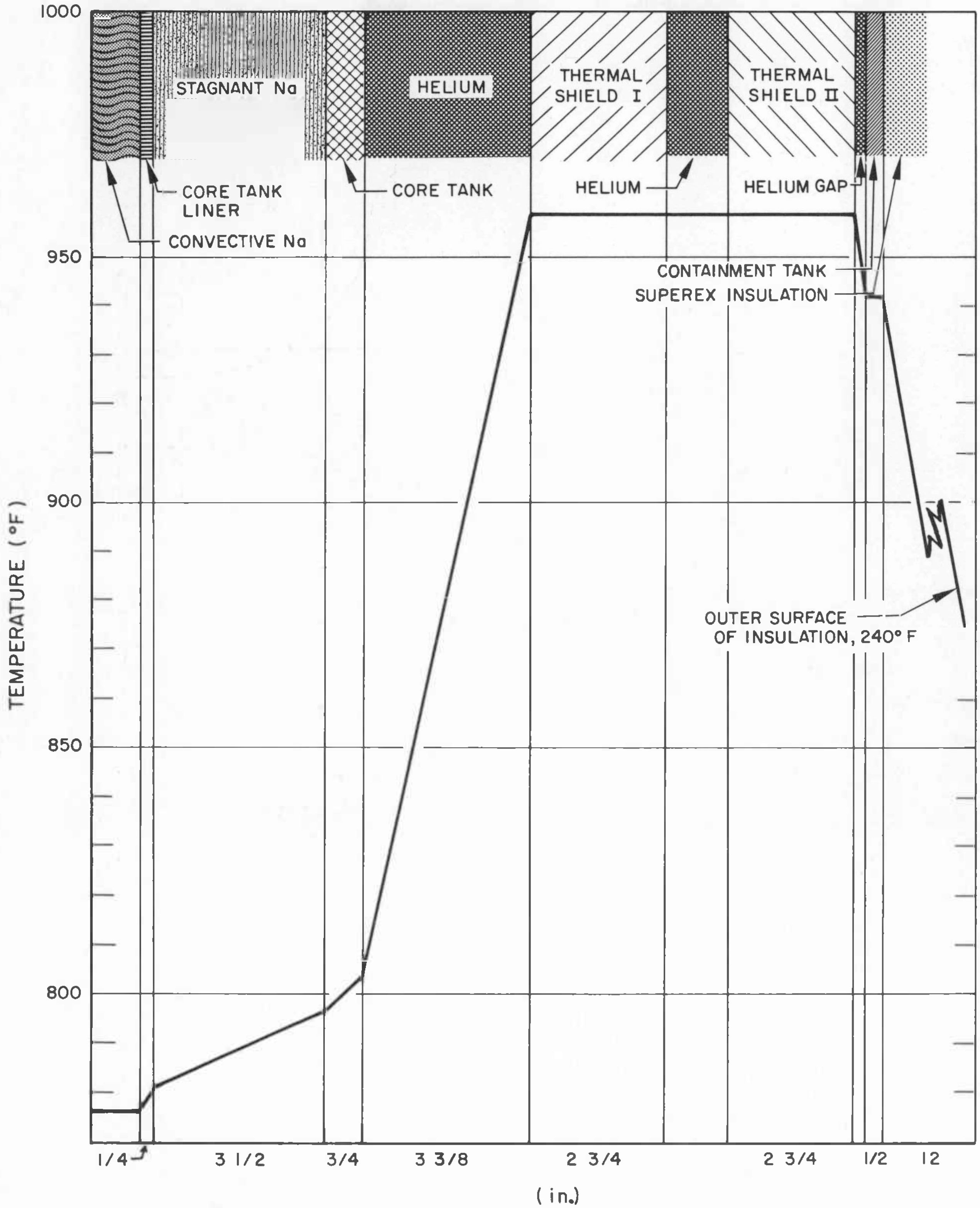

Figure 8. Radial Temperature Profile at Reactor Mid-length

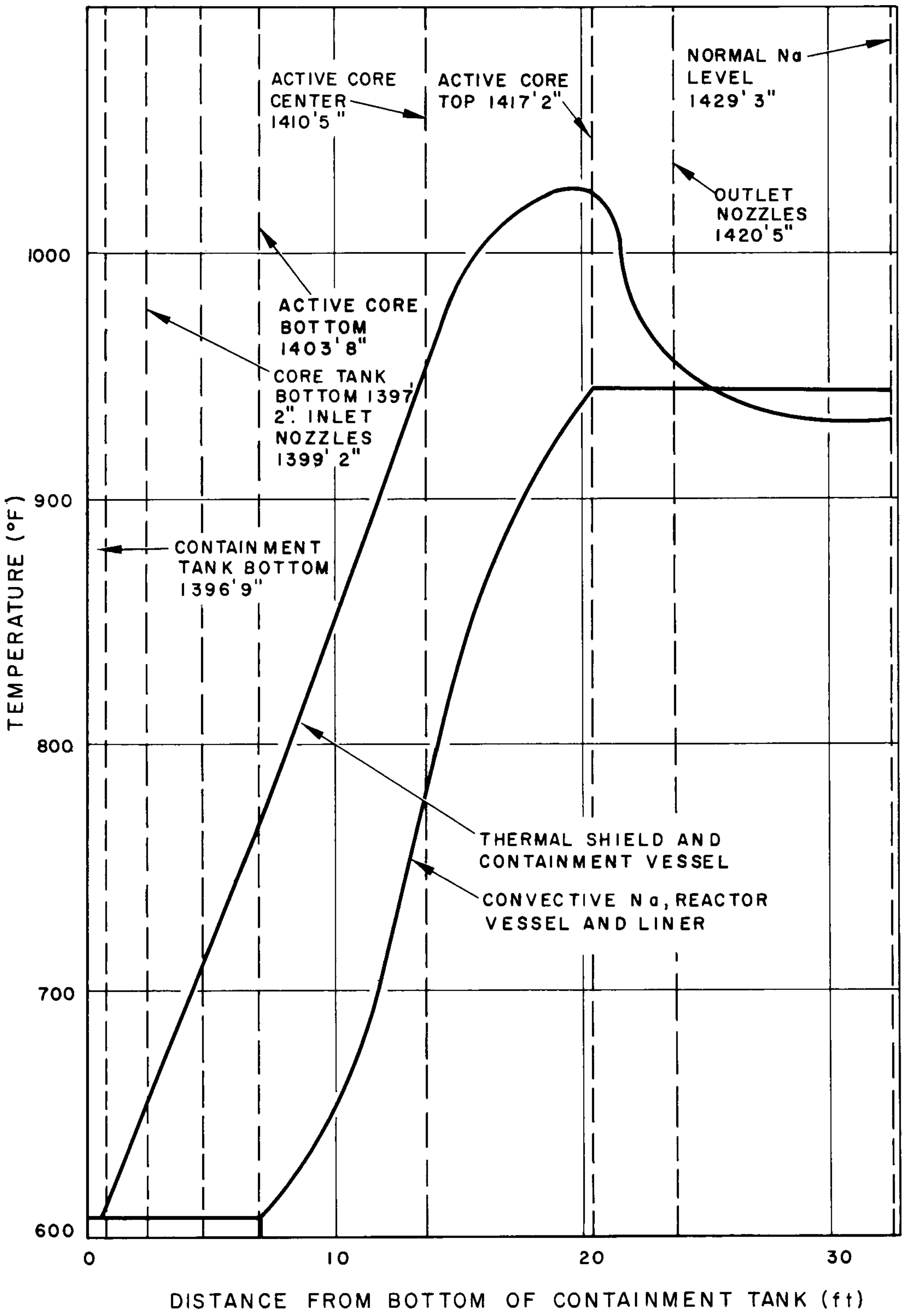

Figure 9. Reactor Structure Axial Temperatures Between Bottom of Outer Vessel and Outlet Nozzles

The mechanical loads on the outer vessel are primarily developed at the lower portion of the vessel. The bottom plate of the vessel transfers the load of the reactor vessel and thermal shields to the reactor support structure. The pedestals of the support structure are located so that the reactor vessel support pads are not in direct alignment. Consequently, bending stresses of 4500 psi are developed in the plate. The allowable bending stress for the material (ASTM-A-212 Grade B) at 950°F is 6750 psi.[4]

The radial and axial temperature distributions in the outer vessel are shown in Figures 8 and 9. The maximum temperature in the outer vessel is ~ 1030°F and occurs slightly below the top of the active core.

The loads acting on the joints between the guard pipes and outer vessel are mainly the radial loads and the longitudinal and circumferential moments. Since the magnitudes of the loads are different at each nozzle, a different stress is applied to each nozzle. The outer vessel may also be subjected to earthquake loading. This will have an effect on the stresses at the joint. In order to take into account the effects of an earthquake on the containment vessel, an acceleration of one g was used. This is considerably in excess of the value of 0.05 g required by the Uniform Building Code.[9] The maximum stress due to the combined effect of the radial load, longitudinal and circumferential moments, and earthquake loads occurs at the joint of the outlet guard pipe to the outer vessel and is 2,800 psi. The allowable bending stresses are based on 1-1/2 times the membrane stress allowable given in the ASME Boiler Code,[2] or 6,750 psi at 950°F.

During normal operating conditions, the outlet nozzles are at ~ 950°F, while the inlet nozzles are at ~ 650°F. The plate material of the outer vessel is SA-387GRA. The allowable bending stress for SA-387GRA, at 950°F is 10,000 psi, while at 650°F, the allowable bending stress is 16,250 psi. Thus, the stresses developed at the joint of the guard pipes and the outer shell are within the allowable.

The outer vessel is also subject to lateral loads that can result from two entirely separate causes. The first is due to a difference in friction forces between the supports and the bottom head of the vessel; the second is due to earthquake loading. The earthquake loads were computed from the coefficients of the Uniform Building Code[9] for Zone I. The maximum lateral load results from the earthquake loading.

Seismic loads also affect the design of the keys and keyways, the walls of the outer vessel, and the thermal shield guides. The walls of the outer tank must absorb the imposed energy loads due to movement of the thermal shields during an earthquake. The flanged T sections retaining the thermal shields are designed to minimize the impact of the thermal shields on the reactor vessel.

Thermal stresses in the outer vessel are developed from two causes. The first is due to a difference in the average temperature of the bottom head and the average temperature of the cylindrical wall. Inasmuch as the difference in average temperatures is ~ 50°F at steady-state conditions, the thermal stress levels are not appreciable. The second condition which results in a thermal shock is due to sodium spillage into the vessel. The vessel is adequate to contain the sodium in the event that the outer vessel filled with sodium at 945°F. The total maximum stress due to the hydrostatic head of sodium is ~ 7900 psi. The vessel is also adequate to resist the thermal shock if this sodium at 945°F is dumped into the annulus between the two vessels instantaneously.

Resistance heaters are located directly below the outer vessel head and are supported by a polished SS reflector sheet which lies on top of the lower support structure insulation. The bottom heaters are used during preheating and to maintain the reactor core and structure temperature at or above 350°F during any prolonged shutdown of the reactor. There are 24 active and 12 spare heaters provided which are rated at 7.5 kw each and will be operated at one-third their rated power.

5. Reactor Support Structure

The total heat generation distribution through the reactor support structure is shown in Figure 10.

The maximum temperature in the support columns during normal operation is 600°F and the average, ~ 375°F. The temperature distribution in the support structure is shown in Figure 11.

The mechanical loads on the support pedestals vary from 25,000 lb at the outermost pedestals to 11,000 lb at the inner ones. The outer pedestals carry a greater load because they must support the weight of the thermal shields and insulation, as well as assist in supporting the weight of the reactor vessel.

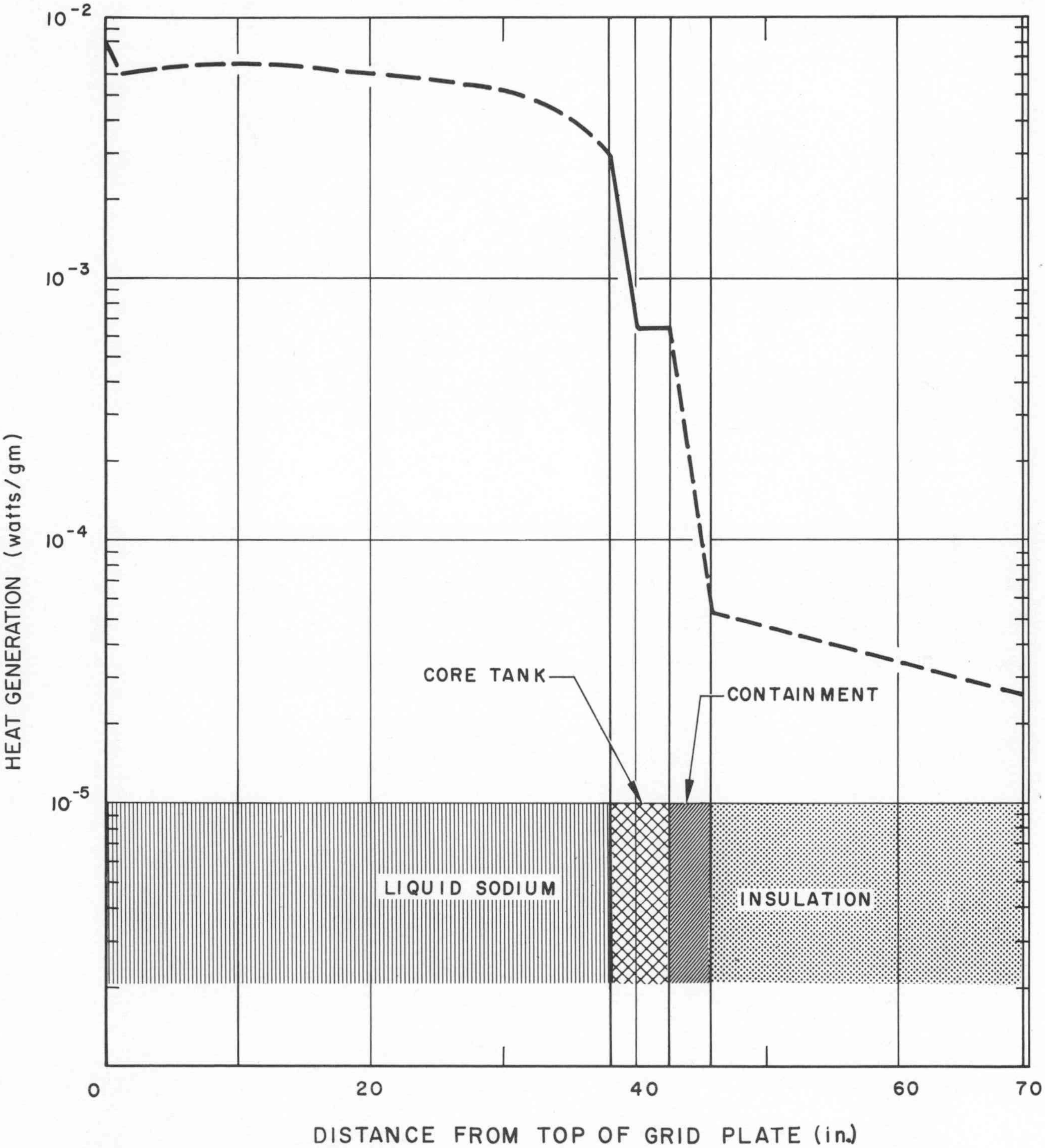

Figure 10. Nuclear Heat Generation in Reactor Support Structure

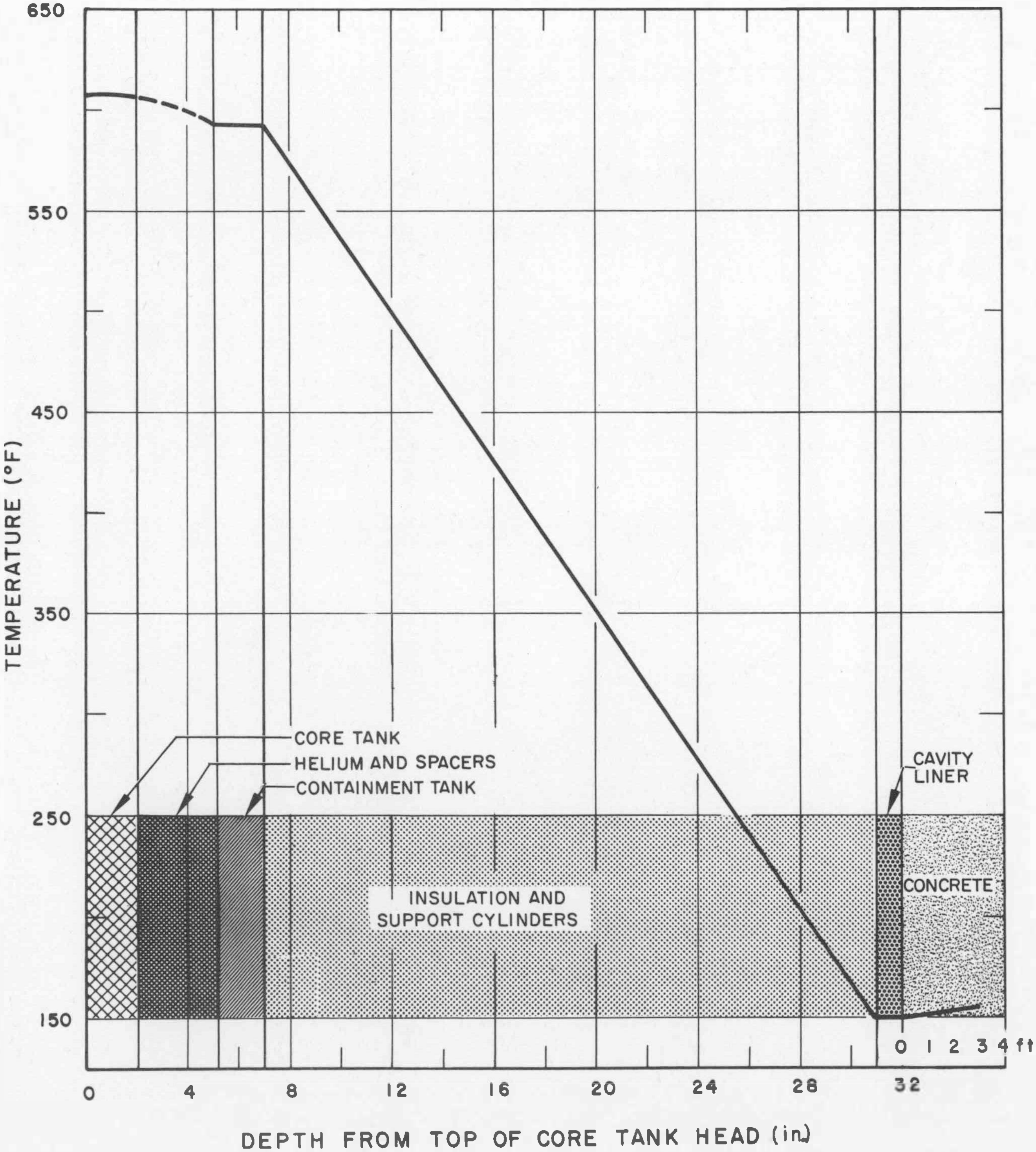

Figure 11. Steady-State Temperatures through Lower Support Structure

Frictional forces are developed because of the relative expansions and contractions of the reactor and containment vessel. These lateral forces induce additional stresses in the support pedestals. The stress level in the pedestal will vary according to the temperature. As the temperature of the components in the reactor cavity increases, the expansion of the reactor vessel relative to the support pedestals decreases the eccentricity of the applied load. The maximum mechanical stress occurs at the base of the pedestal when the reactor is at 350°F and is equal to 12,300 psi. During normal operation, the maximum mechanical stress is approximately 12,000 psi.

During normal operation, the maximum thermal stress occurs at the base of the pedestal and is equal to 9200 psi. This is due to the linear thermal gradient from 607°F at the top of the shim plate to 150°F at the base of the support pedestal. The maximum thermal stress at the base of the support pedestal during the accidental transient condition where the reactor vessel lower plenum changes from 607 to 945°F is equal to ~ 16,200 psi. This is the maximum thermal stress.

During normal operation, the combined thermal and mechanical stresses are equal to 21,500 psi. The allowable stress for SA-106 Grade B below 650°F is 22,500 psi.

The stresses developed in the support pedestals after the accident transient condition resulted in a total stress level of 25,500 psi. These stresses are higher than the allowable value of 22,500 psi but are not serious because of the very low thermal fatigue damage factor.

6. Reactor Cavity Liner and Reactor Cavity Diaphragm Seal

The reactor cavity liner provides the final containment barrier between the reactor and the atmosphere. In the event of a sodium leak between the reactor vessel and the outer vessel the sodium level in the reactor will drop ~ 3 ft which is about 9 ft above the top of the active core. The sodium can be brought up to its original level by the addition of ~ 500 cu ft of sodium from the primary fill tanks. If the outer vessel could not contain the sodium, the level in the reactor would drop to about 2 ft below the top of the active core. However, the addition of the sodium in the primary fill tank would raise the level back up to a point ~ 2 ft above the top of the active core.

The cavity liner heat load is 535,000 Btu/hr. Included is 90,000 Btu/hr for internal heat generation in the liner and the concrete.

The maximum concrete temperature will be 175°F with a nominal maximum of 150°F. The maximum concrete temperature with the loss of one coolant line is 180°F. The average cavity liner temperature under normal operating conditions is 145°F. With the loss of one coolant line this average temperature rises to 160°F.

The bottom plate of the cavity liner transfers the weight of the reactor complex into the concrete foundations. The loading condition on the bottom plate consists essentially of the forces and moments applied through the support pedestals. Because the support pedestals do not anchor into the concrete foundation directly, the cavity liner plate is more highly stressed. The maximum stress in the bottom plate of the cavity liner is ~ 15,000 psi. This compares with the allowable value of 20,000 psi.[4]

The gallery diaphragm seal is designed for an internal pressure of 10 psi with the outlet pipes operating at 1000°F and the inlet pipes at 607°F. The maximum stress due to mechanical loading occurs at the junction between the dished head and the skirt attached to the gallery liner. This stress is equivalent to a normal stress of 18,700 psi, which is less than the allowable bending stress of 20,600 psi. Thermal stresses due to temperature gradients and dissimilar materials were analyzed using the cumulative damage hypothesis as the governing criteria. The maximum cumulative damage occurs at the joint between the guard pipe and the transition cone to the primary pipe. The damage factor in this area is equal to 0.268, which is approximately one-third of the limiting value of 0.8.

7. Core Preheaters

The core preheaters consist of the following:

a) 12 moderator preheaters which have 76.3 watts per foot of length in the moderator element region.

b) 15 moderator and lower plenum preheaters which have 76.3 watts per foot in the moderator region and extend into the lower plenum with 320 watts per foot.

c) 6 moderator and upper plenum preheaters which have 76.3 watts per foot in the moderator region and 490 watts per foot in the upper plenum region.

The preheaters are arranged symmetrically in the core with a total of 42.8 kw in the moderator region, 9.6 kw in the lower plenum, and 29.3 kw in the upper plenum.

IV. FABRICATION AND FIELD ERECTION

A. GENERAL

During the preliminary design phase of the project, one objective was to arrive at a design which would allow considerable latitude for fabrication. The intent was to effectively utilize available fabrication equipment and experience, thereby obtaining a high-quality, low-cost product. It was deemed desirable to obtain a single fabricator for the reactor structure, excluding the cavity liner, who would be responsible for shop fabrication, transportation, and field installation.

Atomics International performed preliminary surveys regarding the transporting of the major components, based upon these items being completely shop fabricated. The major items were vessels 20 to 21 ft in diameter, and 35 ft in length. This investigation revealed that these components could readily be transported. River barges could be used for transportation up the Mississippi and Missouri Rivers, to Plattsmouth or Omaha, Nebraska. Then movement by trucks, using secondary roads to Hallam, Nebraska.

With the above objectives in mind, a preliminary design package was prepared. Trips were made to potential fabricators. The design features of the structure were reviewed and comments requested regarding potential fabrication and field problems. Potential fabricators were also surveyed to determine their capabilities as to shop fabrication, transportation, and field installation. The basis for qualifying a capable fabricator was as follows:

1) High quality tank fabrication capability was desired, preferably with some experience on nuclear work.
2) The fabricator must be capable of performing accurate machining on large diameters, either in their own shop or by subcontracting to some other local shop.
3) A stress relieving facility either existing or to be provided was required.
4) Handling and shipping facilities and experience with large components was required.
5) Field erection capabilities and experience was required.

In general, this survey established the soundness of the design and assured the feasibility of obtaining a number of sound, competitive bids.

The following schedule was established during the preliminary design phase:

Design	2/1/59	11/1/59
Material Procurement	7/1/59	6/1/60
Shop Fabrication	4/1/60	10/1/60
Field Installation	6/1/60	1/1/61

Assurance was obtained that the total time allowed for shop and field work was sufficient. The phasing of work would be determined after selection of a particular fabricator.

A design specification was prepared for the cavity liner establishing pertinent design requirements. This specification was transmitted to Bechtel Corporation, who in turn, prepared the final design drawings and specifications for procurement of this item. This procedure was established so that Bechtel could coordinate the field installation of this component with the concrete work surrounding the reactor structure which was also their responsibility.

Henry Pratt Company was selected as the cavity liner fabricator. Because of the critical requirements established for this component, it was deemed necessary that Atomics International perform all quality control surveillance in the shop and field. All welds were 100% radiographed, dye penetrant and helium leak checked. The final acceptance of this component was a 24-hr pressure decay test. The leakage is limited to one cubic ft in 24 hr when pressurized to 2-1/2 psi.

As final design started, we faced the first and most serious problem of the project. Most reliable sources were predicting a serious steel strike for the period in which we anticipated our major material procurement effort. A decision was reached to expedite the final design package and it was released for bids in January 1959. The fabricators bidding on the project were requested to immediately place a material order with the mills. The bid return date was established so that the unsuccessful bidders could be notified and cancel their material orders without charges. The above procedure was carried out and most of the major steel requirements were obtained before the strike. The only material not obtained before the strike was some Type 405 for the fuel plug sleeves in the loading face shield. Consequently, the loading face shield was delayed in shop fabrication and in delivery to the site.

The following schedule was developed with Baldwin Lima Hamilton Corp., the successful bidder for the reactor structure:

Material Procurement	2/1/59	
Shop Fabrication	4/1/59	2/1/60
Field Installation	4/1/60	12/12/60

With delays that arose from material, and other problems, the project was completed on the following schedule:

Shop Fabrication	4/1/59	1/1/61
Field Installation	4/1/60	10/1/61

This schedule would appear to have delayed the project by approximately 10 months. This was not the case. When it appeared that the loading face shield was to be late in delivery, a change in sequence of work was made. The moderator and reflector elements were installed prior to final placement of the shield, thereby accomplishing the overall combined scope of work within schedule.

B. SHOP FABRICATION

The major fabrication of the various components was almost entirely accomplished in the Baldwin Lima Hamilton Shop. All critical fit-ups were made in the shop, in order to minimize the field work. To provide some understanding for the scope of work, the following is a description of the size, weight, and some tolerances of the major components which were shop fabricated:

1) The reactor vessel with internals weighed 78 tons, its shipping weight without internals is 60 tons. The diameter is 19 ft held to ±1/4 in. The length is 33 ft 2 in.

2) The outer vessel weighs 50.5 tons. The diameter is 20 ft 8 in. held to ±1/4 in. The length is 35 ft 4 in.

3) The thermal shield panels were ~ 6 ft wide by 12 ft long by 2-3/4 in. thick. The total weight of the shield panels is 225 tons.

4) The loading face shield as completed in the shop without shielding concrete weighed 60 tons. The maximum diameter at the top was 21 ft. The clearances for fit into the cavity liner were held to 1/8 in. The overall height is 5 ft 8 in.

A number of photographs, Figures 12 to 21, are included to show the various components in work.

The shop work in general moved along in a very satisfactory manner.

The major problems were related to the loading face shield. The first, as mentioned previously, was due to the inability to obtain suitable Type 405 material for the fuel plug sleeves. Type 304 SS was substituted for the Type 405. The second was in developing a procedure for machine welding of the sleeves to the loading face shield. The machine weld procedure was abandoned and all welding had to be performed by hand. These two items caused a major delay in the shop schedule.

The last component to be released from engineering was the reactor vessel bellows. Atomics International placed a separate contract for this component. Uniflex Corporation, of Los Angeles, was the successful bidder.

All of the material for the bellows was ultrasonically inspected, all welding was dye penetrant inspected and where possible, radiographed. The final assembly was helium-leak checked. The major problem encountered was in performing the final helium leak check, due to the difficulty of securing a seal in order to obtain a full vacuum on the bellows for helium probing.

C. TRANSPORTATION

All of the components fabricated by Baldwin Lima Hamilton, excluding the loading face shield, were ready for shipment in March of 1960. A route survey was made to determine the best route from the Missouri River to Hallam. Plattsmouth, Nebraska, was selected for unloading the components and a contract was let for trucking the components to Hallam.

The components at Baldwin Lima Hamilton were cocooned and loaded on a barge. The trip by barge was made down the inland water-way, across Florida, and up the Mississippi and Missouri Rivers. The barge was pulled onto the bank at Plattsmouth and the components unloaded on specially designed trailers. The overland trip to Hallam was approximately 80 miles.

As the caravan approached the end of the first day's travel, the vehicle transporting the reactor vessel upset and dropped the vessel. The vessel broke from its skid and rolled ~ 350 ft into a cornfield. A cursory review was made and there was no obvious damage to the vessel. An intensive effort was then made to plan and execute its recovery. The reactor vessel arrived at the Hallam site ~ 3 weeks after the accident occurred. Figure 22 shows the vessel in transit, the reinforcing of one of the many bridges en route, the reactor vessel, and the reactor vessel transport skid in the cornfield.

In order to be sure the vessel had suffered no significant damage, an intensive program of visual, dimensional, dye penetrant, radiography, and helium leak testing was undertaken as soon as the vessel was set up within the reactor building. This investigation showed that no damage was incurred from the accident, which would in any way affect the integrity or functional requirements of the reactor vessel. This work was performed by Pittsburg Testing Laboratory.

The shop fabrication of the loading face shield was completed in December 1960. The shield was shipped in February 1961. This delay was necessary since the Missouri River was not expected to be navigable until March 1961. Transportation was over the same route as previously established and the shield arrived at the Hallam site in April 1961. Figure 23 shows the shield being unloaded at the Hallam site.

Transportation of the reactor bellows from Los Angeles to the Hallam site was accomplished by railroad. This required a special low-load car to obtain the necessary clearances. No difficulties were encountered.

D. FIELD INSTALLATION

The work in the field proceeded in a slow but successful manner. There were only a few major problems.

The main 60-ton building crane had to be used for all of the installation. The major lift was 75 tons and consisted of the outer vessel with approximately half of the insulation installed on the exterior surface. The insulation was installed on the vessel before placing it into the cavity liner, because there is only a 7-in. radial gap between the insulation and the cavity liner wall. The thermal shield panels caused some trouble on installation. These had not been fitted in the shop and some rework had to be done to obtain the proper fit up.

The reactor vessel bellows caused the most serious field problem. In making the closure weld between the lower angle on the bellows assembly and the reactor vessel, the weld shrinkage caused the angle to rotate. This rotation produced considerable distortion in the lower two convolutions of the bellows. An extensive examination was performed, this included a complete nondestructive inspection, as well as additional analytical analysis. The conclusion of this effort was that the bellows in its existing condition would satisfactorily perform its function.

Due to the problem encountered with the loading face shield, it was not on hand when required for the alignment of the upper cavity liner with the grid plate. The alignment was accomplished instead from shop as-built information. The upper cavity liner was welded to the cavity liner and concrete shield poured around it before arrival of the loading face shield. Upon arrival of the loading face shield it was placed into the upper cavity liner. The N-S and E-W centerlines were checked with the reactor grid plate. These centerlines coincided within 0.015 in. The shield was rotated 360° and rechecked. The centerlines zeroed in on the original dimensions. The empty shield was then removed and placed in an adjacent area for the placement of concrete shielding.

After placement of the concrete shielding in the moderator removal plugs and the loading face shield, it was discovered that the required ovality tolerance of the moderator removal plugs had not been held. Machines for grinding the bore of the sleeves and the outside diameter of the plugs were built in the field. These were used to produce the necessary fit.

The last and largest of the assembly problems occurred at the very end of the field erection work. The completed loading face shield had been rolled over the reactor and lowered into place. The final major operation was to rotate the shield to prove clearances and to check the rotation system. The shield could not be rotated through a full revolution and had to be removed. Extensive measurements were then taken to find possible points of interference and to check the axis of revolution. The special boring tool shown in Figure 24 was used to machine the upper cavity liner to a diameter which would return the 1/8 in. radial clearance.

The loading face shield was replaced and the flange which contacts the guide rollers was ground to a runout of only 0.009 in. When the guide rollers were replaced the shield rotated without difficulty.

As mentioned before, the fabrication and installation of all components was performed according to the best known practices, employing all applicable means of non-destructive testing. Due to the vessel configuration and unusual pressure requirements, it was impossible, practically, to perform a proof pressure test on the completed component; this type of test, therefore, was not performed. Atomics International maintained close surveillance of all operations performed in the shop and field. The basic reason for maintaining this high degree of quality in material and workmanship was to assure complete confidence with regard to primary vessel integrity.

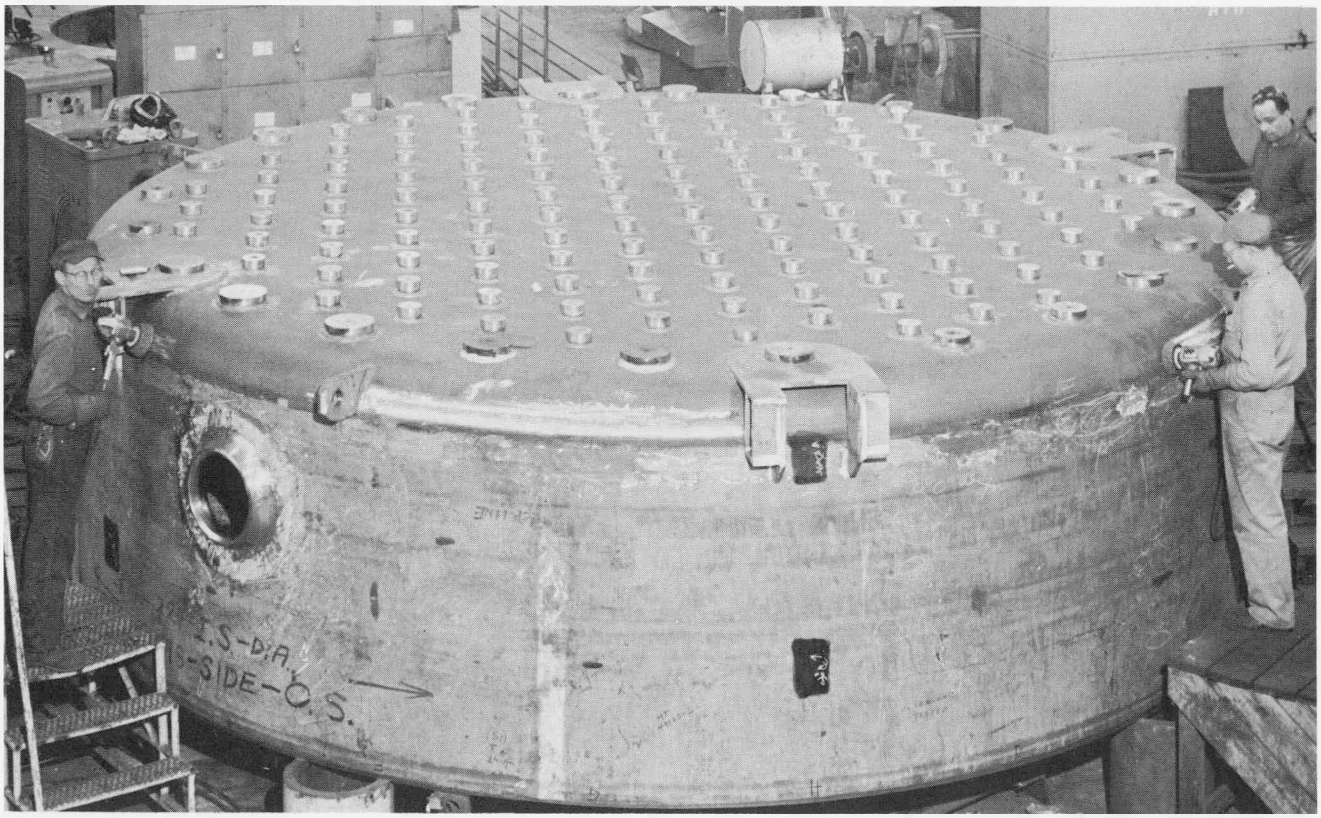

Figure 12. Reactor Vessel, (lower section, 2 in. thick, 228 in. OD, Type 304 SS, showing one sodium inlet nozzle, keyways, and support bosses)

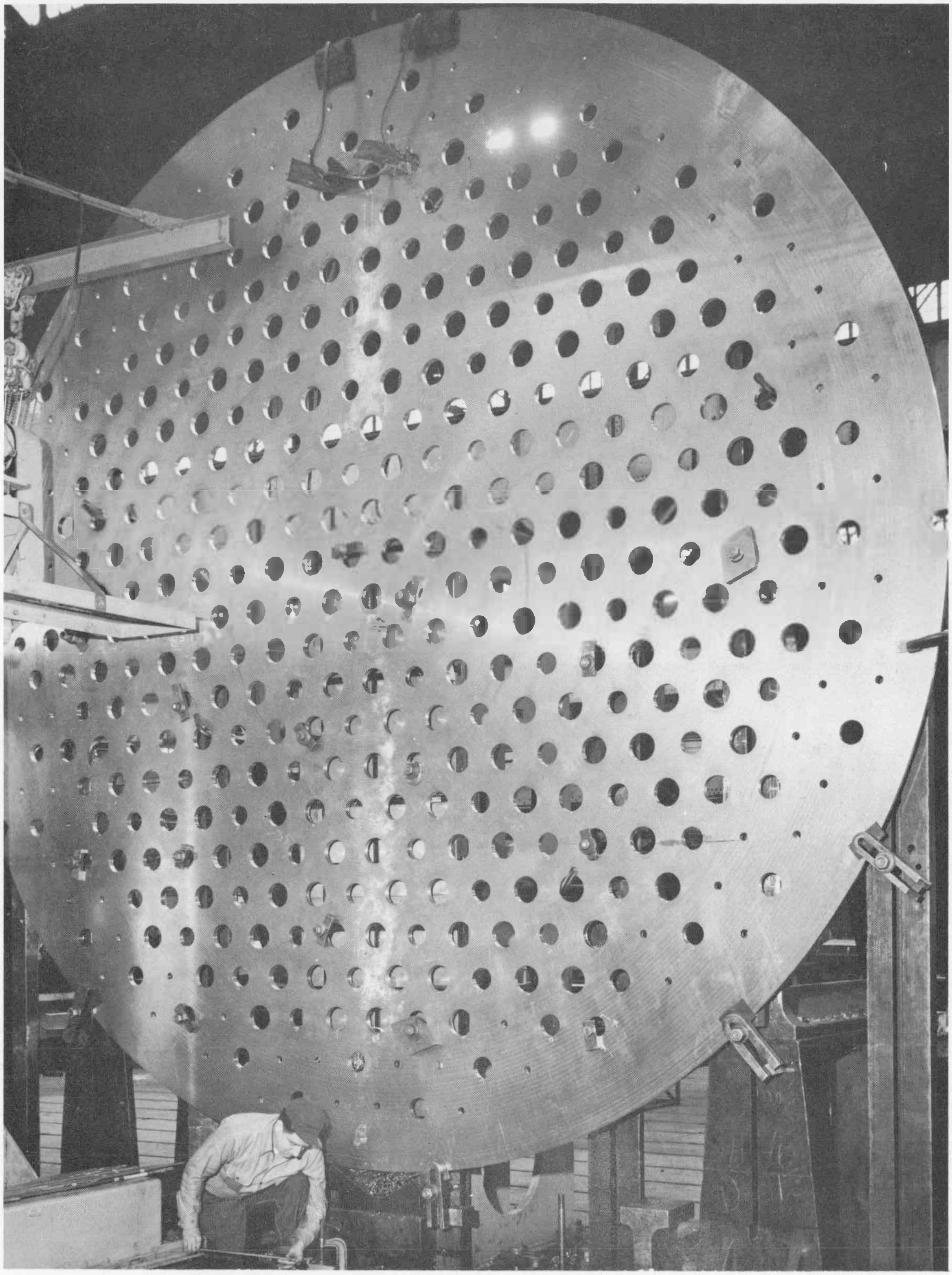

Figure 13. Grid Plate, Finished to 1-in. -Thick, Type 304 SS (Shown at horizontal boring mill for machining of holes for the fuel element and moderator can supports)

Figure 14. Grid Plate with 141 Moderator Can Supports Placed Preparatory to Securing the Supports to the Grid Plate (Other holes are for process tube nozzles.)

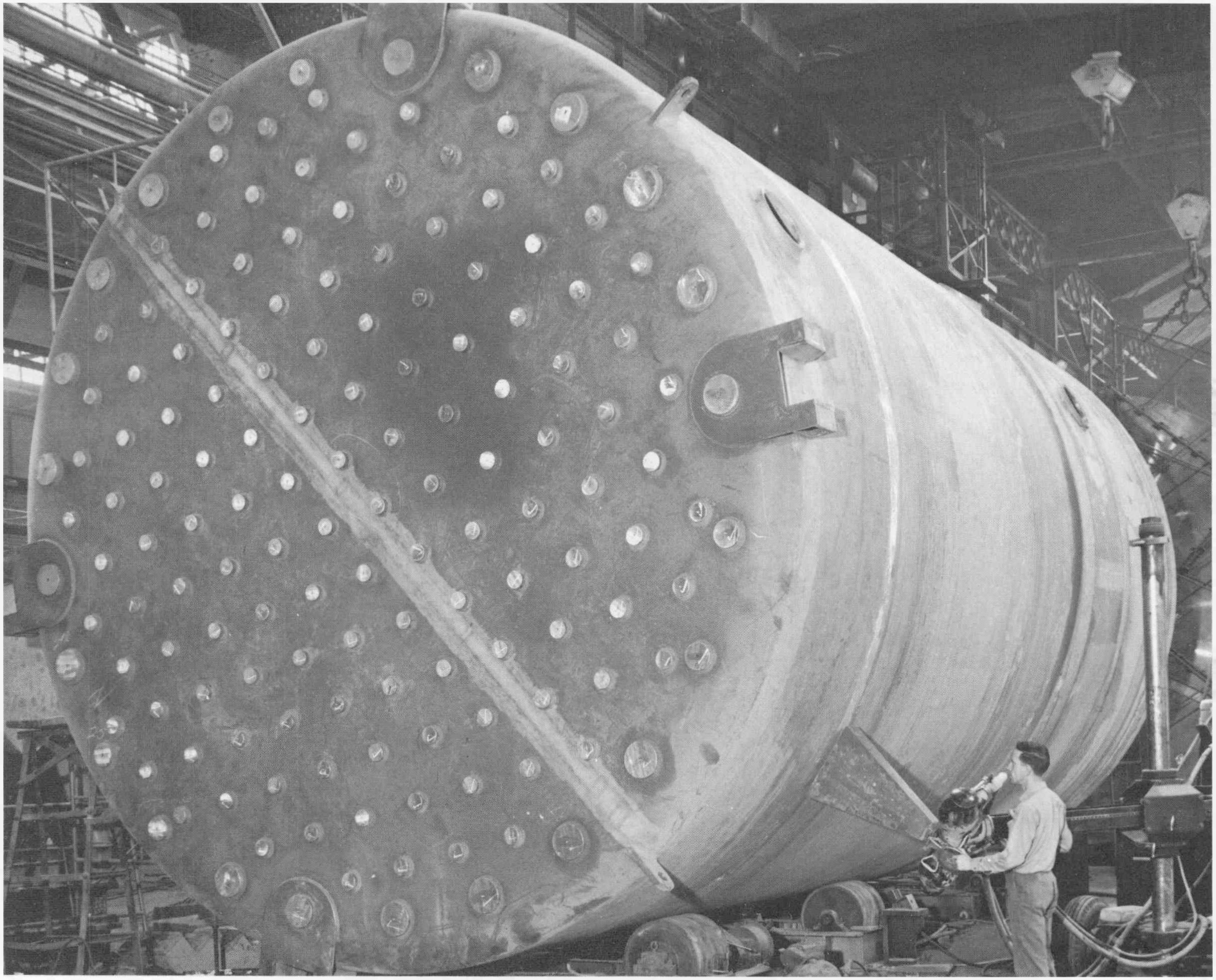

Figure 15. Reactor Vessel of Type 304 SS (View during radiographing of final weld joints, showing support pads and keyways on the bottom of the vessel.)

Figure 16. Containment Vessel Showing 2-in. Bottom Plate and Lower Shell Section. (ASTM - A-212 grade B steel)

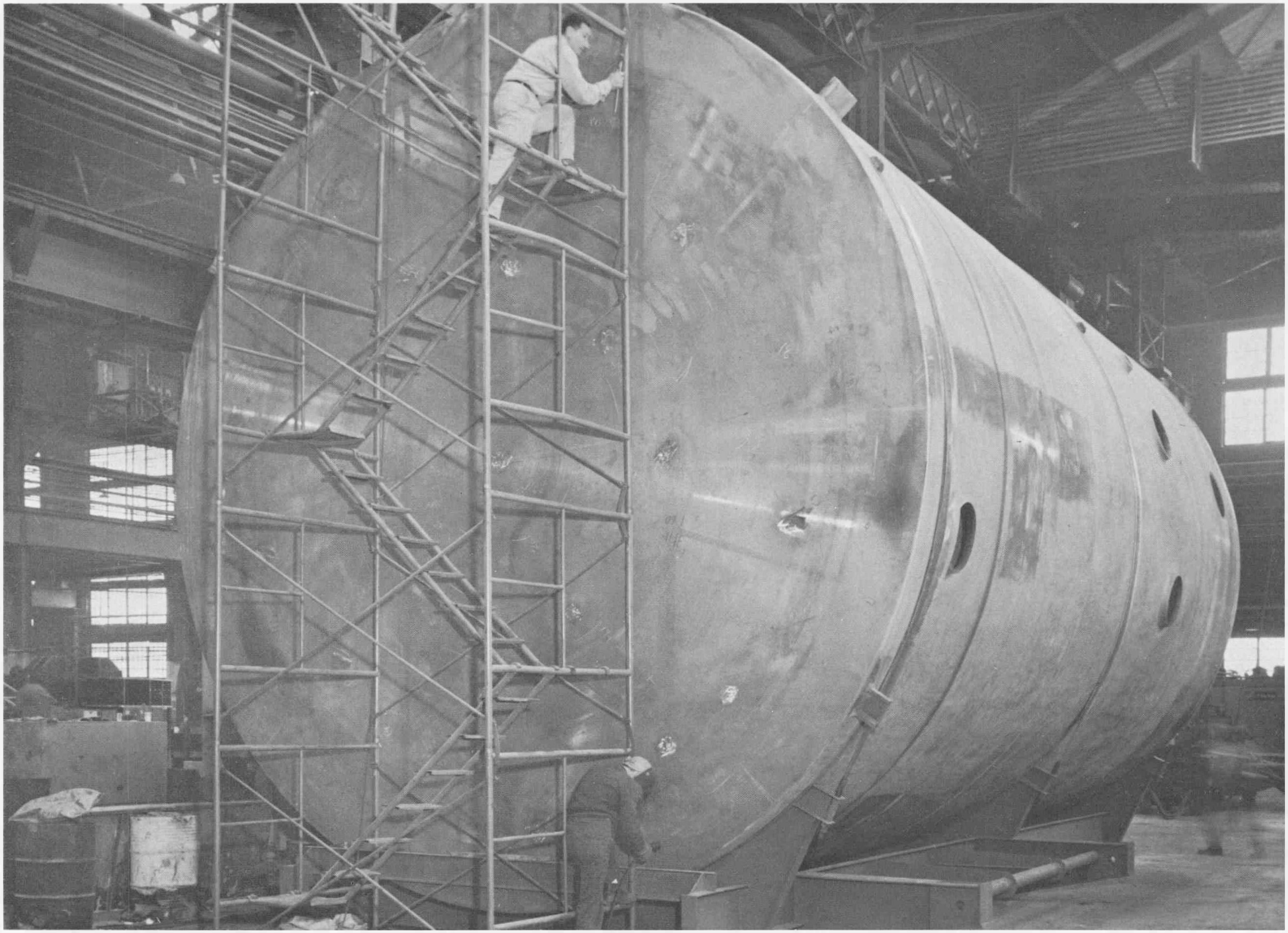

Figure 17. Containment Vessel (View from bottom showing installation of vessel on shipping skid. Vessel is 20 ft 9 in. OD and 35 ft 4-1/2 in. high.)

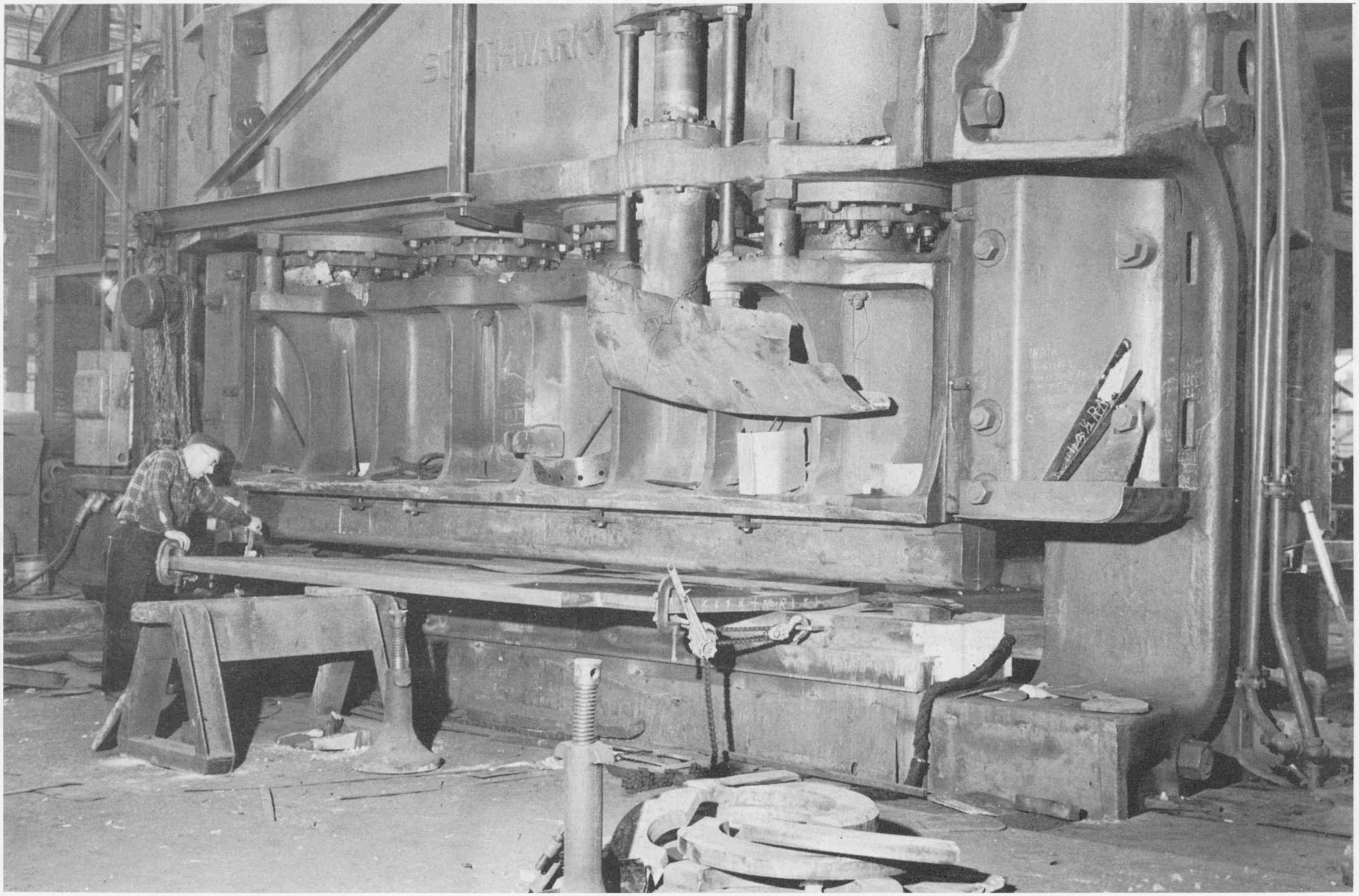

Figure 18. Reactor Thermal Shielding (2-3/4-in. -thick plate to ASTM spec. A-212 Grade B. This is one of about 75 plates with a total weight of over 200 tons. The radius is being formed in a press brake.)

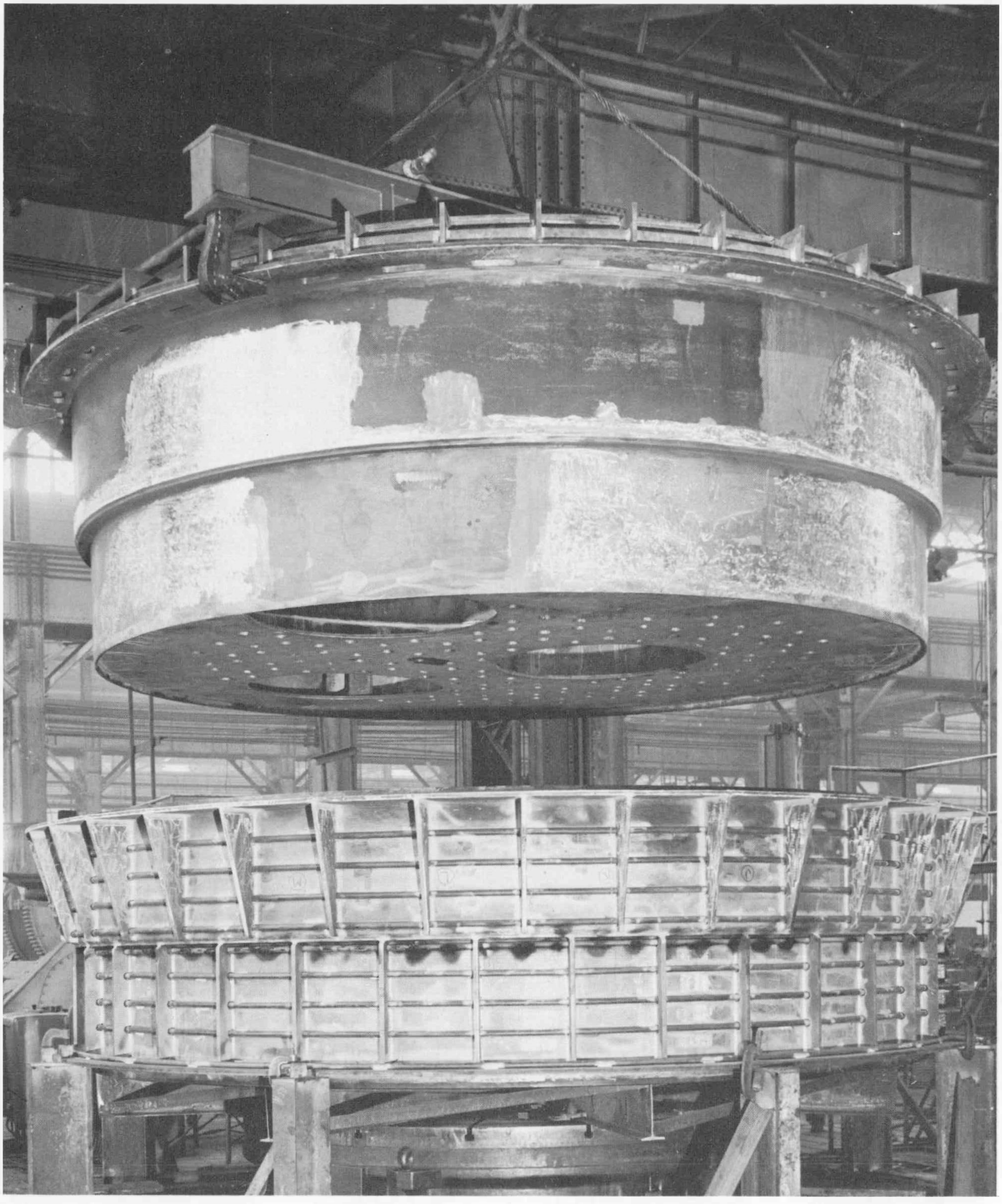

Figure 19. Upper Cavity Liner (Type 405 SS shown with the loading face shield shell being lowered in place to check the fit)

Figure 20. Reactor Vessel Bellows (welding of bellows convolutions in shop)

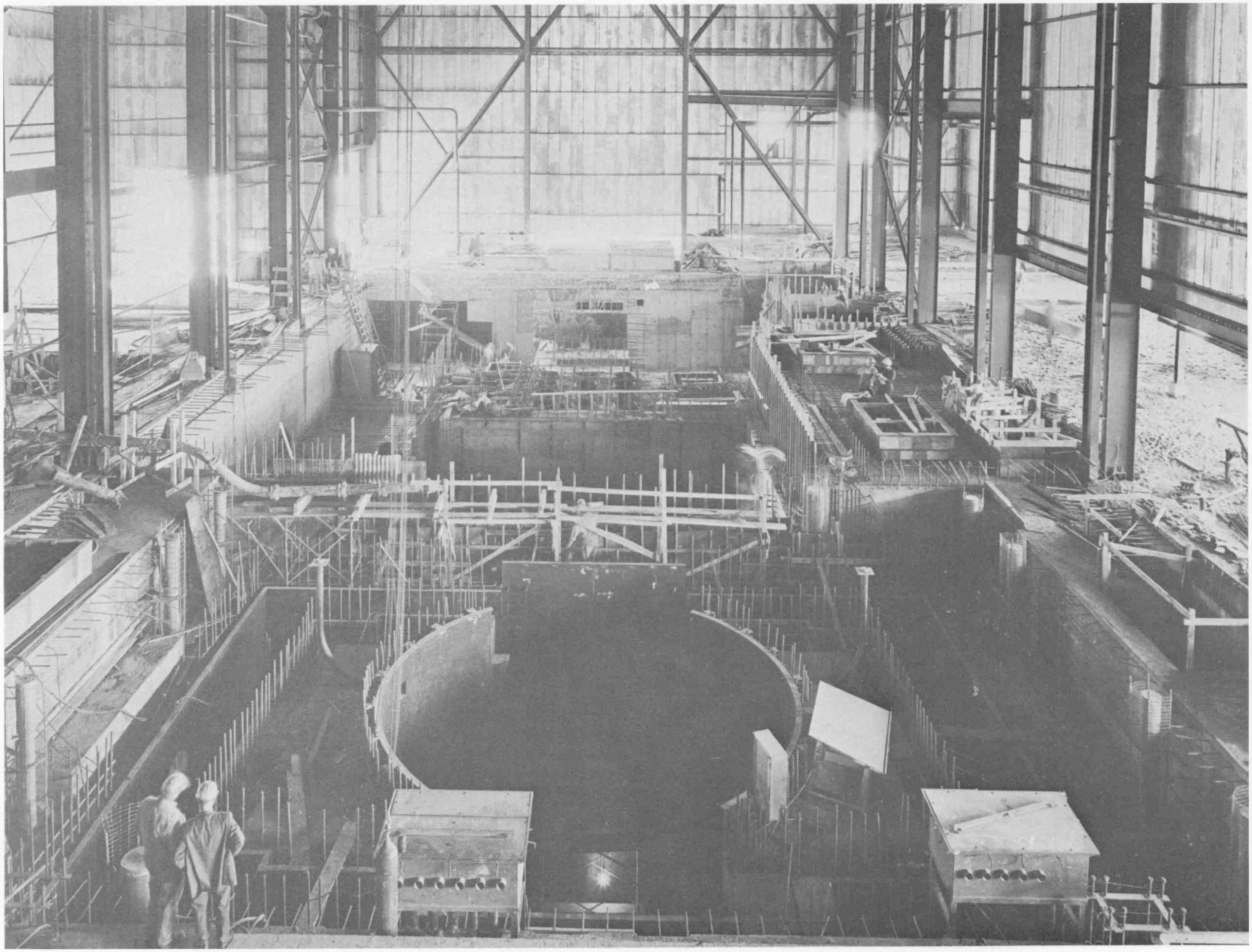

Figure 21. Cavity Liner (prior to reactor structure installation)

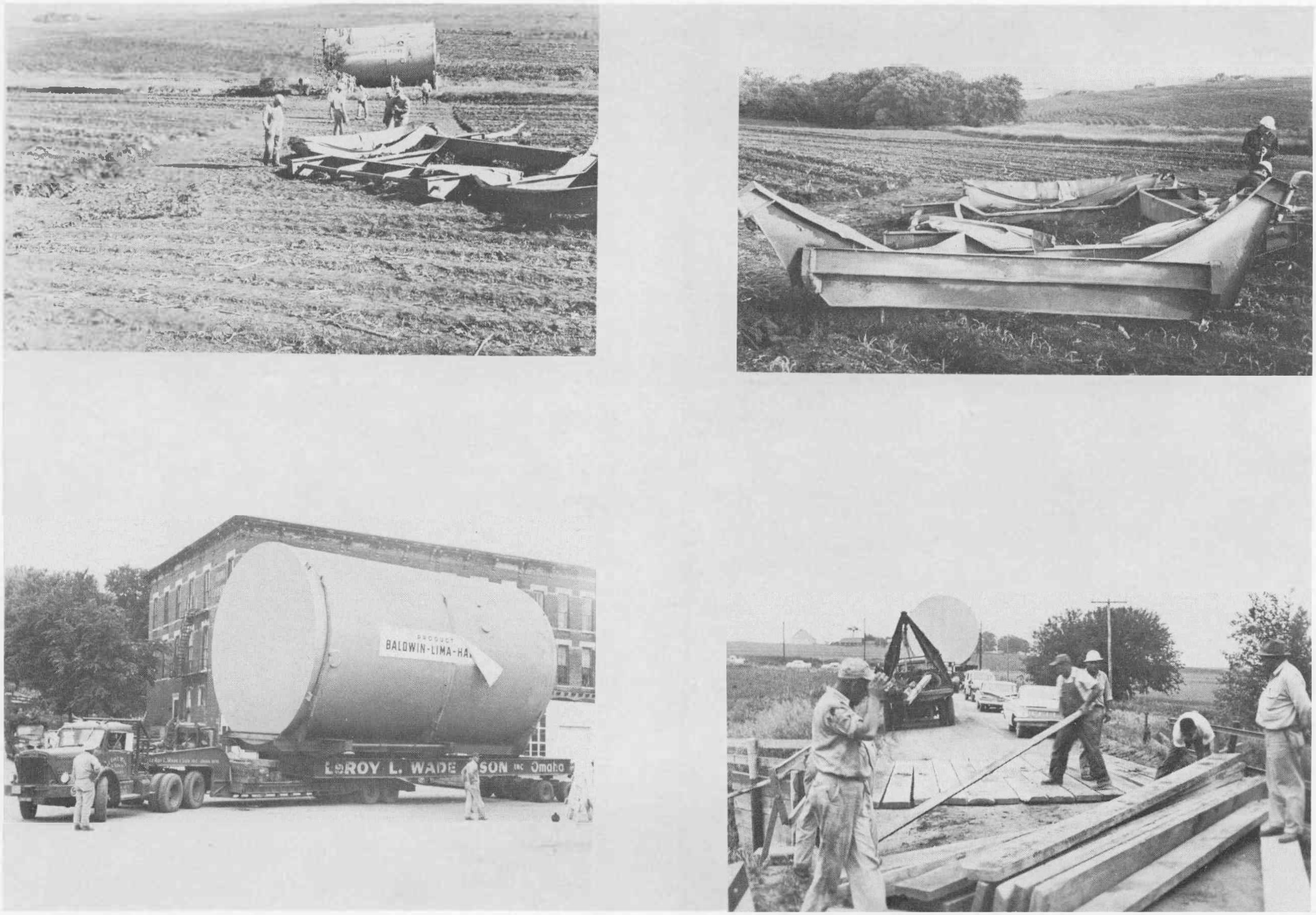

Figure 22. Reactor Vessel Transportation

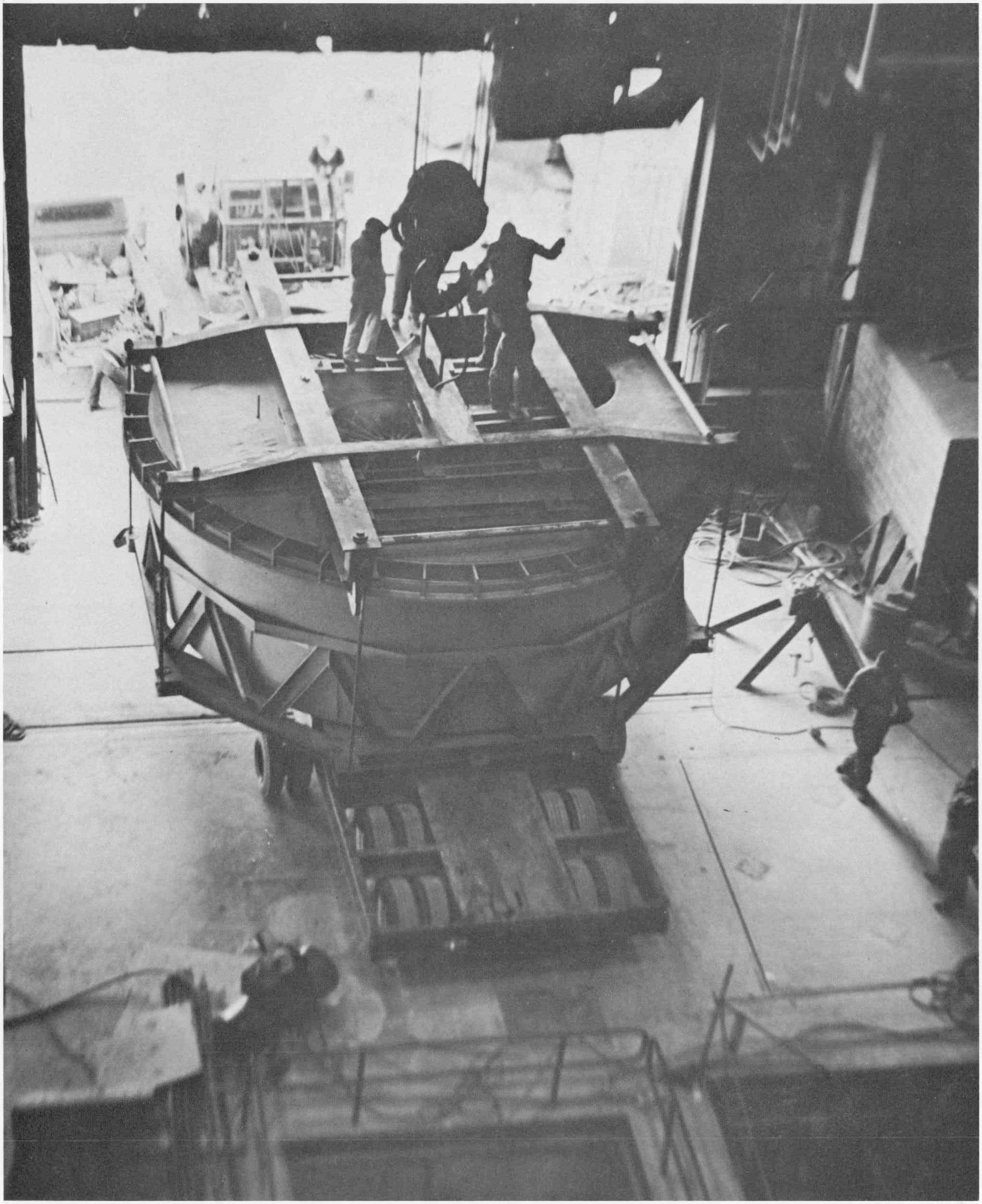

Figure 23. Removing Loading Face Shield From Trailer

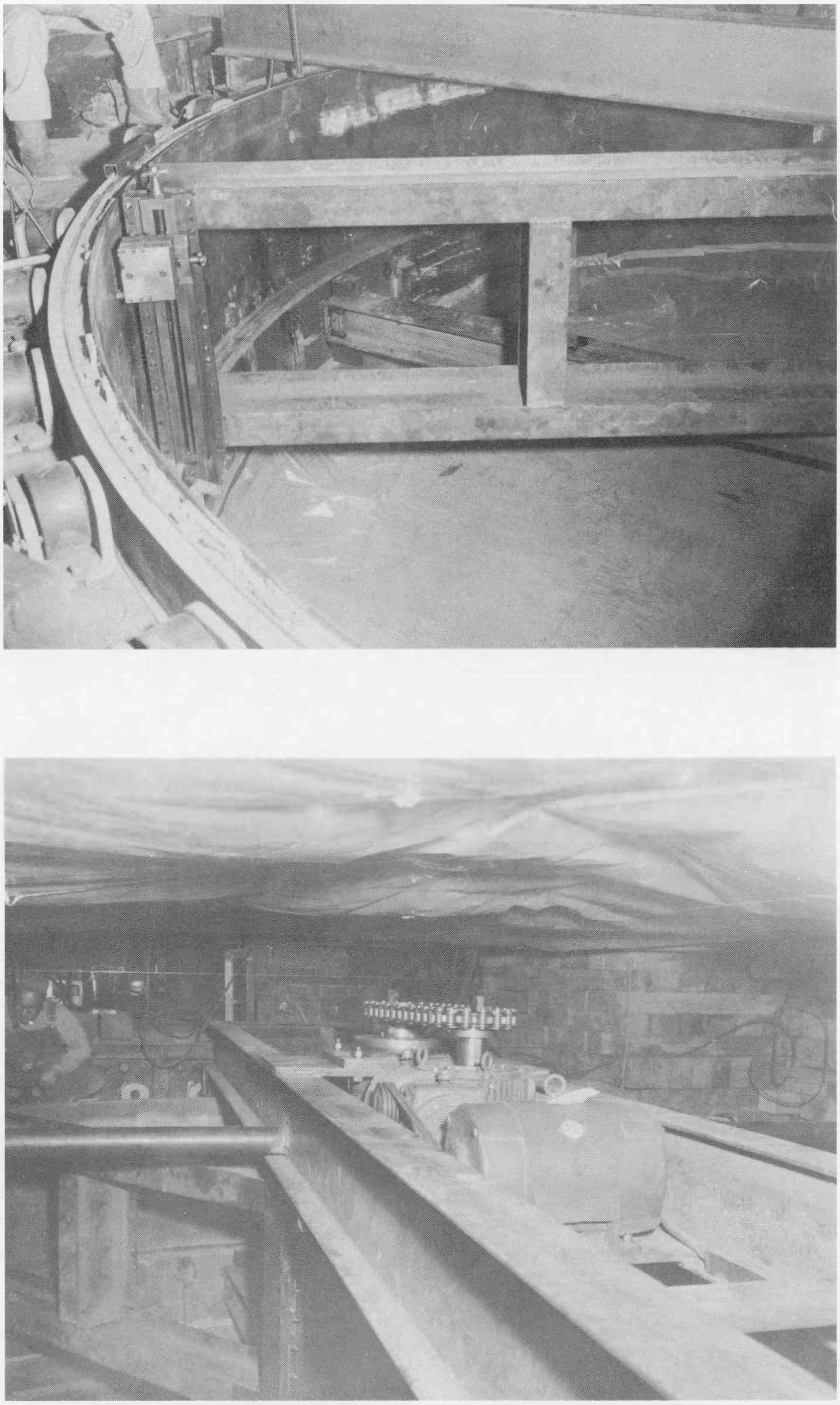

Figure 24. Special Boring Tool For Upper Cavity Liner

V. CONCLUSIONS

The experiences with the design and construction of the HNPF reactor structure have established the following technical facts:

1) The reactor structure, fabricated of austenitic and ferritic SS, can be readily shop fabricated and field erected to provide the required conservative, reliable design.

2) The incorporated design features permit the materials of construction to operate at very low stress levels, primarily due to the low pressure property of liquid sodium, for the HNPF conditions. Future designs could readily operate at higher temperature levels (11 to 1200°F range) with very little design complications.

3) The reactor structure design will allow larger size structures to be fabricated (up to 30 to 35 ft vessels) such that core sizes required for reactors in the 300 to 1000 Mwe range can be economically and practically constructed.

4) More than adequate containment or confinement capability is provided for without adversely affecting the design. Low radioactive gas leakage can be obtained with practical, high quality welding and testing procedures.

5) Design features can readily be incorporated to minimize the effects of any rapid thermal shocks which might occur under casualty conditions without adversely affecting containment or operability.

REFERENCES

1. F.C. Gronemeyer and J.W. Merryman, "75,000 Kilowatts of Electricity by Nuclear Fission at the Hallam Nuclear Power Facility," (ASCE Convention, Reno, Nevada, June 1960)

2. R.J. Beeley and J.E. Mahlmeister, "Operating Experience with the Sodium Reactor Experiment and Its Application to the Hallam Nuclear Power Facility," (Small and Medium Power Reactor Conference, Vienna, 1961)

3. S. Berger, "Gamma Ray Heat Generation in HNPF Biological Shields," NAA-SR-6244

4. ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels, (1956)

5. Bureau of Ships, Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components, (Department of Navy, April 1958)

6. Specifications, (American Institute of Steel Construction, (June 1949)

7. American Concrete Institute, Building Code Requirements for Reinforced Concrete, (ACI 318-56, 1956)

8. "Equipment Specification, Steam Generator," AT 5-354, (December 16, 1958)

9. Uniform Building Code

10. "Supplement to the Preliminary Safeguards Report Based on Uranium-Molybdenum Fuel for the Hallam Nuclear Power Facility," NAA-SR-3379 Supplement I

11. R.D. Chipman, "Stress Analysis of the HNPF Loading Face Shield," NAA-SR-4535

12. Staff, "Final Summary Safeguards Report for the Hallam Nuclear Power Facility," NAA-SR-5700