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EBR-I, MARK III - DESIGN REPORT

by

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INTRODUCTION

The EBR-I reactor was designed to demonstrate the feasibility of power production with a liquid metal-cooled, fast neutron reactor and to demonstrate that such a reactor could breed fissionable material. The reactor first produced power in 1951; during the next four years two experimental cores, Mark I and Mark II, were used to produce 4,000 megawatt hours of heat, a good portion of which was used in the generation of electricity for plant operation.

Experiments with both cores showed, that with this design, a breeding ratio of one is achieved. The experiments further indicated that a practical power plant fueled with uranium-235 might have a breeding ratio of 1.2 and a plutonium-fueled plant a breeding ratio of 1 5

In normal operation the plant was quite stable and largely self-regulating. The reactor system proved to be quite reliable and gave excellent service during almost four years of operation.

It was found, however, that when operating conditions were made severe, e.g., high power at reduced coolant flow, an oscillation in power occurred which became more severe as power was increased or flow decreased. It had been further observed that, although the over-all temperature coefficient of the reactor was negative a prompt positive coefficient was noted which was overcome by a slower negative coefficient

Experiments to investigate these effects were undertaken as a last use of the second core. During the course of, and before the completion of, the experiments, a transient test resulted in the melting of many of the core rods.

Since the question of stability in reactors is of extreme importance to the power reactor business, it was decided to design a third core for EBR-I which might be used to investigate and demonstrate its stability characteristics. In this new design called Mark III, it was decided to eliminate, in as far as practicable, the possiblity of fuel rod bowing, and hence to eliminate the most likely contribution to a prompt positive power coefficient. To investigate the effects of coolant temperature, means are provided for making the coolant flow through core and inner blanket either in parallel or series, as opposed to the series flow of the original design. Provision for extensive temperature measurements are provided. Oscillation techniques will be used to investigate the power coefficients and to point out areas for future study.

A. REACTOR INTERNALS

1. Fuel Rods

In both Mark I and Mark II, the fuel rods were made of stainless steel tubing containing loosely fitted uranium slugs held concentrically with spacers. The annular clearance was filled with NaK as a heat transfer bond.

In order to achieve a more rigid core for Mark III, a new type of EBR-I fuel rod, which gives a good metallurgical bond between core and clad, has been developed. This rod is shown in Figure 1. It consists of a uranium-2% zirconium alloy rod, 0.364 inch in diameter, with a 0.020-inch thick Zircaloy-II cladding made by the coextrusion process. The three 0.046-inch diameter zirconium wire ribs are equispaced and spot welded to the cladding at 1/4-inch intervals. A fuel rod is made up of three pieces welded end to end with a 0.020-inch Zircaloy spacer between them. The center core piece of highly enriched uranium-2% zirconium alloy is 8-1/2 inches long. The upper and lower blanket pieces of natural uranium-2% zirconium alloy are 7-3/4 and 3-9/16 inches long, respectively. On the bottom end is a Zircaloy-II triangularly shaped tip for location and orientation. On the upper end is a Zircaloy-II piece with a threaded hole into which the Type 304 stainless steel handle is screwed and locked. This extension is used for handling purposes and as part of the upper shield. It has a reduced section from blanket to plenum chamber to facilitate flow and a special head on the upper end to receive the handling tool. The maximum capacity of the core is 252 fuel rods containing 60 kg of enriched uranium.

2. Blanket Rods

The design of the blanket rod uranium section is the same as that of the fuel rod with one exception. The blanket rod is made up of one 19-13/16-inch rod of natural uranium-2% zirconium alloy. There are two lengths of blanket rods: a rod with a short handle which is used in the blanket assemblies, and a rod with the same length handle as the fuel rods for use in the core to fill spaces not occupied by fuel rods. The capacity of the blanket zone is 432 rods.

3. Fuel Assemblies

As part of the rigid core concept of the Mark III design, rod assemblies are being used instead of the individual rod scheme used in the original EBR-I design (see Figure 13). As shown in Figure 2, the fuel rod assembly is contained in a hexagonal Type 304 stainless steel tube of 2-7/8 inch outside dimension across flats with a wall thickness of 0.040 inch. At the bottom of the assembly tube is a nozzle which reduces from the hexagonal shape to that of a cylinder. Resting on the nozzle is a rod sheet which supports, locates and orients the rods. The triangular holes in the rod sheet receive the triangular tips of the rods. The round holes around the triangular holes allow the NaK coolant to pass through the sheet and into the spaces between the rods. The middle set of holes on the fuel assembly tube allows the NaK coolant to flow out of the assembly into the outlet plenum chamber. The upper set of holes is for overflow.

Each fuel assembly contains 36 fuel rods plus one tightening rod. The purpose of the tightening rod is to force the fuel rods out against the inside of the hexagonal tube, thus making a rigid assembly. The tightening rod is shown in Figure 3. It is essentially an expandible rod, consisting of an outer split tube which may be expanded by means of a series of Woodruff key-type wedges riding in modified Woodruff key slots in the center shaft. The expansion is actuated by means of a nut on the center rod, at the top of the assembly, which moves the center rod relative to the wedges on the split tube. The ribs on the fuel rods maintain the design spacing and flow area. All of the rods in the fuel assemblies have long handles reaching to the top of the reactor tank. Fuel may be handled either by replacing the whole assembly or by loosening the tightening device and replacing individual rods.

4. Blanket Assemblies

The blanket assemblies are similar in design to the fuel assemblies but with the following exceptions. The blanket assemblies contain only short handle rods. There is a seal plate between the inlet and outlet plenum chamber to prevent short circuit flow. Coolant flow holes are located both above and below the seal plate. Since there are no long rod handles to serve as shielding, the upper two feet of the blanket assembly is made of solid steel. The tightening rod for the blanket assembly does have a long handle which can be operated from the top of the reactor. However, rods in the blanket assembly can only be replaced by removing the whole hexagonal tube, placing the lower section in a shielded hole, removing the flat head screws at the seal plate, and lifting off the upper section, thus exposing the rods.

5. Inner Tank Assembly

Figure 4 shows a cross section taken through the core center line. The inner seven hexagons are fuel assemblies and the outer ring of twelve hexagons are blanket assemblies. Located at the core center line on each of the six flats on the outer periphery of the assemblies is a double wedge-type clamping device used to force the outer assemblies against the center one. The outer, axially moving wedge is made of aluminum bronze and the inner radially moving wedge is of stainless steel. These wedges are operated individually by means of reach rods by screw and nut actuators on the top plate of the inner tank assembly. Also shown on this drawing are the twelve downcomer holes through which the inlet NaK flows under parallel flow conditions, the six tie rods for the lower part of the structure, the antimony-beryllium source, and the oscillating rod location.

The new Mark III design covers only that part of the EBR-I structure contained inside the EBR reactor tank. This structure, known as the Inner Tank Assembly, is shown in the cutaway drawing, Figure 5. This drawing also shows the inner and outer reactor tank, the outer blanket, control and safety rods, inlet and outlet piping and some of the lower shielding, all of which are unchanged. The structure inside of the reactor tank is made up of laminations of Type 304 stainless steel rings which have a cylindrical outside surface and an inner surface forming a hexagonal pattern to fit the hexagonal blanket assembly outline. These stainless steel rings are aligned and held together with tie rods. At the bottom of this structure is a fuel assembly-supporting plate known as the tube sheet, which has circular holes to receive, support and locate the nozzles of the rod assemblies. To enable the outer assemblies to be clamped against the center assembly the clearance between the nozzle and the hole in the tube sheet is small for the center assembly and progressively larger toward the outer part of the plate. The lowest large diameter plate, called the mounting plate, supports the entire inner tank structure on a ledge in the reactor tank. In the region just above the mounting plate and outside the rod assemblies is the inlet plenum chamber. In this plenum chamber is an outer ring baffle which forms an annular flow distributing chamber between the tank wall and the baffle. Between the baffles are located four three-way inlet valves with spool-type discs. Their purpose is to permit changing of core and blanket flow to either parallel or series. These valves may be operated individually from the top plate by means of extension shafts and nut and screw actuators.

Immediately above the inlet plenum chamber is the seal plate. On its outer periphery are two inconel seal rings, similar to piston rings, which seal the area between the inlet and outlet plenum chambers. These rings may be expanded or contracted by a screw and toggle mechanism buried in the seal plate and operated from the reactor top. This design is the same as in the original reactor. The aluminum bronze bushings for the inlet valve shafts are located in the seal plate. On the inner edge of this plate are the seal plate clamps, consisting of six segmental shoes, made of aluminum bronze, which clamp the fuel and blanket assemblies into a rigid bundle and also minimize bypass leakage. These clamps have a hexagonal pattern on their inside edge to fit the outline of the blanket assembly tube bundle. The shoes are moved in a radial direction by a toggle mechanism buried in the seal plate and are operated from the reactor top by a screw and nut actuator. Also in this plate are two throttle valves which are used to control the exit flow from the inner blanket under parallel flow conditions.

On top of the seal plate is a thermal baffle cooled by inlet NaK bled through the seal plate. The purpose of the thermal baffle is to reduce the temperature gradient across the seal plate and thus inhibit the warping of the plate. Inlet NaK is bled through the seal plate by means of a ring of holes near the outside. The NaK then flows radially inward between the top of the seal plate and the thermal baffle and exits at the inner edge of the thermal baffle into the outlet plenum chamber. Without this thermal baffle it would be possible to get a thermal gradient across the plate of about 90°C. With 17.5 gpm of inlet NaK bypassed through the baffle at a total flow of 300 gpm, the calculated gradient is 25°C.

Located above the seal plate is the outlet plenum chamber which collects the flow from the fuel assembly outlet holes and the throttle valves. It is formed by spacer tubes around the tie rods. The outlet coolant line drains this region.

The coolant flow path under each flow condition is as follows:

Under series flow conditions the cool NaK coming in the inlet line is distributed in the annular inlet plenum chamber from which it passes into the inlet valves, which will then be in their "up" position. The NaK enters the upper annular plenum chamber, between the two ring baffles, and then flows into the outer ring of blanket assemblies. The coolant goes down around the blanket rods and into the lower plenum chamber where it is reversed 180 degrees and flows up through the inner seven fuel assemblies to the outlet holes in these assemblies. The flow is then radially outward, through the perforated portion of the blanket assembly, into the outlet plenum chamber and the outlet pipe.

Under parallel flow conditions the inlet valves are in their "down" position. Flow is then from the inlet valve down into another lower annular plenum chamber just above the mounting plate. Here the NaK is distributed to the twelve downcomers and goes down to the lower plenum chamber. Part of the coolant then goes up through the outer ring of blanket assemblies and out of the holes below the assembly seal plate into the upper annular plenum chamber and then through the throttle valves to the outlet plenum chamber. The remainder of the flow goes through the fuel assembly as formerly described. Under this condition of parallel flow the throttle valves may be used to adjust the ratio of blanket flow to core flow in order to attain the desired temperature rise across the blanket.

Proper positioning of the valves for either series or parallel flow is assured by limit switches on the valve-actuating mechanism. These switches prevent operation of the reactor and cause a scram if the reactor is in operation, unless either (1) the inlet valves are fully up and the throttle valves completely closed, thus assuring series flow, or (2) the inlet valves are fully down and the throttle valves are opened to a preset minimum, thus assuring parallel flow. Proper valve arrangement is shown by indicator lights in the control room.

The inner tank assembly above the outlet plenum consists of laminated stainless steel shielding, an overflow plenum at the level of the tank overflow line, and the top plate. On this top plate are located the actuators for the core clamp, seal plate clamps, seal rings, inlet valves, and throttle valves. The actuators are all individually operable with an extension wrench during shutdown. However, the throttle valve actuators are mechanically linked by a roller chain. An extension shaft through the top of the reactor tank and the top shield permits flow adjustments during reactor operation.

Figure 6 shows the completed Inner Tank Assembly.

6. Heat Transfer and Hydraulics

The reactor power, coolant flow, and reactor temperatures will be approximately the same in Mark III as for the EBR-I, Mark I and Mark II, with the original core structure. The internal blanket flow may be throttled in parallel flow, to have a temperature rise equal to that in the core. The flow through the core and blanket will then be proportional to the power generation in each section. It is expected that an appreciable part of the internal blanket will be in the outer rows of rods of the core subassemblies adjoining the blanket flow path. Therefore, the ratio of the power generated in the blanket flow sections to the power in the core flow section will be somewhat smaller than in the first and second loadings.

Table I compares the three loadings as to dimensional, heat transfer, flow, and nuclear data. The flows, temperature and power distribution between core and blanket are assumed to be the same in each case.

Table I

Summary of Design Characteristics, EBR-I

	Mark I	Mark II	Mark III
1. Flow and Temperature Conditions - Reactor Core and Inner Blanket			
Temperature of NaK in, °C	228	228	228
Temperature of NaK out, °C	316	316	316
Flow rate, gpm	292	292	292
Total power produced, kw	1158	1158	1158
Btu/hr	3.95×10^6	3.95×10^6	3.95×10^6
Power produced in core, kw	960	960	960
Power produced in internal blanket, kw	196	196	196
2. Dimensional Data - Reactor Core			
Fuel rod lattice spacing, in.	0.494	0.494	0.450
Jacket O.D.,in.	0.448	0.448	0.404
Fuel slug diameter, in.	0.364	0.384	0.364
NaK bond thickness, in.	0.020	0.010	0.000
Jacket, 0.020-inch wall	347 SS Ribbed	347 SS Plain	Zircaloy-II Ribbed
Cross-sectional area for coolant flow, ft^2	0.1008	0.1008	0.0905
Coolant velocity, fps	6.5	6.5	7.4
NaK flow area per lattice triangle, in^2	0.0248	0.0248	0.0204
Percent flow area in lattice	23.5	23.5	23.2
NaK flow area per rod in lattice, in^2	0.0496	0.0496	0.0408
Per cent NaK flow area of total area	28.2	28.2	26.5
Core volume, liters	5.9	6.1	6.07
Total area of core section, ft^2	0.338	0.338	0.349
Total fuel rod surface area, ft^2	15.68	16.2	16.4
3. Heat Transfer Data - Reactor Core			
Average heat flux, cal/sec-cm^2	15.7	15.2	15.0
Btu/hr/ft^2	209,000	202,000	200,000
Average power density kw/liter	172	167	158
Average specific power kw/kg	18.1	18.4	18.3
Ratio maximum/average power	1.25	1.35	1.35
Maximum heat flux, cal/sec-cm^2	19.6	20.5	20.3
Btu/hr/ft^2	262,000	273,000	270,000
Maximum specific power, kw/kg	22.6	24.9	24.7
Temperature difference in U slug center to surface, maximum, °C	69.2	81.0	71.8
Temperature difference across NaK bond, maximum, °C	17.5	9.5	0
Temperature difference across jacket, maximum, °C	17.2	20.0	23.0
Temperature difference across coolant film, maximum, °C	11.9	12.5	12.3
Total temperature difference, NaK coolant to slug center at hottest point of slug, °C	113	120	105
NaK coolant temperature at hottest point of slug, °C	287	287	287
Maximum slug temperature, °C	400	407	392

Table I (Continued)

	Mark I	Mark II	Mark III
4. Inner Blanket			
Total surface area of rods, ft^2	56.2	56.2	30.4
Average heat flux, cal/sec-cm^2	0.89	0.89	1.70
Btu/hr/ft^2	11,900	11,900	22,500
Total cross-sectional area of blanket section, ft^2	1.02	1.02	0.599
Uranium area, ft^2	0.57	0.57	0.312
Cross-sectional area for coolant flow, ft^2	0.368	0.368	0.155
Coolant velocity, series flow, fps	1.77	1.77	4.3
5. Outer Blanket			
Inlet air temperature, °C	20	20	20
Outlet air temperature, °C	108	108	108
Air flow rate, ft^3/min	5,800	5,800	5,800
Power produced in outer blanket, kw	213	213	213
6. Nuclear Data			
Critical mass (wet, cold), kg	51.5	48.2	47.5
Core composition, Volume %			
U	48.2	48.9	49.5
SS Type 304	15.3	15.3	7.3
NaK	36.50	32.9	25.6
Zr	-	2.9	17.6
Radial Inner Blanket Composition, Volume %			
U	70.8	70.8	48.9
SS Type 304	9.3	9.3	7.3
NaK	19.9	19.9	25.6
Zr	-	-	18.2
Upper and Lower Blanket Composition, Volume %			
U	48.2	48.9	48.9
SS Type 304	15.3	15.3	7.3
NaK	36.5	32.9	25.6
Zr	-	2.9	18.2

Calculated flow resistances and bypass leakage rates for full flow in series and parallel are given in Table II. The flow resistance through this core will be considerably greater than through the original EBR-I core. As the flow through the reactor is from a gravity feed tank, the available driving head is limited. In series flow there should still be available about 3.5 feet of NaK excess head over that required for full flow through the reactor.

Figure 7 shows a graph of pressure drop versus vertical distance along a fuel assembly. Figure 8 shows a pressure drop versus flow rate curve. The measurements from which these data were obtained were run on a mockup fuel assembly using water at 64°F. The water data was then correlated to NaK.

The major bypass flows will be the seal plate clamp shoe leakage and the seal plate thermal baffle flow. The total leakage will be an appreciable percentage of the reactor flow. The actual flow through the blanket and core sections may be found from a heat balance determining the total reactor power and the coolant temperature rises through the core and blanket sections.

Table II

EBR-I, Mark III, Pressure Drop and Bypass Leakage Data

	Ft of NaK	Leakage, gpm
1. Pressure Drop, Series Flow, 300 gpm		
Inlet plus distribution annulus	4.25	
Inlet valves	8.65	
Blanket rods	3.16	
Core rods	8.60	
Total	23.9	
2. Pressure Drop, Parallel Flow, 300 gpm		
Inlet plus distribution annulus	4.25	
Inlet valves	8.65	
Downcomer	3.28	
Core and blanket rods	4.91	
Total	21.1	
3. Available Head		
Gravity tank overflow	132.78	
Reactor outlet	-105.35	
Net head available	27.43	
4. Excess Head		
Series flow	3.53	
Parallel flow	6.33	
5. Bypass Leakage, Series Flow		
Seal plate shoes	12.1	10.17
Seal shoe rods	6.05	2.48
Inlet valve rods	12.1	3.49
Core clamp rods	6.6	1.59
Three open thermocouple holes in seal plate	6.6	.39
Blanket assembly tightening rod handle	12.1	2.36
Seal plate thermal baffle	12.1	17.50
Blanket thermocouple assembly	6.6	2.09
Total Leakage		40.07

7. Reactor Physics

Critical Mass

The most accurate estimate of the cold critical mass of the EBR-I, Mark III, is obtained from the critical mass of the Mark II loading by extrapolation through critical experiments performed in the Argonne fast reactor critical facility, ZPR-III.

An assembly closely simulating the composition and geometry of the EBR-I, Mark III, was set up in the ZPR facility. The closeness of the mock up to the composition and geometry of the actual reactor may be seen by comparing Figures 9 and 10. The minor differences shown, due to the finite size of the building blocks available to ZPR-III, were corrected for by substitution measurements. One major difference existed. Since the sodium-potassium eutectic used as coolant was not available for use in ZPR-III, aluminum was used to mock up the coolant. The critical mass of the assembly was 46.5 kg of U^{235}.

To avoid complications arising from the substitution of aluminum for NaK, a similarly careful mock up of the EBR-I, Mark II, reactor was constructed using the same aluminum for NaK equivalence used previously. The critical mass for this assembly was 46.25 kg of U^{235}.

Since the cold, wet critical mass of the EBR-I, Mark II, as determined in the reactor was 48.2 kg we arrive at the cold, wet critical mass for the Mark III loading as

$$48.2 \times \frac{46.5}{46.25} = 48.5 \text{ kg } U^{235} .$$

The measured cold, wet critical mass on EBR-I, Mark III, is 47.5 kg U^{235}.

Steady State Operating Characteristics

The close similarity of the EBR-I, Mark II and III, cores in both loading and geometry indicates very little difference in most of the operating characteristics of the two reactors Control rod, safety block, and outer cup worths will be the same for the two, to well within the accuracy of calculation. The worth of fuel rods at the edge of the assembly is the same when corrected for U^{235} content of the rods. (Measurements of worth of uranium-235 versus void at the edge of the core in the two mock ups gave 430 ih/kg of U^{235} in Mark II, and 443 ih/kg of U^{235} in Mark III, a change of ~3%.)

Control and safety rod speeds and rate of reactivity insertion of Mark II and Mark III are similar. The following table is based on measured speeds of the mechanisms and reactivity calibrations of the safety devices. Reactivities were obtained from the delayed neutron data of LA-2118, adjusted to account for U^{238} fission using $\beta_i \times 10^3$ = 0.234, 1.377, 1.257, 2.760, 0.974, 0.228.

Reactivity Insertion

	Speed, Inches per Second	Reactivity Change, $\frac{\Delta k}{k}$ per Second
Control Rods - 4 (withdrawal)	0.64	1.4×10^{-5} (per rod)
Safety Rods - 8	0.64	1.4×10^{-5} (per rod)
Safety Block - 1	5	4.9×10^{-4} (avg)
Cup (80" to 30") - 1	0.32	-
Cup (30" to 4-1/4") - 1	0.095	6.7×10^{-4} (max. est.)
Cup (4-1/4" to 0") - 1	0.005	1.2×10^{-5} (max.)

Scram Speeds

	Time to Initiate Motion, Seconds	Total Time for Indicated Travel, Seconds	Reactivity Change, Per Cent $\frac{\Delta k}{k}$
Control Rods - 4		No Scram Provision	
Safety Rods - 8	0.085	0.38 to 16"	0.20
Safety Block - 1	0.15	0.35 to 6"	0.06
		0.275 to 4-1/4"	0.80
Cup - 1	0.1	0.46 to 12"	>4.00
		Total Travel	8.00

One change in static properties is expected. The static temperature coefficient of the Mark III reactor will be reduced by the effect of the lower expansion coefficient of the zirconium jacketed fuel rods. Preliminary measurements of the expansion coefficient of the rods indicate that the expansion coefficient will be about 70% of that of the unrestrained uranium rods used in Mark II. At a maximum this effect will reduce the isothermal coefficient of Mark III to about -1.23 ih/°C as opposed to the -1.37 ih/°C in Mark II. The measured value of the isothermal temperature coefficient on the Mark III core, within the range 30°C-210°C, is -1.22 ih/°C.

8. Temperature Instrumentation

The Mark III design includes an extensive system for measuring temperature. Thermocouples are located in the structure, plenum chambers, fuel and blanket rods and in the coolant channels between the rods.

There are eight permanent duplicate thermocouple locations in the structure at points where thermal expansions might lead to reactivity changes, as follows:

- S-1 In vertical center of tube sheet, adjacent to center fuel assembly hole.
- S-2 In vertical center of tube sheet near outside edge.
- S-3 In structural ring at vertical core center line close to downcomer.
- S-4 In structural ring at vertical core center line away from downcomer.
- S-5 In two-inch thick seal plate, 1/4 inch below top, near inner edge.
- S-6 In seal plate, 1/4 inch above bottom, near inner edge.
- S-7 In seal plate, 1 inch below top, near outer edge.
- S-8 In 3-inch thick mounting plate, 1-1/2 inches from top, 1-3/4 inches from outer edge.

There are four locations in plenum chambers, as follows:

- P-9 In outer inlet plenum, at entrance to inlet valve nearest inlet line.
- P-10 In upper inlet plenum near north throttle valve.
- P-11 In upper inlet plenum near south throttle valve.
- P-12 In outlet plenum near outlet line.

In addition there are thermocouples in the inlet and outlet NaK lines outside of the reactor tank.

Figure 4 shows four of the structural thermocouple holes. This drawing also shows the possible locations for thermocouples in fuel and blanket rods and in coolant channels between rods. Three fuel assemblies and one blanket assembly have been modified to receive thermocouples in the positions shown, thus allowing a full radial temperature traverse. Thermocouples of various lengths allow axial traverses in the coolant passages between rods.

The thermocouples are of the stainless steel sheathed type with MgO insulation and duplex iron-constantan wire. The hot junction is welded to the inside of the sheath at the tip. All metal temperatures are obtained by sheath tip contact with a NaK heat transfer bond. The rod and core coolant couples each have a sheath diameter of 0.040 inch. The structural couples are each 1/16 in. in diameter and the plenum chamber couples are 1/8 in. in diameter.

There are two types of fuel and blanket thermocouple rods. As shown in Figure 11, one for center rod temperatures, has a central axial hole terminating at the core centerline. The other has an inverted "Y" shaped hole, the two prongs of which terminate at the uranium clad interface, thus giving a temperature gradient across a rod. Vent holes are provided in the thermocouple rods to assure the presence of a NaK heat transfer bond at the sheath tip. Each sheathed thermocouple has a shielded flexible lead wire with a push on type connector.

On the reactor tank inside wall above the top plate of the inner tank assembly is located a gas-tight thermocouple connector box containing 48 hermetically sealed glass-Kovar-type connectors with iron and constantan contacts. Duplex iron-constantan wire leads from the connectors through a nozzle in the tank shell via a special conduit to a patch board in the control room. The patch board also has iron and constantan contacts with 48 input thermocouple connections and 6 output potentiometer connections.

Six thermocouple amplifiers are provided, as well as a six-channel, heated stylus, fast recorder for both experimental temperature and neutron flux measurements.

B. REACTOR EXTERNALS

1. Reactor Vessel

The reactor vessel, or tank, is shown in both Figures 5 and 13. (The core and inner blanket of Mark I and II are shown in Figure 13 for reference.) It is double walled, as is the piping which leads from the reactor vessel through the reactor shield. The part of the reactor vessel surrounding the reactor core has an inside diameter of 15.87 inches and a length of 28 inches. Above this small diameter part the reactor vessel increases in diameter and is filled with shielding material, mostly steel. The whole reactor vessel rests on the shoulder formed by the change in diameter; thus the reactor core itself projects below the point of support as a smooth cylinder. The stainless steel plate on which the reactor vessel stands is slotted to provide cooling air, which first flows over the outside cover of the reactor tank and then through the slots, thus cooling the reactor support. It then is drawn through the outer blanket. The small diameter part of the reactor tank consists of a stainless steel vessel of 5/16-inch wall thickness, made by deep drawing. It is surrounded by a second tank made of Inconel, 1/16-inch thick. The second tank fits snugly on ribs which have been formed in the Inconel. The upper portion of the reactor vessel also is double walled. The double-walled construction serves a number of purposes: the gas space between the two walls provides some thermal insulation, it gives a method for testing for integrity of the inner vessel at any time, and, finally, in the event that the inner vessel should develop a leak, the outer vessel would prevent serious consequences.

2. Outer Blanket

The outer blanket consists of 84 keystone-shaped bricks of natural uranium, each weighing about 100 pounds (see Figure 13). These bricks are jacketed in stainless steel of 0.020-inch thickness. The bricks fit together in a stable array; each is provided with a recess on top into which fit projections on the brick above it. Air-cooling of the bricks is obtained by drawing air through five holes which are fitted with sleeves in close contact with the uranium. In order to increase the area for heat transfer, each sleeve carries fins on its inside. Each brick also provides a passage for a control rod and air is similarly drawn through this passage. Figure 14 is a photograph of the outer blanket in which an inner aluminum cylinder is present but in which an outer aluminum cylinder has been removed. The purpose of the aluminum cylinders is to provide additional stability.

3. Controls

There are twelve control rods, each 2 inches in diameter, made of natural uranium jacketed in stainless steel. These move vertically in the outer blanket bricks. Eight of these control rods normally are used as safety rods. Their time of travel out of the blanket is short: 0.85 second to initiate motion, 0.29 second to reach 16 inches. The remaining four normally are used for the running controls and can be positioned with considerable accuracy. Their maximum speed is 0.64 inch per second. The whole outer blanket is mounted on an elevator which is hydraulically driven. Figure 15 shown the main platform of this elevator, which carries a shield section on which the outer blanket rests. Below is shown the drive motors which activate the twelve safety and control rods. The elevator can be driven upward at a maximum speed of 0.32 inch per second from 80 inches, its fully down position, to 30 inches, then at a maximum speed of 0.095 inch per second to 4-1/4 inches. These speeds may be reduced by adjustment of the mechanism but may not be changed from the control room. The last 4-1/4 inches of travel is mechanically controlled, permitting location of the outer blanket around the reactor core with a precision of 0.001 inch. The maximum speed is 0.005 inch per second. For shutdown, the outer blanket, the shield plug on which it rests, and the elevator can be dropped quickly: 0.1 second to initiate motion, 0.56 second for 12 inches of travel. The handling of the outer blanket for replacement of bricks is done by lowering the elevator and lifting the outer blanket from the shield plug with a handling dolly and transporting it into a neighboring shielded room. There it is disassembled with a remotely operated manipulator. In addition to the rods, a part of the outer blanket in the form of a cylindrical block is arranged so that it can be driven out of the bottom of the outer blanket with pneumatic force.

A full list of interlocks on the reactor follows:

Immediate Scram Interlocks	Cause of Trip
1. Period A Negative	Excessive Negative Period
2. Period A Positive	Excessive Positive Period
3. Period B Negative	Excessive Negative Period
4. Period B Positive	Excessive Positive Period
5. Reactor Coolant Flow	Low Flow for Power Operation (Interlock may be removed for low power operation)
6. Reactor Inlet Temperature	High Temperature
7. Reactor Outlet Temperature	High Temperature
8. Exchanger Outlet Temperature	High Temperature

Immediate Scram Interlocks (Cont'd.)	Cause of Trip
9. Fuel Rod Temperature	High Temperature
10. Elevator Jacks	Leaves Jacks
11. Gravity Tank Outlet Shut Off Valve	Valve Closed
12. Gravity Tank Drain Valve	Valve Open
13. Reactor Overflow Valve	Valve Closed
14. Receiving Tank Overflow Valve	Valve Closed
15. Flux Level #1	High Power
16. Flux Level #2	High Power
17. Blanket Rod Temperature	High Temperature
18. Reactor Assembly Inlet Valves	Valves not completely up or down
19. Reactor Assembly Throttle Valves	Valves not in Correct Position for Series or Parallel Flow

Two-Minute Delayed Scram Interlocks	Cause of Trip
Pumped Coolant Flow	Low Flow
Circulating Gas Flow	Low Flow
Receiving Tank Level	High Level
Compressed Air Supply	Low Pressure
Elevator Hydraulic Pressure	Low Pressure
Auto Control	Malfunction
Outer Blanket Air	Low Flow

Alarm Only	Cause of Alarm
1. Secondary Coolant Flow	Low Flow
2. Gas Pressure Pump No. 1	Low Pressure
3. Primary Blanket Gas Pressure	Low Pressure
4. Secondary Blanket Gas Pressure	Low Pressure
5. Gas Pressure Secondary Pump	Low Pressure
6. Pump No. 1 Under Speed	Low Speed
7. Expansion Tank Level	Low Level

Alarm Only (Cont'd.)	Cause of Alarm
8. Secondary Surge Tank Level	Low Level
9. Reactor Overflow	Overflow
10. Pump No. 1 High Level	High Level
11. Secondary Pump High Level	High Level
12. Gas Trap High Level	High Level
13. Liquid Heater Overtemperature	High Temperature
14. Boiler Pump Cooling Water	Low Pressure
15. Gas Supply	Low Pressure
16. Exhauster Water	Low Pressure
17. Gravity Tank Level	Low Level
18. Pump Temperature	High Temperature
19. Secondary Pump Under Speed	Low Speed
20. Smoke Detector	Smoke
21. Smoke Detector Trouble	Detector Not Operating
22. Transfer Pump Probe	Leak in Pump Walls
23. Electromagnetic Pump Probe	Leak in Pump Walls

4. Nuclear Instrumentation

The only change in nuclear instrumentation from the previous design has been the provision of new safety circuits, period circuits and a NaK-activity monitor circuit. A block diagram of the nuclear instrumentation is shown in Figure 12. A brief description of the circuits follows.

Flux Level Trip Circuits

The safety circuits are ANL Model CD-118 Safety Trip Circuits. They have a range of 10^{-12} to 10^{-4} ampere in decade steps and a trip circuit which is adjustable over one decade from the front panel. In normal operation there are three units with their recorders installed, two of which are in operation and one unit serves as a stand-by.

Period Circuits

The period meters are ANL Model CD-125A and Model CD-125B. The Model CD-125A has a log current range of 10^{-11} to 10^{-3} ampere, a period range of ∞ to $\pm$ 15 seconds, and a variable positive period trip of 75 seconds to 15 seconds, adjustable from the front panel. The Model CD-125B has a log current range of 10^{-11} to 10^{-3} ampere, a period range of 00 to $\pm$ 5 seconds, and a fixed 5-second positive period trip. Both units have an internally adjustable negative period trip. One Brown strip chart recorder is provided for recording periods from either Period A or Period B circuits.

Startup Circuits

Vibrating Reed

The vibrating reed is an Applied Physics Laboratory Model 30. Two input resistors of 10^{10} ohms and 10^{7} ohms are provided in the electrometer head, giving the instrument a range of 10^{-13} to 10^{-7} ampere. The output of the vibrating reed amplifier is recorded on a Brown strip chart recorder.

Fission Circuit

Another startup circuit consists of a fission counter, preamplifier, pulse amplifier and Audio Pulse Monitor. The radial position is on the inside edge of the thermal column.

Galvanometers and Automatic Control

The power and differential galvanometers are conventional Leeds and Northrup light beam galvanometers. The automatic control is a relay-type control system actuated by photo-cells mounted on the differential galvanometer scale.

Exhaust Air Activity Monitor

The exhaust air activity monitor consists of an ionization chamber and a linear amplifier having a range of 10^{-12} to 10^{-10} ampere. The output of the amplifier is recorded on a Brown strip chart recorder.

NaK-Activity Monitor

The NaK-activity monitor measures activity in the gravity feed tank by means of a photomultiplier and plastic crystal and a log amplifier having a range of 10^{-9} to 10^{-3} ampere. The output of the amplifier is recorded on a Brown strip chart recorder.

Ion Chambers

All of the ion chambers are filled with natural BF_3. The galvanometer and vibrating reed circuit chambers are seven inches in diameter and the safety and period circuit chambers are four inches in diameter. All chambers are located in radial holes in the reactor shield.

Chamber Power Supplies

All of the ionization chambers, with the exception of the two galvanometers, are supplied from ANL Model V-93 Chamber Voltage Supplies. The galvanometer chambers and lights are supplied with batteries so that they may continue to function in case of a power failure. A voltmeter and selector switch are provided on the front panel to monitor the chamber supply voltages.

C. POWER PLANT DESIGN

The reactor operates at a maximum power of 1400 kw. An air flow of 5,800 cubic feet per minute maintains the blanket bricks below a temperature of 200 °C. The flow of NaK through the reactor is normally 290 gallons per minute.

1. NaK Systems

The external systems outside of the reactor tank remain unchanged. The primary and secondary NaK systems are shown in Figure 16.

The sodium-potassium coolant is supplied to the reactor from a high, constant level tank, flows through the reactor where it is heated, through a heat exchanger where it is cooled by the secondary NaK system, and thence to a lower tank. Either a mechanical or an electromagnetic pump returns it to the high tank. In the secondary system the pump draws the NaK from a reservoir, circulates it through the primary heat exchanger and the steam generator, and returns it to the reservoir.

The NaK-to-NaK heat exchanger is of a conventional shell and tube design with all joints welded. Primary flow is through the tubes with secondary flow counter-current through the shell.

2. Steam System

A diagram of the steam system is shown in Figure 17. The steam generator is divided into three components: the economizer, the boiler, and the superheater. Flow of NaK through these units is counter-current to the flow of water and steam. Heat transfer tubes in each component are similar and consist of a composite assembly of inner nickel, intermediate copper, and outer nickel tubes. These tubes were assembled by mechanically drawing together and thermally diffusion bonding them for good heat transfer. Total wall thickness of the tube is 5/16 inch, of which 3/16 inch is nickel. An outer stainless steel tube makes up the shell of the heat exchanger, and a bellows is used to allow for differential thermal expansion. Thus each heat exchanger is of a single tube in a shell type, where NaK flow is in the shell side and water or steam is in the tube.

In order to limit the quantity of water in the system and to increase the heat transfer rate, a forced circulation, falling film-type boiler is used. This utilizes the above described heat exchangers in a vertical position. A water film is established at the upper end of the internal tube on its inner surface by means of a baffle. It then runs to the bottom where excess water and generated steam are piped into a drum. Steam is led through a separator in the drum out to the superheater which consists of horizontal heat exchangers with the NaK in the shell side.

The economizer is also a horizontal unit and serves to heat the feed water from the deaerating tank to steaming temperature before injection into the boiler drum.

Table III gives pertinent information concerning the heat exchanger and steam generator.

Steam which is produced in the steam generator is used to drive a turbogenerator set which may provide power for the building and area. Rather than try to control reactor power by load requirements, it is held at constant power and excess steam is generated. A back pressure-regulating valve maintains constant steam main pressure by unloading the excess steam directly to the condenser. The efficiency of this plant is found to be about 17%. This low value may be attributed to the small size of the plant and to the fact that it is operated at only part load.

Table III

Data on the EBR Heat Removal System

1. Intermediate Heat Exchanger

Over-all length	14 ft 8 in
Shell outside diameter	17 in
Flow type	Double pass shell and tube
Number of tubes	102
Type of tubes	3/4 in. OD, 16 gage, hairpin
Tube material	"A" nickel
Outside area of tubes	495 ft^2

2. Heat Exchanger Tube Used in Superheater, Boiler, and Economizer

Effective length	9 ft 6-3/4 in
Outside diameter	2-5/8 in
Inside diameter	2 in

3. Superheater

Number of heat exchangers	4
Arrangement	Series with countercurrent flow
Total inside area of tube	20 ft^2
Shell size	5 in. IPS
Shell side cross-sectional flow area	14.6 in^2
NaK velocity	6.15 ft/sec
Steam velocity, average	58.8 ft/sec

4. Economizer

Number of heat exchangers	9
Arrangement	Series with countercurrent flow
Total inside tube area	45 ft^2
Shell size	5 in. IPS
Shell side cross-sectional flow area	14.6 in^2
NaK velocity	6.15 ft/sec
Water velocity in annulus around 1-3/4-in. OD baffle	3.43 ft/sec

5. Boiler

Number of heat exchangers	18
Arrangement	Parallel with countercurrent flow
Total inside tube area	90 ft^2
Shell side cross-sectional flow area per tube	1.98 in^2
Shell size	3 in IPS
NaK velocity	2.51 ft/sec
Water flow rate	13 times steam rate

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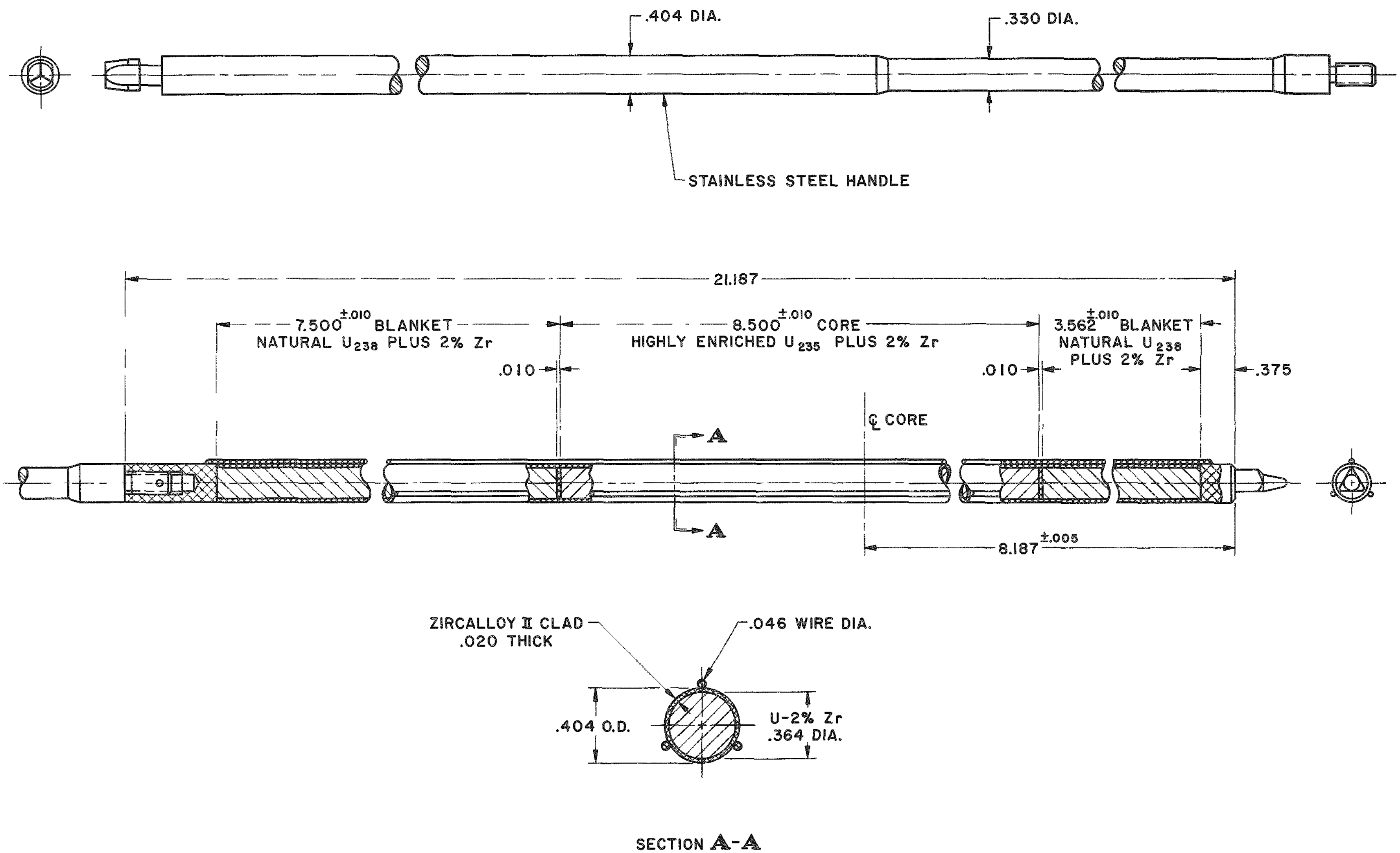

FIG. 1. FUEL ROD EBR I, MARK III

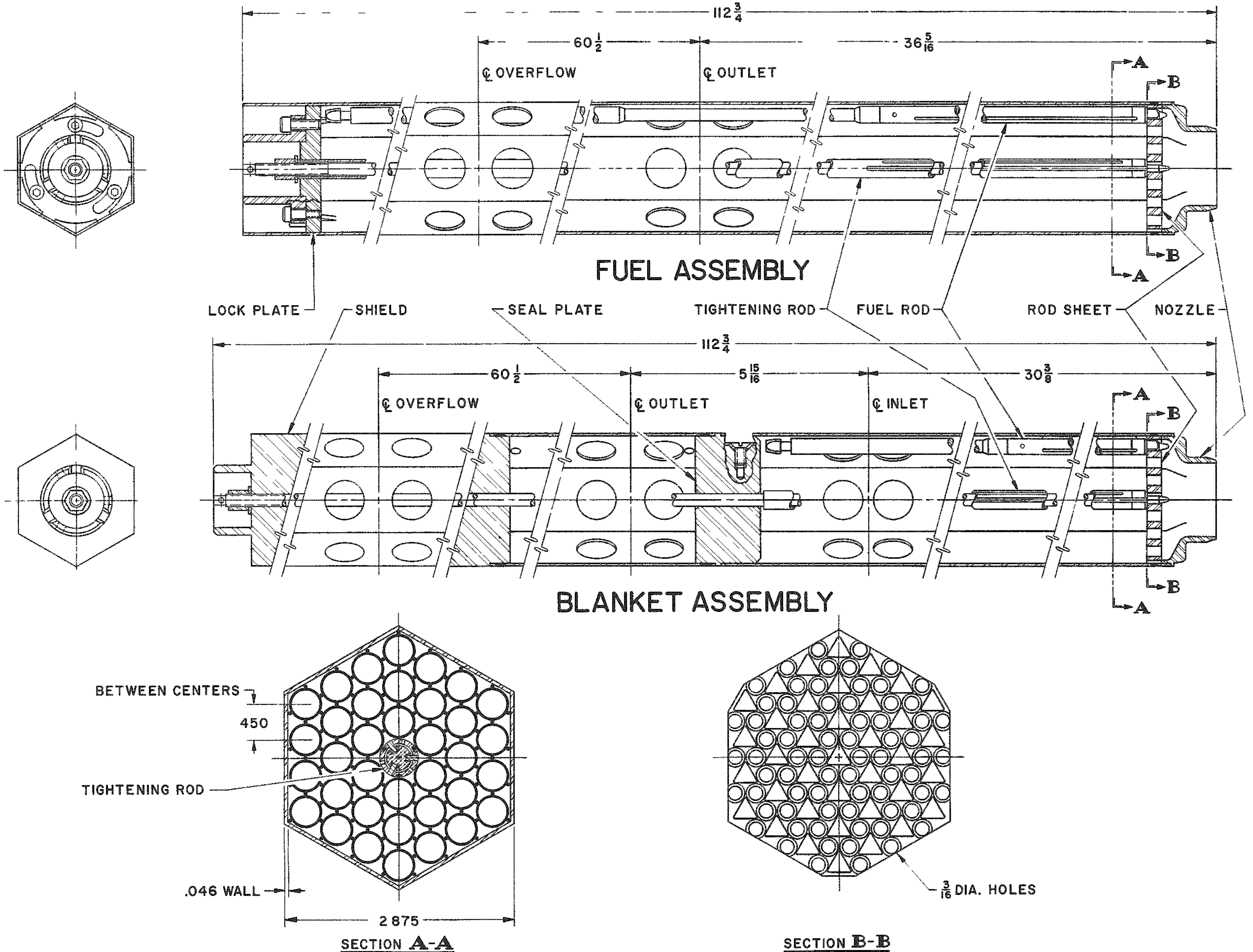
112 3/4
60 1/2
36 5/16
℄ OVERFLOW
℄ OUTLET
A
B
FUEL ASSEMBLY
LOCK PLATE
SHIELD
SEAL PLATE
TIGHTENING ROD
FUEL ROD
ROD SHEET
NOZZLE
112 3/4
60 1/2
5 15/16
30 3/8
℄ OVERFLOW
℄ OUTLET
℄ INLET
BLANKET ASSEMBLY
BETWEEN CENTERS
450
TIGHTENING ROD
.046 WALL
2 875
SECTION A-A
3/16 DIA. HOLES
SECTION B-B

FIG. 2

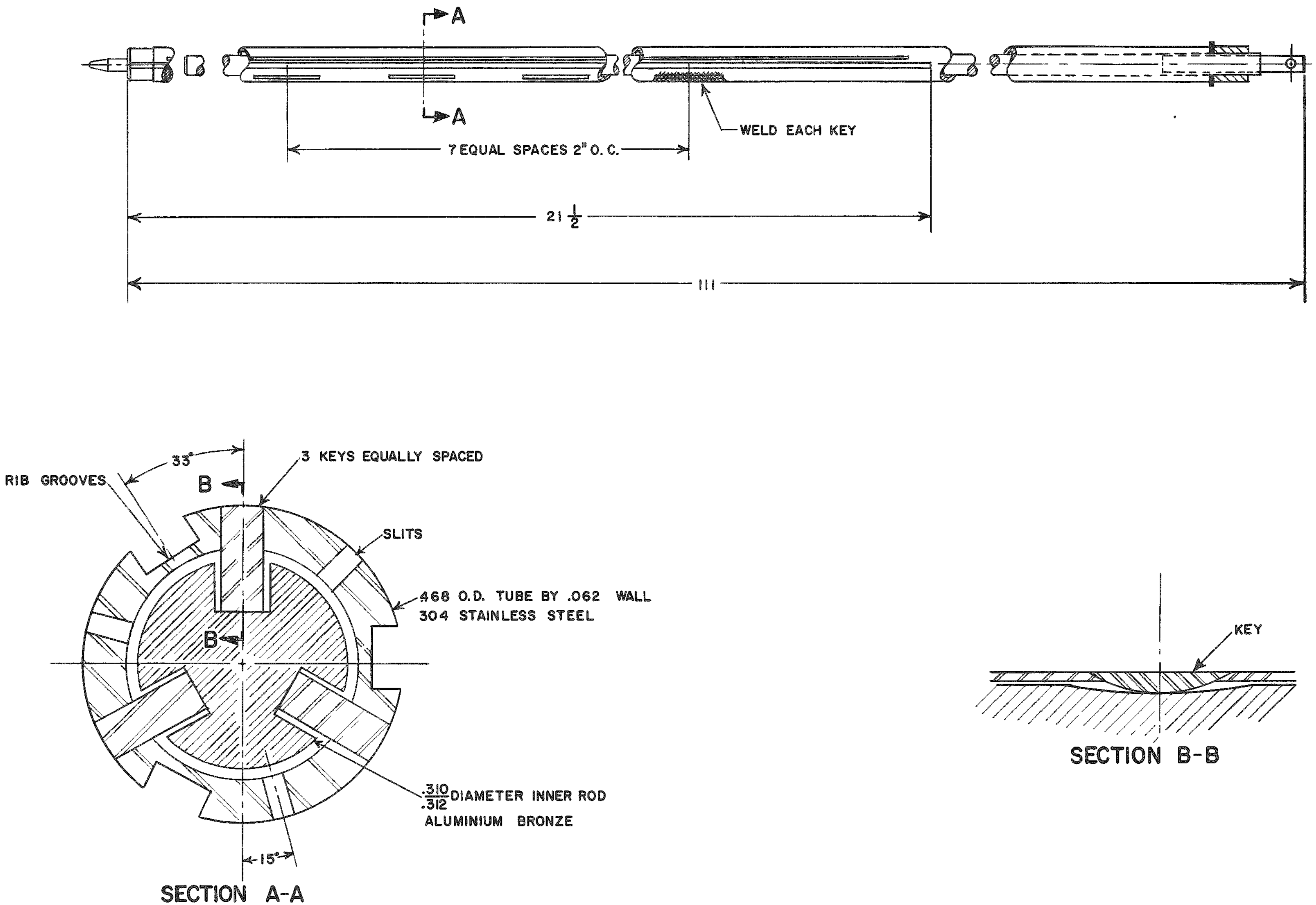

FIG. 3. FUEL TIGHTENING ROD, EBR I, MK III

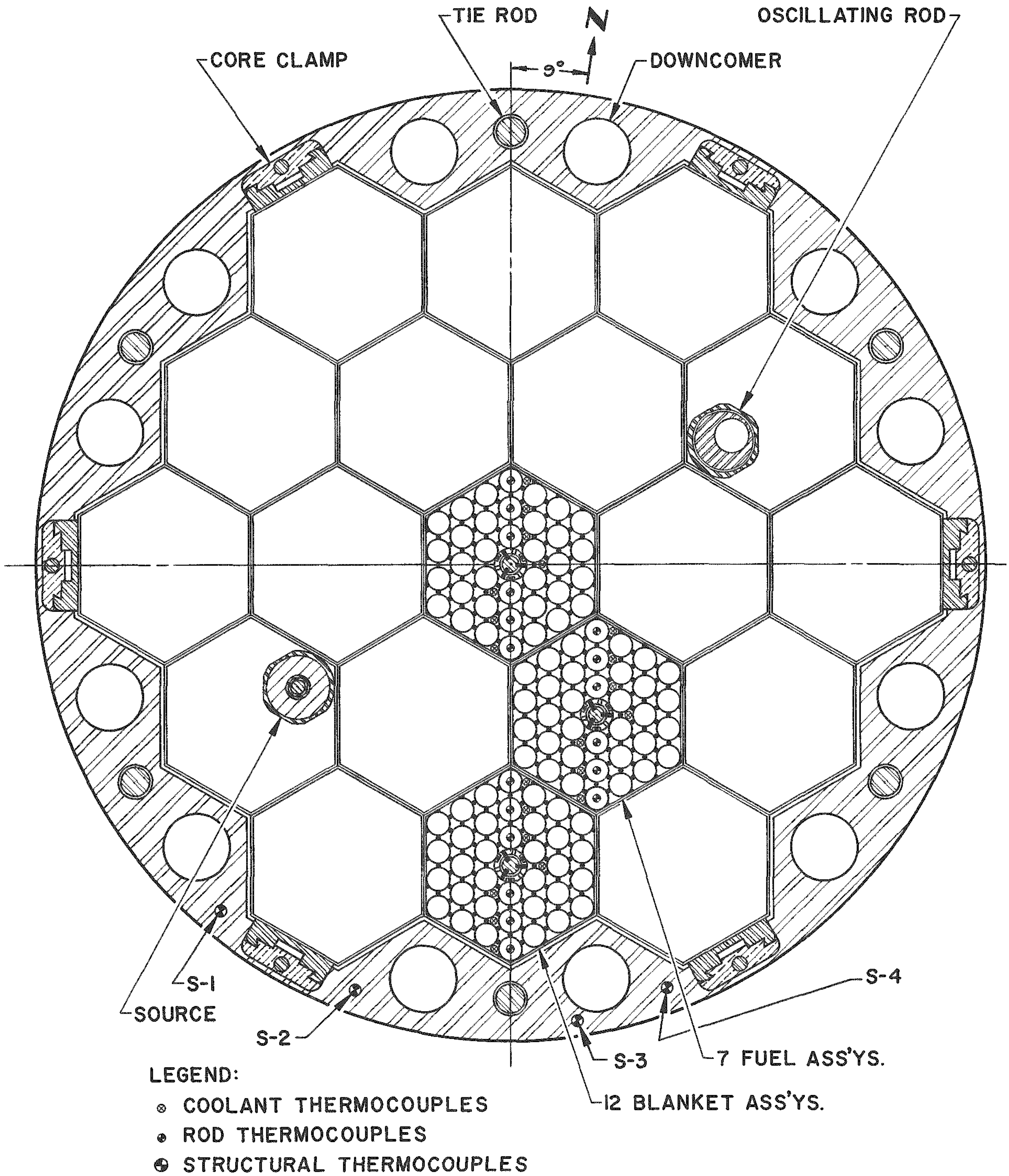

FIG. 4. CROSS SECTION THRU CENTERLINE OF CORE EBR I, MARK III

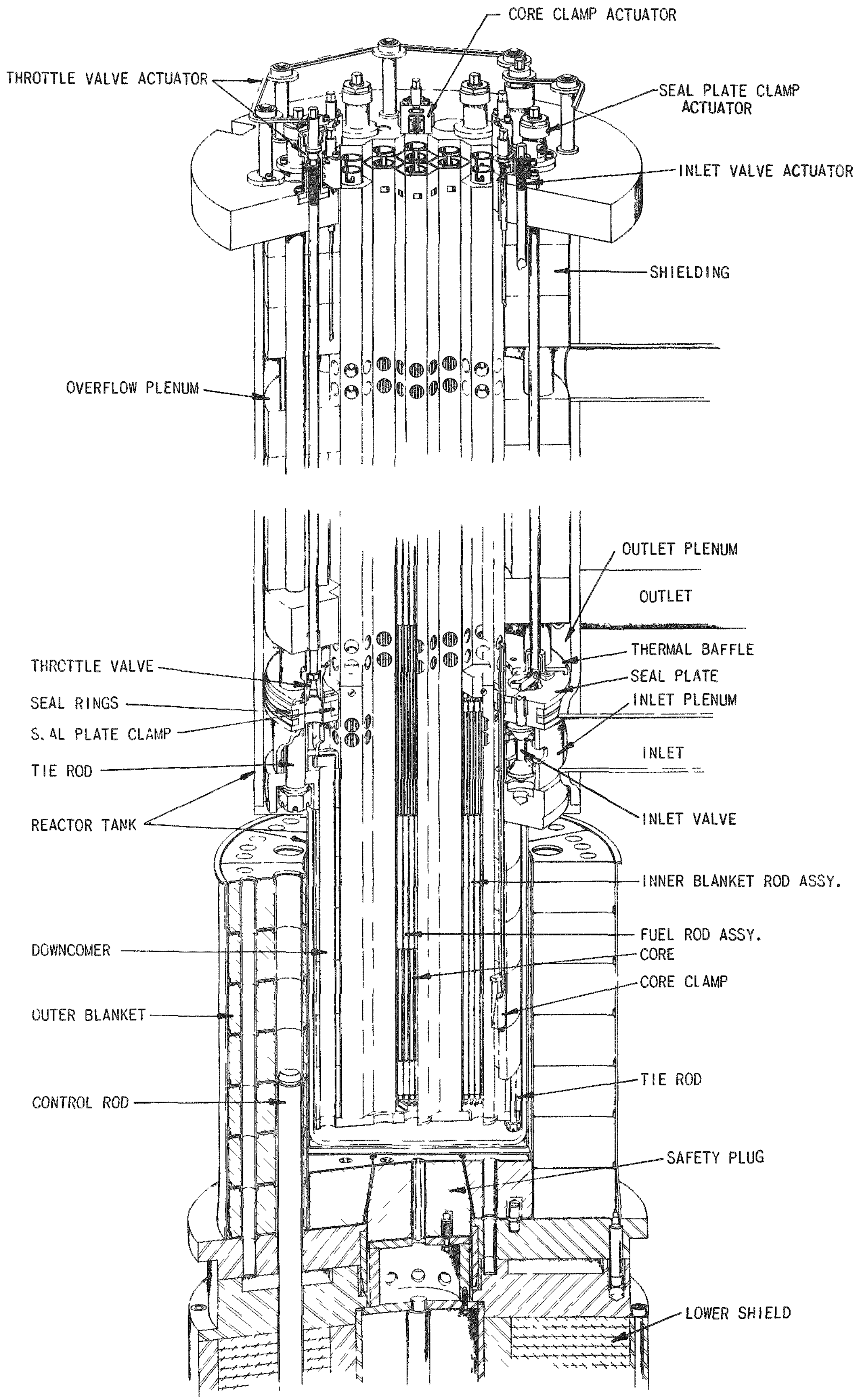

FIG. 5. INNER TANK ASSEMBLY EBR I, MARK III

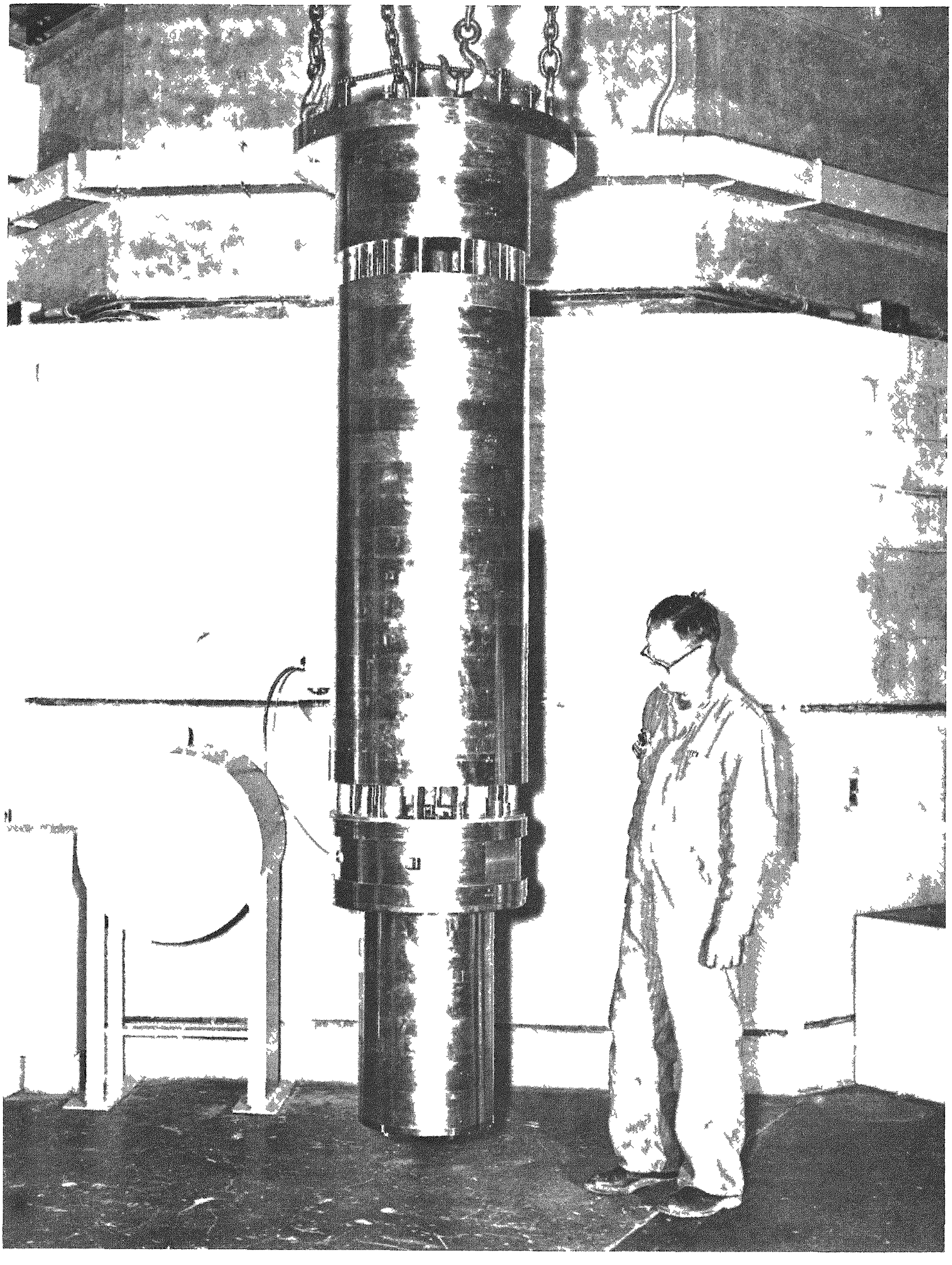

FIG. 6. COMPLETED INNER TANK ASSEMBLY, REACTOR SHIELD BEHIND

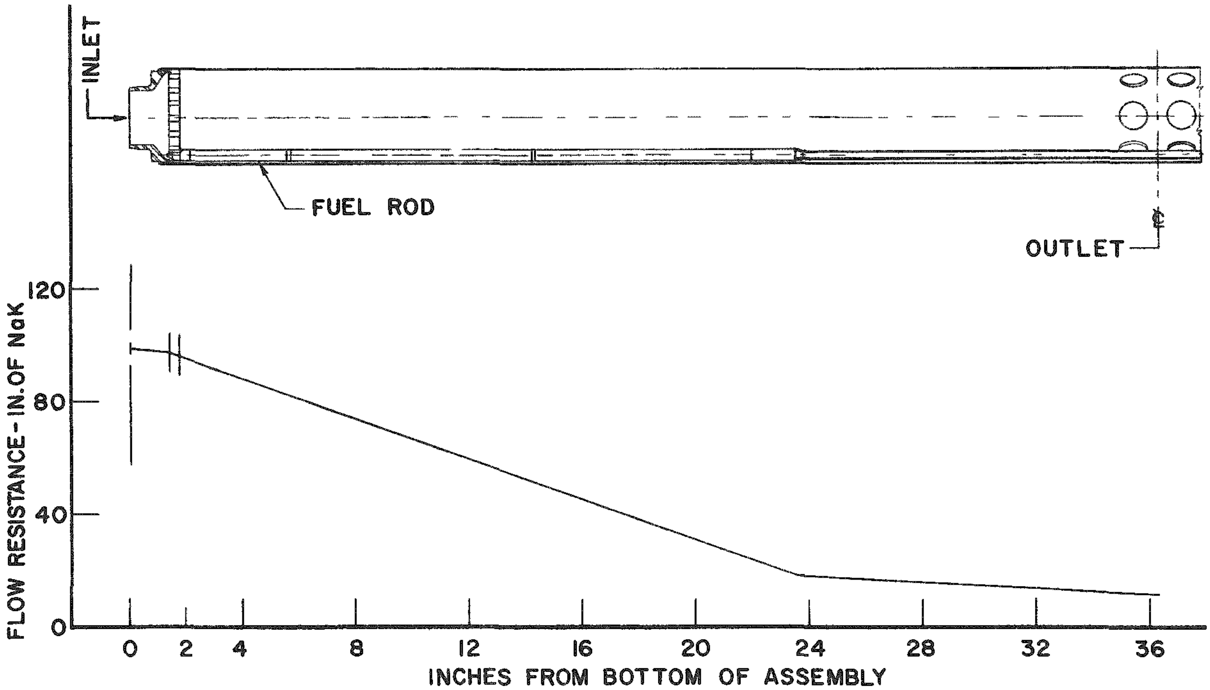

FIGURE 7
FLOW RESISTANCE VS. DISTANCE
EBR I, MARK III. FUEL ASSEMBLY

300 G.P.M. THROUGH REACTOR-NaK AT 300° C.

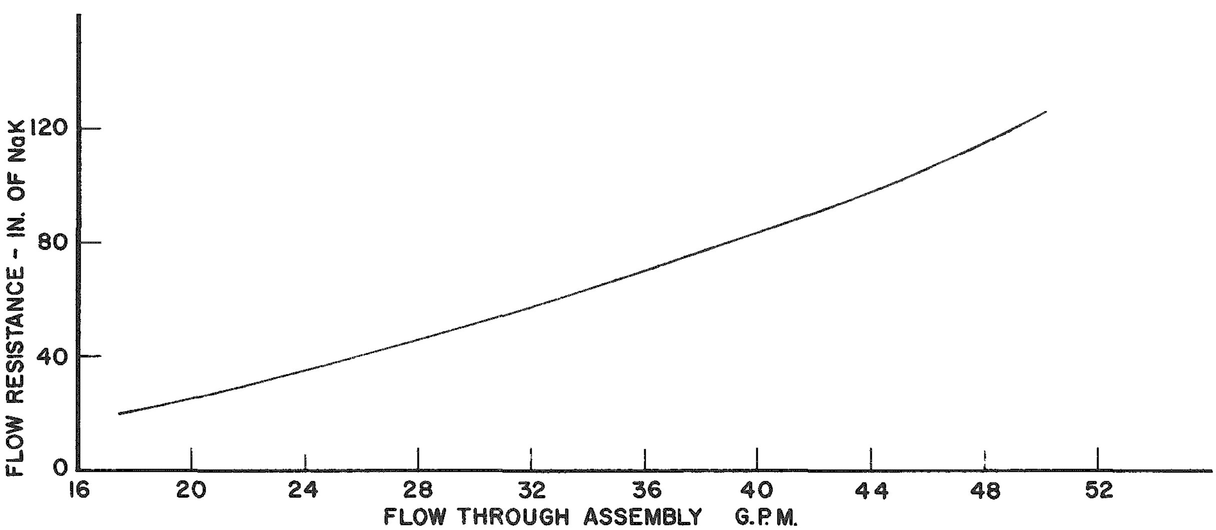

FIGURE 8
FLOW RESISTANCE VS. FLOW
EBR I - MARK III - FUEL ASSEMBLY

NaK 300° C
300 G.P.M. REACTOR FLOW = 42.8 GPM/CORE ASSEMBLY
= 25 GPM./BLANKET ASSEMBLY

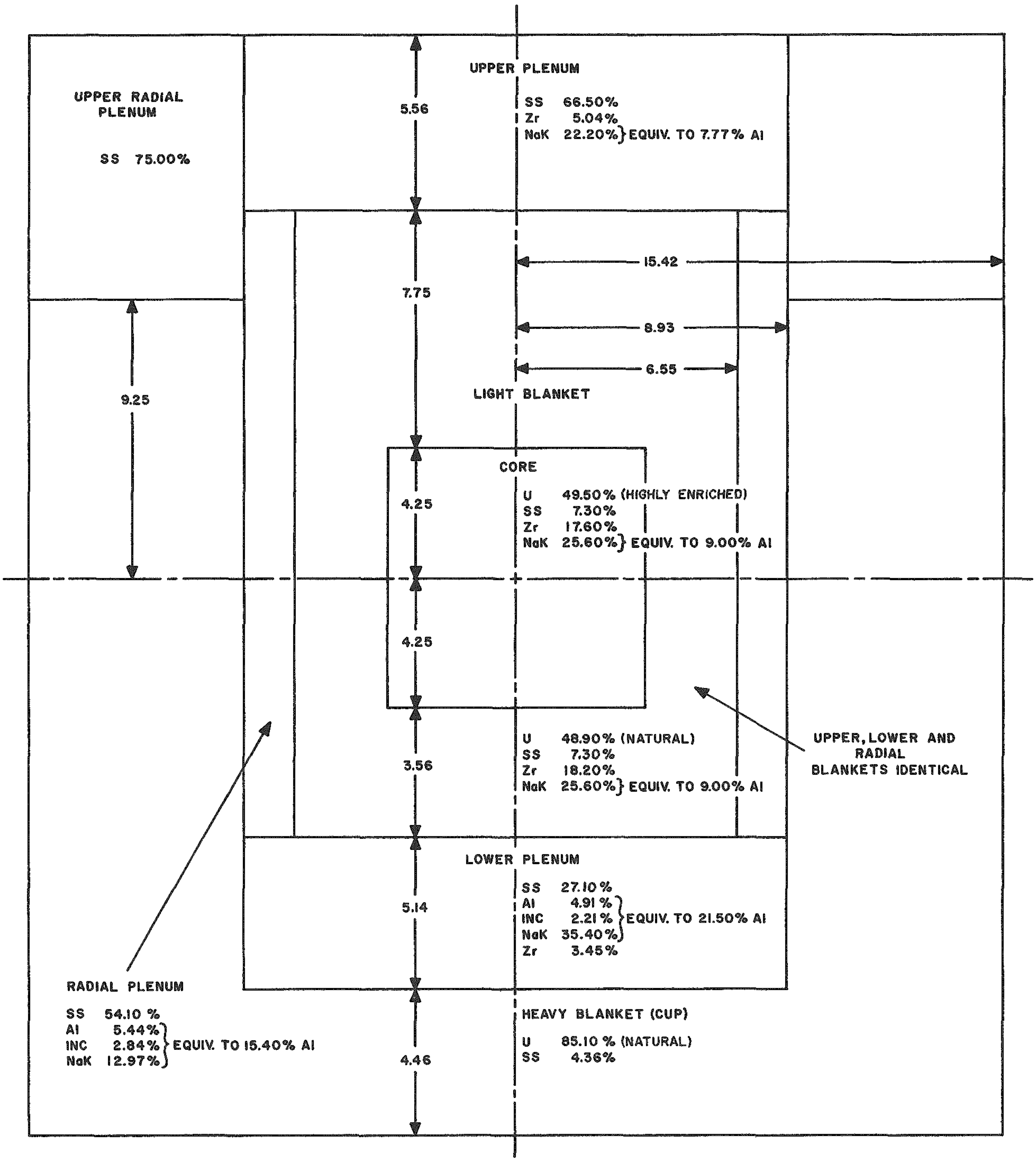

FIG. 9. COMPOSITION AND DIMENSIONS, EBR-I, MARK III REACTOR

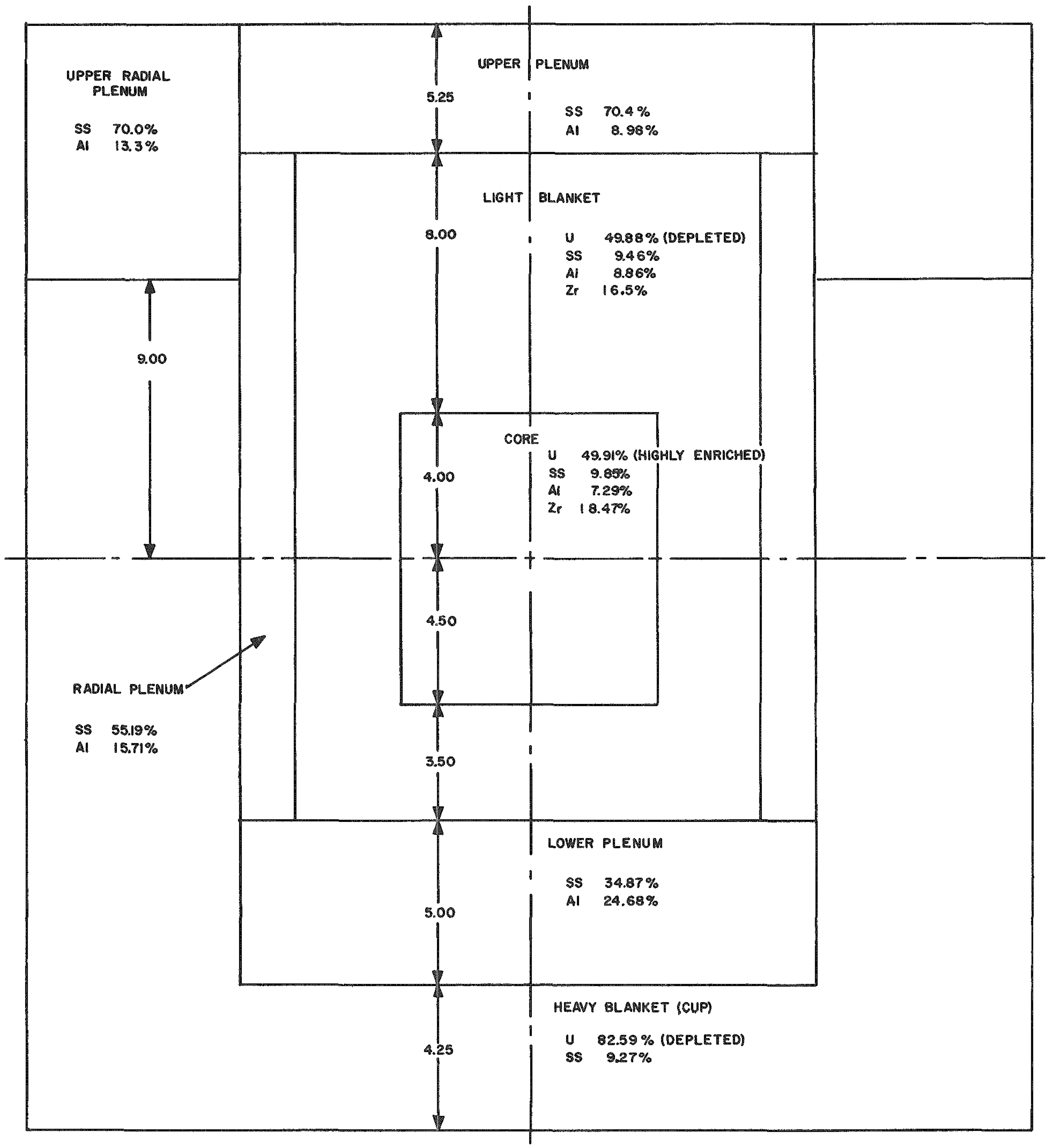

FIG. 10. COMPOSITION AND DIMENSIONS, EBR-I, MARK III MOCKUP

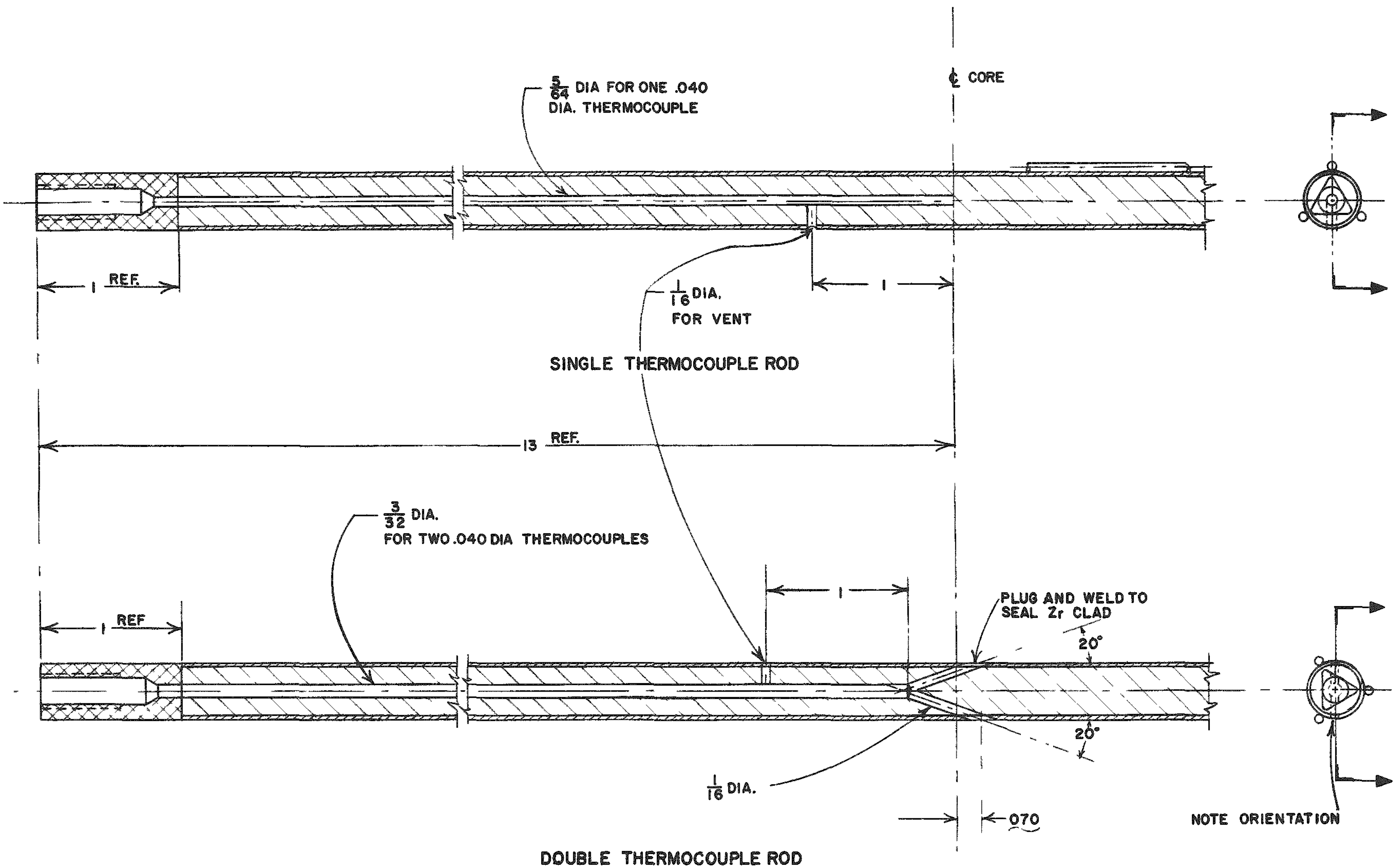

FIG. 11. THERMOCOUPLE RODS, EBR I MARK III

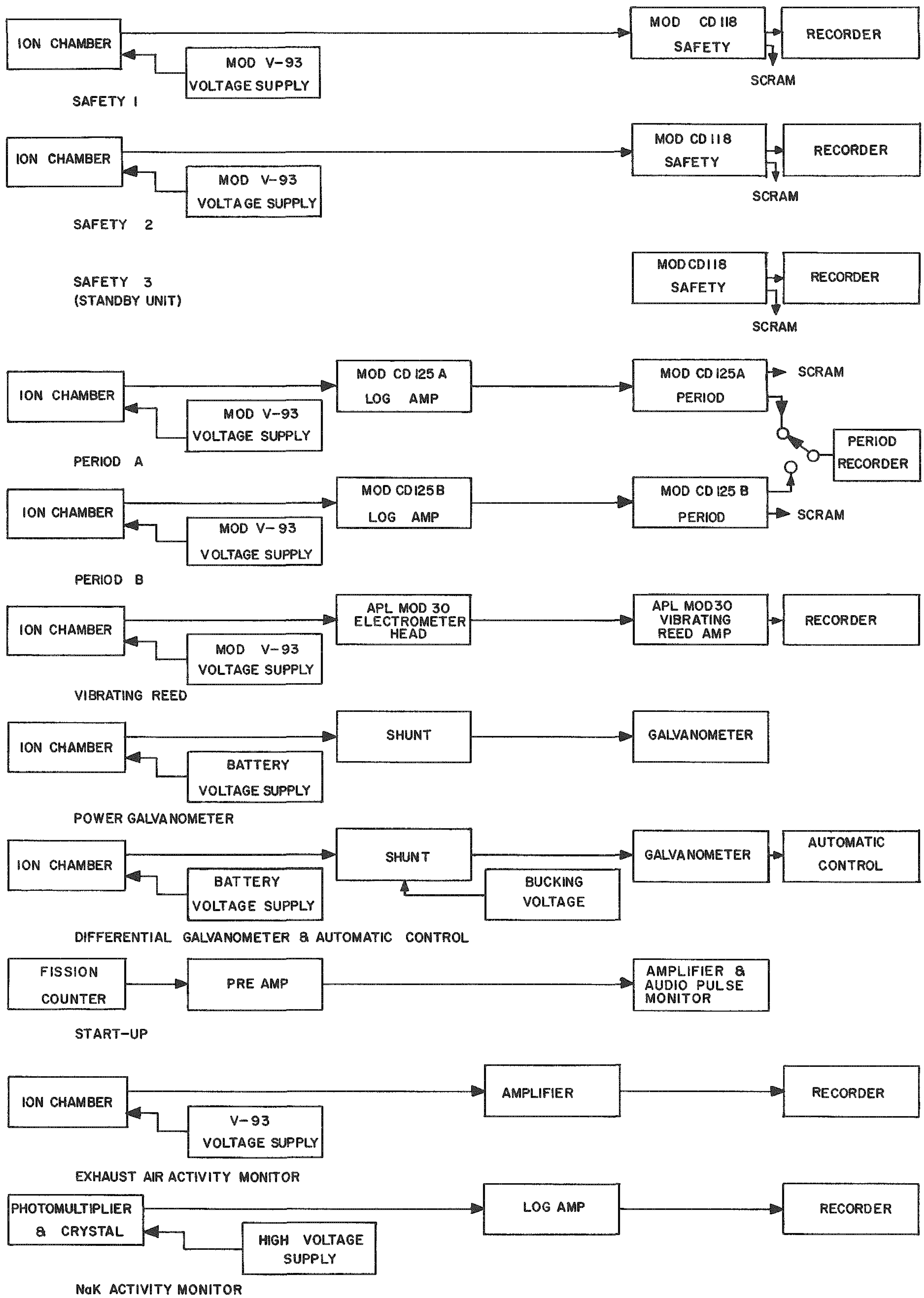

FIG. 12. EBR I MARK III, NUCLEAR INSTRUMENTATION

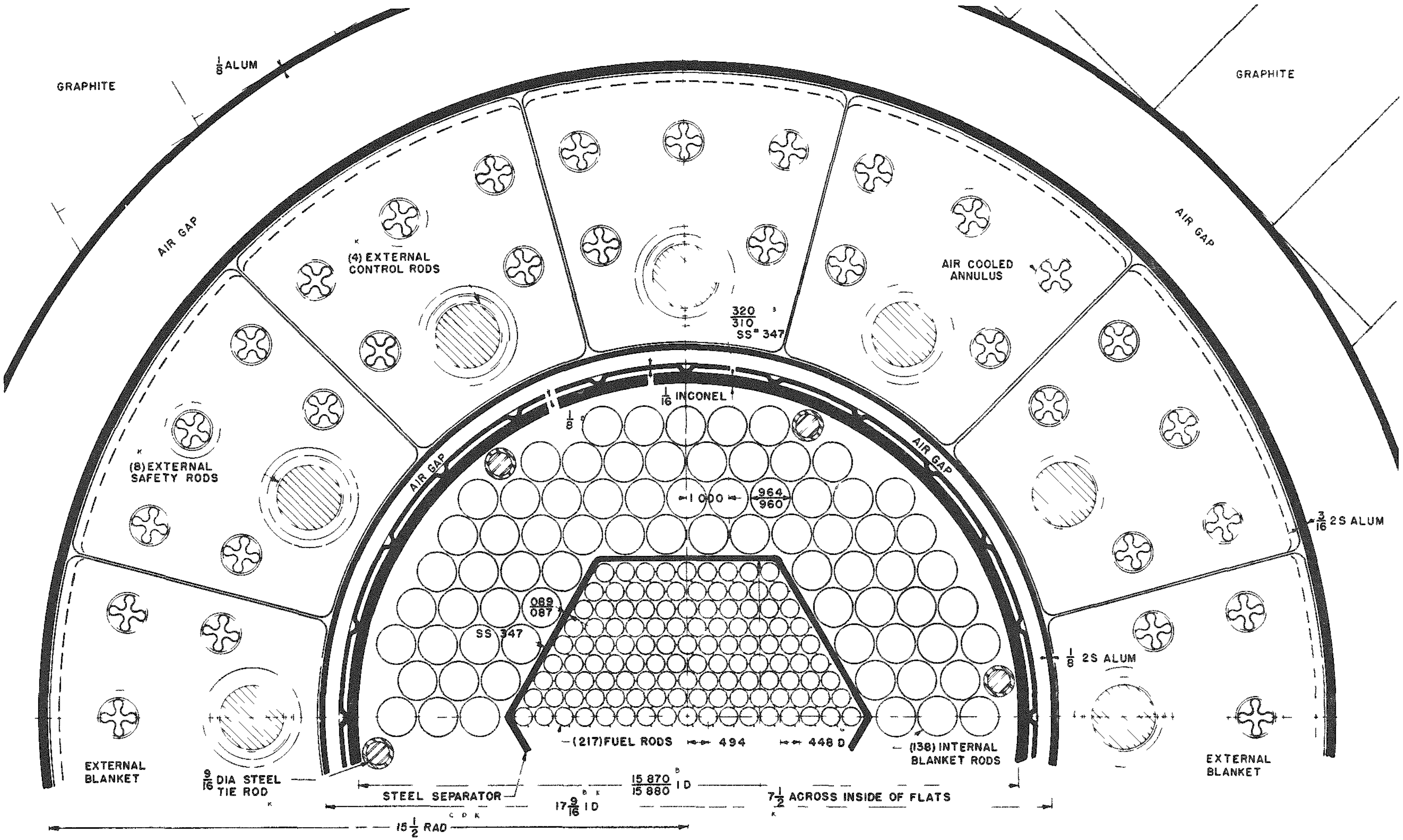

FIG. 13. HORIZONTAL CROSS SECTION AT MIDPLANE, MARK I AND MARK II

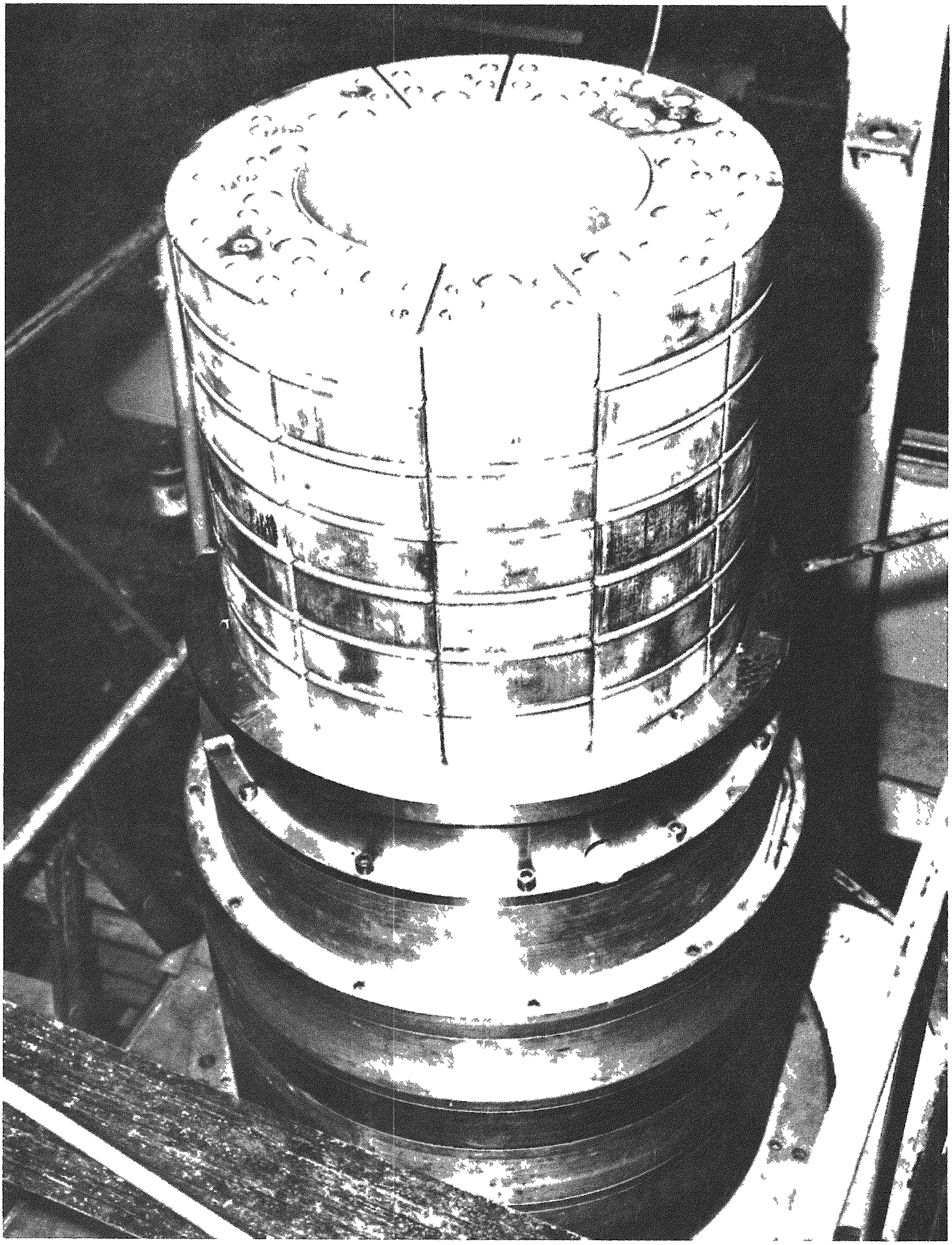

FIG. 14. OUTER BLANKET OF REACTOR

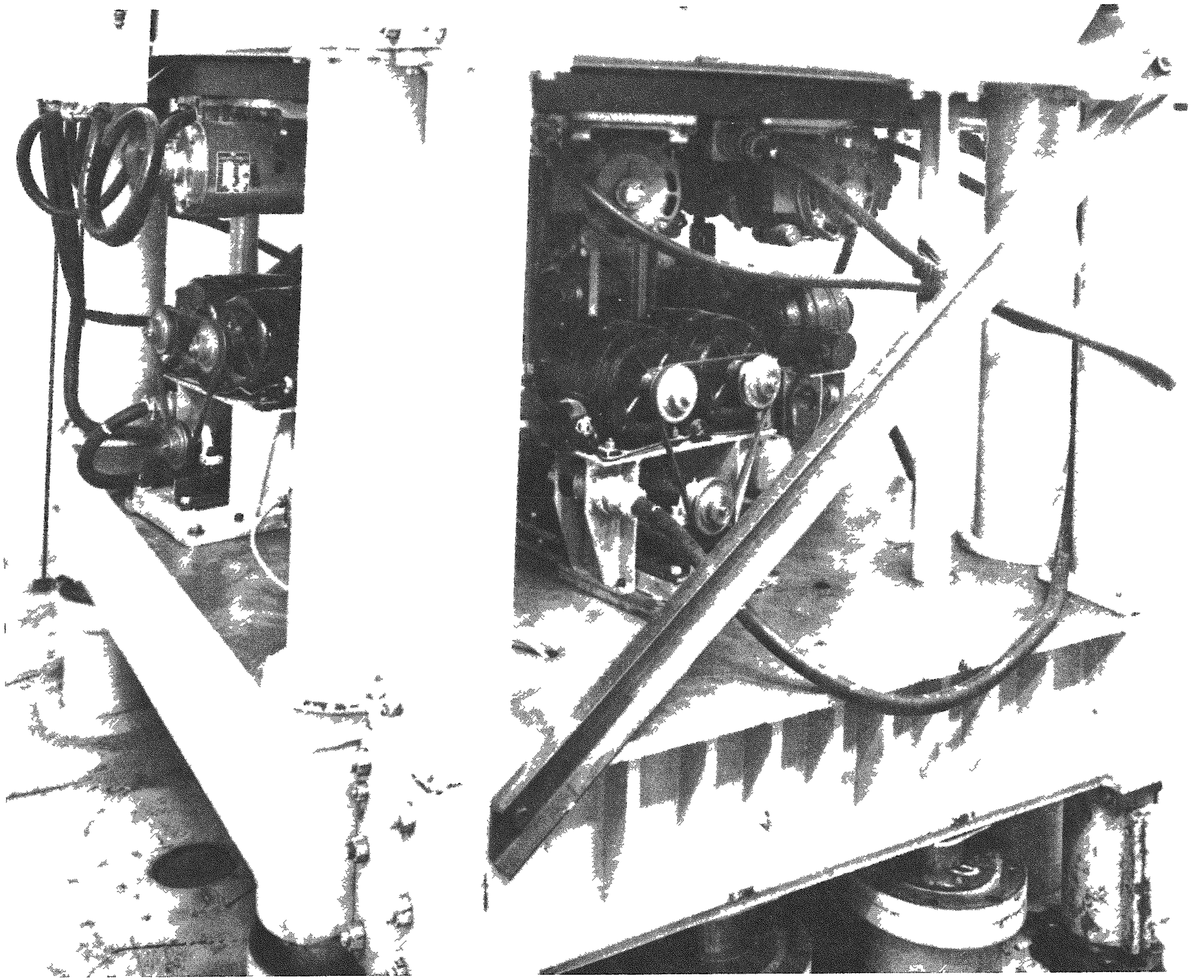

FIG. 15. OUTER BLANKET ELEVATOR AND CONTROL DRIVE MOTORS

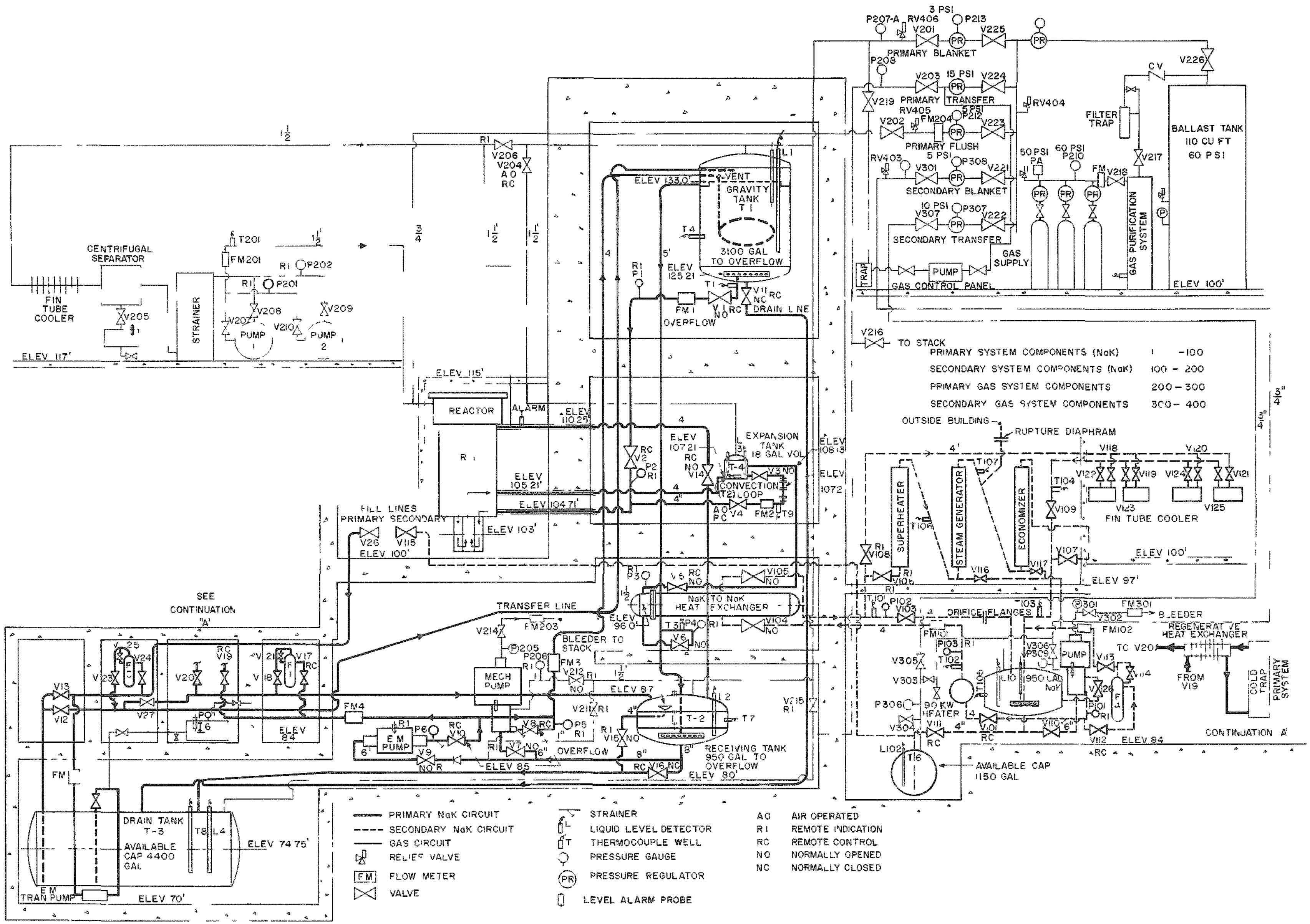

FIG. 16. NaK SYSTEMS, EBR I

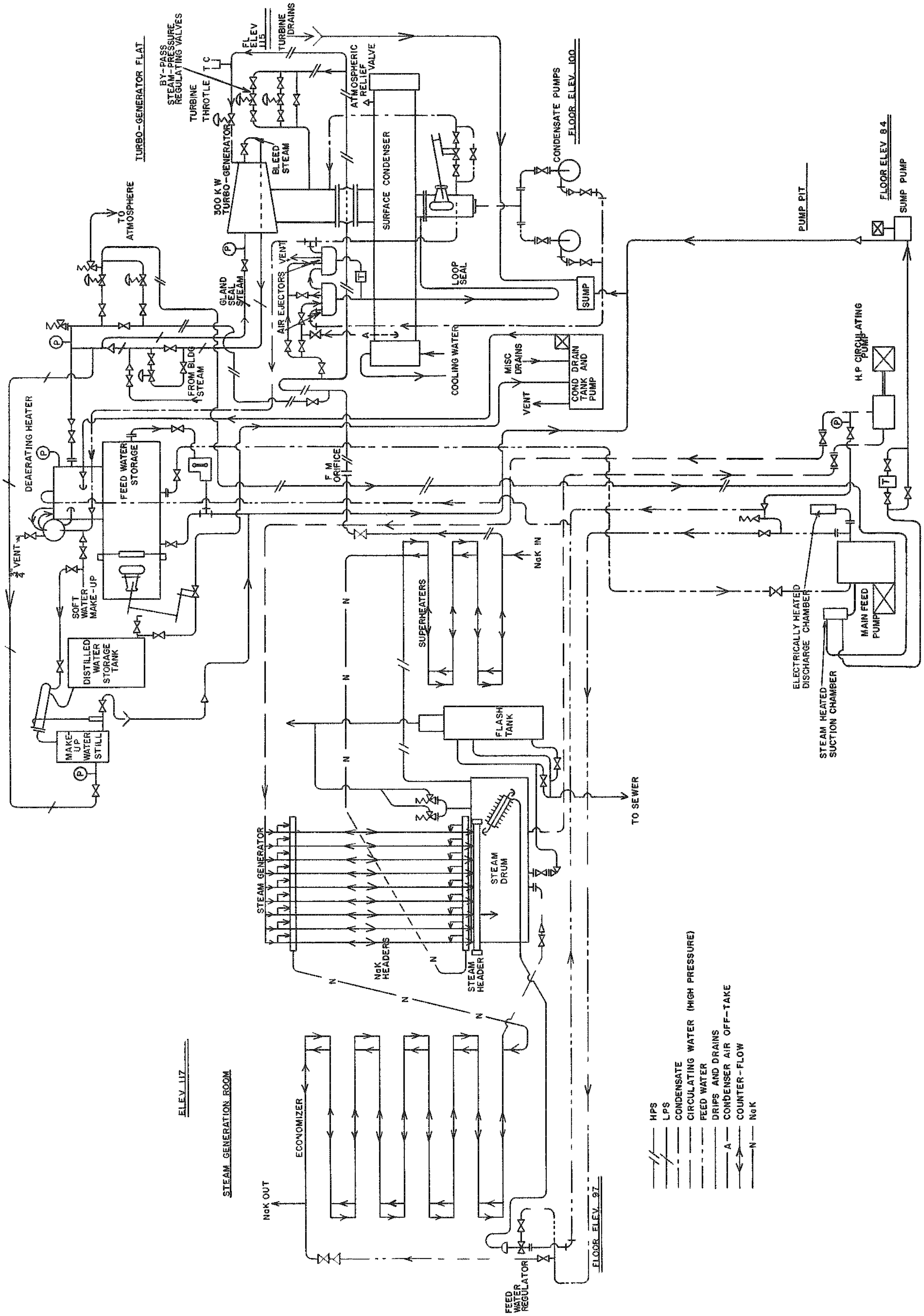

FIG. 17. STEAM SYSTEM DIAGRAM EBR I