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OPERATING EXPERIENCE WITH THE SODIUM REACTOR EXPERIMENT AND ITS APPLICATION TO THE HALLAM NUCLEAR POWER FACILITY

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OPERATING EXPERIENCE WITH THE SODIUM REACTOR EXPERIMENT AND ITS APPLICATION TO THE HALLAM NUCLEAR POWER FACILITY

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ABSTRACT

The Sodium Reactor Experiment (SRE) was designed and constructed by Atomics International, a Division of North American Aviation, Inc., as part of a program of the United States Atomic Energy Commission to demonstrate the technical and economic feasibility of sodium cooled graphite moderated reactors for central station power. The operating experience from this developmental reactor has provided valuable data for the design of the Hallam Nuclear Power Facility (HNPF). The HNPF is currently under construction at Hallam, Nebraska, as part of the Sheldon Power Station being constructed by the Consumers Public Power District of Nebraska.

Operation of the SRE revealed areas in which design modifications were required or would lead to improved performance. These modifications are predominantly concerned with the cooling systems. Difficulties with the horizontal U-tube intermediate (sodium-to-sodium) heat exchanger were experienced. A vertical unit which will avoid these difficulties will be used in HNPF. The SRE sodium pumps, which were adapted from hot oil pumps through the addition of a freeze-seal, gave considerable difficulty and required modifications. The freeze-seal was initially cooled with tetralin. A leak developed which permitted the tetralin to enter the reactor and partially plug some of the fuel channels. This resulted in local overheating and damage to 13 of the 43 fuel elements in the core. The HNPF loops will use a free-surface pump which does not require a

freeze-seal. During the SRE cleanup, following the fuel element damage, it was found that the SRE fuel handling machine would not permit handling of broken elements. Both the fuel elements and the handling machine for the Hallam Nuclear Power Facility are of a modified design so that this difficulty should not arise. In addition, it was found necessary to control the coolant flow in SRE so that system temperatures could be maintained after a reactor scram, thus avoiding excessive thermal stresses in the reactor and system components. Such flow control has also been incorporated in the HNPF design.

Special HNPF features which were incorporated subsequent to the SRE fuel element damage include the addition of an activity monitoring system for the core cover gas, the complete elimination of tetralin for auxiliary cooling of plant equipment, provision of instrumentation for fuel elements, and addition of carbon traps in the primary sodium system.

Special tests and experiments on SRE have demonstrated unusual reactor stability and capability for rapid power changes. Similar characteristics are predicted for HNPF.

The experiences gained from the SRE and the HNPF development programs which have been incorporated in the design features of the HNPF should result in a facility of high reliability and long life.

INTRODUCTION

Development of the Sodium Graphite Reactor (SGR) was initiated in 1949 at Atomics International when studies of this concept and limited experimental work on materials were initiated to determine its applicability to combined plutonium and power production. In 1951 the studies were reoriented to emphasize power production, and all subsequent work has been done with the objective of developing the SGR to produce economical electric power in central station plants.

The features of the SGR which make it attractive as a heat source for modern steam turbines are its high operating temperature and the excellent heat transfer characteristics of the sodium coolant. The high temperature is available with essentially atmospheric pressure on the reactor and sodium systems. The low pressure, together with lack of energetic chemical reactions among reactor materials and negligible corrosion of ordinary structural materials by liquid sodium make large, high power reactors entirely practical. A negative power coefficient of reactivity, the considerable difference between the normal operating temperature and the boiling temperature of sodium, the high heat capacity of the pools of sodium coolant at the reactor inlet and outlet, and the natural convection cooling characteristics combine to make the reactor unusually safe to operate.

A considerable amount of research and development effort, particularly on sodium components and fuel materials, was required to bring the SGR concept to the point at which a reactor experiment could usefully be constructed. Construction of the Sodium Reactor Experiment (SRE) was started on April 25, 1955, and completed in February, 1957. The first wet critical experiment was performed on April 25, 1957. The purposes of the SRE were:

1. To provide a facility for irradiation of fuel materials of interest to sodium cooled reactors, in the environment (temperature and coolant) for which they were intended and in adequate quantities to provide statistically meaningful data.

2. To determine the static and kinetic neutron behavior of this reactor type and establish its safety characteristics.

3. To test and demonstrate sodium system components which were in general adapted from conventional designs and commercially available.

4. To demonstrate maintenance techniques and general plant operability.

Operating experience with the SRE has yielded information on most of these areas of uncertainty which existed during the design phase. This experience has been applied in engineering the Hallam Nuclear Power Facility (HNPF). The HNPF, under construction for the Consumers Public Power District of Nebraska, is the first SGR to be built primarily for power production. The HNPF is a 75,000 kwe plant, still small for economic nuclear power production but utilizing sodium components of the scale required for large plants. Because of the factor of 12 scale-up from SRE, a research and development program has been required on components. The HNPF program was initiated in November, 1957; site construction began in April 1959, and the plant is scheduled for initial operation in mid-1962.

SRE DESCRIPTION

The SRE has been described in detail elsewhere,[1] and only those features pertinent to this paper will be discussed here. The major features of the reactor are shown in Figures 1 and 2.

The moderator consists of hexagonal logs of graphite canned in zirconium and submerged in a pool of sodium. An axial process channel through the graphite log contains the fuel elements. Some of the moderator assemblies have a scallop along the edge of the hexagon. When three such assemblies are stacked together a channel is formed into which a control rod or instruments may be inserted. All moderator assemblies are supported and located on the grid plate. A typical moderator assembly is shown in Figure 3.

The fuel element consists of a cluster of 3/4 inch rods suspended in the central channels of the moderator cans and supported from the top shield by means of a hanger rod. The fuel rods are spaced from each other and from the process channel by a 0.091 inch diameter wire wrapped in a spiral around the peripheral rods. (Figure 4).

The eight shim and safety rods operate dry in closed thimbles in the scalloped corner channels of the moderator matrix with the drive mechanisms mounted on the top shield. The neutron absorbing material is boron nickel alloy containing about 2 percent boron.

The reactor primary cooling system consists of two separate cooling loops, a 20 Mw loop for power operation and a 1 Mw auxiliary loop for emergency cooling. A schematic flow diagram is presented in Figure 5. The components within the dotted lines of the figures are within shielded vaults with a nitrogen atmosphere. The induced Na^{24} activity makes a secondary loop desirable in order to avoid radiolytic decomposition of water in the steam generator and to isolate the reactor in the event of a major leak on the steam generator.

Coolant flow through the core is upward from a plenum under the grid plate, passing through the fuel channels to the pool above the core. Additional flow for moderator cooling is from a distributor pipe located above the grid plate. Moderator coolant flows upward between moderator cans to the upper pool.

The heat sink is a steam generator and a 7500 kw turbine-generator system provided by the Southern California Edison Company.

HNPF DESCRIPTION

The HNPF, Figure 6, incorporates a 240 Mwt sodium reactor and associated equipment to generate steam of sufficient quantity to produce 75,000 net kw electric power. The steam, at 800 psig and 800°F will be used in a 100,000 kw modified turbine, which is also shared with the Conventional Facility part of the Sheldon Power Station. The Conventional Facility, which will be completed before the reactor, includes a fossil fuel fired boiler and auxiliaries. The nuclear facility has been described in detail[2,3] elsewhere and only pertinent features will be discussed.

The reactor building, Figures 7 and 8, houses the reactor, the heat transfer system, reactor service system and equipment, and miscellaneous maintenance and service areas. The building design conforms to all applicable building codes. The building is of steel frame construction with insulated roof deck and metal siding. The building is 275 ft. long by 116 ft. wide. The 116 ft. width of the building comprises two bays, one which is 81 ft. wide by 70 ft. high over the reactor and an auxiliary bay 35 ft. wide by 51 ft. high. Overhead cranes of 60 tons and 25 tons, respectively, service each of the bays. The operating floor of the reactor plant is at finished plant grade elevation of 1440 ft.

Beneath the operating floor, in shielded compartments, are the sodium graphite reactor, and radioactive portion of the sodium heat transfer system, fuel cleaning and storage facilities, the maintenance cell, and piping and service equipment. These shielded compartments are of regular concrete having adequate thickness to fulfill the shielding requirements. High density concrete is used above the reactor and in the operating face of the maintenance cell.

The compartments housing radioactive sodium equipment are lined with steel to prevent spalling of concrete in the event of a sodium leak. The concrete access plugs to these areas are gasketed to minimize nitrogen leakage, and all plugs are stepped to preclude radiation streaming. A railroad spur track enters the high bay area at the south end of the building to facilitate the delivery of heavy components and for shipment of spent fuel elements to the reprocessing plant. The sodium melt station is adjacent to the railroad spur at the south end of the low bay area.

The steam generators and associated components of the nonradioactive portion of the sodium heat transfer system are housed in three separate enclosures at the north end of the high bay area. The high bay bridge crane has coverage over the steam generator rooms.

A lead-shielded fuel handling cask is mounted on a gantry which operates in the reactor room between the reactor and the fuel cleaning and storage area. The control rod drive mechanisms are mounted on movable support structure. The drive mechanism can be disconnected from the control rods at the upper surface of the reactor loading face shield and the entire assembly can be moved away, leaving the shield face clear for fuel handling operations.

Adjoining the reactor plant at the north end is a control room serving both the conventional plant and the nuclear facility. Adjoining the reactor plant to the west are offices, the equipment decontamination room and the plant ventilation equipment room.

The HNPF reactor core, which is basically similar to that of the SRE is shown in Figure 9. The reactor structure, which is designed for operation with 1000°F sodium, is below the operating floor level and provides shielding and containment as well as support for the core components. The core components, located within the reactor vessel are accessible through the loading face shield.

The major core components are the fuel elements, moderator and reflector elements, and control rods. Fuel is suspended from the top shield in a matrix of moderator cans. The fuel and other core elements are individually loaded through the top shield by the fuel handling cask. The hexagonal moderator cans, which are submerged in sodium, are scalloped along each edge as shown in Figure 10. When stacked together three scalloped corners form a circle in which process channels for fuel and control rod thimbles are located. All moderator cans are supported and located upon grid plate pedestals which transmit the can weight to the core support structure.

The moderator and reflector elements are identical and are made up of machined graphite blocks canned in 0.016 inch stainless steel as shown in Figure 10. Each element is a prism of hexagonal cross section, measuring 16 in. across the flats, and is scalloped at the corners. The elements are 17 ft. in height. A total of 141 elements makes up the graphite assembly. Design features provide for support of the stainless steel sheath and for gettering of the thermally and radiolytically released gases which will evolve from the graphite.

The fuel element, Figure 11, consists of a cluster of 18 fuel rods assembled within a zircaloy-2 process tube. The fuel rod is comprised of 0.500 in. diameter U-Mo slugs in 0.010 in. wall, 0.660 in. OD, type 304 stainless steel tubing with an annular sodium bond. The active length of fuel is 13-1/2 ft.; each rod is provided with 21 in. of expansion space at the top of the tube. The fuel elements are suspended from the top shield; the suspension hardware incorporates a variable orifice for the control of sodium flow in each fuel element.

The HNPF control rods are combined shim-safety rods which regulate the reactor power level and provide for emergency setback or scram as required by the protective system. Each rod (Figure 12) consists of an absorber assembly, actuator assembly and drive. The absorber assembly, which is suspended from the loading face shield, includes a zircaloy-2 thimble, snubber, and poison column. The poison column consists of a 12 ft. can made of Hastalloy-X which is filled with a mixture of gadolinium and samarium oxides. These absorber assemblies are interchangable with the fuel elements, allowing placement as desired within the central core region.

The sodium heat transfer system (Figure 13) consists of three parallel circuits originating at the reactor and terminating at the steam generators. Each circuit consists of a radioactive primary loop and a nonradioactive secondary sodium loop, with exchange of thermal energy through an intermediate heat exchanger.

The three primary sodium loops are in parallel and common to the reactor. The intermediate heat exchangers are located in individual shielded cells. In each primary sodium loop there is a variable speed primary pump located in the cell with the intermediate heat exchanger.

Blocking and throttling valves are also provided. The drive for the pump is located outside of the cell above the reactor floor level.

The secondary sodium system consists of three parallel loops each originating on the shell side of an intermediate heat exchanger and terminating at the sodium side of a steam generator. Each loop contains a variable speed secondary sodium pump, expansion tank, throttling valve, and a radiation detector. Only that portion of the piping in a loop which connects to an intermediate heat exchanger extends into the shielded cells. The remainder of the loop is in an accessible, unshielded area above floor level.

The HNPF pumps, Figure 14, are similar in construction to conventional vertically-mounted, centrifugal pumps, utilizing a hydrostatic (sodium) shaft bearing. An oil-fired, double face shaft seal is provided to seal the helium cover gas over the free sodium surface in the pump. The pumps, which are provided with 16 in. suction and 14 in. discharge lines, are rated at 7200 gpm and 160 ft. of sodium head.

The sodium-to-sodium intermediate heat exchangers are mounted as shown in Figure 15. The exchangers are vertical to reduce stratification problems at low flow rates. The exchangers are designed for a pressure of 100 psig at 1000°F.

The steam generator, Figure 16, consists of an evaporator, moisture eliminator and a superheater. Construction is of the shell and tube type, utilizing duplex tubes to prevent interleakage between the sodium and the water or steam. The third fluid is helium, provided to monitor the duplex tubes for leaks.

The main primary sodium valves are split-wedge block valves and venturi ball-type throttle valves as shown in Figure 17. Each valve has a frozen sodium stem seal. The seal is cooled by circulating nitrogen. The valves are of conventional design except that the stems are longer than standard to accommodate the freeze seal assembly.

The sodium service systems provide the means to fill, drain, flush, and purify the sodium in the reactor core tank and the sodium heat transfer systems.

The fuel handling machine, Figure 18, is provided with features which permit fuel handling in an inert atmosphere.

SRE POWER OPERATIONS AND REACTOR CHARACTERISTICS

The SRE is a multipurpose development facility with the production of power of secondary concern. In spite of this more than 15,000,000 kilowatt-hours of electrical energy were produced in the first two years of operation. Its operating characteristics have in general proven to be very satisfactory. This fact must be borne in mind since this paper, by its nature, must emphasize the difficulties and their solutions.

The power operating history is summarized in bar chart form in Figure 19. The chart covers the two year period from 1 July 1957 through 30 June 1959. Fuel damage was experienced in July 1959 and most of the following year was spent in modifications and recovery operations.

Most of the difficulties experienced with the SRE have been concerned with the cooling system. These include stress problems resulting from lack of after-scram flow control, high stresses in the intermediate heat exchanger due to recirculation, and some problems with pump seals, and valves. These will be discussed later with reference to the HNPF design.

A number of tests have been performed on the SRE to determine the kinetic behavior of the core and the response of the entire system to imposed conditions.

The kinetic stability of the reactor was conclusively demonstrated by a series of oscillator tests.[4] The reactor has an isothermal temperature coefficient which is positive up to about 800°F. This is due to the moderator contribution to the coefficient. However, the moderator coefficient has an associated thermal time constant of about 10 minutes while the fuel plus coolant coefficient has a time constant of about 10 seconds. The result is a power coefficient which is negative and which provides the SRE with its high degree of stability and self-regulation.

The static stability of the reactor was demonstrated in a test in which integrating timers showed only 3.5 minutes of control rod motion during a 144 hour period.

The Hallam reactor has a calculated stability characteristic similar to that of the SRE. A major advantage accrues to a power reactor with these characteristics: no shim rods are required to compensate for a change in reactivity when the reactor is brought up to operating temperature.

Other tests on SRE, which have important implications for Hallam or other SGR power plants, demonstrated its ability to follow power demand quickly. The tests show that load changes at a rate of 20% per minute are entirely feasible. Flow control tests have demonstrated that the reactor temperature gradient can be maintained for at least two hours after a shutdown; with the gradient established the reactor can be restarted in about twenty minutes.

A variety of other tests, particularly on components, have been performed on the SRE which have provided information to aid the design of the Hallam facility. A brief summary of the operating characteristics of SRE and HNPF is presented in Table I.

Table I

SRE and HNPF Design Parameters

	HNPF	SRE	SRE
Thermal Power (Mw)	240	20	20
Electrical Power (Mw)	76	6	6
Steam Conditions (psig/°F)	800/825	600/825	600/825
Fuel Materials	U-10 Mo	U	Th-U
Fuel Loading (kg U)	27,600	3,000	1030 (alloy)
Sodium Temperatures			
Primary outlet (°F)	945	960	960
Primary inlet (°F)	602	500	500
Secondary outlet (°F)	900	900	900
Secondary inlet (°F)	557	440	440
Sodium Flow			
Primary (lb/hr)	8,400,000	510,000	510,000
Secondary (lb/hr)	8,400,000	510,000	510,000
Maximum Heat Flux Btu/hr-ft^2	350,000	340,000	420,000
Specific Power (kw/kg-25)	256	240	275

SRE EXPERIENCE AND APPLICATION TO COMPONENT DESIGN FEATURES OF HNPF

Operational experience with the SRE has provided insight for design improvements which have been incorporated into modified SRE components and components for the HNPF. Components which have been improved are: after-scram flow control, the intermediate heat exchanger, sodium pumps, sodium valves, fuel handling machine, cover gas monitoring system, fuel element design and outlet sodium temperature monitoring system.

Coolant Flow Control - During the initial operations with the SRE, unanticipated results involving the heat transfer system were observed. Perhaps the most important of these was excessive thermal convection flow in the primary sodium system following a reactor scram. Because of the large temperature rise across the core (460°F) it is necessary to match the coolant flow rate to the afterglow power level in order to avoid temperature fluctuations and resulting thermal stresses. It was found on the SRE that normal flow decay was not sufficiently fast to avoid a rapid reduction in temperature of the upper sections of the reactor. To correct this situation eddy-current brakes were added to both the primary and secondary cooling circuits.[5] The improvement in stability of the reactor outlet temperature is shown in Figure 20. On the basis of this experience, the major components for the HNPF were physically oriented to provide approximately the correct convection sodium flow through the core from any one of the three sodium circuits to remove the afterflow energy. Control of the convective flow is obtained by automatic positioning of throttling valves as shown in Figure 17.

Heat Exchangers - Another unexpected result from the SRE operation was a tendency for the sodium to stratify in the intermediate heat exchanger at low flow. The exchanger is a horizontal U-tube unit of conventional design.

Following a reactor scram, with the sodium flow reduced to a low value, it was found that a temperature difference of more than 200°F could develop between the top and bottom of the heat exchanger shell. This apparently was due to stratification of the secondary sodium on the shell side and to recirculation of the primary sodium from the top to the bottom tubes. The resulting thermal stresses produced some buckling of the heat exchanger shell. This problem is avoided in the HNPF design by vertical orientation, as shown in Figure 15. The difficulty with the SRE intermediate heat exchanger has not yet been corrected. However, a new unit of higher capacity has been ordered which should not be susceptible to this difficulty.

The performance of the SRE intermediate heat exchanger departed from design in another respect;[6] the log mean temperature difference was observed to be about 50% higher than was anticipated because of poor baffling, particularly in the bend area of the U-tubes. The temperature variation on the shell side on the exchanger is shown in Figure 21. As the figure indicates, the heat transfer surface in the bend area is not utilized to any appreciable extent.

Sodium Pumps - The sodium pumps in the SRE are of the freeze-seal type in which the liquid sodium is permitted to enter a narrow annulus between the shaft and housing and is frozen there by an auxiliary coolant. The sodium seal is continuously sheared by rotation of the shaft. A similar freeze seal used on the pump casing of the main secondary sodium pump is shown in Figure 22. The main primary pump is of similar design except that the housing is extended vertically to permit operation of the drive motor outside of the shielded gallery. The freeze-seals on the SRE sodium pumps have been sources of considerable operational difficulty due to shaft binding, sodium extrusion and gas in-leakage. The temperature gradient in the seal provides a diffusion trap for sodium oxide which collects in the seal area. On three occasions, twice with the main primary pump and once with the main secondary pump, material failures have permitted leakage of the auxiliary coolant, tetralin, into the sodium systems. On the last occasions a sufficient quantity of tetralin was admitted into the primary system to produce severe plugging of the fuel channels in the reactor. The resultant overheating of the fuel elements caused failure of the fuel cladding on 13 of the 43 elements in the core.[7]

The SRE pumps have been modified to permit the use of NaK for the auxiliary coolant on the freeze seals. This will avoid the introduction of a non-compatible material into the sodium systems.

The HNPF sodium pump shown in Figure 4 utilizes hydrostatic sodium bearing and requires no auxiliary coolant. This feature eliminates the possibility of introducing foreign material into the primary coolant through this component.

Sodium Valves - Valve performance in the SRE has been generally satisfactory. Both freeze-seal and bellows-seal valves have been used without major difficulty. Four of the bellows-seal valves in the sodium service system have had to be replaced because of damage to the bellows resulting from extrusion of solid sodium during pre-heating operations or to valve operation when the bellows was partially filled with solid sodium. No difficulties have been encountered with the freeze-seal valves. The desire to remove any possibility of contamination in HNPF led to changes to the sodium valves so that cooling is provided for the freeze-seal valve stems by circulating the nitrogen atmosphere in the cells that contain the components. Vapor traps, freeze traps and plugging meters are cooled in the same manner.

Fuel Handling Machine - In its role as a fuel testing facility, the SRE has required extensive fuel handling for inspection of elements. The original fuel handling machine performed this task satisfactorily as long as the elements were undamaged. The fuel elements which were damaged when tetralin entered the primary system could not be handled satisfactorily and it was necessary to build a modified machine. Jamming of the SRE fuel handling machine occurred when broken spacer wires from the fuel element became entangled in the hoist chains. A modified machine has been built which avoids these difficulties. In the HNPF machine (Figure 18) two guide tubes are provided which run the full height of the fuel handling machine. Thus, any joint or damaged item being handled cannot get out of position and jam the internal mechanisms. Gaps between sections are limited to 1/16 inch maximum and all joints are smoothly chamfered. The machine is designed such that the entire internal mechanism can be lowered out of the shielded body sections for maintenance. A maintenance cell is included in the fuel handling area of the HNPF (Figures 7 and 8) which has the capability of accomplishing this operation.

Cover Gas Activity Monitor - During power run No. 14, several fuel elements in the SRE were severely damaged. This situation was not known until the reactor was shut down to inspect the fuel. Earlier detection would have minimized the damage, perhaps to failure of a single rod. One of the modifications to the SRE following the fuel element damage incident was installation of a radiation detection system to monitor continuously the Xe^{133} radioactivity in the helium cover gas over the reactor. Some background activity will always exist in the cover gas from minute leaks in the cladding, but these are not harmful. Any major cladding failure in the future, such as opening of a crack, should be detected within seconds. The HNPF reactor is provided with a similar detection system.

Fuel Element Design - The fuel damage of power run No. 14 led to modifications of the SRE fuel element. A screen was added at the inlet to provide for collection of particulate matter from the coolant. This modification should greatly reduce the probability of collecting such material in the fuel element where its insulating properties could result in local overheating of the element. The clearance between the element and the process channel was also increased, by reducing the number of fuel rods from seven to five, to reduce the possibility of binding in the event of extensive fuel swelling. The thermocouples which measure the temperature of the sodium leaving each fuel channel have been lowered in the channel to avoid the thermal effect of the upper sodium pool. The number of thermocouples within the fuel rods has been increased.

A fuel element of sodium bonded uranium slugs within 304 stainless steel tubes is also utilized in the HNPF fuel element (Figure 11). Detailed investigations of the SRE fuel damage have strengthened the belief that this economical design can be used successfully in the HNPF. Fuel elements, instrumented with thermocouples embedded in the fuel at different locations, are also provided. Readouts for these thermocouples will be located within the control room.

Outlet Sodium Temperature Control and Monitoring Systems - The SRE fuel elements are designed with a fixed orifice plate for control of sodium flow. The matching of the sodium flow with power generation to obtain a uniform outlet temperature across the reactor has been only moderately successful.

Flow control through the HNPF element (Figure 11) by providing a variable orifice which can be adjusted from the loading face shield. In addition, the reactor is designed with an outlet temperature monitoring system for all core positions. This system consists of two multi-point scanner-recorders and six continuous recorders located in the control room. The recorders are equipped with high and low temperature alarms. With this system the outlet temperature of each channel is observed every six minutes and as many as six channels may be continuously monitored.

Monitor Handling - The SRE fuel damage resulted in lodging of portions of the fuel elements in the moderator elements. The removal and replacement of the damaged moderator elements through the loading face shield was successfully concluded without incident. The HNPF is provided with complete moderator element removal equipment, as well as the necessary design features in the loading face shield and core support structure to accomplish this operation. The fuel handling cask, with special components, is used for this operation.

Plant Control System - The plant control system for the HNPF provides automatic operation and load following between 15 and 100% of design power. It provides for semi-automatic control capability at all power levels and maintains all temperatures and pressures within their design limits at all times.

A diagram of the plant control system is shown in Figure 23. During normal operation an increase in plant load which initiates opening of the turbine inlet valves will result in an increased steam flow and decreased steam pressure. Signals representing steam flow and steam pressure are immediately transmitted to both the primary and the secondary sodium pumps, increasing the pump speeds proportionately. The increase in sodium flow through the reactor acts on the neutron flux controller to withdraw the shim-regulating rods, increasing the reactor power level. These actions represent the initial response of the control system to a change in plant load. Re-adjusting (reset) signals then come into play to prevent cummulative errors in these initial responses from driving the controlled variables out of limits. The secondary sodium flow rates are reset to maintain steam pressure at its set point. The primary sodium pump speeds are reset to maintain the programmed reactor inlet sodium temperature at its variable set point. The shim-regulating rods are reset to maintain fuel channel exit temperature at its set point.

Currently the SRE control system is load-forcing, the turbine accepting any steam rate set by the SRE. A load-following system similar to that for HNPF is being installed however, which will permit the power level to be established by the turbine throttle.

Loading Face Shield Coolant - The desire to eliminate tetralin from the HNPF also led to a modification in the coolant for the loading face shield. A 230 psi nitrogen cooling system was designed for this component with a special circulating system. Modification of the SRE loading face shield to incorporate gas cooling was not feasible, but kerosene has been used to replace tetralin.

SUMMARY

The SRE has played a major role in developing the information required for engineering of the HNPF. The operation of the reactor has provided the necessary experience for better design of the large-scale components and systems for the Hallam reactor. The fuel damage in the SRE, although requiring most of a year for modification and recovery, has been highly informative. A means of avoiding such occurrences and successful recovery techniques have been developed which will assist in operation and maintenance of all sodium cooled reactors. It is noteworthy that none of the personnel involved in the SRE recovery operations received more radiation than the accepted tolerances.

The data from SRE, together with the information obtained from the HNPF research and development program on the much larger Hallam components, have yielded a plant design which should have a high reliability and a long plant life.

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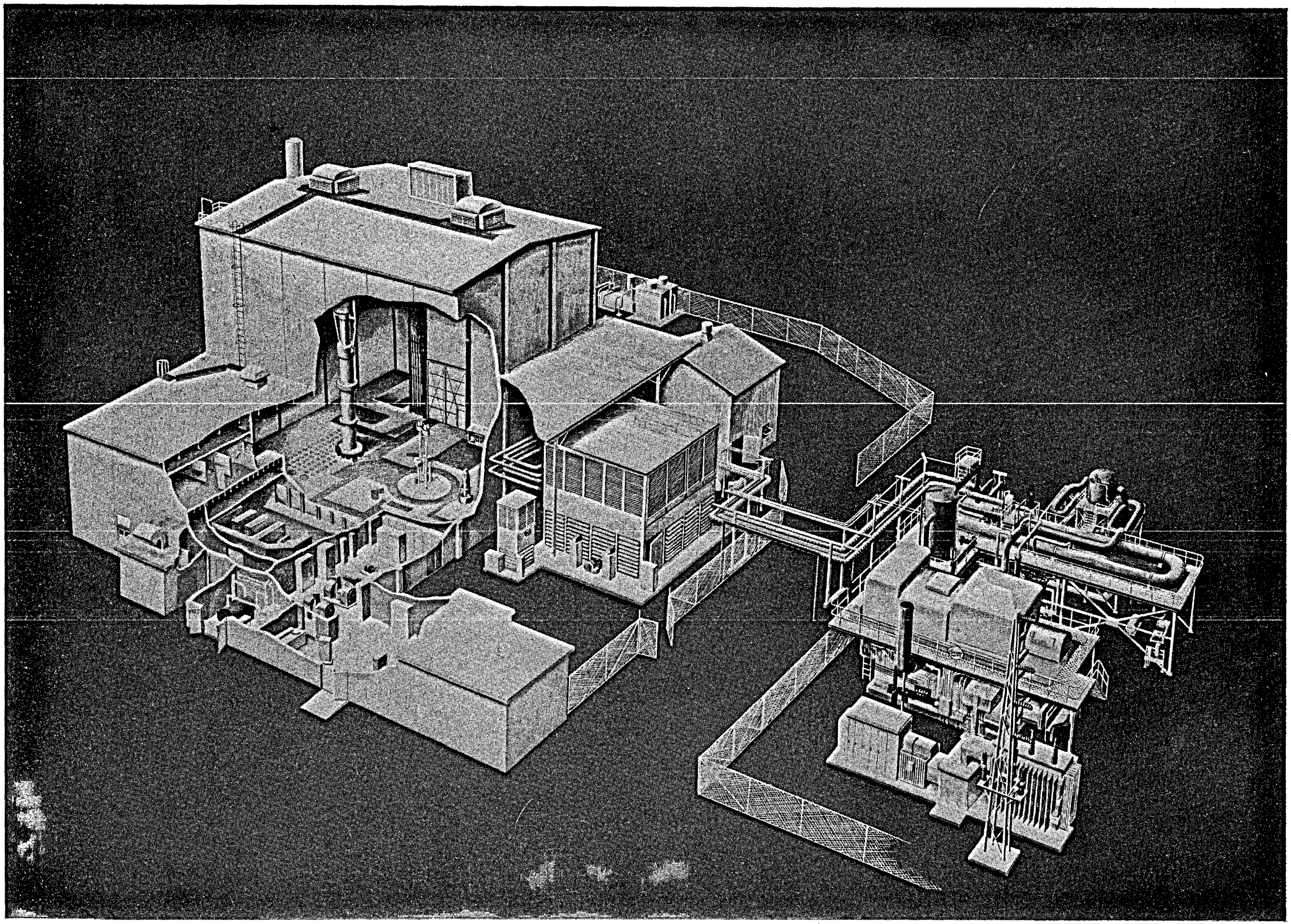

Figure 1. SRE Plant Cutaway

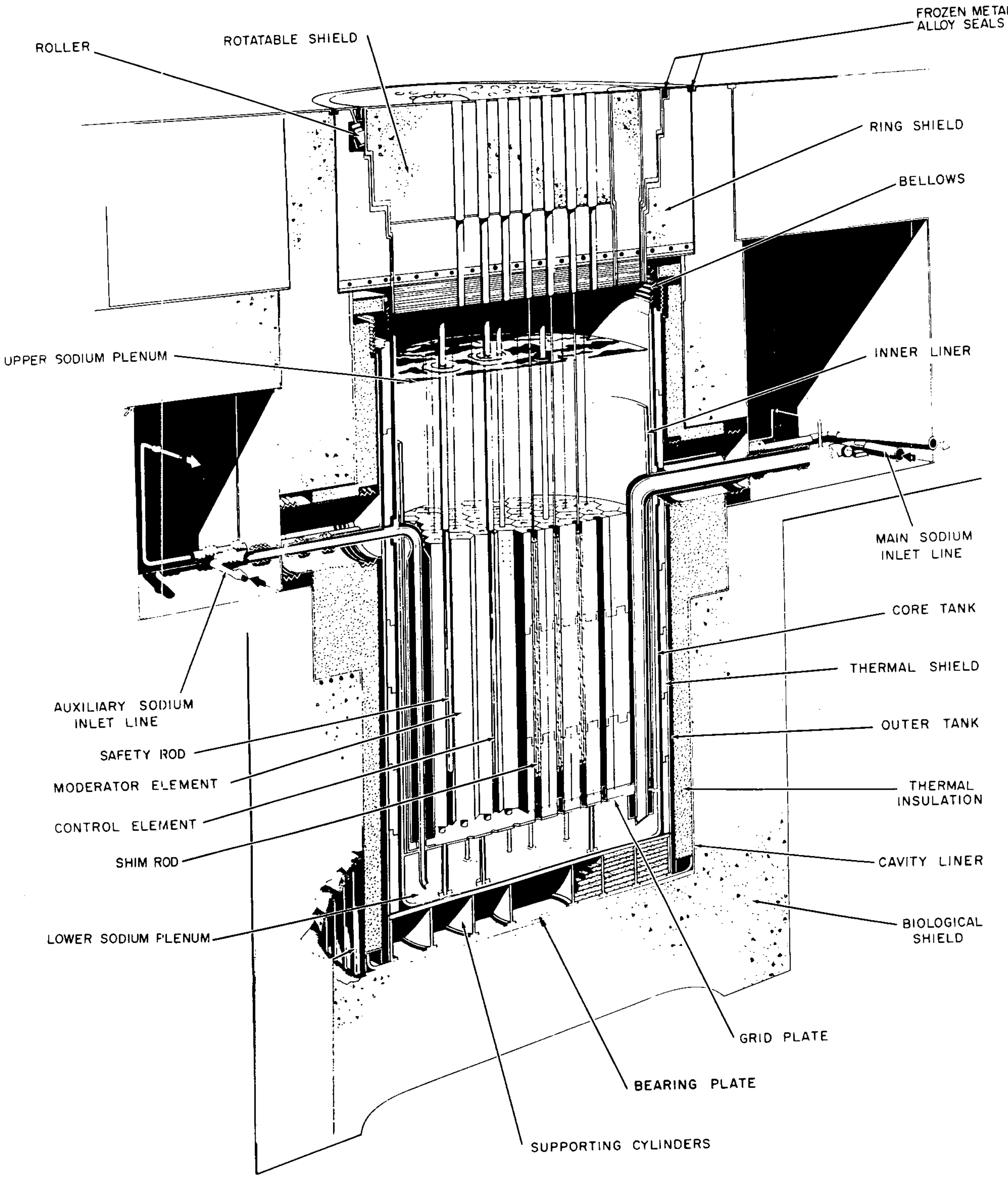

Figure 2. Cutaway View of SRE Reactor

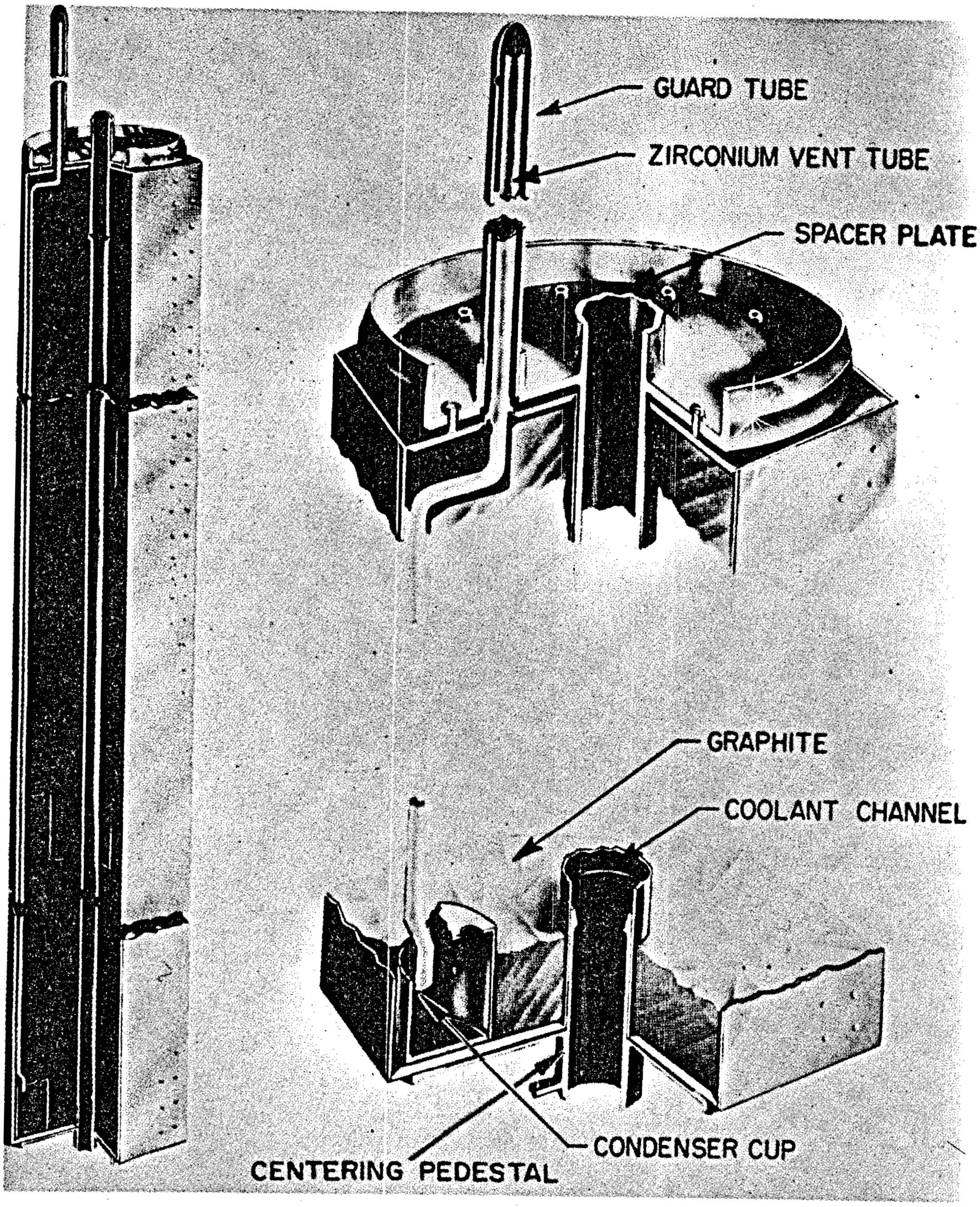

Figure 3. SRE Moderator Assembly

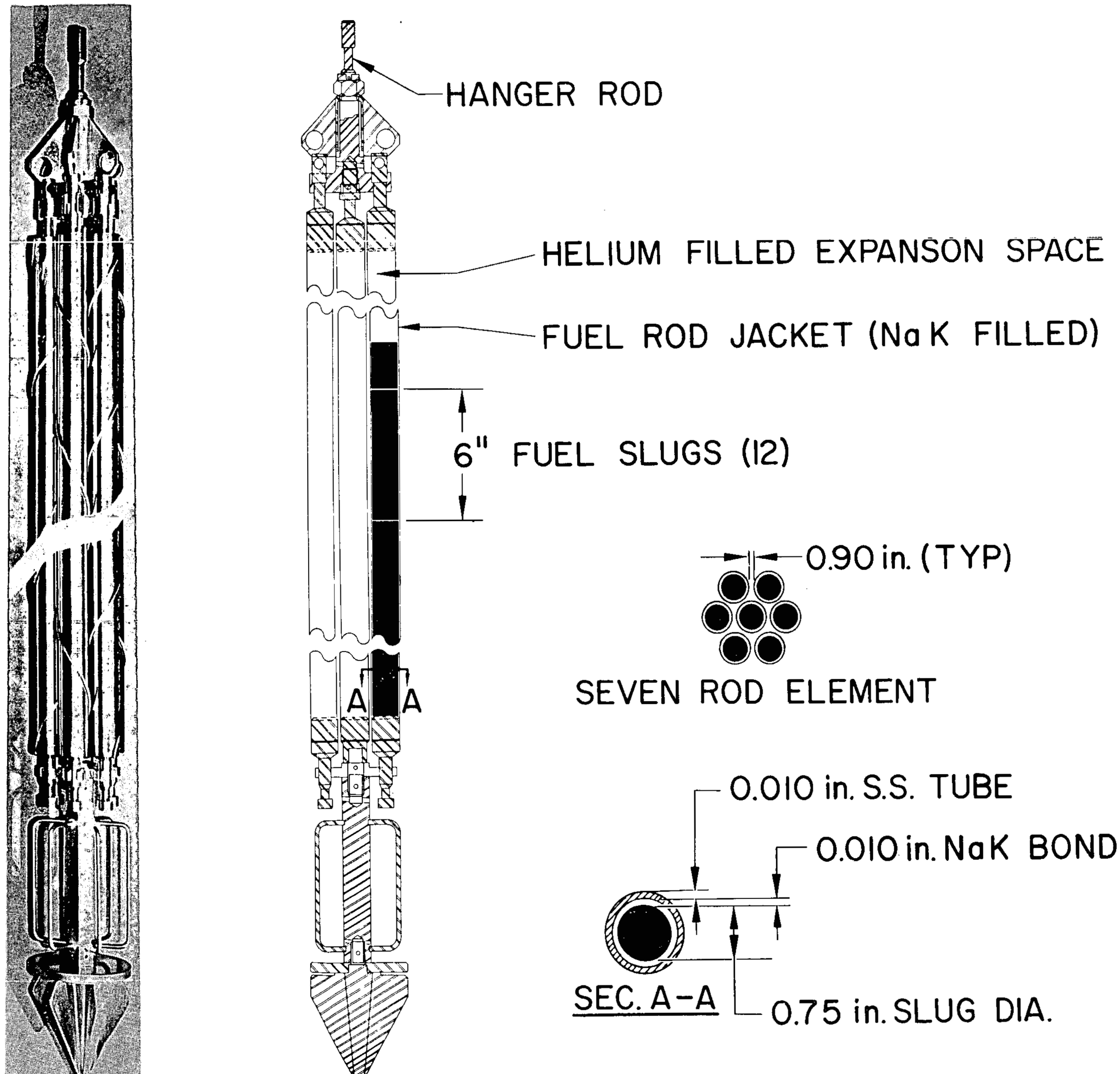

Figure 4. SRE Fuel Element

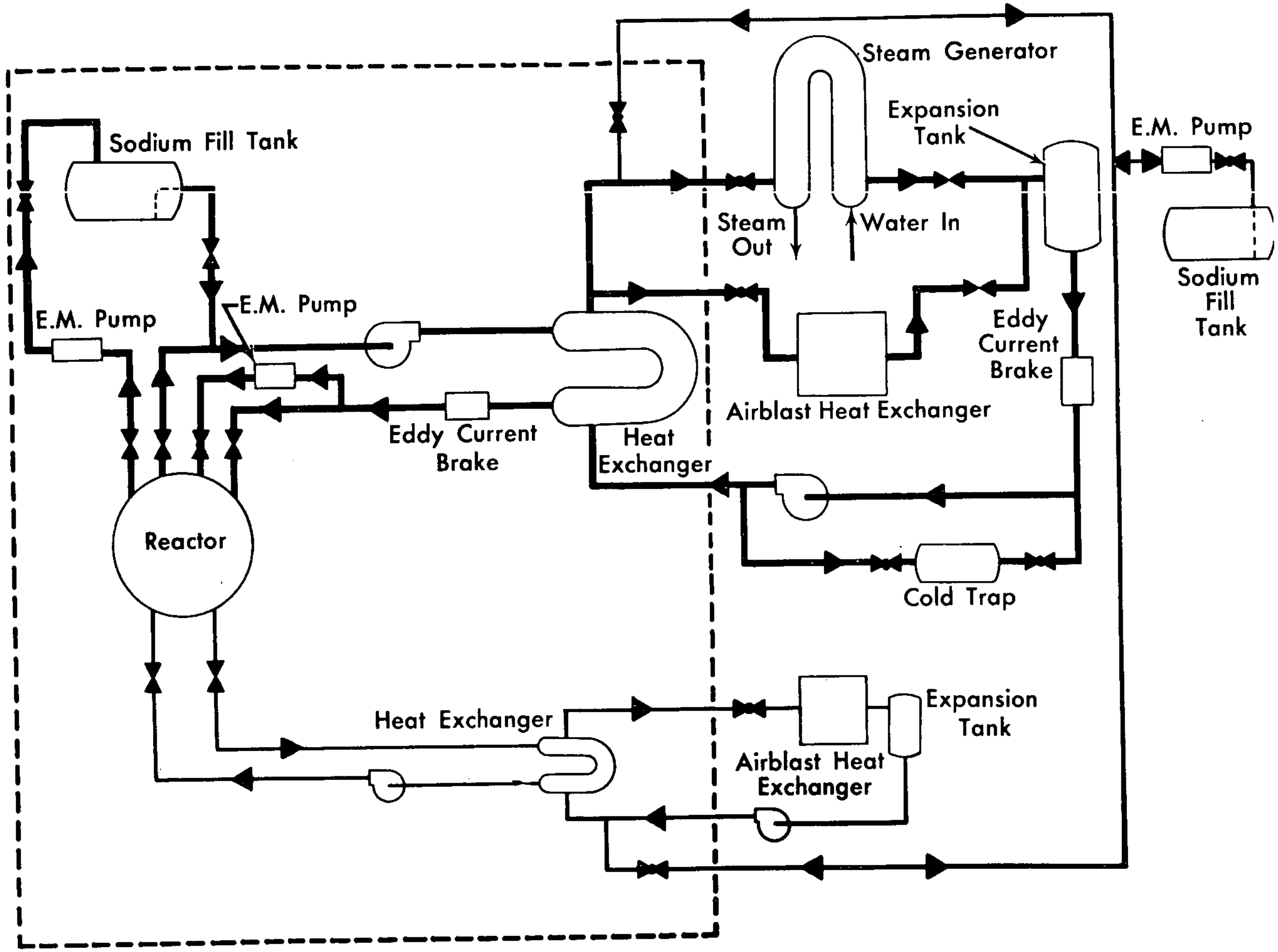

Figure 5. SRE Schematic Flow

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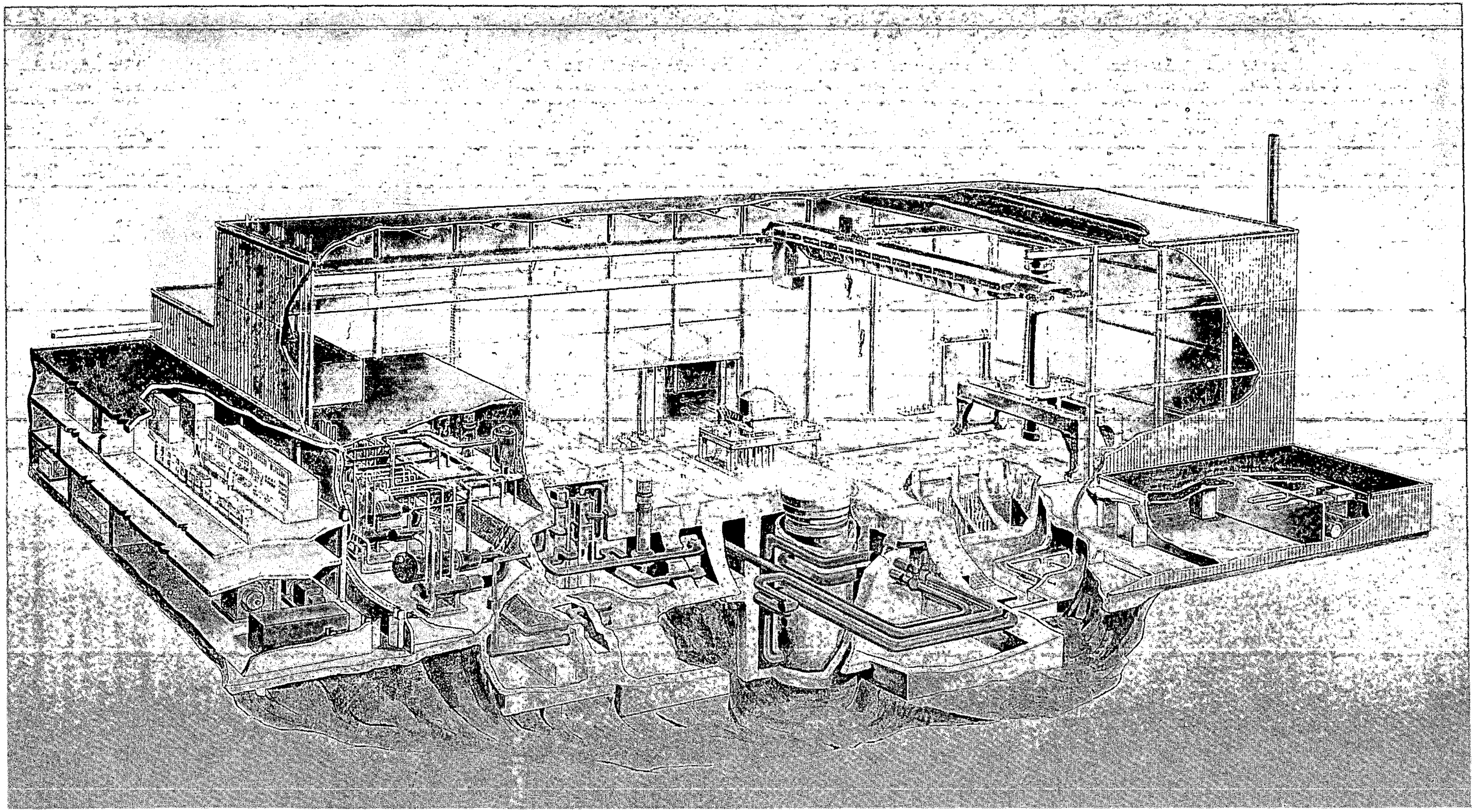

Figure 6. HNPF Reactor

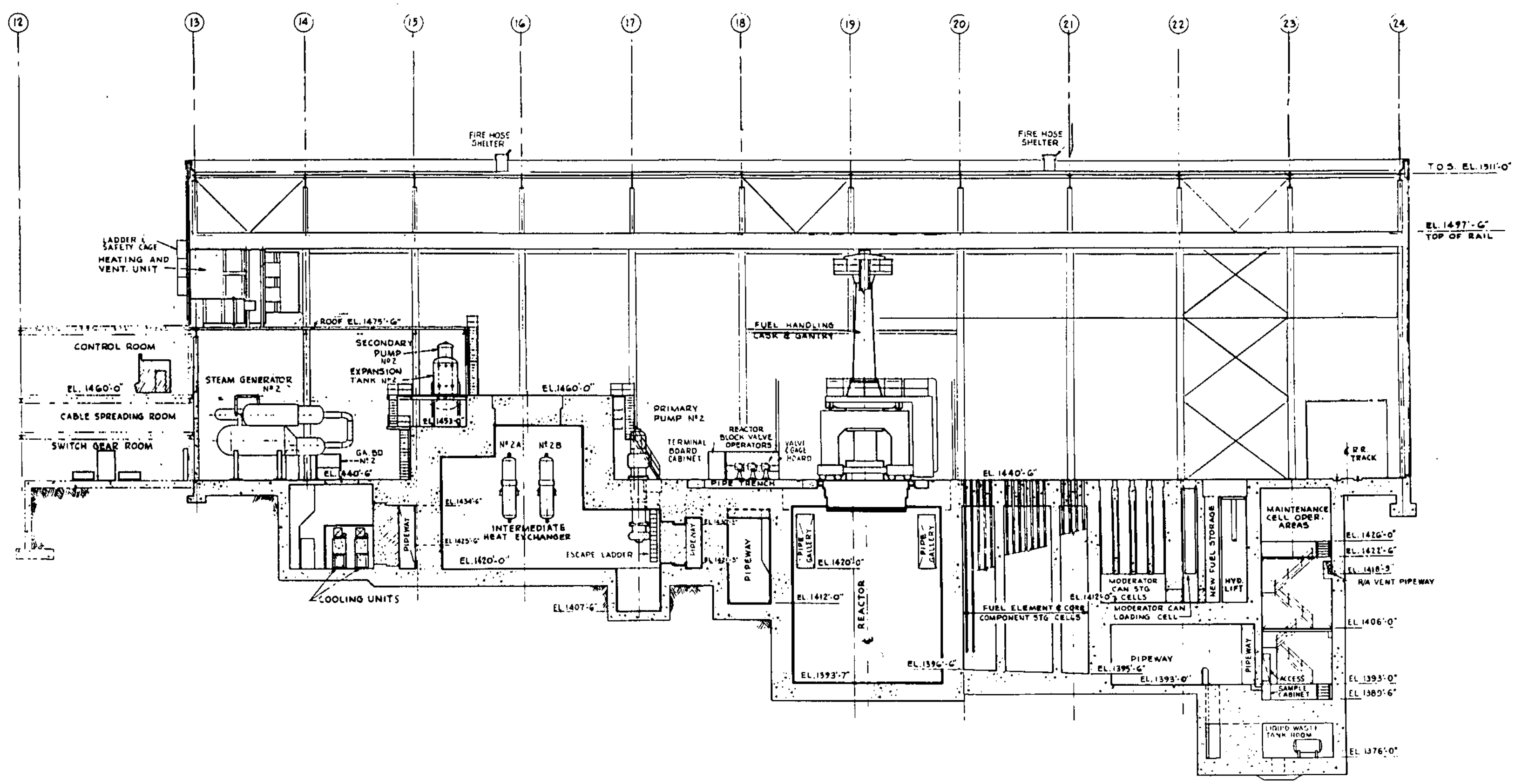

Figure 7. HNPF Vertical Section through Reactor Building

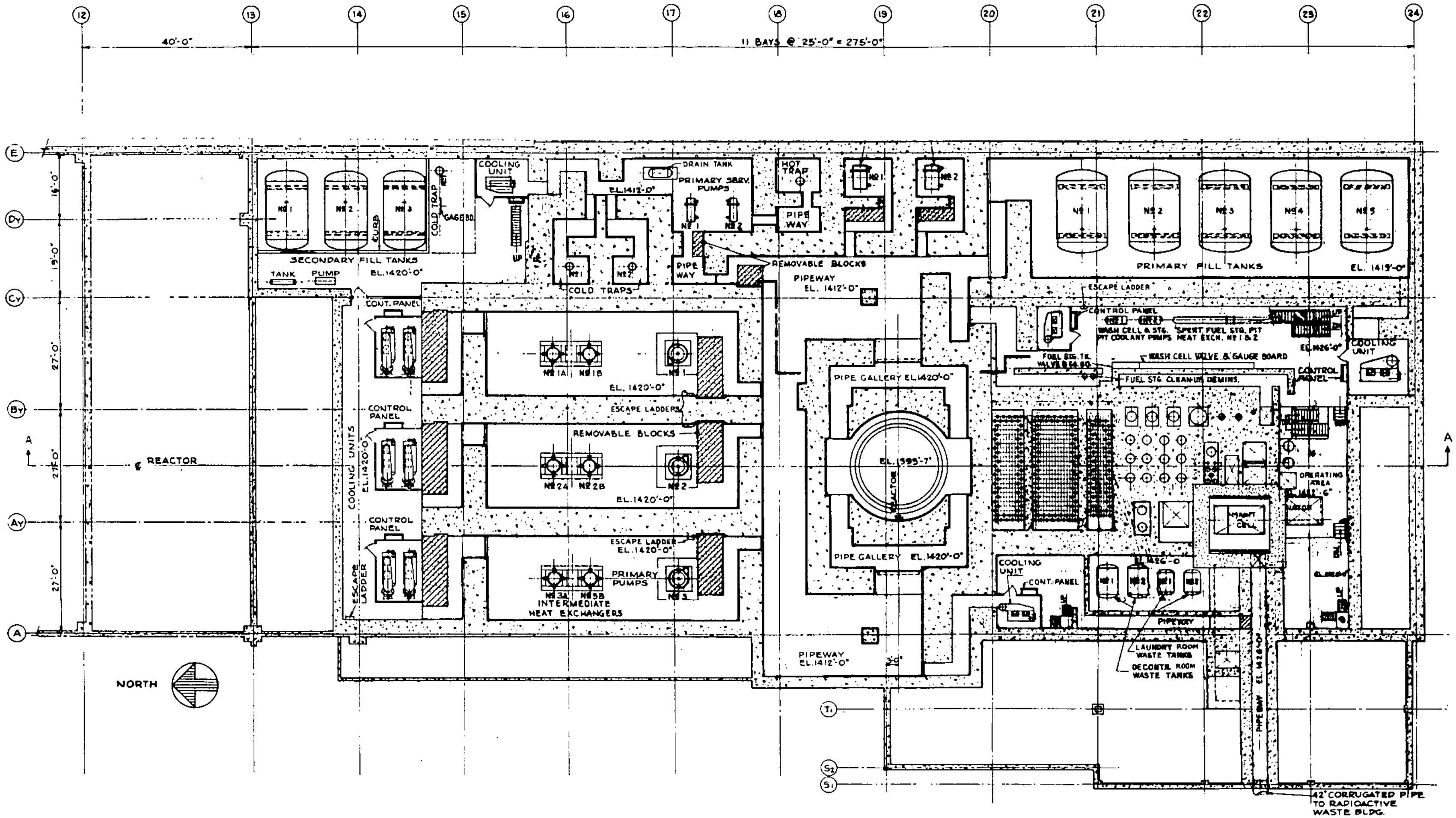

Figure 8. HNPF - Reactor Building Layout Plan Below Elevation of 1440 ft.

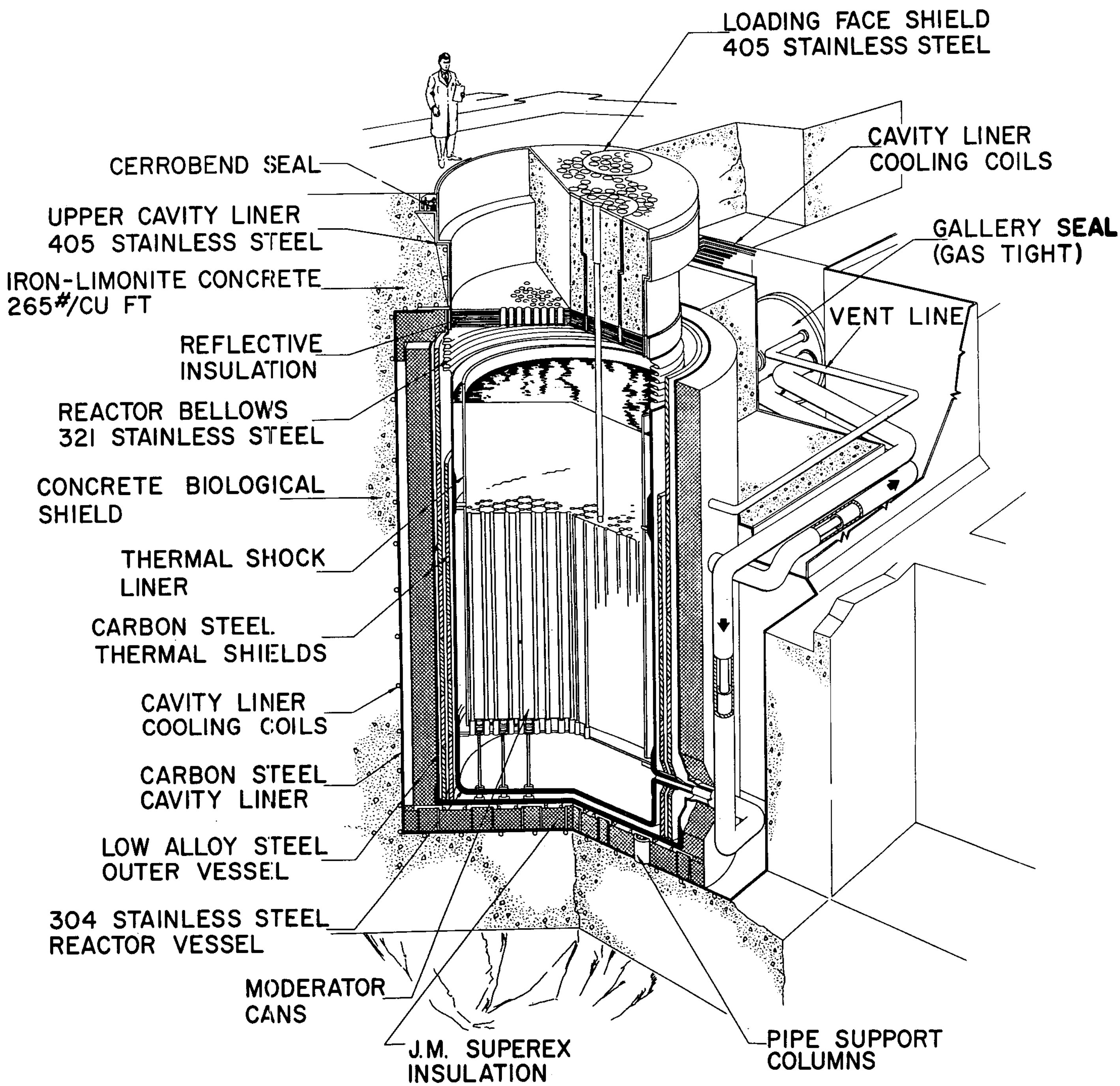

Figure 9. HNPF - Sectional Perspective View of the Reactor Core

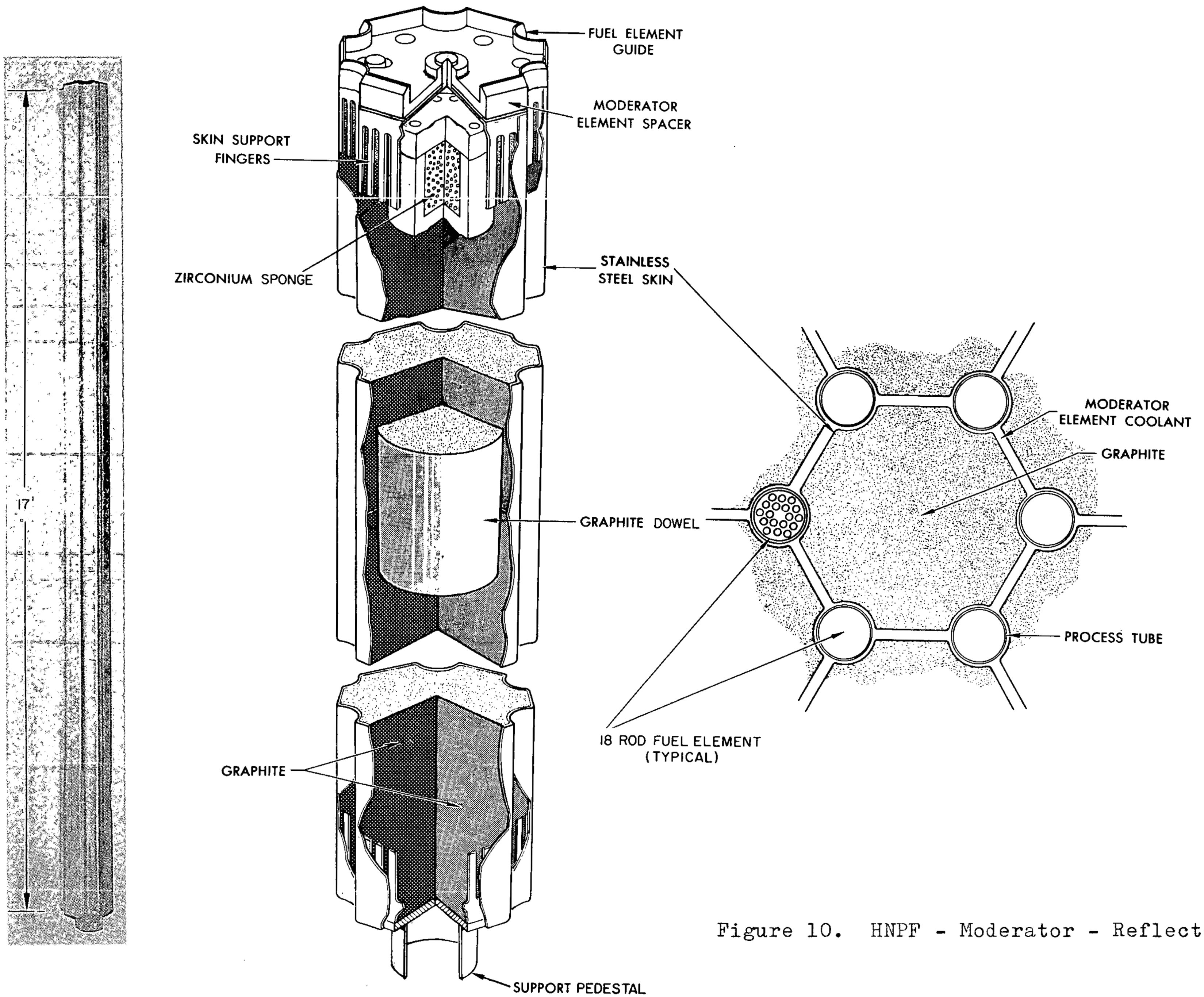

Figure 10. HNPF - Moderator - Reflector Element

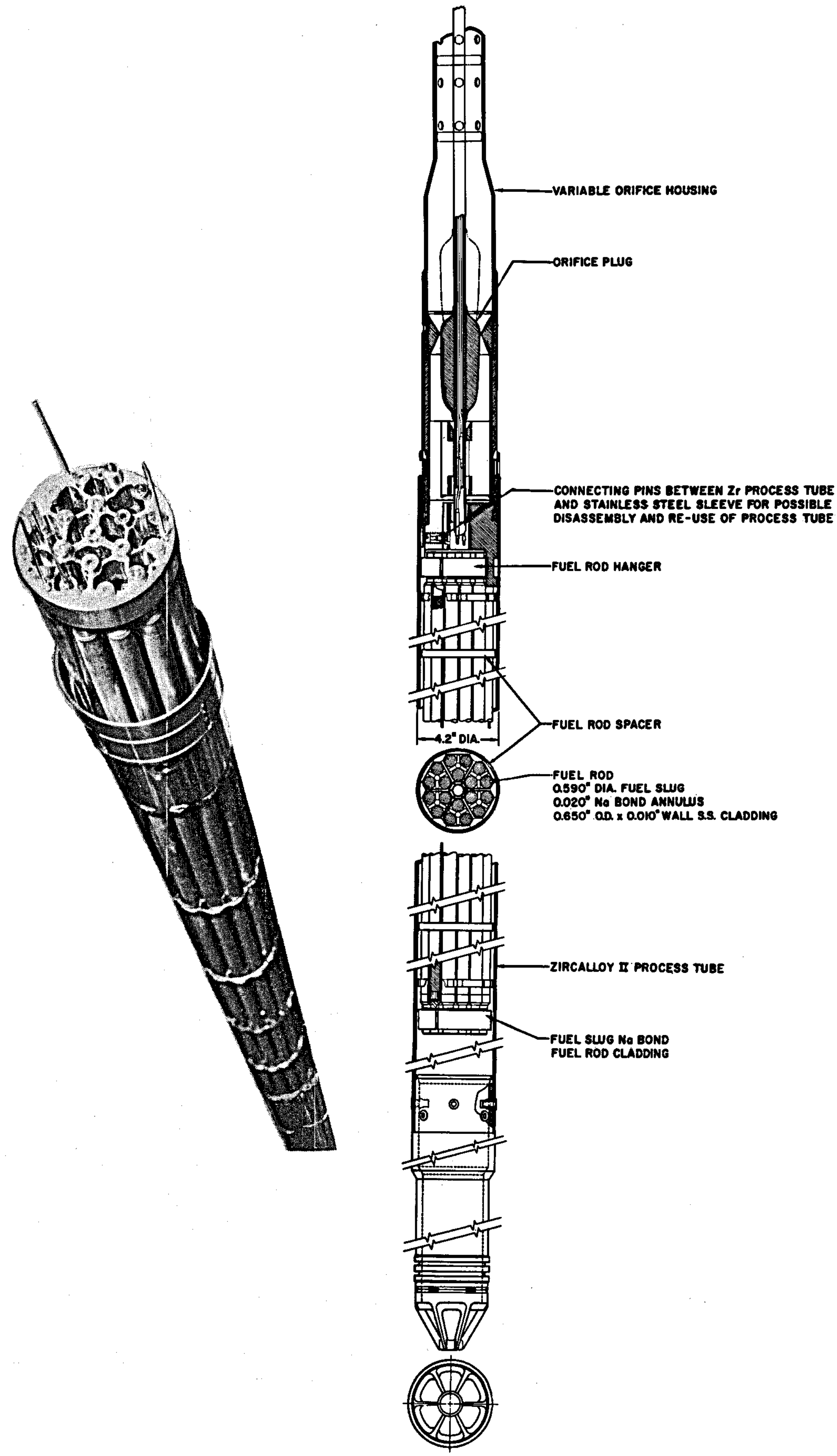

Figure 11. HNPF - Fuel Element Assembly

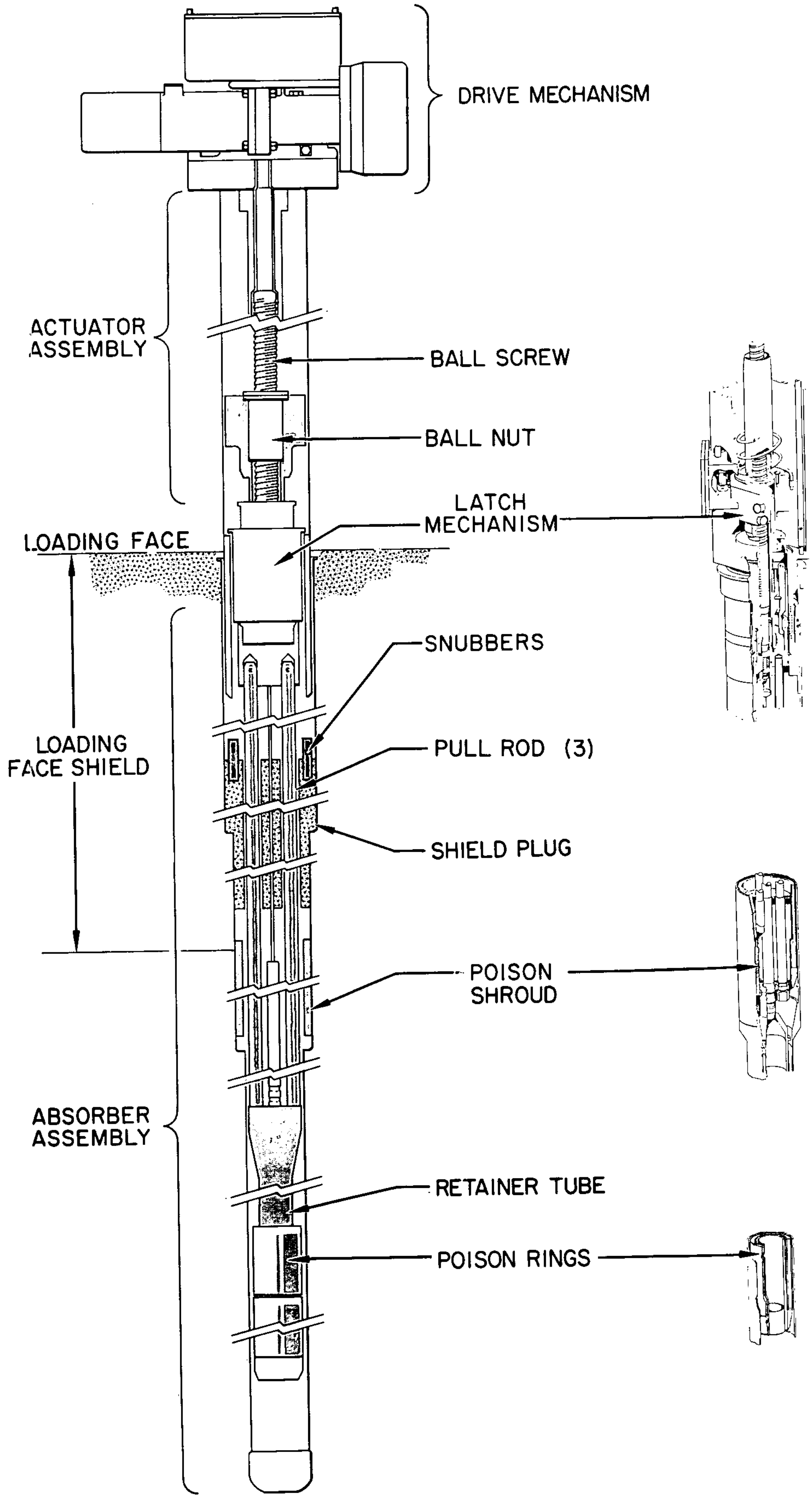

Figure 12. HNPF - Shim-Safety Rod

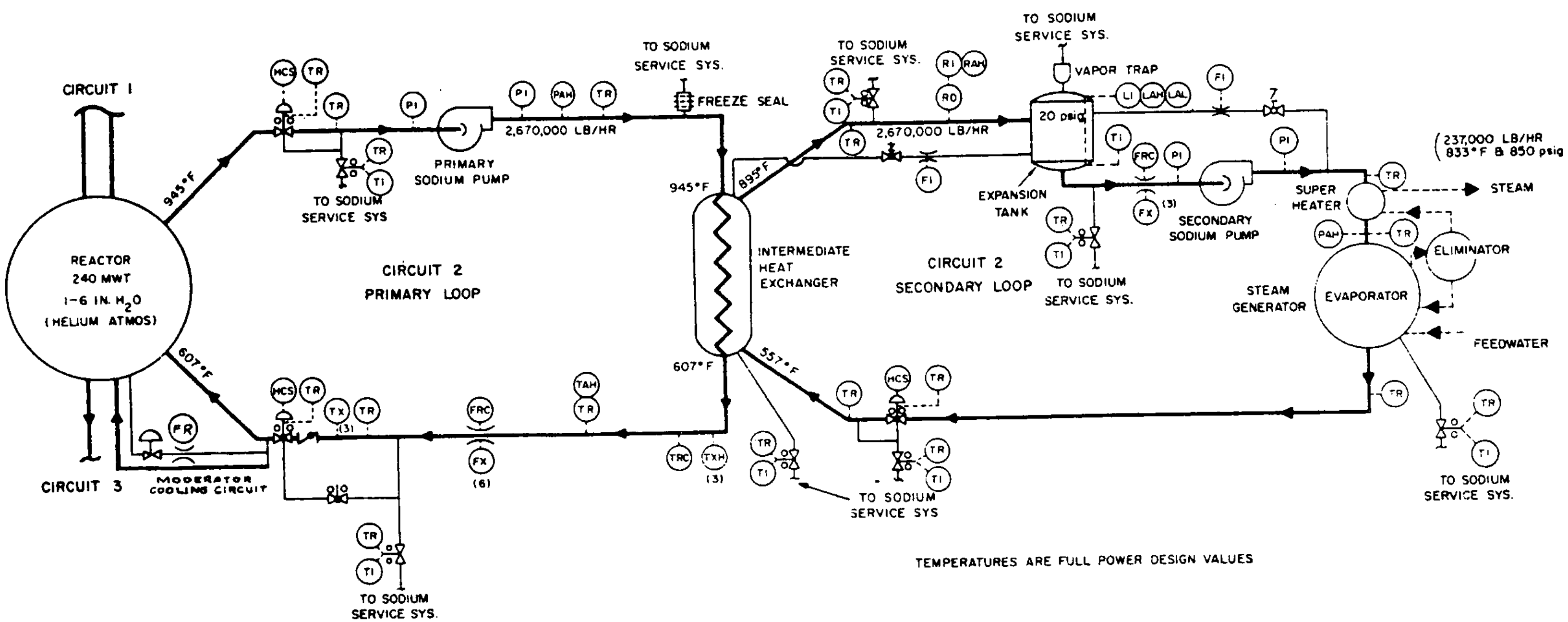

Figure 13. HNPF - Sodium Heat Transfer Circuit

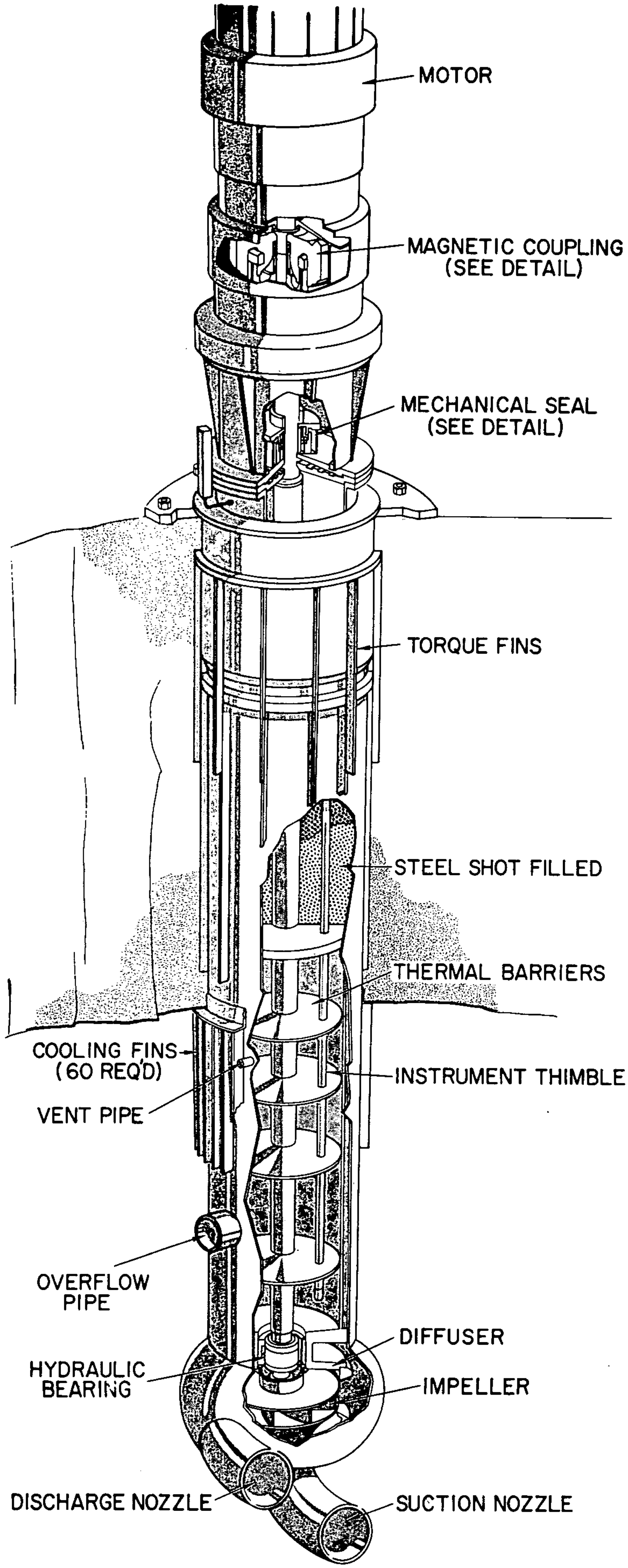

Figure 14. HNPF - Primary Sodium Circulating Pump

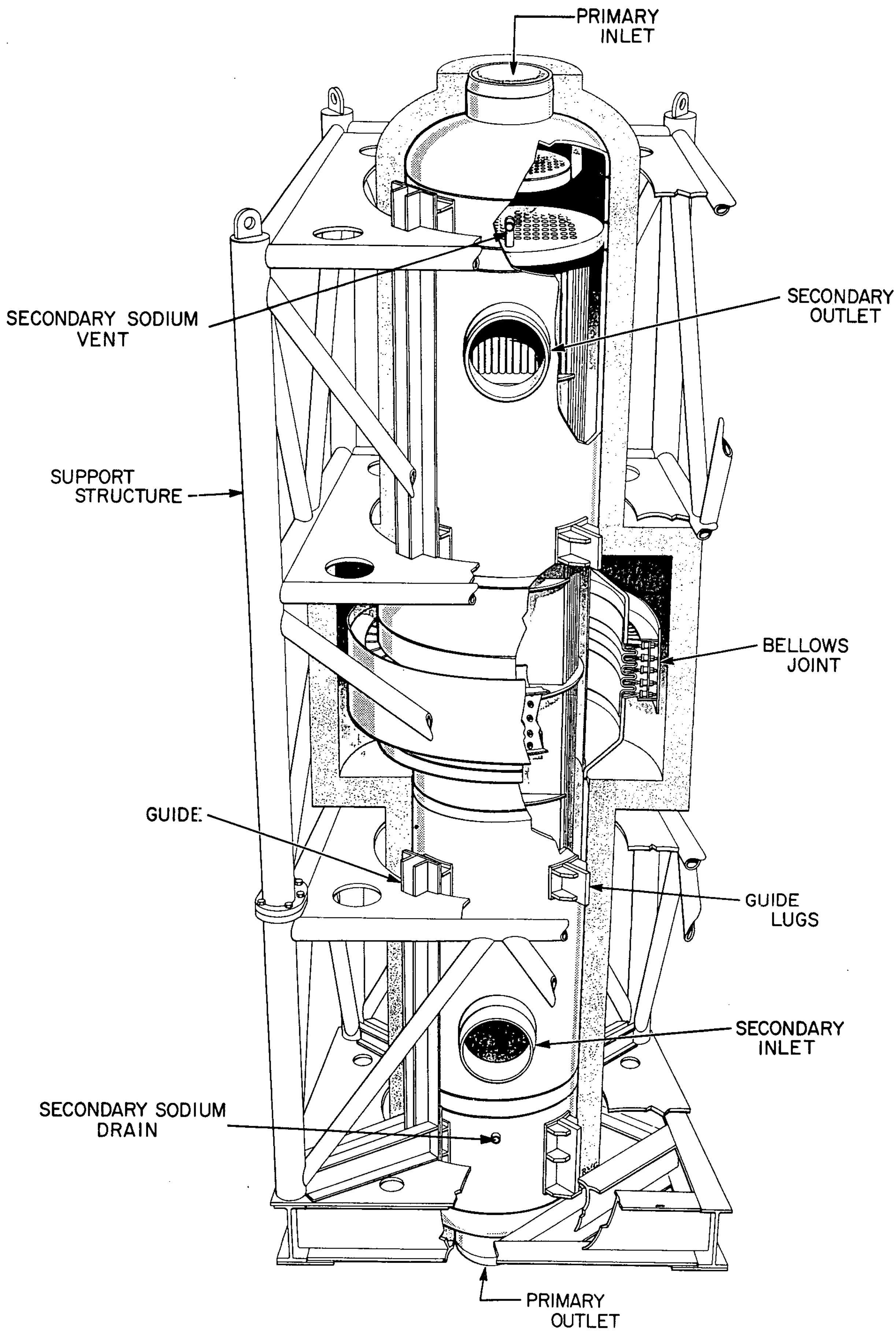

Figure 15. HNPF - Intermediate Heat Exchanger

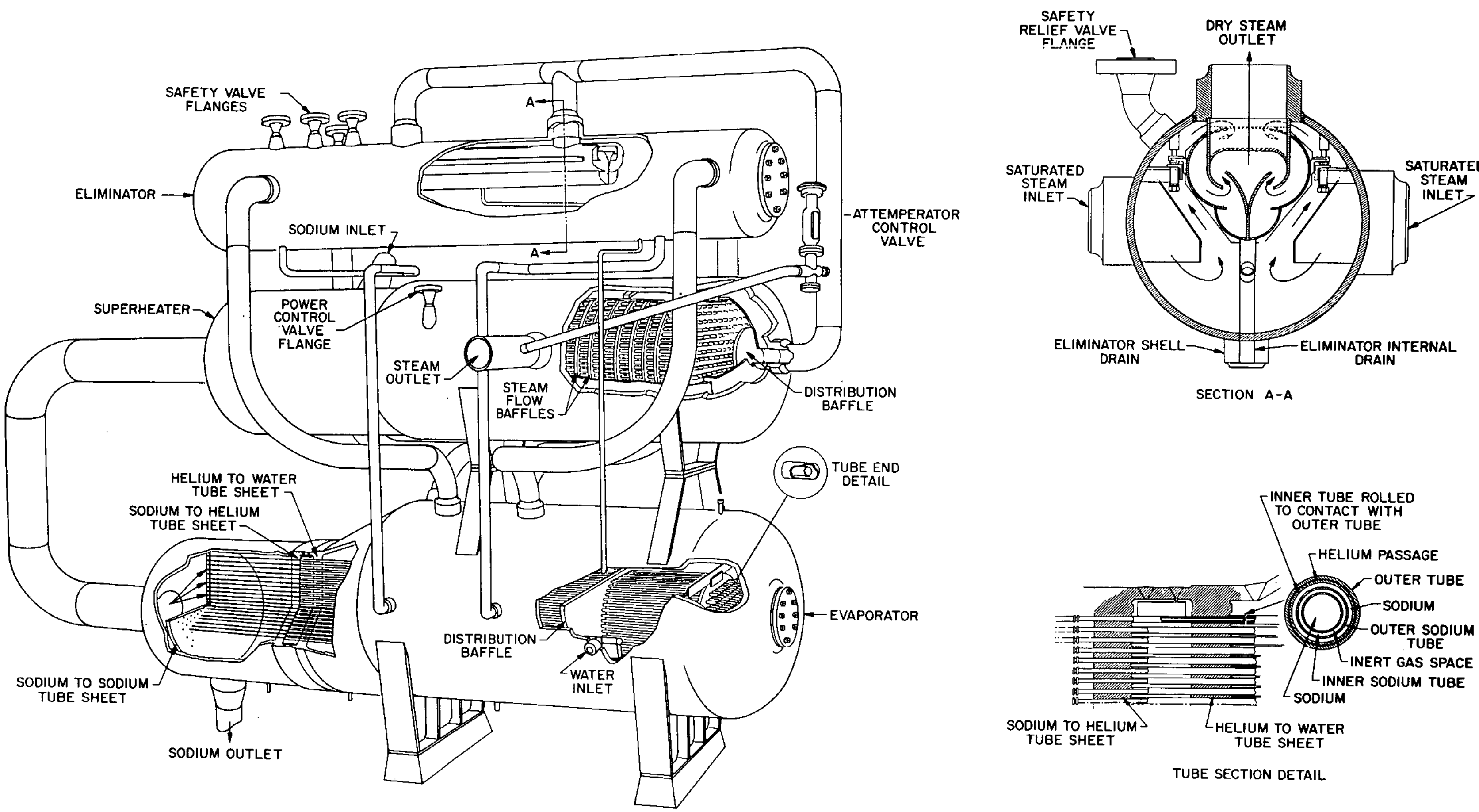

Figure 16. HNPF - Steam Generator

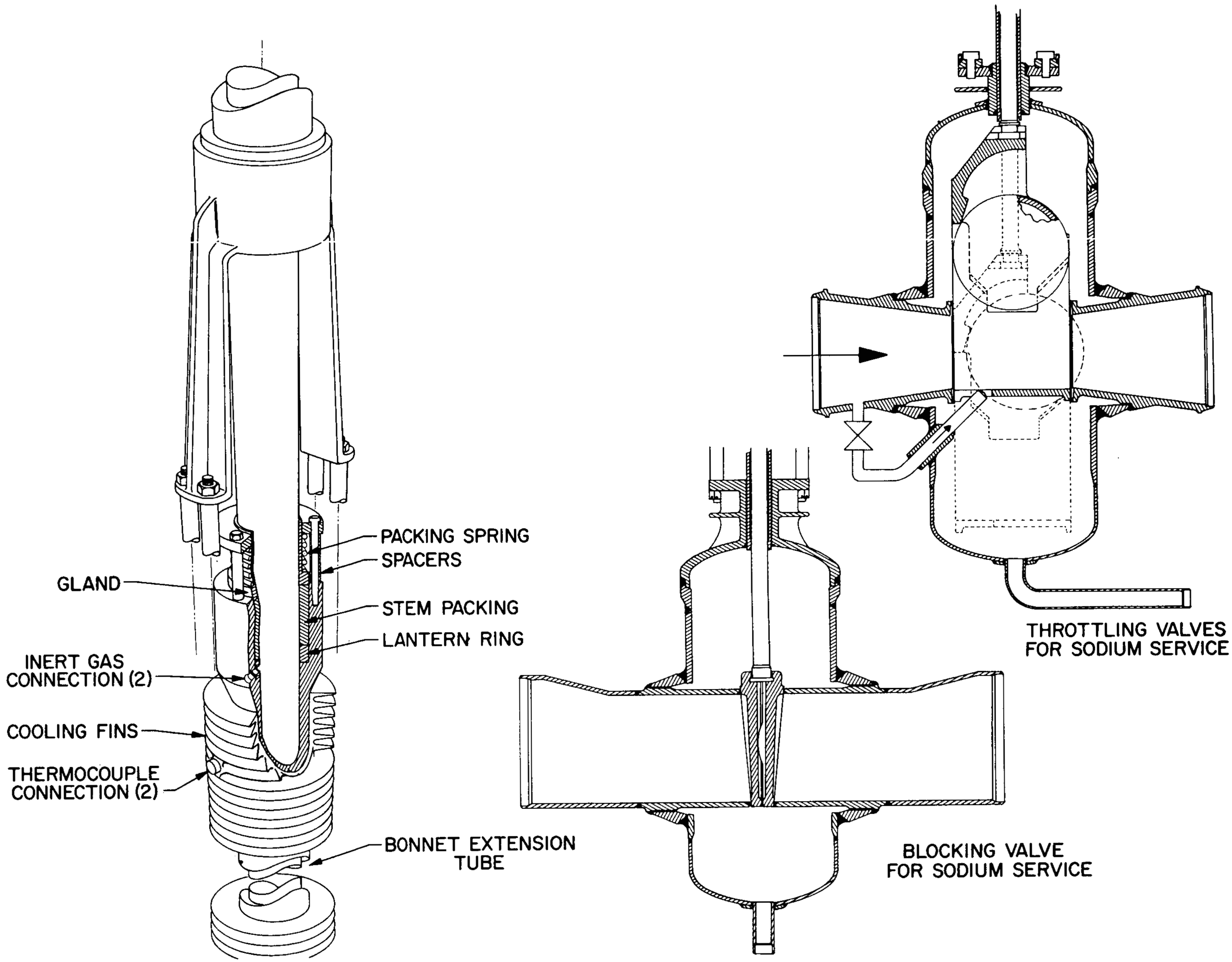

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Figure 17. HNPF - Freeze Seal Valves

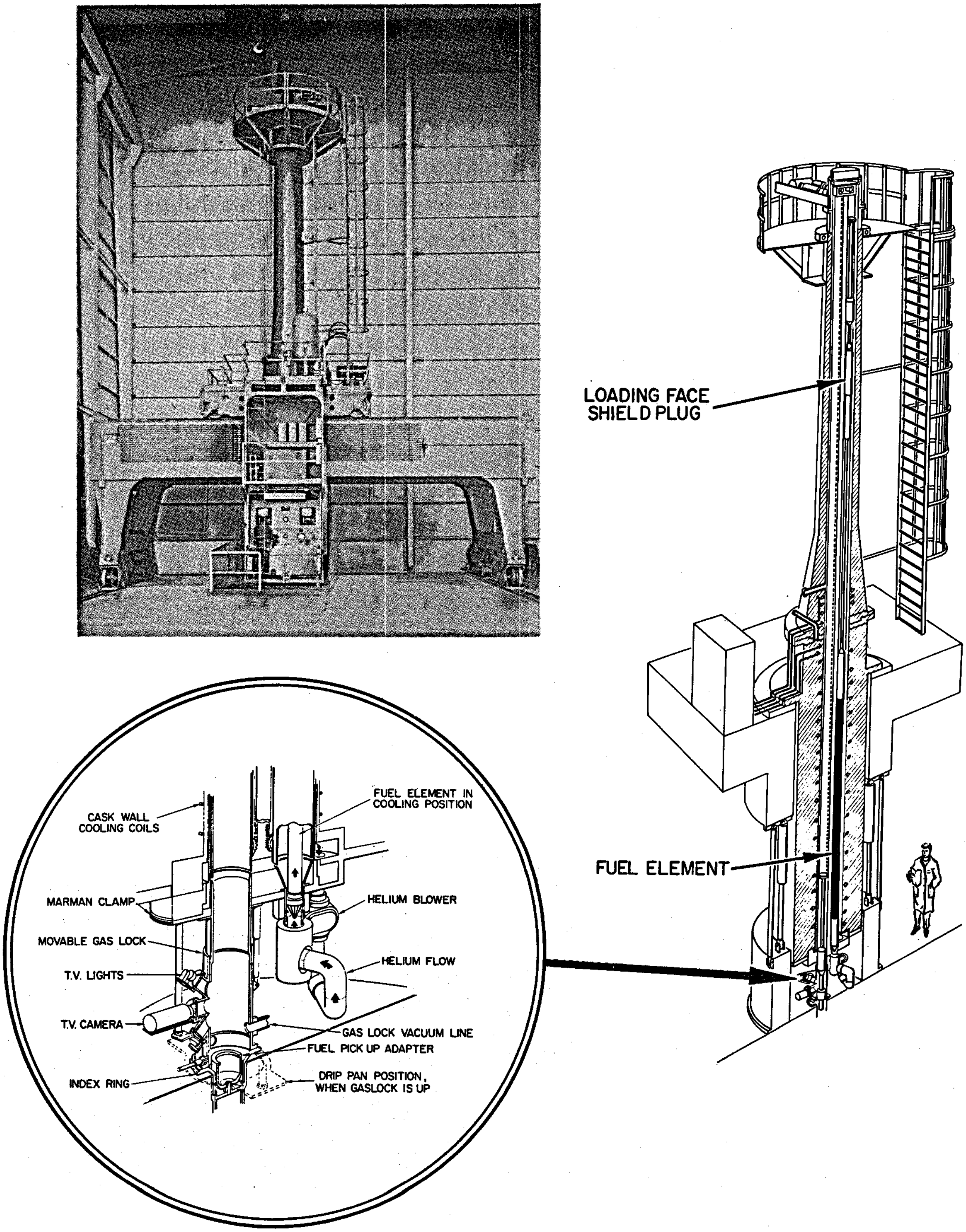

Figure 18. HNPF - Fuel Handling Machine

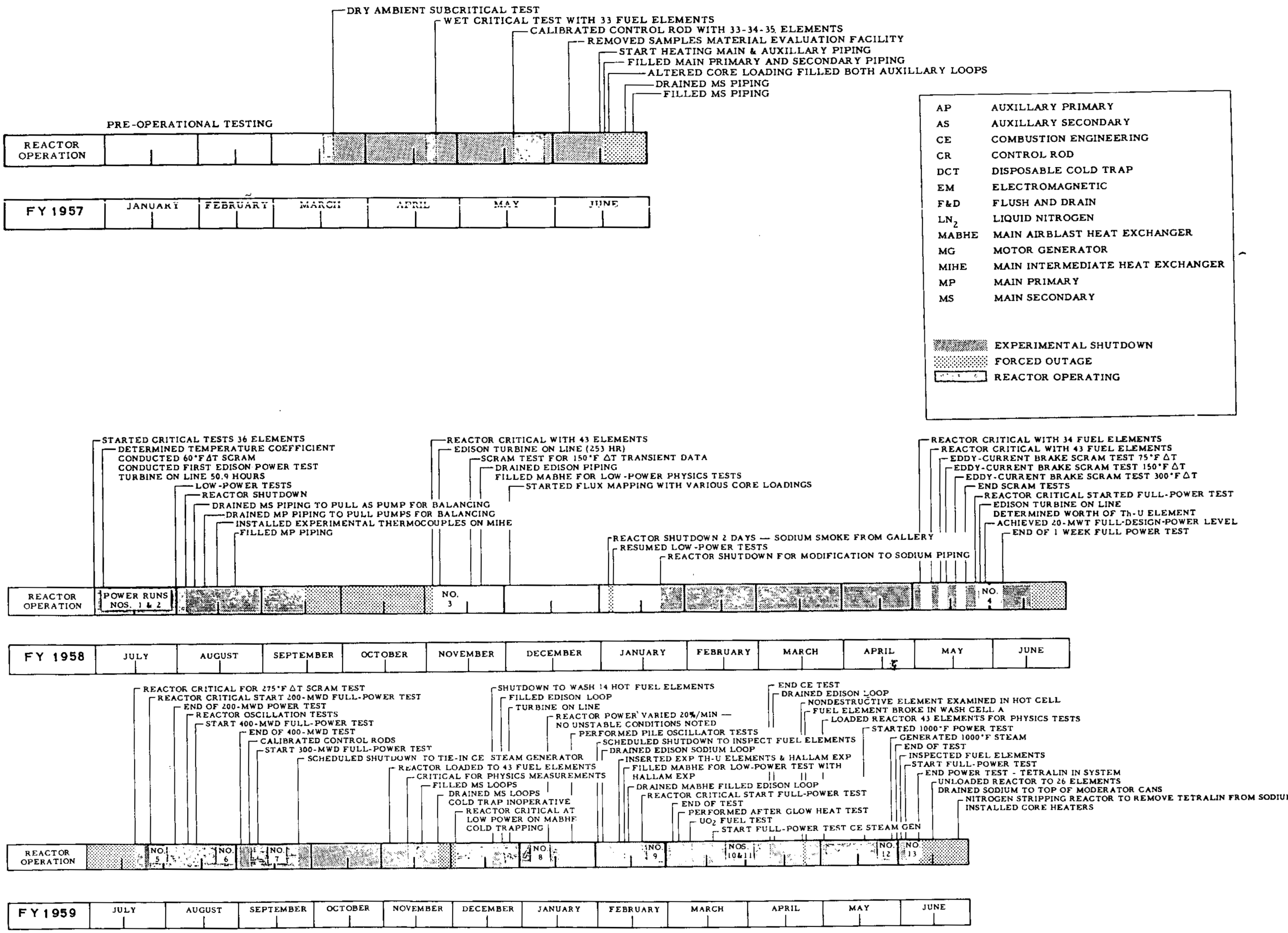

Figure 19. SRE - Operating History

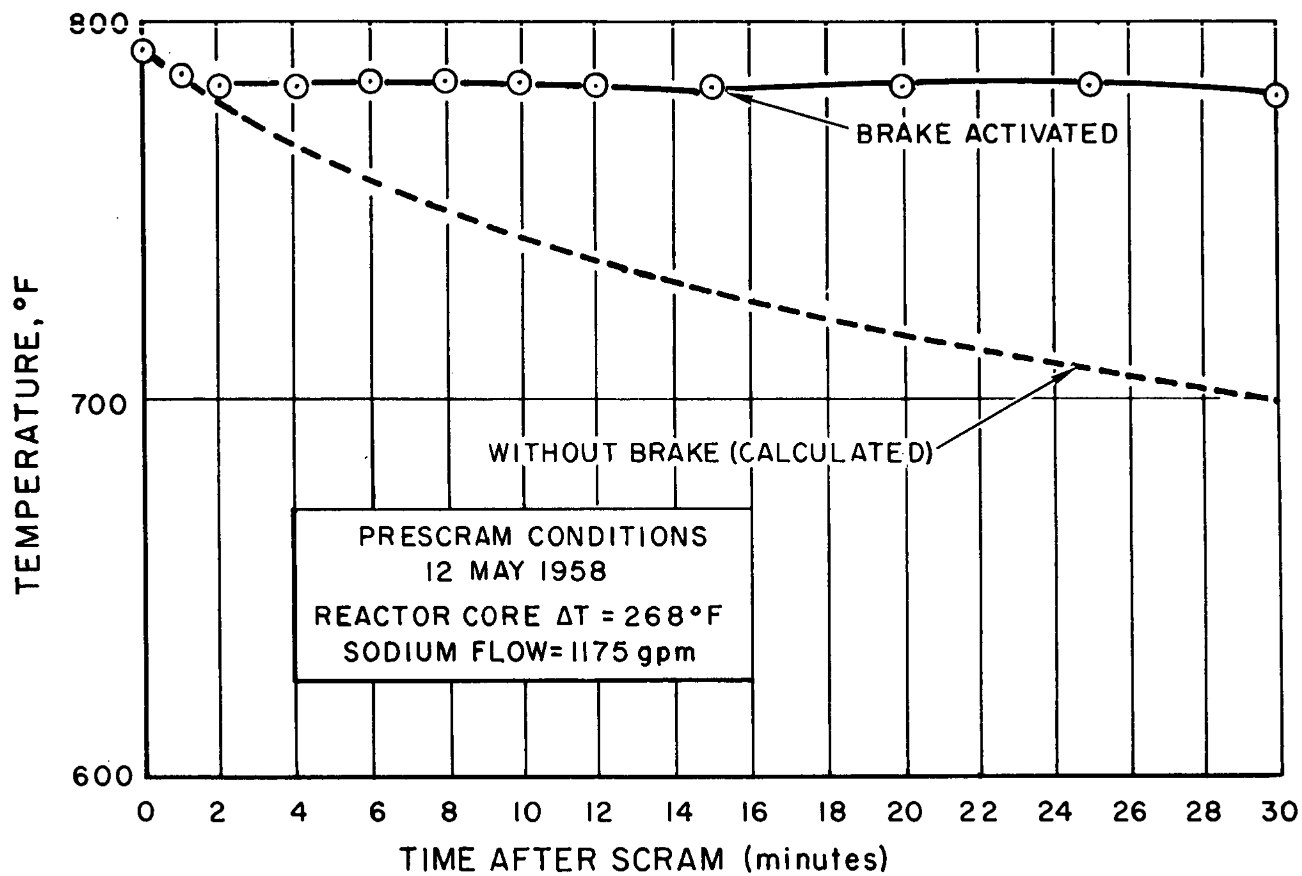

Figure 20. SRE - Brake Effect on Reactor Outlet Temperature After Scram

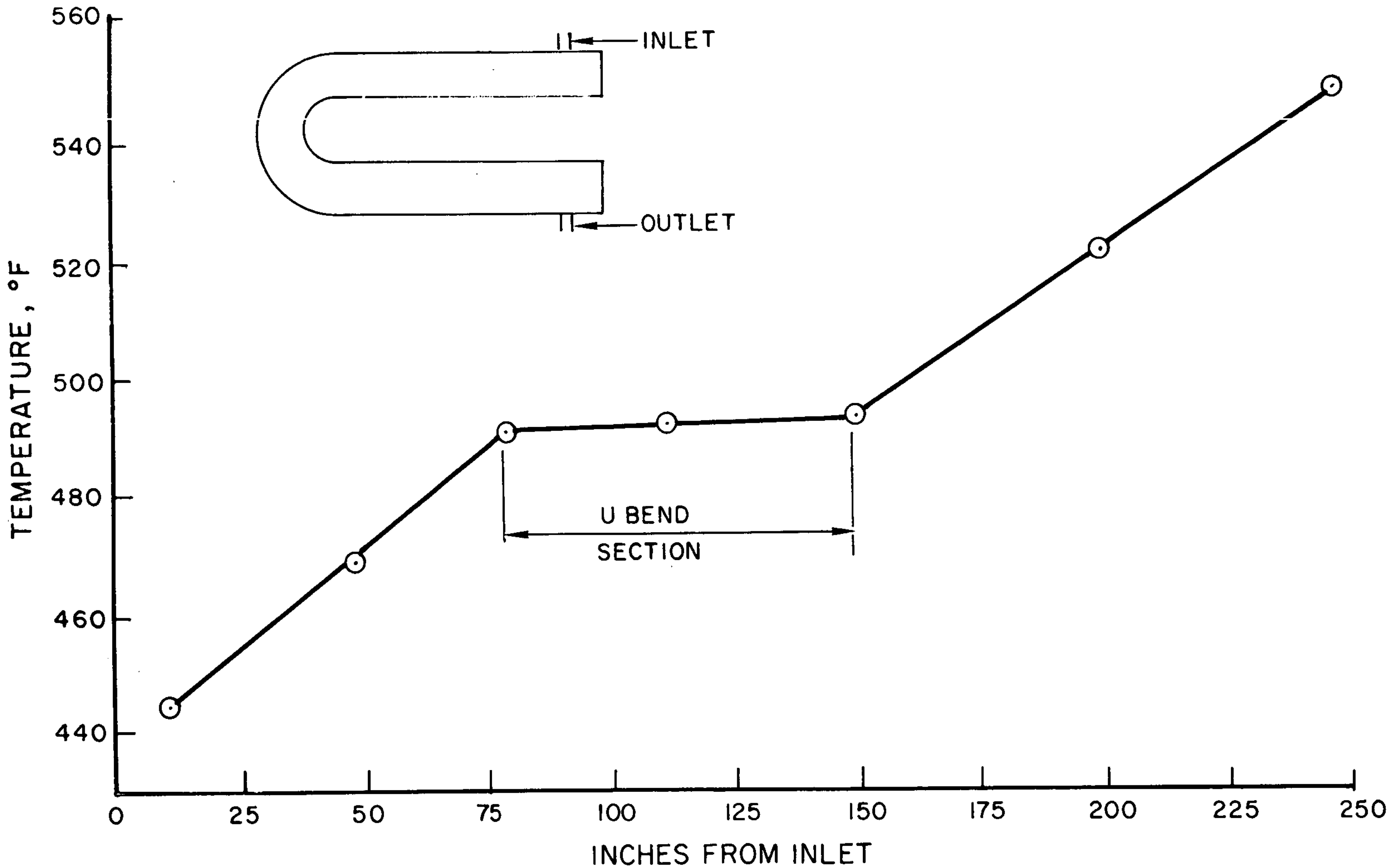

Figure 21. SRE - Temperature Distribution Around Shell of Intermediate Heat Exchanger

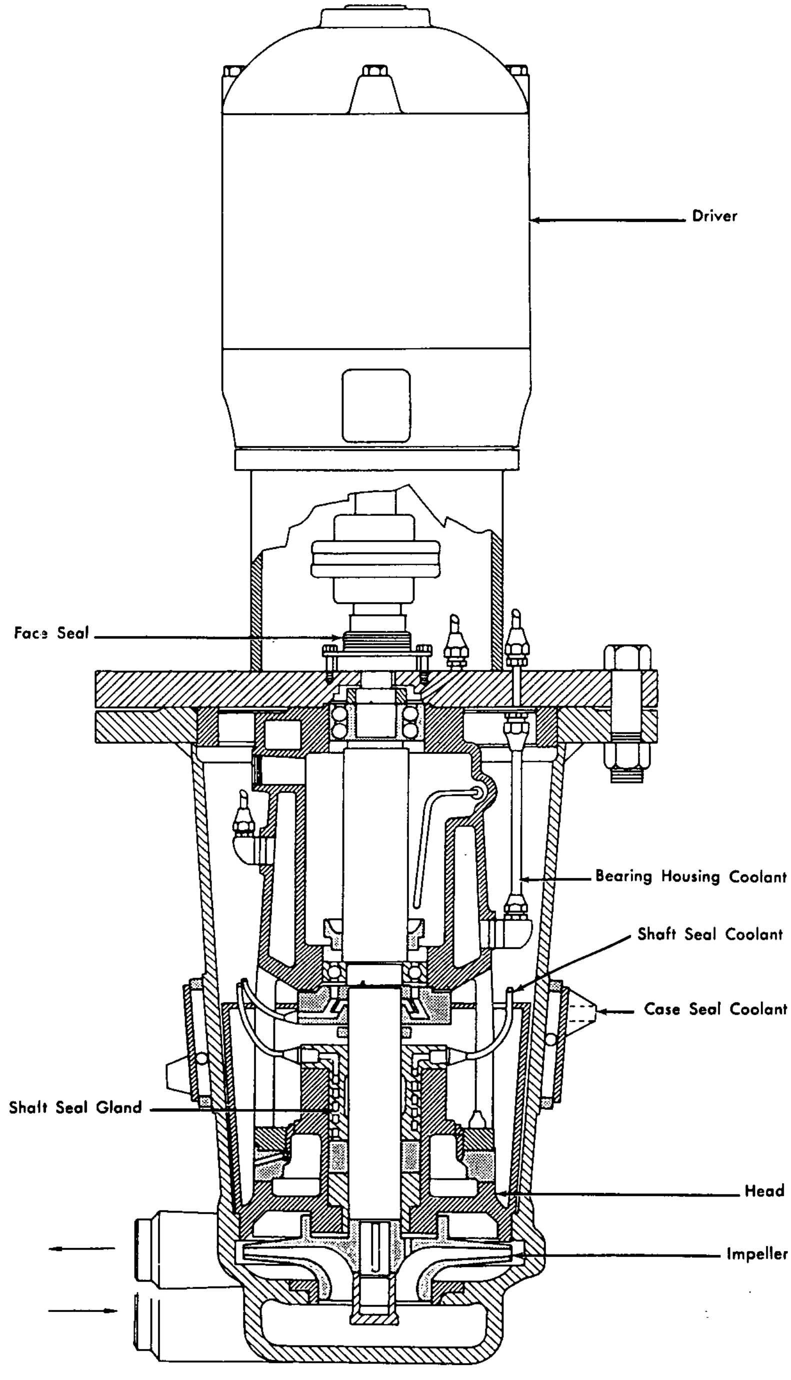

Figure 22. SRE - Secondary Sodium Pump

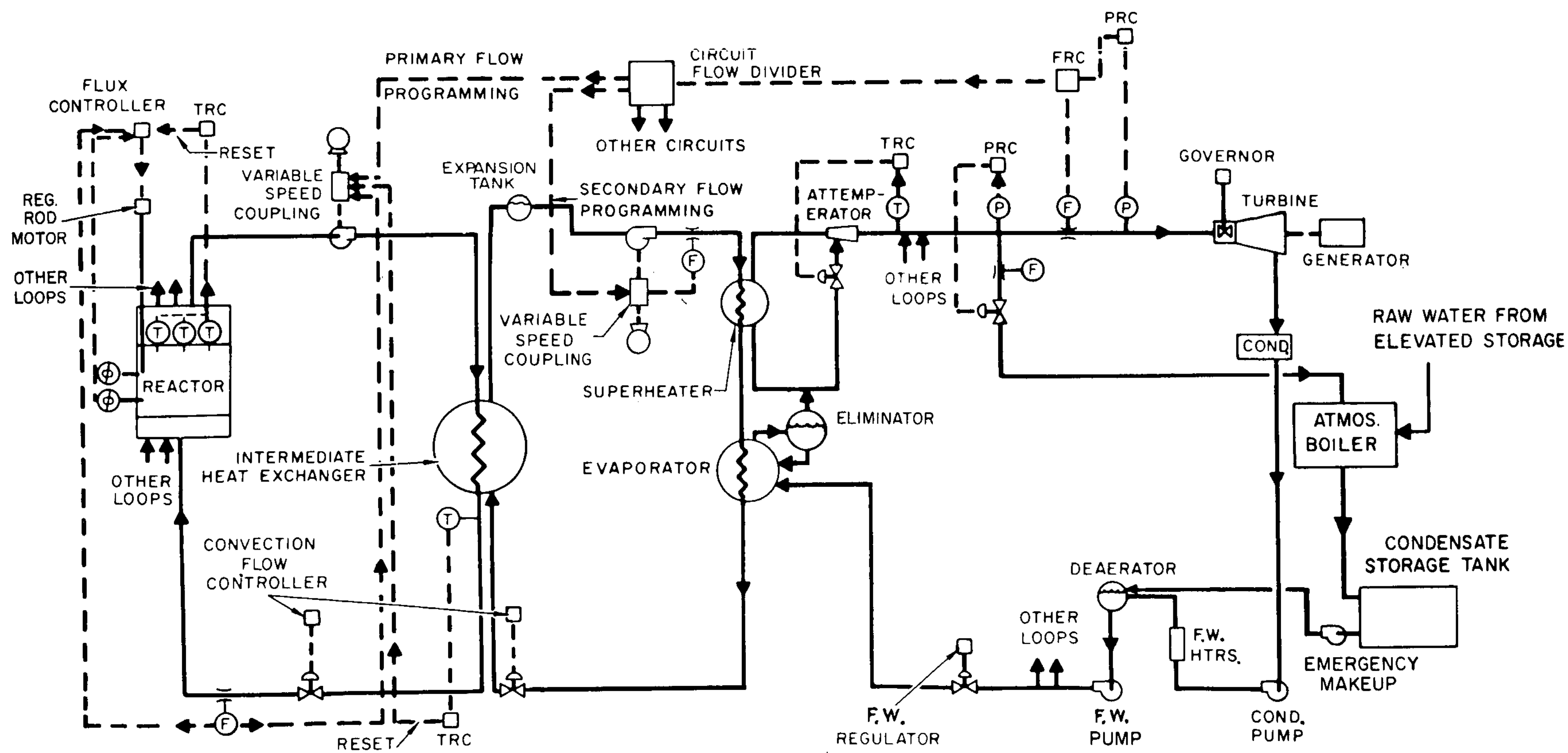

Figure 23. HNPF - Plant Control System