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NAA-SR-1028

REACTORS—POWER

UNITED STATES ATOMIC ENERGY COMMISSION

A SODIUM COOLED, GRAPHITE MODERATED, LOW ENRICHMENT URANIUM REACTOR FOR THE PRODUCTION OF USEFUL POWER

By
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ABSTRACT

A design study is presented for a sodium cooled, graphite moderated power reactor utilizing low enrichment uranium fuel. The design is characterized by dependence on existing technology and the use of standard, or nearly standard, components. The reactor has a nominal rating of 167 thermal megawatts, and a plant comprising three such reactors for a total output of 500 thermal megawatts is described. Sodium in a secondary, non-radioactive, circulation system carries the heat to a steam generator at 910° F and is returned at 420° F. Steam conditions at the turbine throttle are 600 psig and 825° F. Cost of the complete reactor power plant, consisting of the three reactors and one 150-megawatt turbogenerator, is estimated to be approximately $43, 165, 000. (auth)

This report is based upon studies conducted for the Atomic Energy Commission under Contract AT-11-1-GEN-8.

I. INTRODUCTION

During the period preceding the fall of 1952, the study of power reactors was directed toward the objective of producing minimum cost plutonium for military purposes. It was shown (Refs. 1, 2, 3 and 4) that substantial savings in the cost of plutonium could be obtained by reactors which simultaneously produced plutonium and useful power. In studies conducted at North American Aviation, Inc. (NAA), it was determined that of those reactor systems considered "engineeringly feasible," the most promising system was one using sodium coolant, low enrichment uranium fuel, and graphite moderator. A reactor of this type was described in Ref. 5. It was noted that "this reactor has great potentials.....especially if the emphasis on the end purpose changes from plutonium production to power."

In December, 1952, there was a shift of emphasis in the AEC power reactor program from reactors combining the production of plutonium with useful power to reactors designed solely to produce heat for the generation of electrical power. With this change of objective in mind, a review was made at NAA of the earlier engineering work on various reactor systems. It was concluded that, with certain modifications, the sodium-graphite, low enrichment-uranium system found most desirable for plutonium and power also appeared to be a very promising prospect for power only.

To better establish the objectives of a development program for the sodium-graphite type of reactor and to obtain more realistic estimates of construction and operating costs, work was undertaken to prepare a preliminary design of a reactor capable of the production of 167 thermal mw. The present report discusses the design features of this reactor, and of a central station plant utilizing three such reactors for a total output of approximately 150 electrical mw. The cost of the complete reactor plant, as well as the cost of electric power produced, is estimated.

The particular sodium-graphite reactor utilized in the plant described in this report has been designated the "P-12" in NAA notation. The design arrangement has been referred to as the "thru-tube," wherein vertical tubes

carrying the sodium coolant penetrate vertically through the graphite stack. These tubes are manifolded at each end into headers carrying the sodium, and the fuel elements are suspended within the tubes in the coolant stream. Preliminary design studies have been carried out on other basic sodium-graphite reactor arrangements (see Ref. 6). One arrangement in particular, known as the "tank-type," may have significant advantages over the "thru-tube." The P-12 reactor described in this report, however, appears to be a very workable application of the "thru-tube" arrangement and possesses a number of desirable features.

II. DESIGN FEATURES OF THE REACTOR COMPLEX

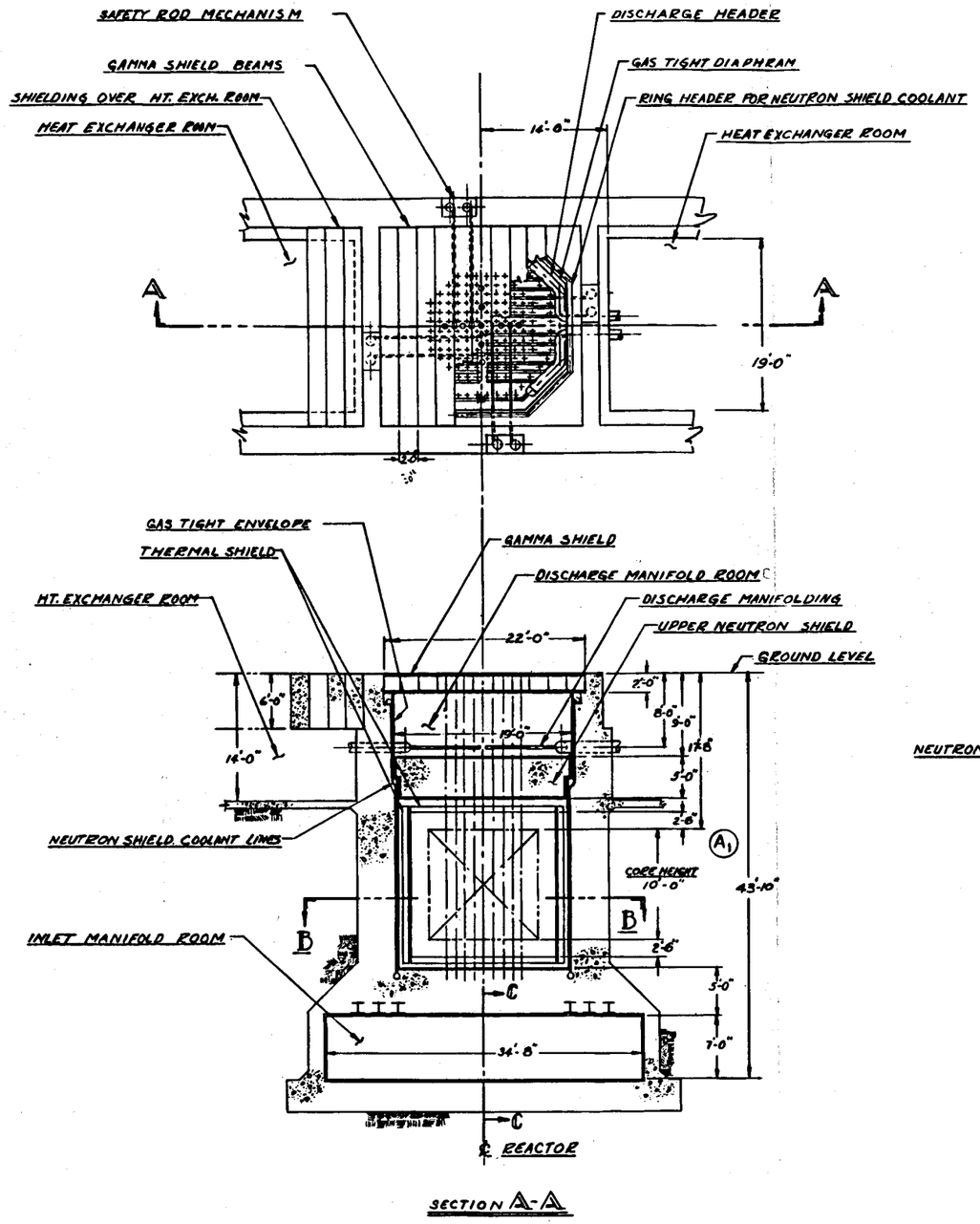
A. Reactor

1. General - The P-12 reactor consists of an octagonal graphite stack surrounded by an iron thermal shield, a layer of thermal insulation and a concrete neutron shield. Coolant tubes pass vertically through the graphite stack and through the top and bottom neutron shields.

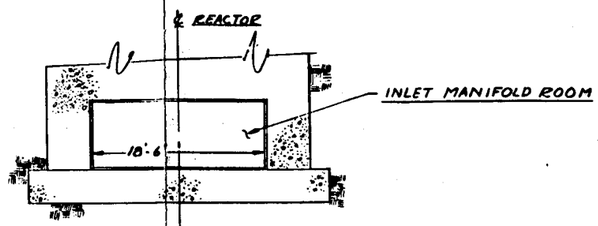
The structure will be located largely below ground. This arrangement simplifies the reactor building and the shielding. Vertical and horizontal sections of the reactor and individual structural details are shown in Figs. 1 and 2.

2. Core and Reflector - The essential elements of the core are: The fuel (low enrichment uranium alloy), and moderator (graphite), a coolant (sodium), and some structural material (Type 304L stainless steel) for isolating the coolant from the fuel and moderator. The moderator represents the major portion of the bulk and weight of the core. It is therefore the graphite which determines the size of the reactor and dictates much of the structural design. A reflector surrounds the core on all sides. It allows the use of a fuel with lower U^{235} content for a given size core. In addition, it contributes a certain shielding effect. It is constructed of graphite similar to that used for the moderator.

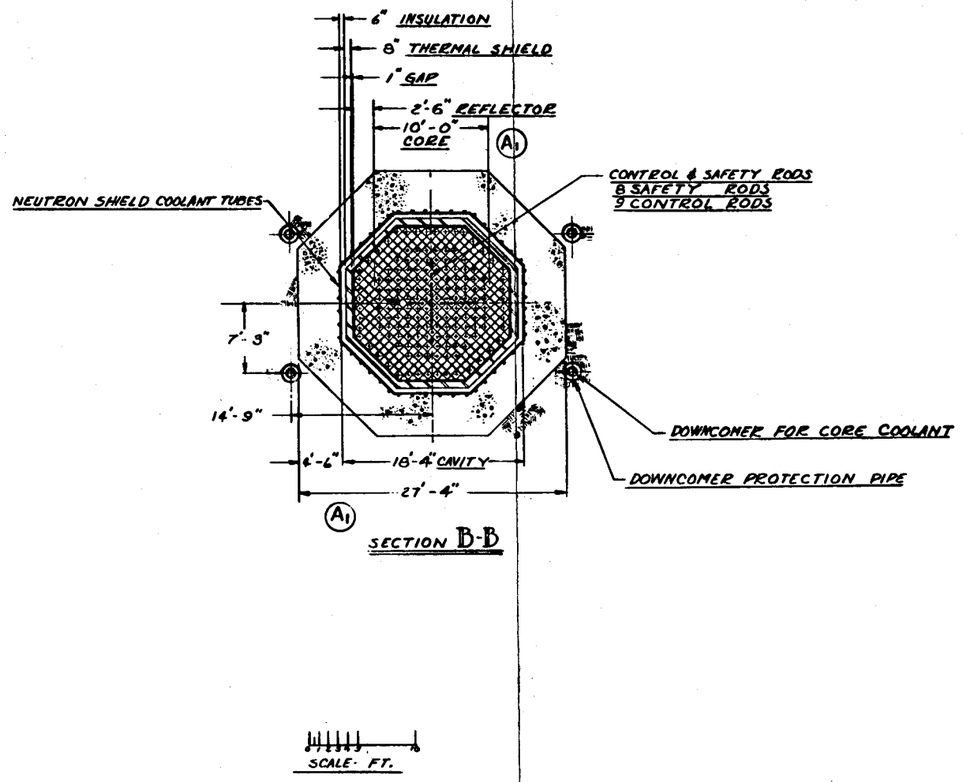
For reasons of insulation, and more particularly radiation shielding, the reactor core is located approximately at the geometrical center of the reactor



SECTION A-A

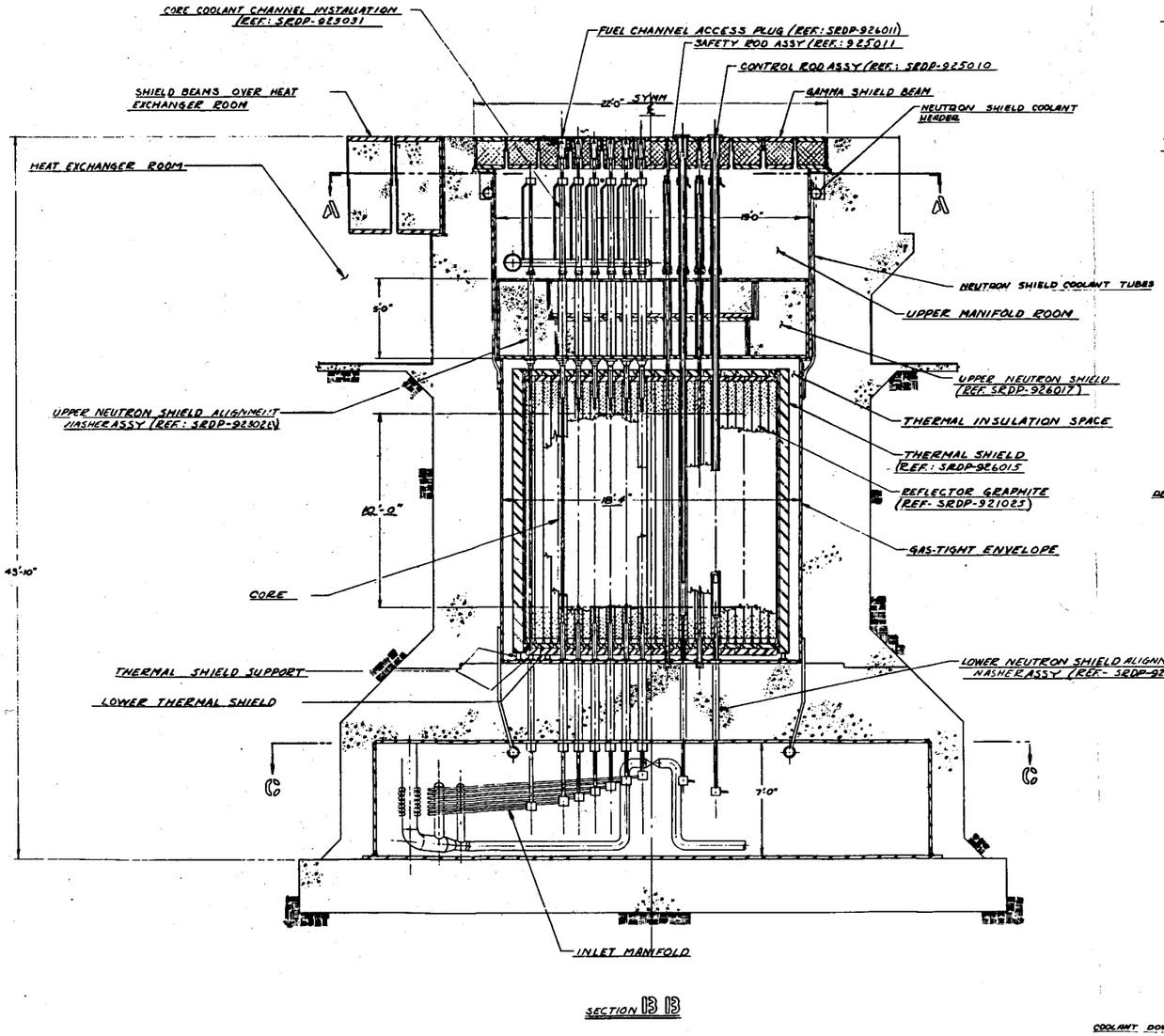


SECTION C-C



SECTION B-B

Fig. 1. Basic Dimensions



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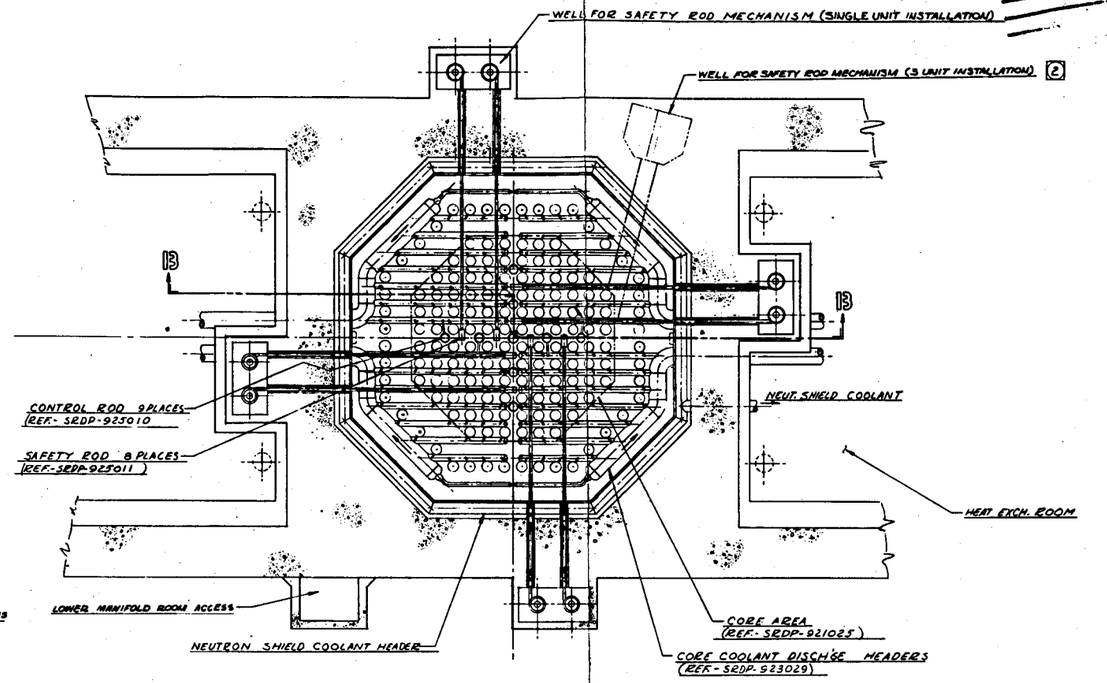

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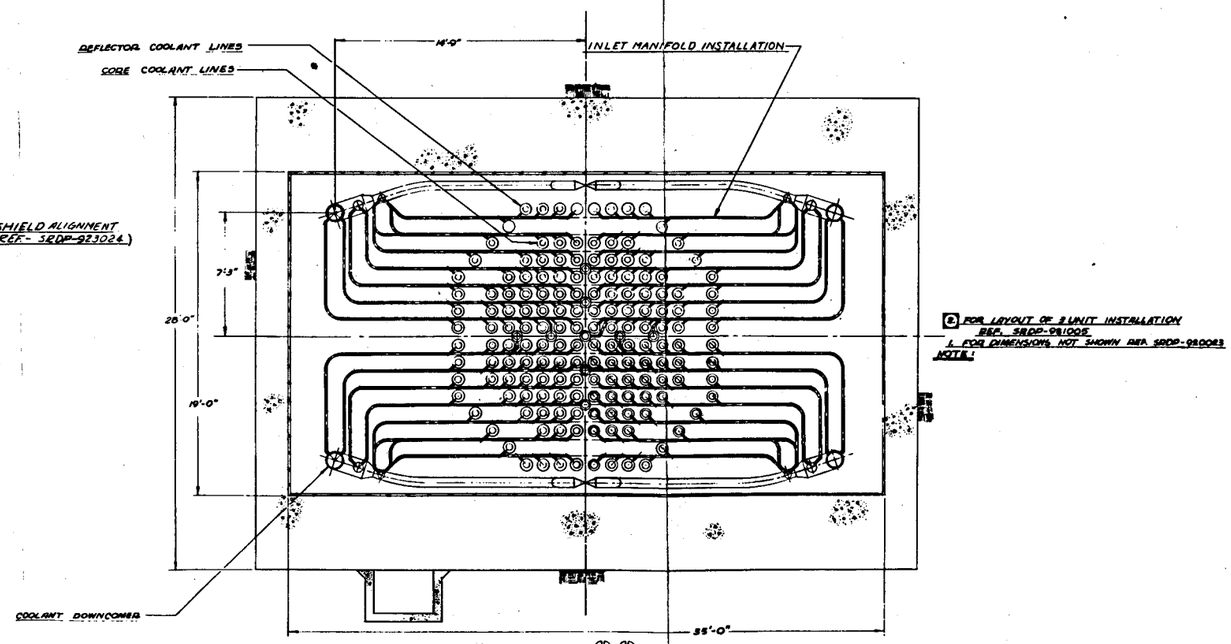


SECTION C C
(CORE COOLANT MANIFOLDING ONLY)

SCALE IN FEET

Fig. 2. General Arrangement

structure. The core, or active part of the reactor, is a right octagonal prism 10 ft across the flats and 10 ft high. With the addition of 2 ft 6 in. of graphite reflector on all faces the dimensions of the prism become 15 ft across the flats and 15 ft high. The core and reflector are made up of independent vertical graphite columns 7.00 in. by 7.00 in. in cross section 15 ft high (see Figs. 3 and 4). Each column is formed from four blocks (three 4-ft long and one 3-ft long) placed one above the other as shown in a detail of Fig. 3. The cold clearance between the columns is approximately 0.070 in.

Alternate graphite columns in the core section are bored to contain the vertical coolant tube assemblies. These are described in detail below. At selected locations in the core additional holes are provided in the graphite for the control rods, safety rods and instrument thimbles. The graphite columns in the surrounding reflector are solid. The graphite columns are supported from below by the lower neutron shield. The weight of the columns is transmitted to the lower neutron shield by individual pipe-like supports which extend thru the thermal shield and thermal insulation. At their upper extremities the columns are located with reference to the upper neutron shield. The bored columns are guided by tubular assemblies extending thru the thermal insulation and thermal shield; the unbored columns are guided by adjacent bored columns thru the thermal shield plates.

Of the various sizes and shapes of graphite columns which could be used with the 10-in. lattice spacing, the 7.00 in. by 7.00 in. square cross section was selected on the basis of earthquake loads and internal thermal stresses. The large moment of inertia associated with a large cross section is desirable in minimizing stresses during earthquakes; a smaller cross section, on the other hand, is effective in reducing internal stresses in the graphite which result from the thermal gradients established during operation.

The heat generation in the graphite due to gamma attenuation and neutron moderation amounts to approximately 7 per cent of the total heat generated in the reactor. This heat is removed by conduction through either the coolant tubes or the thermal shield to the main sodium coolant stream. During full power operation the maximum graphite temperature is computed to be 1950° F. This calculation takes into account a decrease in the thermal conductivity of the graphite due to radiation damage and probably represents an upper limit.

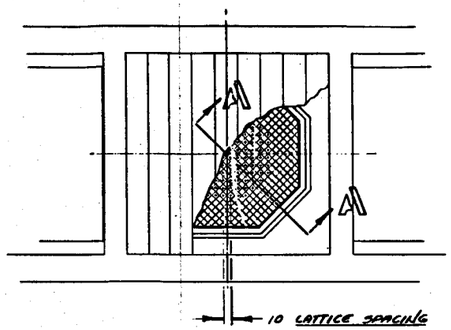
The graphite column is allowed to grow vertically under thermal expansion. Since the columns are supported independently from a cold surface and are separated by a gap, the graphite may also grow in a horizontal plane without disturbing the alignment of the lattice. Dimensional changes in the graphite are expected to be small since radiation damage to the crystal structure is expected to be largely annealed out at the operating temperature (Ref. 7). The graphite in the core and reflector is protected against oxidation or other chemical action by a helium atmosphere.

3. Coolant Tubes - The term "core coolant tube assembly" applies to only those elements of the cooling system located in the active core region. There are a total of 120 of these assemblies; their function is first, to channel coolant sodium around the contained fuel element, and second, to exclude sodium from the moderator material. Each core coolant tube assembly is arranged with its axis vertical and consists of a coolant tube, a top and a bottom extension, and a guard tube. The coolant tube is 304L stainless steel, 3.055 in. ID; the wall thickness, 0.035 in., and the length, 15 ft. This is a drawn seamless tube which is joined at its upper and lower extremities to the top extension and the bottom extension, respectively. The latter are assemblies of heavy wall seamless pipe (or tubing) and fittings of 304L stainless steel (see Figs. 5, 6, 7, 8, 9, and 10).

Surrounding the coolant tube and reducer assembly is a fluted guard tube of zirconium or zirconium alloy. In case of a crack or fissure in the stainless steel coolant tube, the zirconium guard tube prevents sodium from impinging on the graphite by channeling it to the bottom of the reactor where it is drained out of the core structure. The zirconium tube is pre-formed from strip and is wrapped around the coolant tube. The edges of the strip are then seam-welded to form the guard tube.

The reflector coolant tubes are located in the outermost graphite columns. They are used to remove heat from the reflector as well as from the thermal shield.

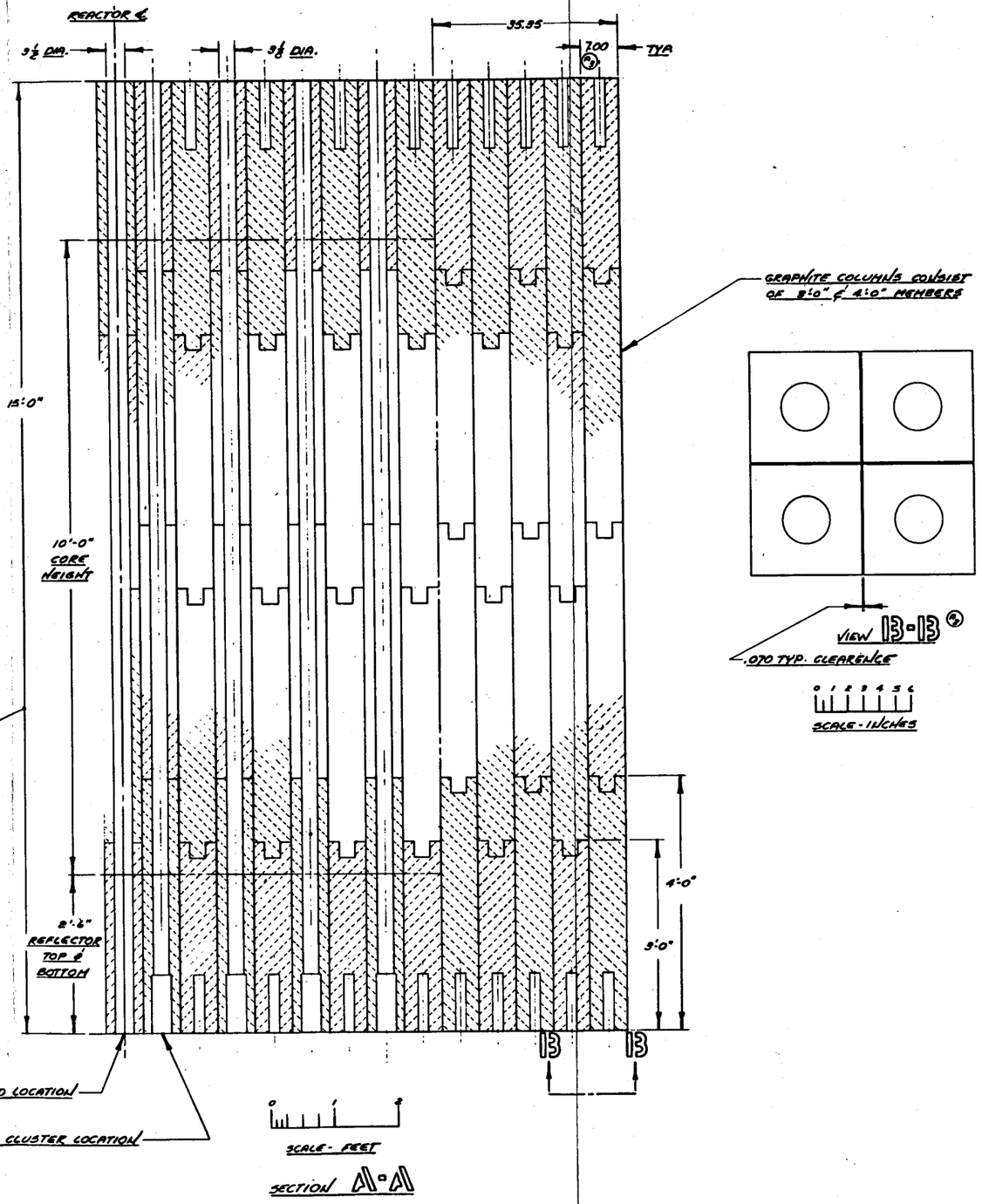
Type 304L stainless steel has been selected for the coolant tubes for its compatibility with sodium and for its strength, workability and welding characteristics (Ref. 8). Zirconium has been selected for the guard tube for high-temperature strength and low neutron capture cross section.



PLAN VIEW

VERTICAL CROSS SECTION

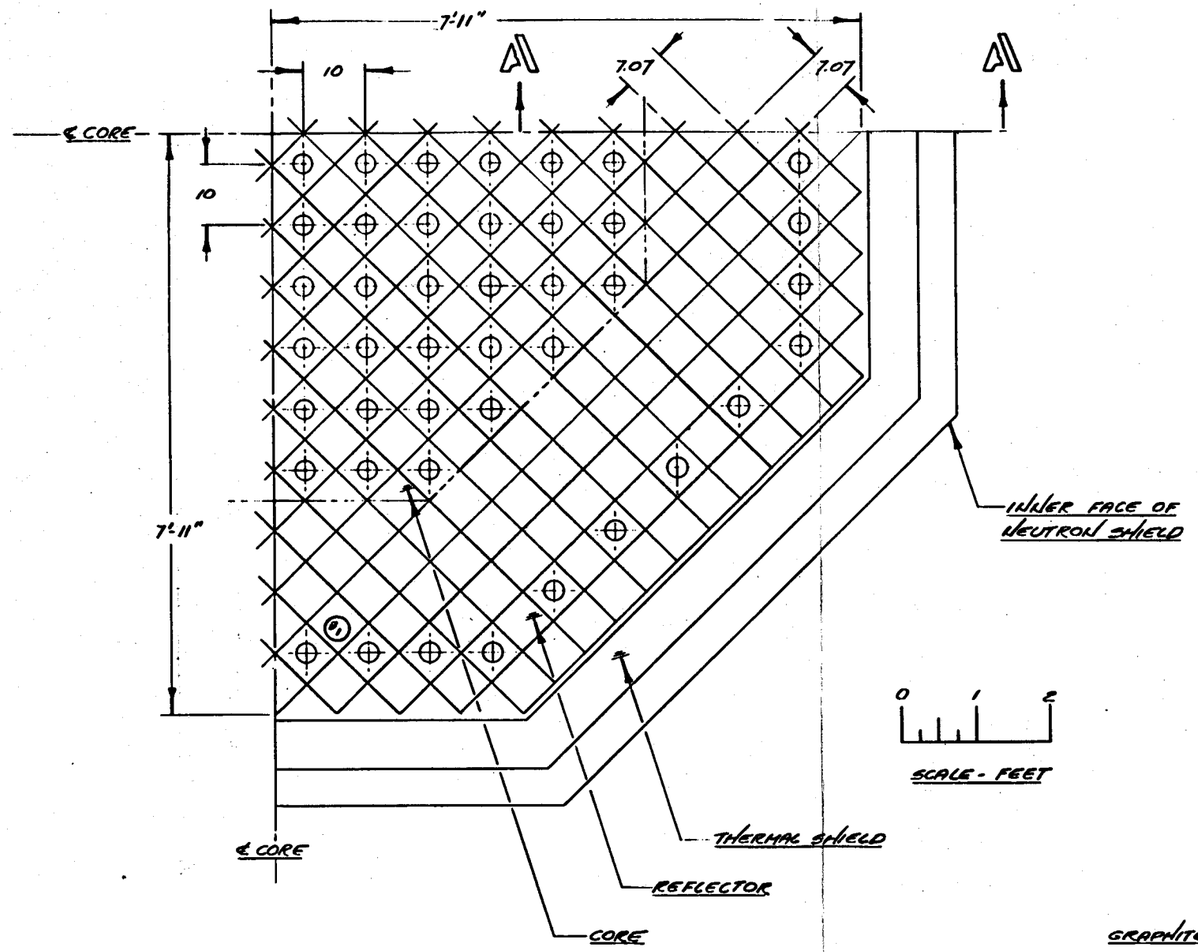
SCALE - FEET



SECTION A-A

SCALE - INCHES

Fig. 3. Core Graphite Arrangement



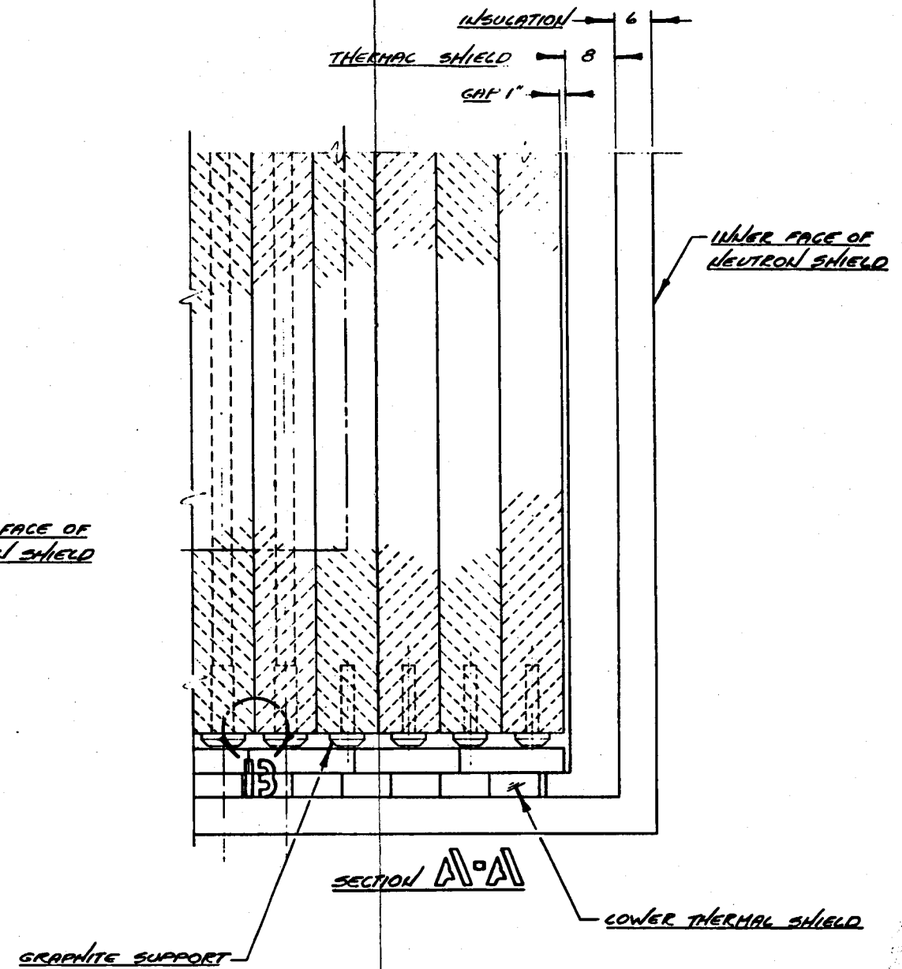


Fig. 4. Lattice Pattern

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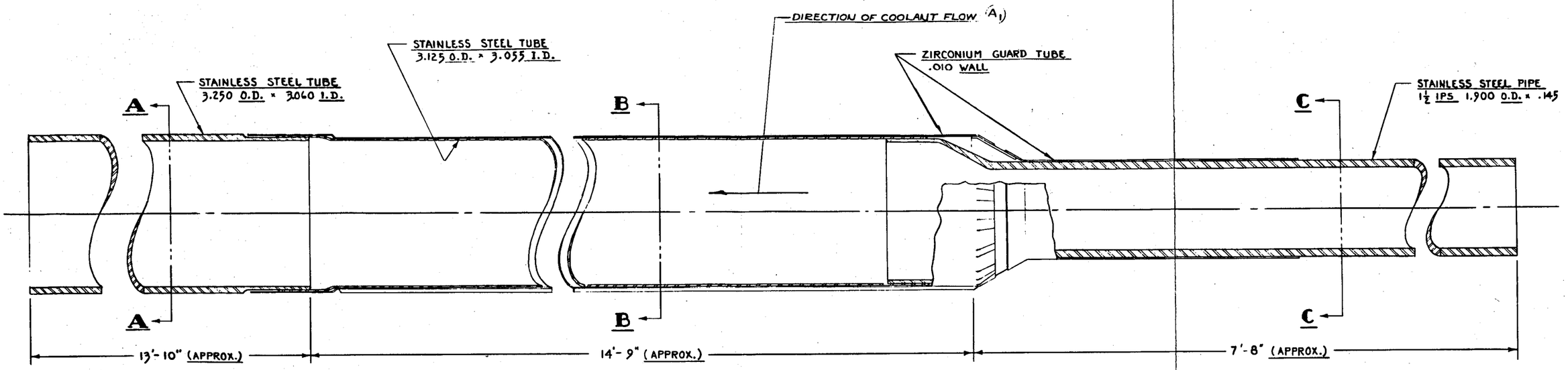

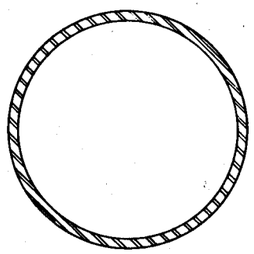

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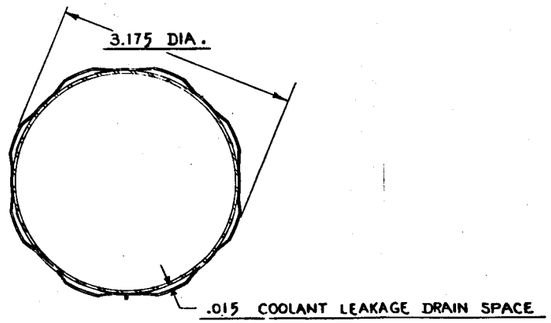

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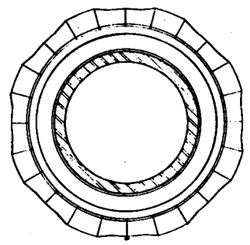

SECTION **C-C**

Fig. 5. Coolant Channel Assembly Proposal

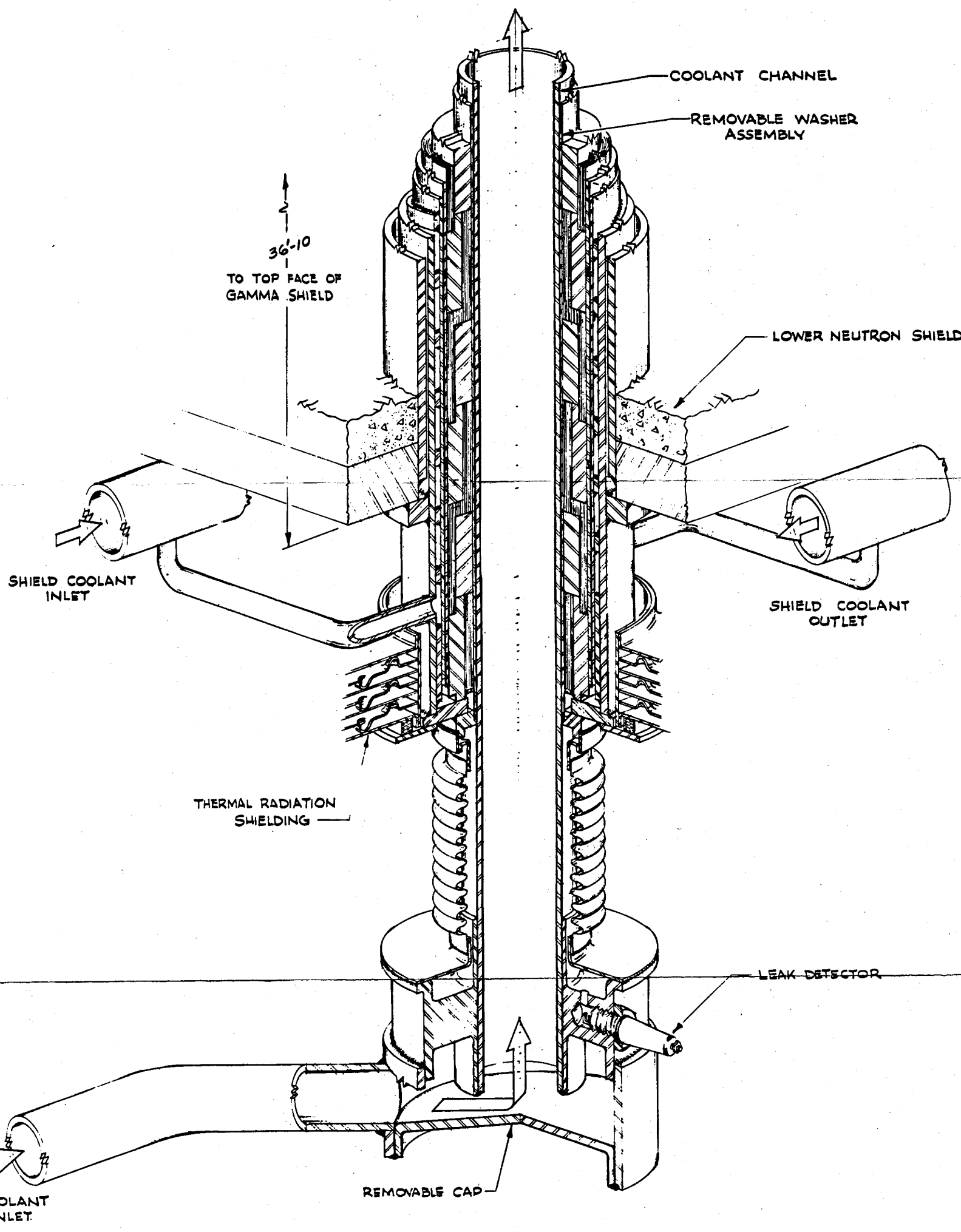

Fig. 6. Coolant Channel Perspective Section thru Inlet Manifold

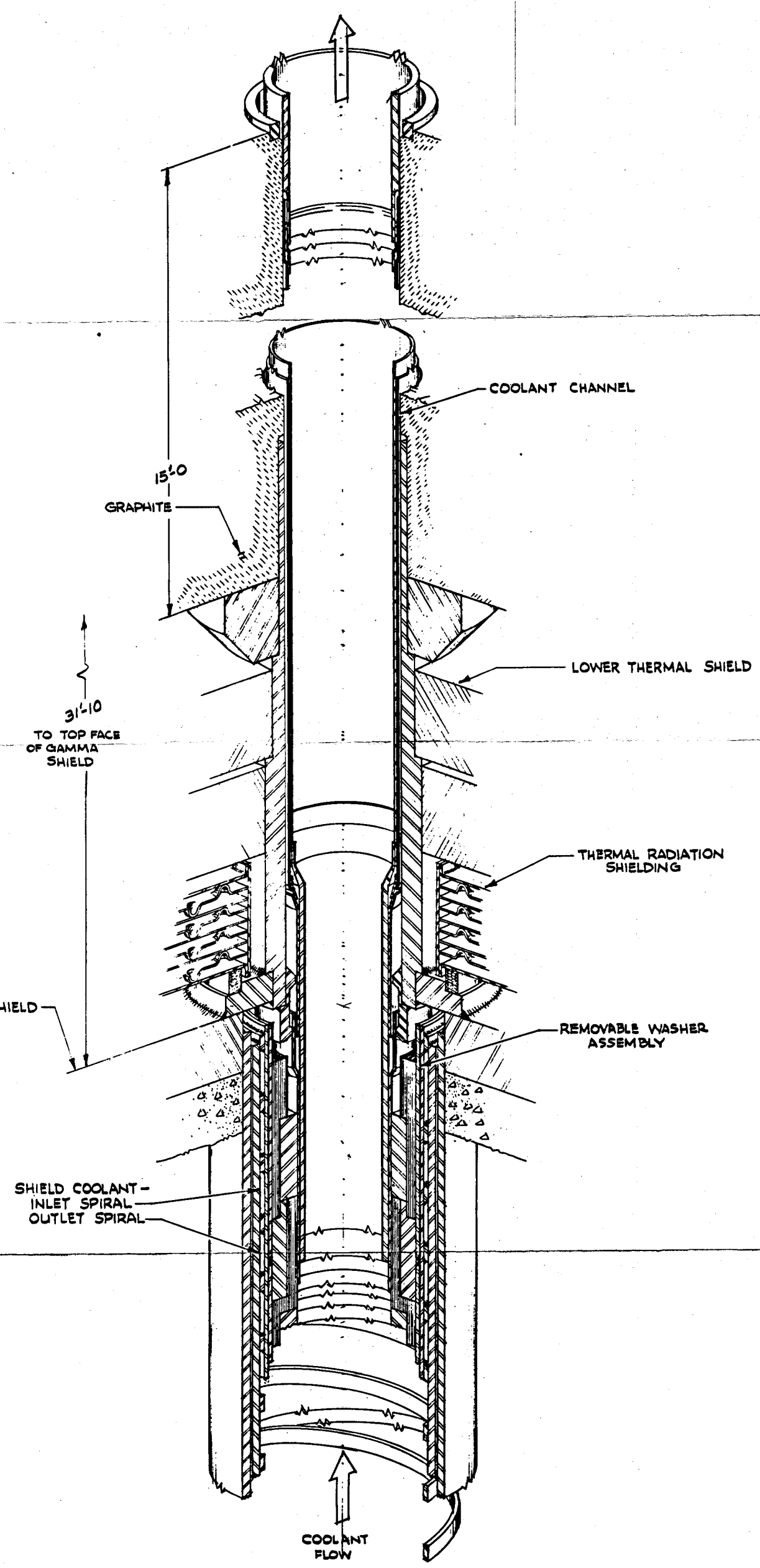

Fig. 7. Coolant Channel Perspective Section thru Core and Lower Shield

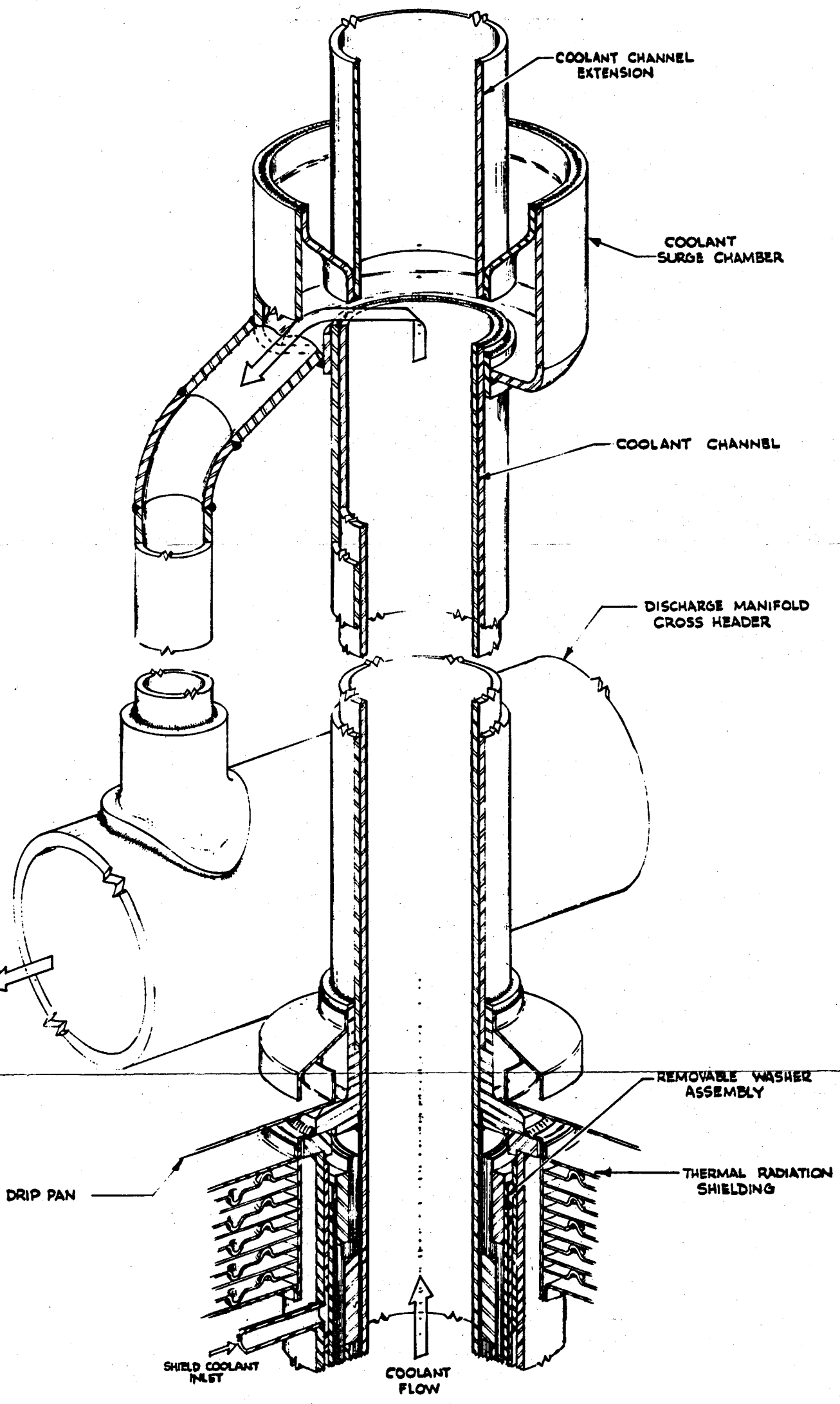

Fig. 8. Coolant Channel Perspective Section thru Outlet Manifold

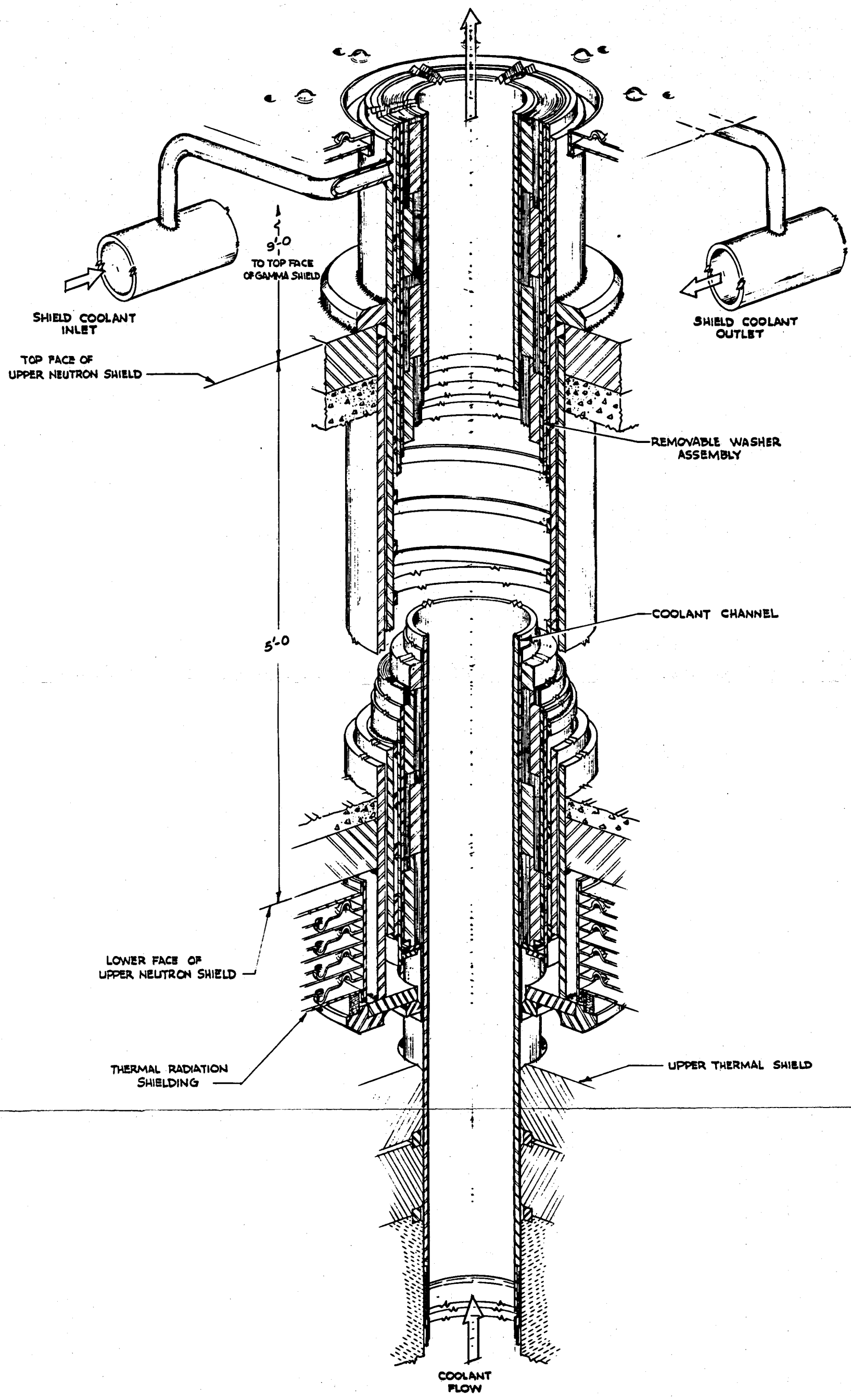

Fig. 9. Coolant Channel Perspective Section thru Neutron Shield

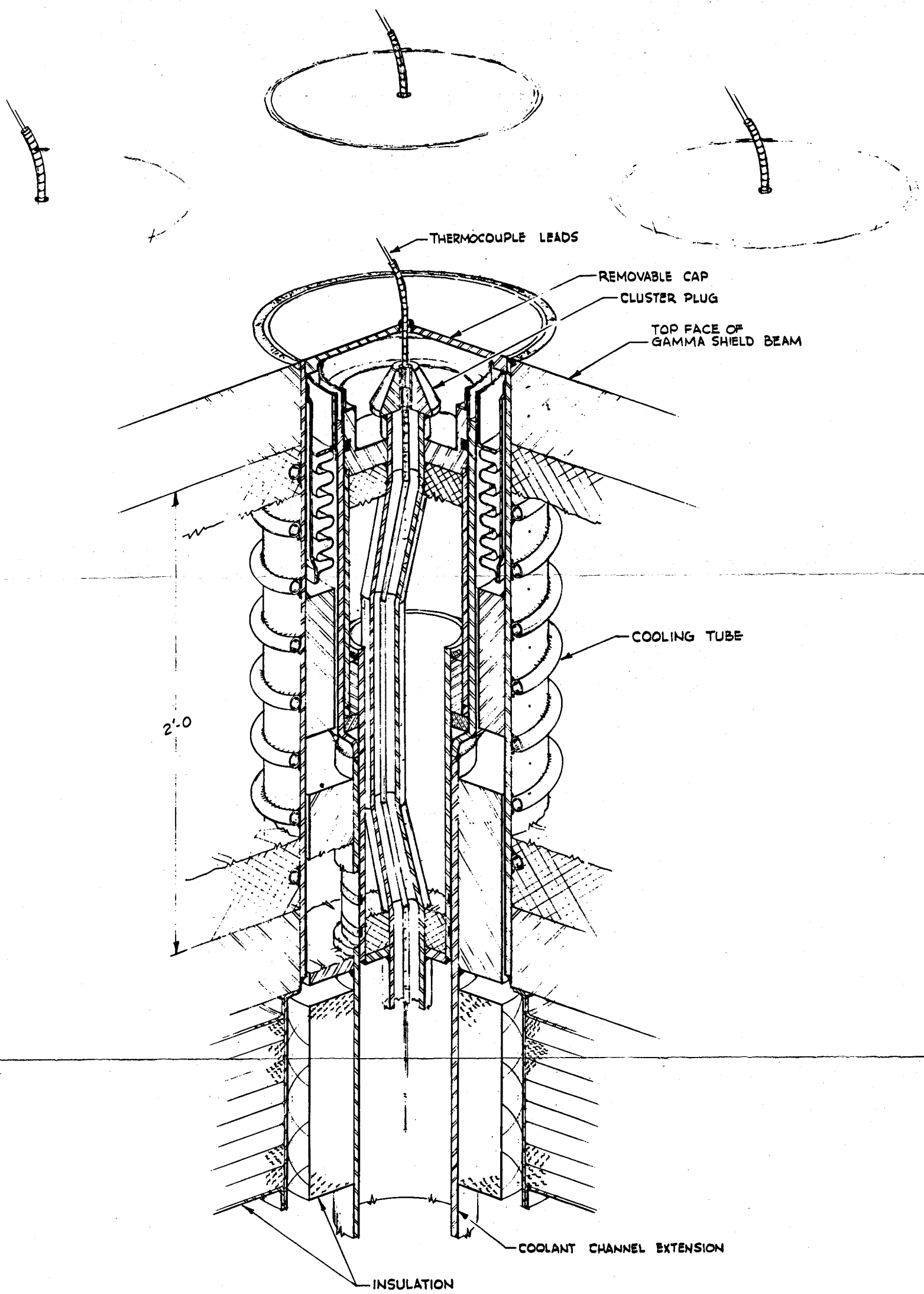

Fig. 10. Coolant Channel Perspective Section thru Gamma Shield

The manner in which the coolant tube assemblies are connected to the cooling system is described below in Section II-B-2.

4. Fuel Elements - The fuel elements are designed to permit heat to be generated in and extracted from the fuel at high temperature while at the same time preventing escape of the fission fragments. The rod cluster is the basic configuration of the fuel elements. A 19-rod cluster is used. Each fuel rod is made up of ten uranium-alloy fuel slugs 0.455-in. diameter by 12-in. long in a Type 304L stainless steel jacket 0.475-in. ID by 0.010-in. wall. The jacket is closed at top and bottom with a plug and, except for a helium filled expansion space, is filled with sodium. The average thickness of the sodium annulus between the fuel slug and jacket is 0.010 in.

The rods are joined to a common hanger rod from which they are suspended in the coolant tubes. The rods are bound together at their lower extremities and throughout their length with a spirally wrapped wire of 304L stainless steel. Suspended from the lower extremity of the center rod in each cluster is a plug and orifice assembly. This is a stainless steel plug with helical passages for the flow of coolant. It hangs in the coolant tube opposite the thermal shield and performs functions similar to those of the thermal shield. At the lower end of the plug there is provision for attaching an orifice plate which throttles the flow of coolant to the fuel element. The plate is sized according to the location of the element in the reactor.

The fuel slugs are made of 2 weight per cent zirconium-uranium alloy. This alloy is cast, rolled, and machined to the dimensions given above. The alloy is heat-treated for maximum dimensional stability under irradiation.

The uranium is slightly enriched in the fissionable isotope, U^{235}, and is therefore the product of a diffusion plant. Initial and final values of fuel enrichment (before and after irradiation) are given in Table I, Performance and Design Data.

The zirconium-uranium alloy has better dimensional stability under irradiation than that attainable with the best heat-treated uranium. The 2 weight per cent zirconium-alloy is therefore preferred although it has a slightly lower alpha to beta phase transition temperature.

TABLE I

PERFORMANCE AND DESIGN DATA

A. Reactor

1. Reactor Type - Thermal heterogeneous

2. Reactor Operating Data-

Nominal output, (maximum rated)	167 mw (thermal)
Nominal specific power (average), fuel	11.1 mw/ton of U metal (788 kw/kg - 25)
Nominal specific power; (average) moderator (not including reflector)	4.26 mw/ton graphite
Average thermal flux in fuel (uranium metal)	$3 \times 10^{13} \frac{\text{neutrons}}{\text{sec-cm}^2}$
Peak thermal flux in fuel	$6 \times 10^{13} \frac{\text{neutrons}}{\text{sec-cm}^2}$
Fuel enrichment	
Initial enrichment (fraction U^{235})	1.55 per cent
Enrichment for criticality (hot, poisoned, all control rods withdrawn)	1.45 per cent
Average burnup (in individual fuel element at time of reprocessing)	$3800 \frac{\text{mwd}}{\text{ton}}$
Average burnup (fuel in reactor-at equilibrium)	$1770 \frac{\text{mwd}}{\text{ton}}$
Enrichment of irradiated fuel	1.09 per cent

3. Materials

Fuel	Uranium metal, slightly enriched in U^{235} alloyed with 2 weight per cent zirconium
Moderator	Graphite
Core structural material	304L stainless steel
Primary coolant (core)	Sodium
Secondary coolant	Sodium
Reflector	Graphite
Thermal shield	Mild steel
Biological shield	Barytes concrete
Gamma shield	Lead

TABLE I (Continued)

4. Core Nuclear Data	
Lattice type	Square
Lattice spacing	10.0 inches x 10.0 inches
Number of fuel cells	120
Arrangement of fuel cells	Approximates a circle
Core diameter (including reflector)	15.3 ft
Active core height	10.0 ft
Core composition, per cent by volume	
Graphite	92.083
Sodium	3.951
Type 304L stainless steel	0.6293
Uranium metal	3.089
Zirconium	0.100
Void	0.1477
U^{235} content (initial loading) (R = 0.0155)	212 kg
Lattice constants (for criticality at 400° C neutron temperature) (R = 0.0145)	
η (for fuel cell)	1.464
ϵ	1.047
p	0.8040
f	0.9021
k_∞	1.0992
L_f^2	339 cm^2
L^2	177 cm^2
B^2	181 x 10^{-6} cm^{-2}
Initial conversion ratio (U^{235} into Pu^{239})	0.70

TABLE I (Continued)

5. Control and Safety Rod Data	
Number of control rods	9
Shim rods	8
Regulating rods	1
Δk in regulating rods	1.27 per cent
Δk in shim rods	
Inner group (4)	4.38 per cent
Outer group (4)	2.67 per cent
Total Δk in control rod system	7.5 per cent
Maximum withdrawal rate	24 in./min
Number of safety rods	8
Number in safeguard bank	4
Δk in safeguard bank (outer 4 rods)	3.6 per cent
Total Δk in safety rod system	7.6 per cent
6. Fuel Element Data	
Type of fuel element	19 rod cluster
Number of elements in reactor	120
Fuel slug material	U-Zr alloy (2 weight per cent zirconium)
Diameter (of U-metal slug)	0.455 inch
Length (of U-metal slug)	12 inches
Initial enrichment	1.55 per cent
Weight of U (one reactor)	13,655 kg
Fuel element length	10 ft
Jacket	0.475 inch ID x 0.010 inch SS
Bond	0.010 inch Na

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TABLE I (Continued)

7. Coolant Tube Data

	Inner Tube	Outer Tube
Inside diameter	3 055 inchs	
Wall thickness	0.035 inch	0.010 inch
Material	304L SS	Zirconium

B. Cooling System Data

1. Reactor

	Core	Reflector & Thermal Shield
Coolant	Na	Na
Temperature in, °F	450	450
Mixed mean temperature out, °F	940	800
Lowest temperature out, °F	889	- -
Highest temperature out, °F	1050	- -
Max. coolant velocity, ft/sec	10	- -
Pressure drop through coolant tube, psi	8	- -
Flow rate, lbs/hr x 10^{-6}	3.82	0.105
Hottest 19-rod cluster, mw	1.95	- -
Max. heat flux, Btu/hr ft^2	373,000	- -
Max. fuel temperature, °F	1,200	- -
Total reactor power, mw	167	- -

2. Primary Loops

Number of parallel circuits per reactor	4
Inlet temperature to primary heat exchanger	940° F
Outlet temperature from primary heat exchanger	450° F
Flow rate in each circuit	9.55 x 10^5 lbs/hr
Number of pumps in each circuit	2

TABLE I (Continued)

3. Secondary Loops	
Number of parallel circuits per reactor	2
Inlet temperature to superheater	910° F
Outlet temperature from economizer	420° F
Flow rate in each circuit	1.91×10^6 lbs/hr
Number of pumps in each circuit	4
C. Power Plant (single reactor system)	
1. General	
Number of units in parallel	2
Drum pressure	615 psia
Saturation temperature	492.5° F
Total temperature	825° F
Feedwater temperature	280° F
Total steam rate	486,200 lbs/hr
Total heat rate	167 megawatts
2. Superheater	
Steam flow rate	486,200 lbs/hr
Steam inlet temperature	492.5° F
Steam outlet temperature	825° F
Sodium flow rate	1.91×10^6 lbs/hr
Sodium inlet temperature	910° F
Sodium outlet temperature	815° F
3. Evaporator	
Water inlet temperature	485° F
Steam outlet temperature	492.5° F
Sodium inlet temperature	815° F
Sodium outlet temperature	505° F
4. Economizer	
Water inlet temperature	280° F
Water outlet temperature	470° F
Sodium inlet temperature	505° F
Sodium outlet temperature	420° F

Sodium was selected for the bonding material because of its high thermal conductivity, relatively low neutron absorption cross section, resistance to radiation damage, and fairly short half life (15 hours for Na^{24}). As in the case of the coolant tube, Type 304L stainless steel was selected for the jacket material for its compatability with sodium and its availability, strength, and workability.

5. Thermal Shield - The thermal shield is designed to attenuate the incident gamma rays and neutrons so that the escaping radiation produces small, and almost negligible, thermal and radiation damage effects in the neutron shield. The side thermal shield is constructed of low carbon steel (Type 1010). It is 8 inches thick and encloses on all sides the graphite stack which constitutes the core and reflector. The thermal shield is a right cylinder of octagonal section made up of interlocking composite steel slabs 2-feet wide by 8-inches thick by 16-feet long arranged with the long dimension vertical. The slabs are made in four pieces, two 3-1/2 inch slabs and two 1/2-inch plates (see Fig. 11). The assembly is bolted together. The slabs are supported individually from the lower neutron shield and guided by two pins anchored in the side neutron shield. Heat generated in the slabs is removed by sodium which flows through coolant channels located in the reflector graphite.

The bottom thermal shield is also 8-inches thick. It consists of two layers of 20-inch by 20-inch by 4-inch interlocking blocks. The pipe-like columns which support the graphite of the core and reflector go through the thermal shield blocks and carry their weight (see Fig. 12). Heat generated in the blocks is removed by sodium in the coolant tubes.

The top thermal shield is made up of two layers of 10-inch by 20-inch by 4-inch thick interlocking blocks which rest on the top of the graphite columns. Heat generated in these blocks is removed by conduction to the sodium which flows through the coolant tubes. The arrangement is shown in Fig. 13.

6. Thermal Insulation - The temperature on the outer face of the thermal shield is not expected to exceed 900° F; the maximum allowable temperature on the inner face of the neutron shield is taken to be 150° F.

The thermal insulation is composed of multiple layers of aluminum foil in an atmosphere of helium. The foil is formed so that individual sheets are

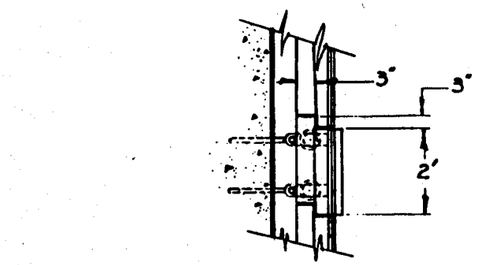
separated from each other by thin layers of helium gas. The forming of the aluminum is such that the layers of gas are further broken up into cells to reduce the circulation of the helium. The aluminum foil constitutes the radiation shield while the helium furnishes the resistance to heat flow by conduction. Convection is made negligible by proper sizing of the gas cells.

The thermal insulation surrounds the thermal shield and, on the vertical faces of the reactor, is attached to it. At the bottom the thermal insulation rests on the neutron shield; at the top of the reactor it is attached to the underside of the upper neutron shield beams.

Aluminum was chosen for its low emissivity, its availability and low cost, and has an adequately high melting point. A gas was chosen for use with the aluminum because lower thermal conductivity could be obtained than with solid insulating materials suitable for this operating temperature. Helium is used in preference to a gas with lower conductivity since it is desirable for the core atmosphere.

7. Container (Gas-tight Envelope) - The envelope material is mild steel, sheet or strip 1/2-inch thick which is joined by welding. The side walls of the container are attached to the inside neutron shield wall. The bottom of the container rests on the top of the lower neutron shield. The steel plates which enclose the upper neutron shield are welded to each other and to the side wall of the container to constitute the top of the container.

8. Neutron Shield - The neutron shield surrounds the entire reactor structure. The principal material is concrete; structural strength is increased by use of steel as required. The lower neutron shield is designed to reduce the radiation level in the inlet manifold room below so that it may be made accessible to service personnel while the reactor is shutdown. While the reactor is operating at power, the important consideration is the neutron current which leaks down through the shield. Over the life of the reactor the activity which this current induces in the steel in the manifold room ("local gammas") must not be hazardous to personnel after a reasonable cooling period. While the reactor is shutdown the requirement is that the afterglow gammas emanating from the core fuel and thermal shield must be so attenuated that the sum of this radiation and the "local gammas" is below tolerance for a short time exposure of, say, a few hours.



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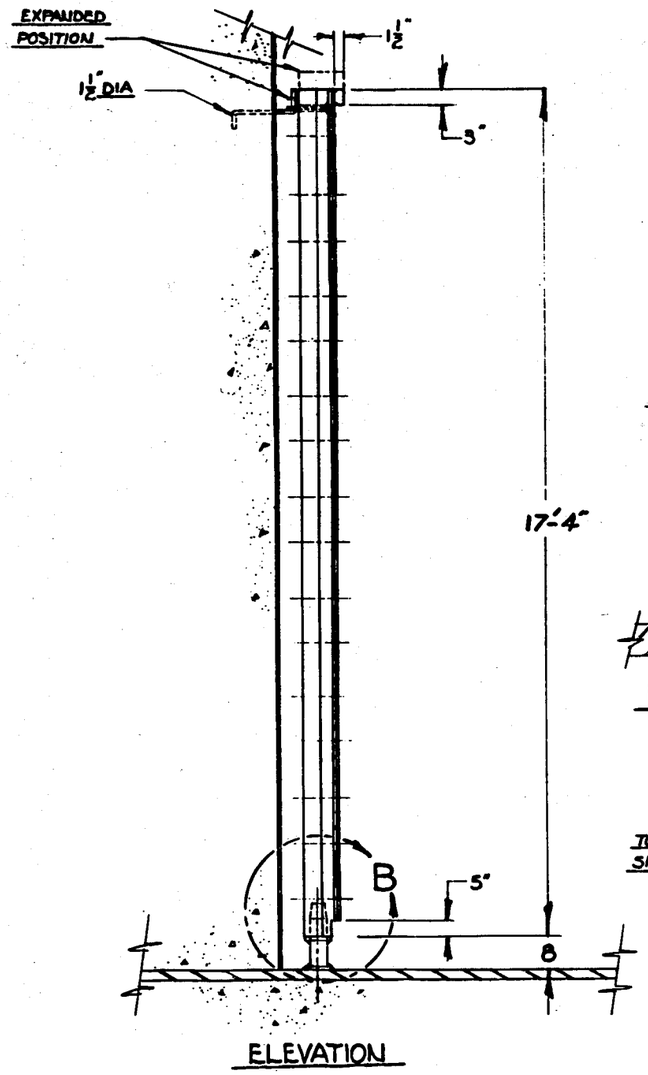

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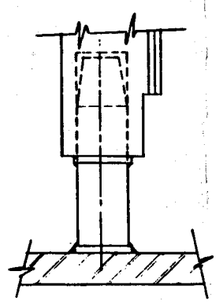
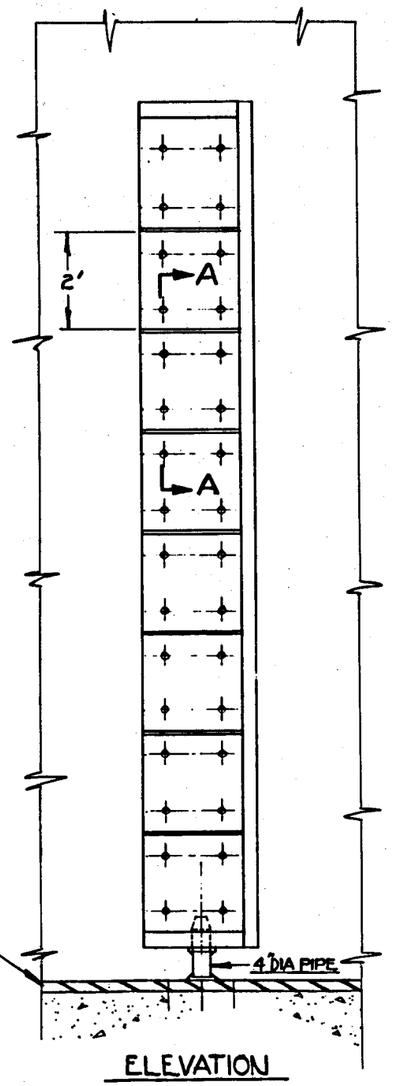


DETAIL-B



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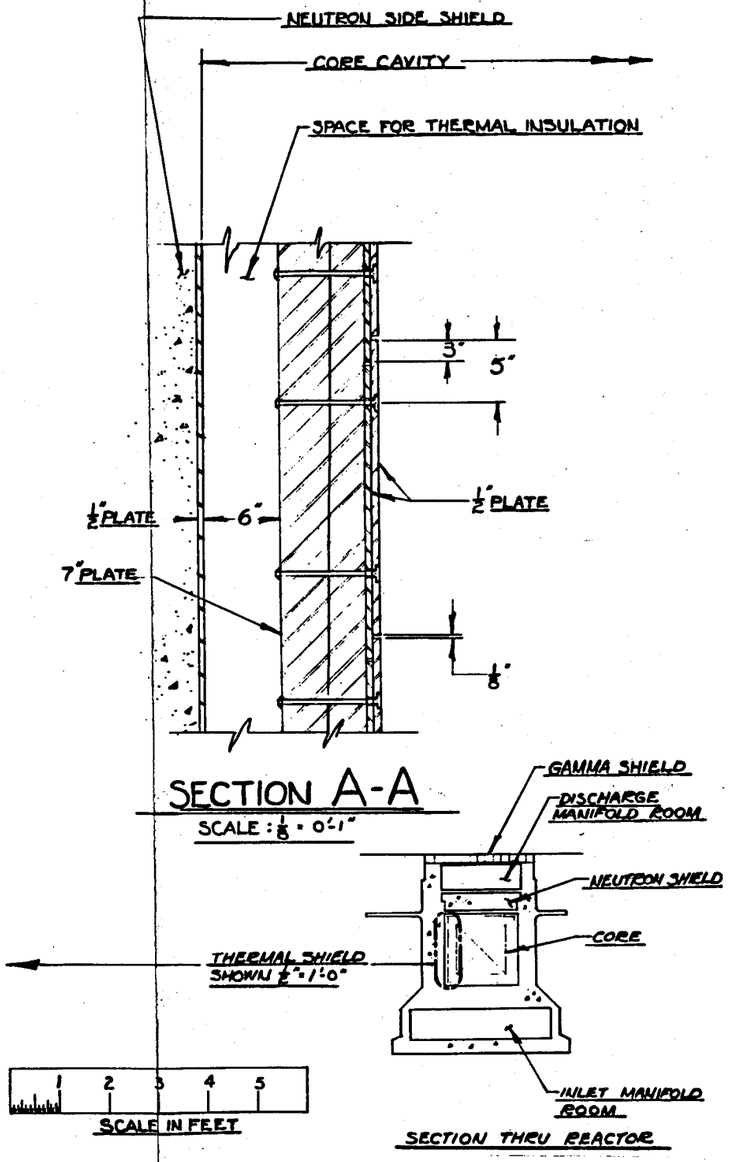


SECTION A-A

SCALE : ⅛" = 0'-1"

SECTION THRU REACTOR

Fig. 11. Thermal Shield Side

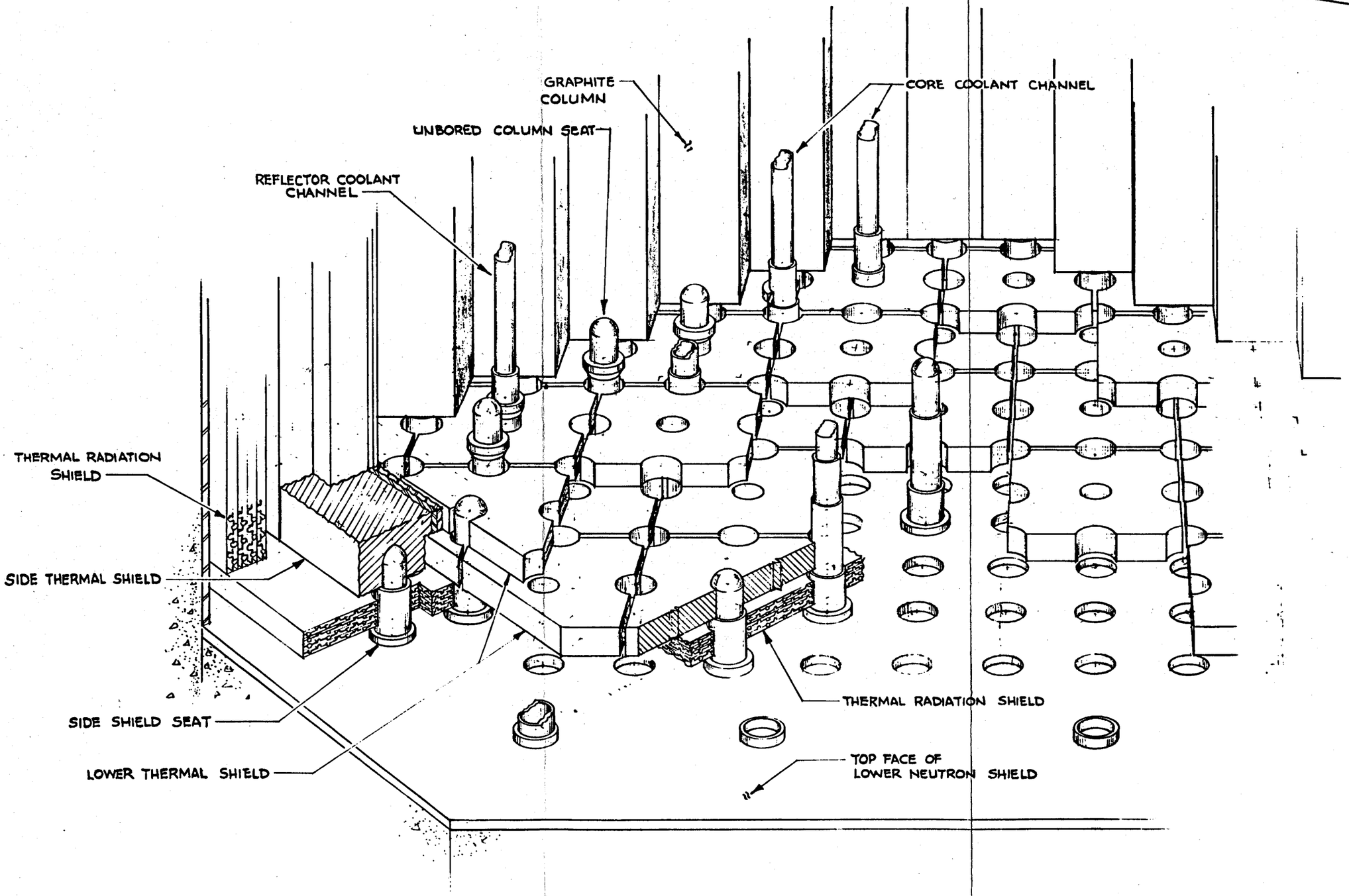


Fig. 12. Thermal Shield Lower Perspective

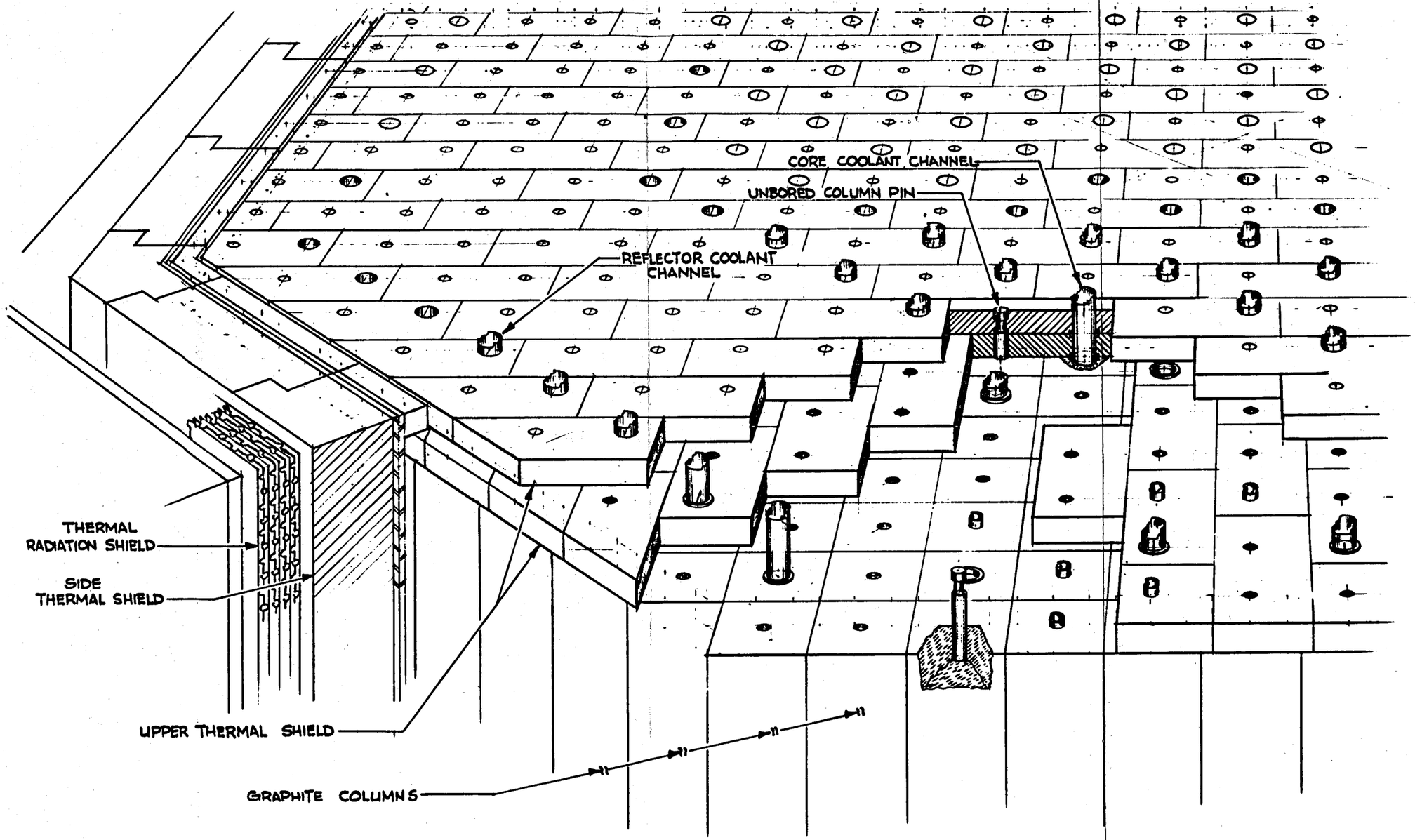


Fig. 13. Thermal Shield Upper Perspective

As a structural member, the lower neutron shield supports and positions the core and reflector graphite, the thermal shield, the thermal insulation, and the bottom of the helium envelope. In addition, it acts as a guide for the coolant channel assemblies. It is cooled to less than 150° F to maintain the lattice spacing in the core substantially fixed and to prevent loss of water from the concrete.

The lower neutron shield is fixed in position and is supported on opposite sides by foundation blocks which constitute two walls of the inlet manifold room. It is a reinforced concrete member approximately 4-feet thick and contains approximately 200 sleeves for the coolant channel assemblies which traverse the core, reflector, and thermal shield. These sleeves are tied into the helium envelope on the upper side of the neutron shield and into a sheet steel membrane on the bottom of this shield. The latter membrane is a portion of the gas-tight steel envelope lining the inlet manifold room. The sleeves have an inside diameter of approximately 5-5/8 inches; the heavy wall extensions of the coolant tubes have an OD of 2 inches where they pass through the sleeves. The annular space between sleeves and coolant tube is occupied by a removable shield and cooling jacket assembly. As a structural element this assembly is designed to compensate for any misalignment between the centerline of the sleeve and the centerline of the coolant tube and to carry the transverse load applied to the coolant tube by connecting piping below. Ring and donut assemblies of iron and thermal insulation material permit the alignment; a graphite ring at the lower end is the bearing surface. The iron rings acts as radiation shielding elements. The donuts of thermal insulating material minimize loss of heat by radiation and conduction from the coolant tube. A cooling jacket surrounds the ring and donut assembly. An organic coolant circulating in this jacket limits the maximum temperature in the concrete to 150° F. Most of the heat in the concrete flows in by conduction; only a small portion is generated by the absorption of radiation.

Barytes concrete was selected for the lower neutron shield for its high specific gravity, high hydrogen density and high strength. Steel structural members provide additional strength, facility of fabrication, and make for accurate alignment of the pipeline columns which support and locate the graphite. The steel also acts as the form for the concrete. In the shield and cooling

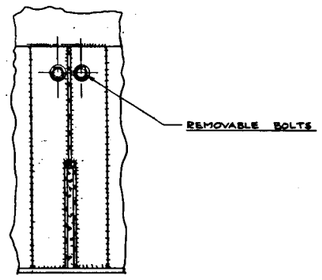
jacket assemblies the rings are made of iron to prevent streaming of gamma rays from the core and fuel element. The thermal insulation, multiple layers of aluminum foil with helium between, was selected for its high resistance to flow of heat by radiation or conduction. The organic coolant will be kerosene or similar petroleum fraction selected for its low chemical reaction rate with sodium, low cost and good resistance to radiation damage.

The side neutron shield is a monolithic wall of reinforced concrete. It has an octagonal section, and the walls are 4-ft thick. It is so designed that no danger exists from irradiation of materials outside the reactor structure and subsequent carrying off of these materials in the ground water. The shield acts as a retaining wall for the surrounding area and a foundation for the upper neutron shield. It also provides passages and shielding for the main inlet sodium lines.

Since gamma ray attenuation is not a serious consideration, ordinary aggregate may be used in place of barytes aggregate. Ordinary concrete meets the principal requirements for the side neutron shield material of high hydrogen density, high strength, availability and low cost.

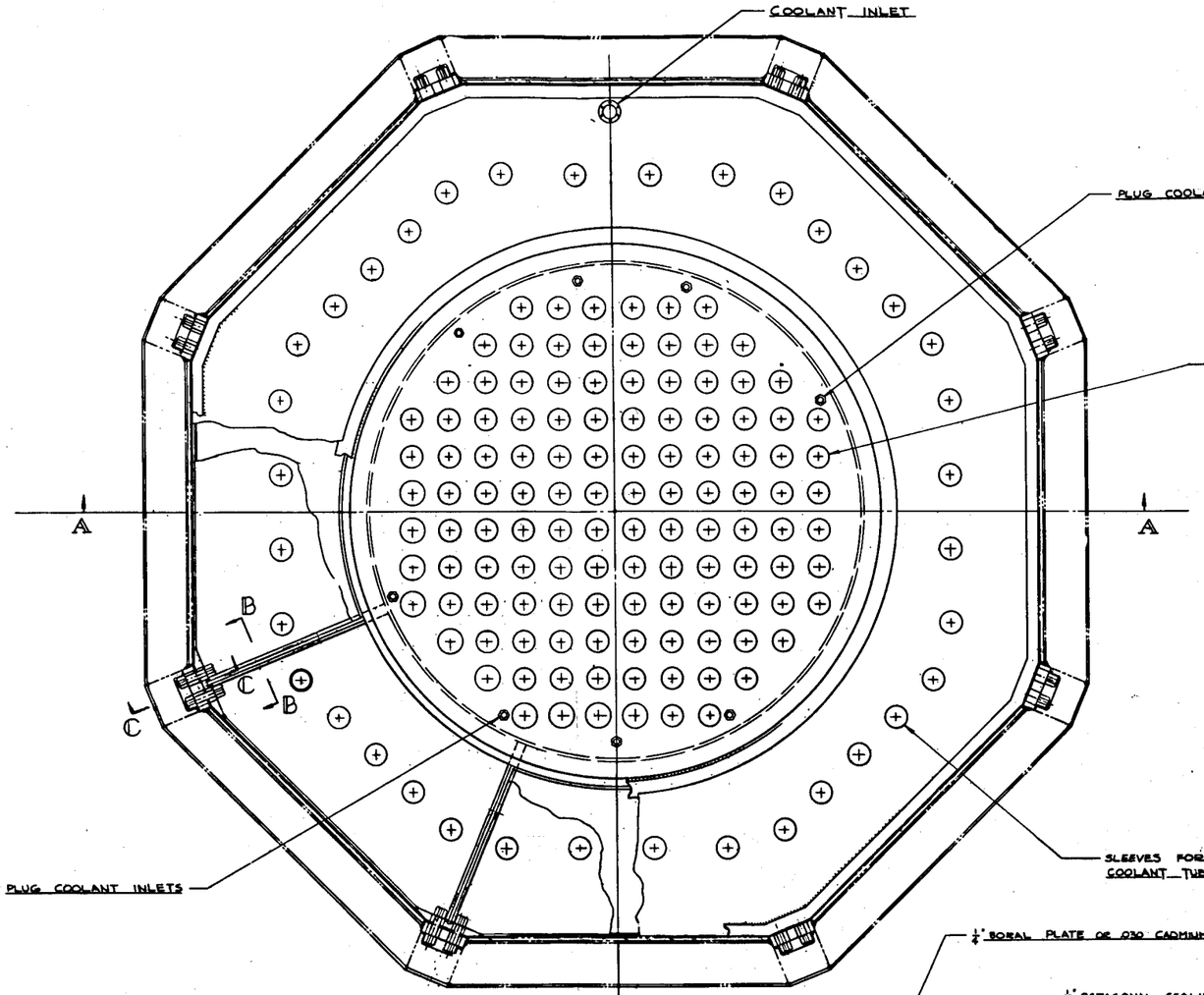
Like the lower neutron shield, the upper neutron shield has shielding and structural functions. When the reactor is operating at power, it must attenuate the neutron flux to a level acceptable from the standpoint of induced radioactivity in the upper manifold room and, in conjunction with the gamma shield above, it must reduce the neutron flux to tolerance for operating personnel in the loading room above the reactor. When the reactor is shutdown, the upper neutron shield must attenuate the afterglow gammas sufficiently to permit servicing of the upper manifold system with the gamma shield removed. As a structural element of the reactor the upper neutron shield supports its own weight and the weight of the discharge manifold, coolant tubes, and fuel elements. It also guides the graphite columns of the core and reflector, as well as the iron slabs of the side thermal shield.

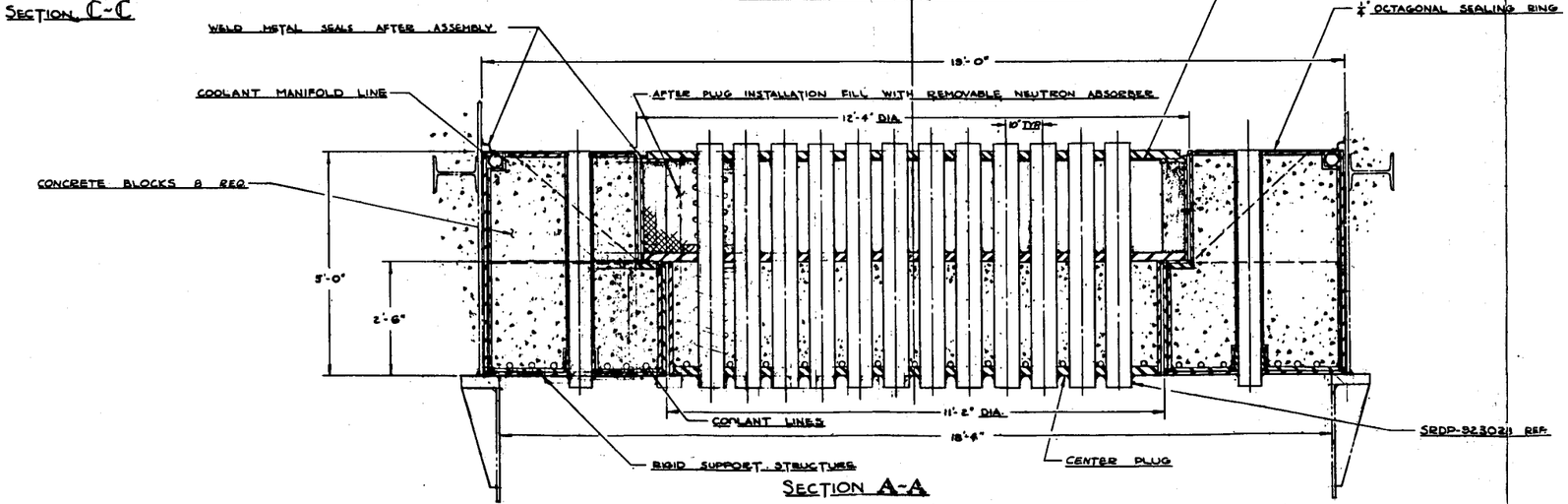
The upper neutron shield is a large circular plug made of steel and barytes concrete. It covers the reactor core, reflector, thermal shield, and thermal insulation (see Fig. 14). Its load is transmitted through the side neutron shield to the reactor foundation. The plug contains rows of sleeves for



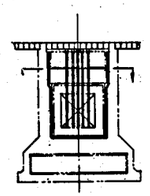
VIEW B B

SECTION C-C





SECTION A-A



REACTOR
0 5 10 20
SCALE FT

Fig. 14. Neutron Shield

the coolant tubes. Additional sleeves are provided interstitally for control and safety rods and instrument thimbles. The sleeves accommodate a shield and cooling jacket assembly which surrounds each coolant tube. Each assembly is removable from above and performs the same function as its counterpart in the lower neutron shield. Materials and construction details for these assemblies are also similar.

The required layers of thermal insulation above the top thermal shield are attached to the underside of the upper neutron shield. Short pipe-like columns extend through the thermal insulation. These engage the upper thermal shield plates which in turn are keyed to pairs of graphite columns. In this manner the graphite columns are positioned with respect to a cold structure (the upper neutron shield), and the lattice spacing remains the same at all times.

The upper neutron shield is the anchor point for the cooling tube assemblies (see Figs. 14 and 15). The heavy wall upper tube extensions are held fixed with respect to the top side of the neutron shield.

Heat generated in the steel and concrete and also the heat flowing into the shield beams through the surrounding insulation is removed by the organic liquid flowing in the cooling jackets surrounding the coolant tubes. The materials in the upper neutron shield are the same as those in the lower neutron shield since they perform similar functions.

9. Gamma Shield - A gamma shield is required above the upper manifold room to protect personnel from the sodium gamma activity. The neutron and gamma leakages from the core are stopped by the upper neutron shield leaving only the sodium decay gammas in the piping and upper manifold assembly for the gamma shield to attenuate. This shield is composed of beams 2 feet deep by 2 feet wide by 22 feet long. The beams are made of steel and lead. After installation the edges of the beams are welded together to form a gas tight envelope so that a helium atmosphere may be maintained in the upper manifold room. The gamma shield is cooled by an organic coolant (see Fig. 16). The gamma shield also supports the fuel elements by means of the fuel element hanger rod. The top of the fuel element support has provision for welding to the gamma shield should this be necessary to form a gas tight seal.

B. Cooling System

1. General - The main coolant removes not only the heat generated in the core but also most of the heat generated in the reflector and thermal shield. The heat generated in the neutron shield, on the other hand, is removed by a separate, low-temperature cooling system described below under "auxiliaries." This heat is wasted as is that lost through radiation and conduction from piping, vessels, etc.

The cooling system is made up of a primary sodium system and a secondary sodium system which are coupled through the intermediate heat exchangers. The sodium in the primary system becomes radioactive under neutron irradiation in the reactor core and must therefore be shielded. In the secondary system the sodium is not radioactive, and therefore, presents no radiological hazard in the steam generating system (or other load).

2. Primary Sodium System - The primary sodium system extracts the heat from the core, reflection; and thermal shield and delivers it to the secondary sodium system through the intermediate heat exchangers. It may be described as a single-flow (as opposed to split-flow) system in which the coolant makes a single pass through the reactor. The primary system is made up of four substantially identical loops, each of which is associated with one quadrant of the reactor (see Figs. 17 and 18). In each loop there are two pumps in parallel. The sodium flows from the pump discharge through a downcomer into the inlet manifold which is located below the reactor (see Figs. 2 and 6). From this point the sodium flows up through the coolant tubes located in the core and reflector graphite structure. On the top side of the upper neutron shield the sodium flows from the individual coolant tubes into the upper manifold (see Fig. 19). From the upper manifold the sodium flows through the intermediate heat exchanger to return to the pump suction. All four primary sodium loops are cross-connected to improve reliability under emergency conditions and to simplify connections to the sodium service system, the surge tanks, and the helium service system.

The primary sodium system (see Fig. 20) is designed to operate with a maximum sodium velocity of 10 ft/sec and a top operating temperature of 1050° F. The system is arranged for natural convection in case of failure of

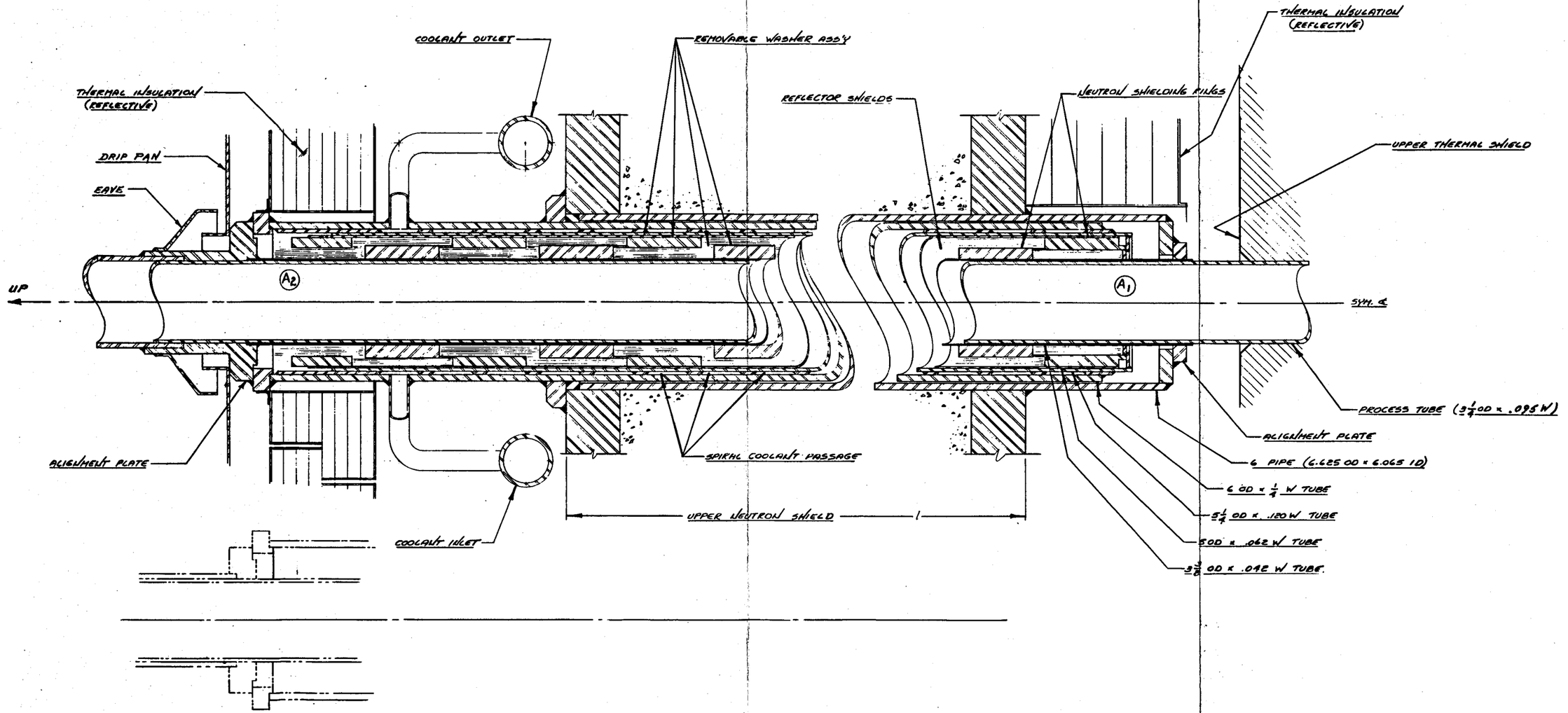


Fig. 15. Shield Alignment Washer Assembly Upper Removable

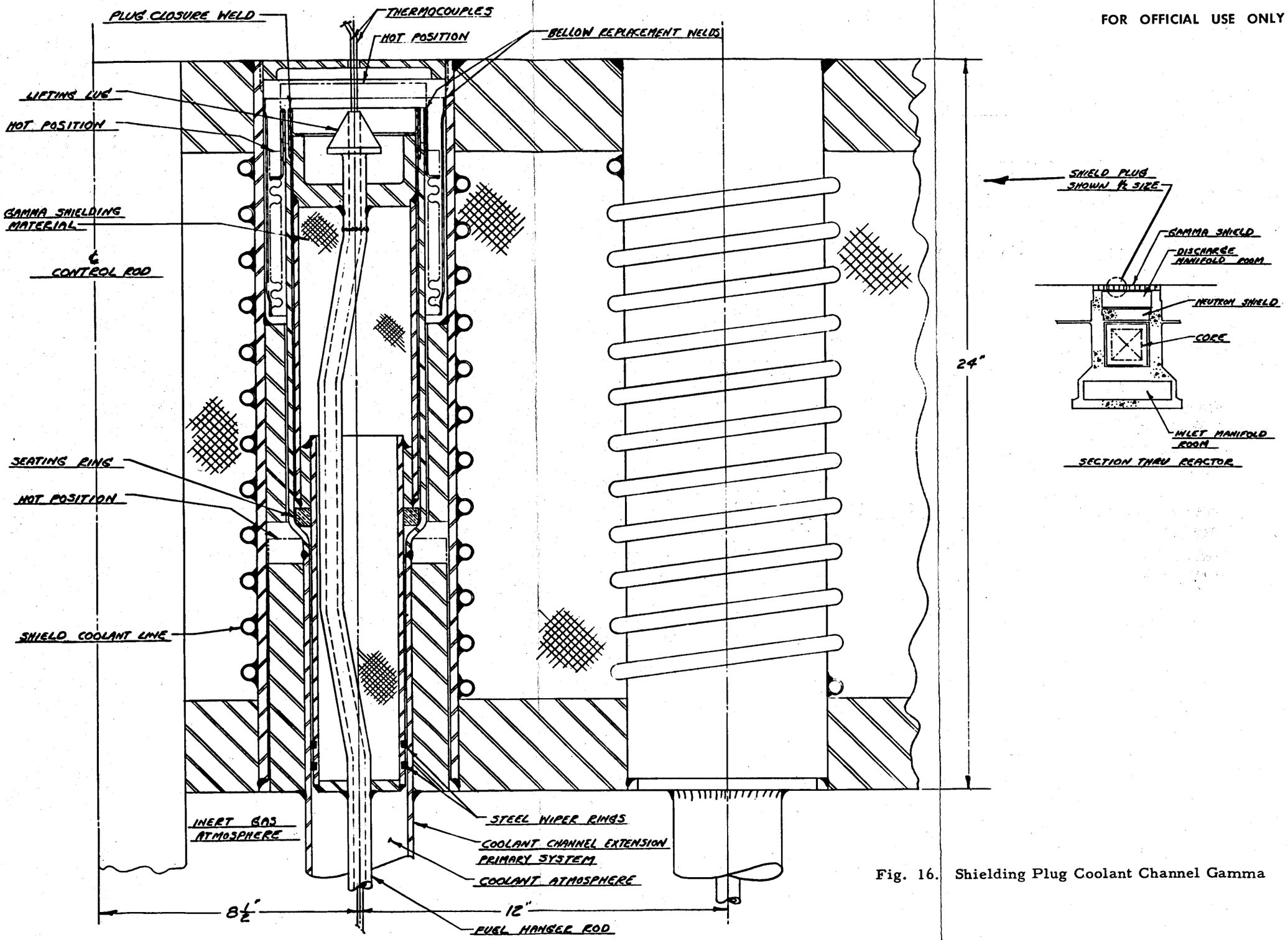


Fig. 16. Shielding Plug Coolant Channel Gamma

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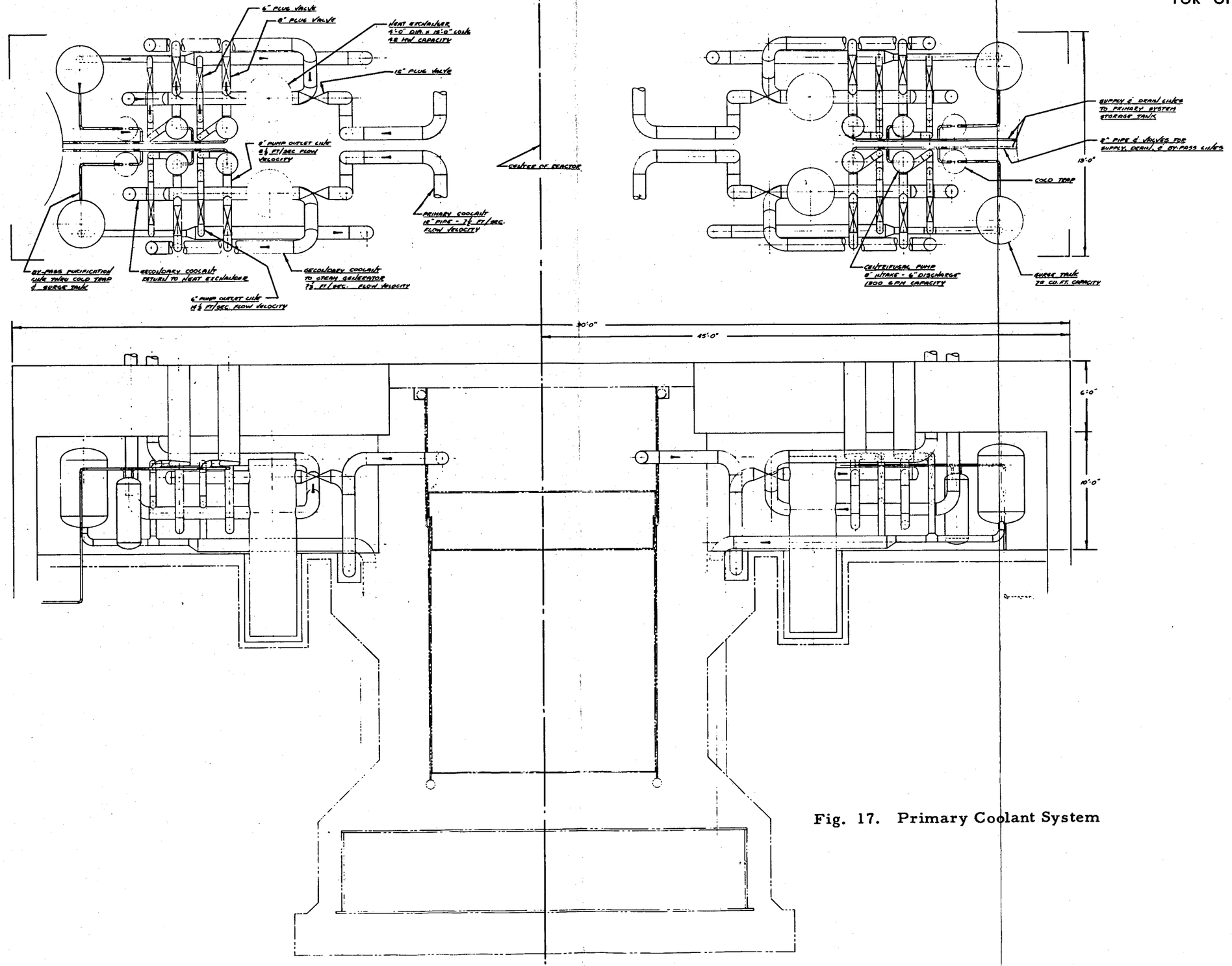


Fig. 17. Primary Coolant System

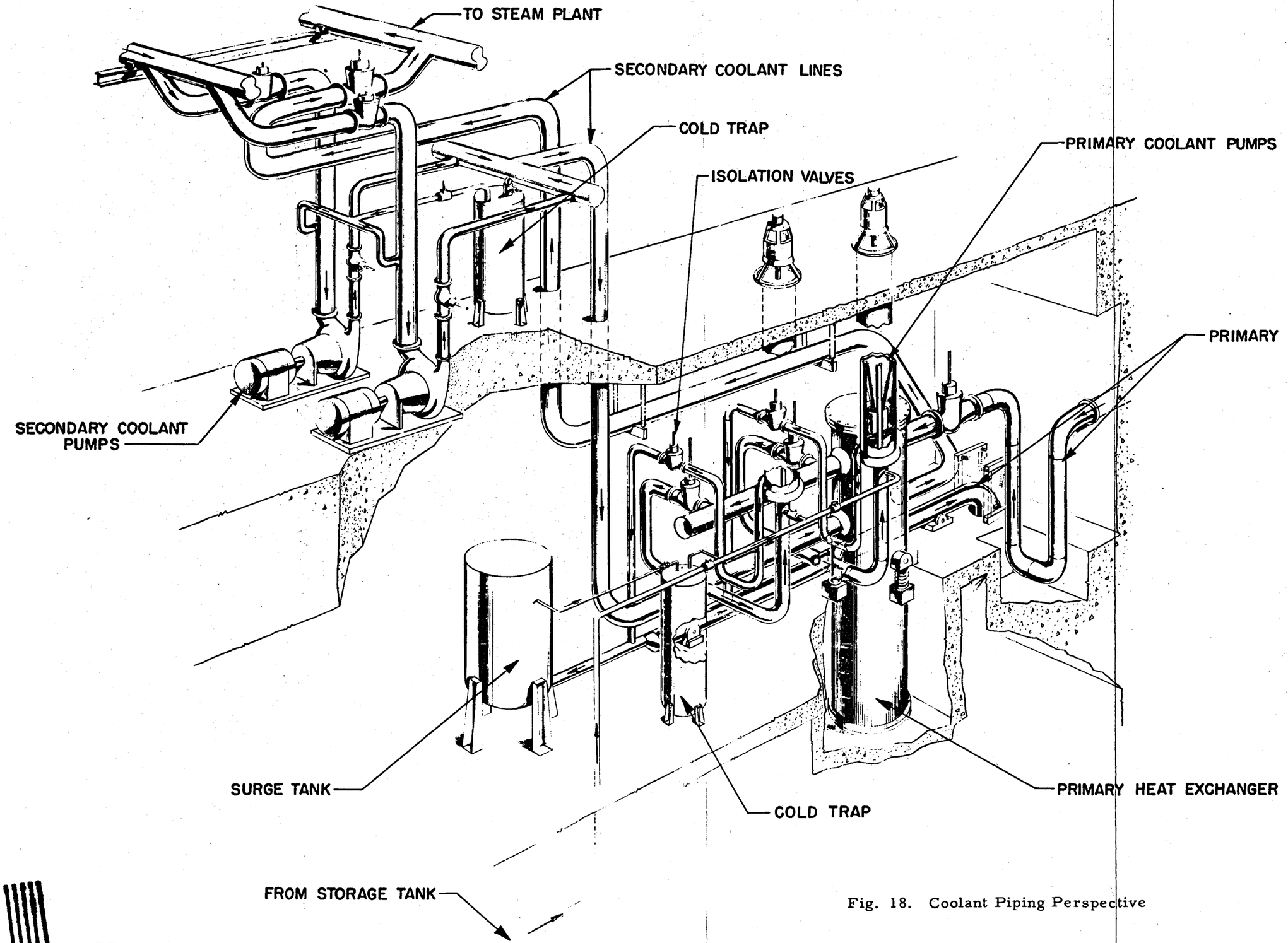
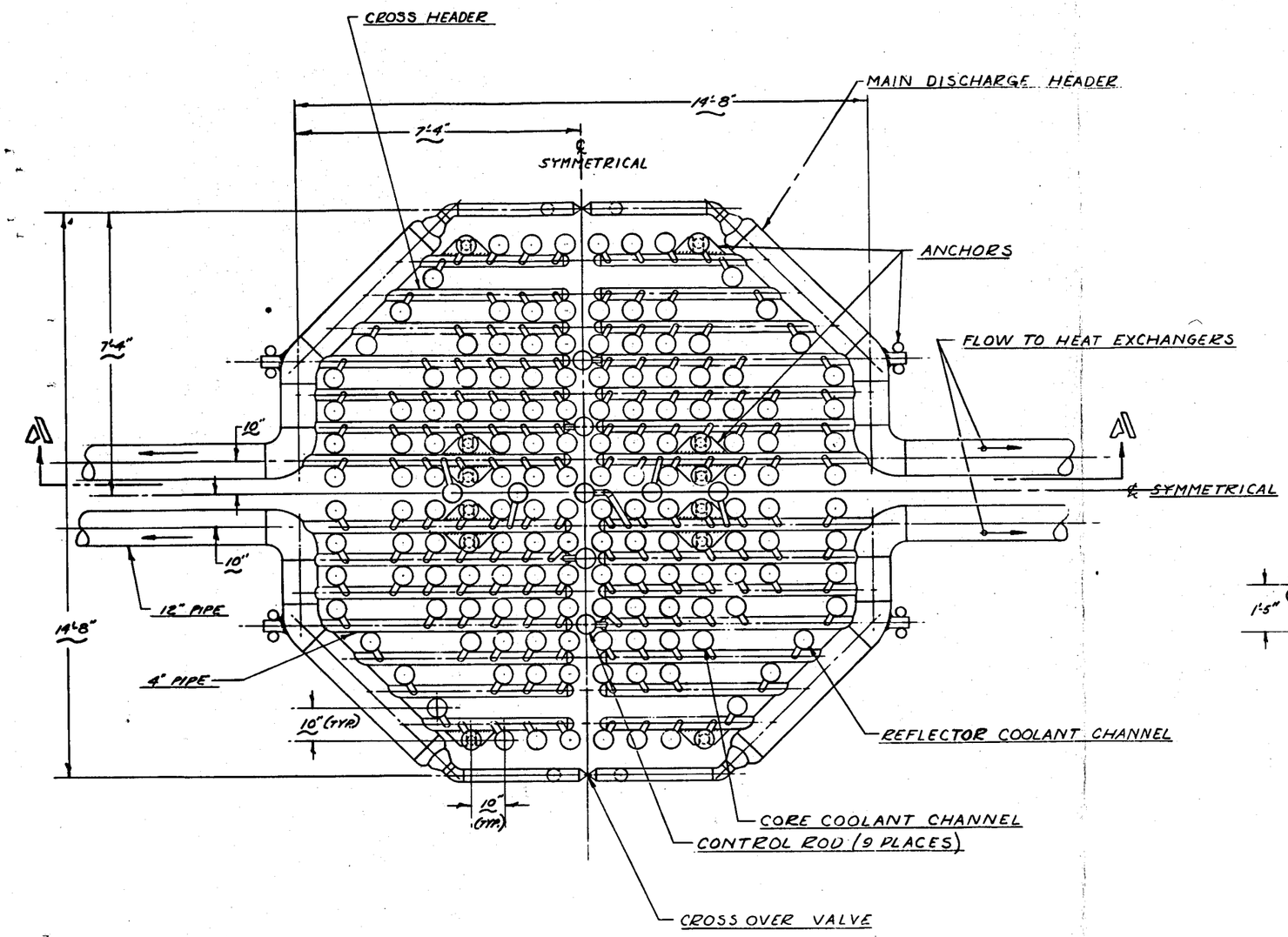
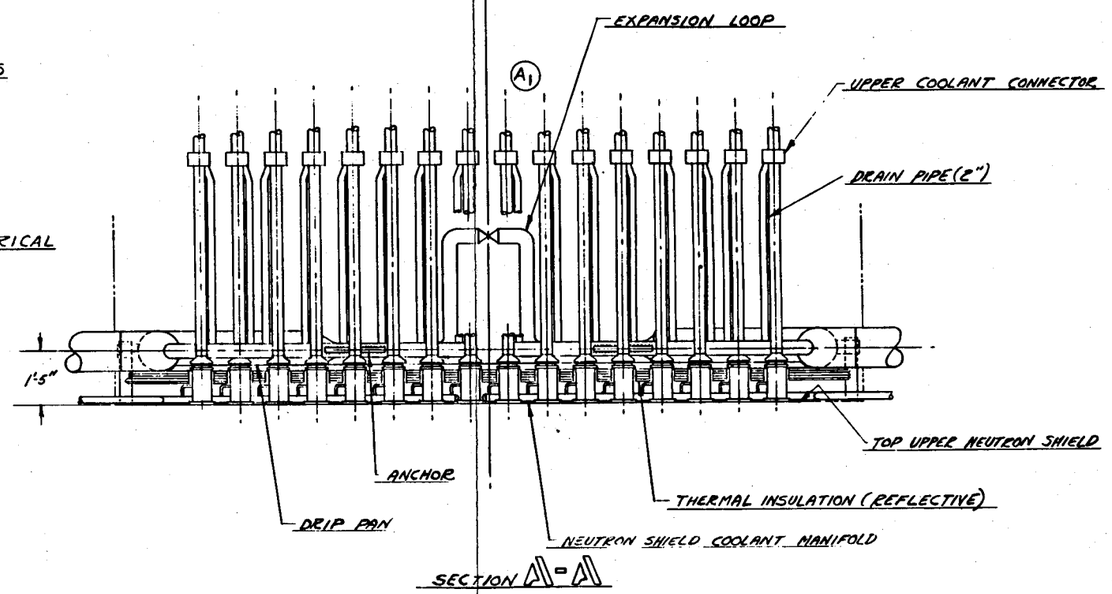
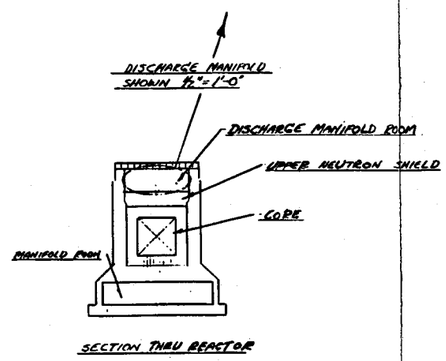


Fig. 18. Coolant Piping Perspective







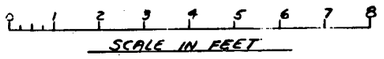


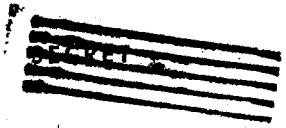
Fig. 19. Discharge Manifold

the pumps. It is possible to remove as much as 2 per cent of full power in this manner without reaching excessive fuel and coolant temperatures. All material in the sodium system is Type 304L stainless steel. Welding is used exclusively in fabrication in the shop and assembly in the field; there are no screwed, bolted or brazed joints. The sodium system likewise contains no bellows in contact with the coolant. Because of the radioactivity induced in the sodium, that portion of the primary sodium system not inside the reactor is surrounded by a gamma shield. The heat exchangers, primary piping, and pumps are located in the heat exchanger room which is adjacent to the upper manifold room (see Fig. 1). This room is covered by 6-foot deep removable concrete shielding beams.

3. Secondary Sodium System - The secondary sodium system receives its heat from the primary sodium through the intermediate heat exchangers. Unlike the primary sodium, the secondary sodium will not be radioactive. This, basically, is the only reason for having a secondary, or intermediate, sodium loop -- so there will be no danger of a chemical reaction between radioactive sodium and water.

While there are four parallel primary loops, each serving a quandrant of the reactor, there are only two parallel loops in the secondary. The secondary sodium operates between 420° F and 910° F with a flow rate of 1,910,000 lbs/hr in each loop. The heat is given up to steam in the superheaters, evaporators, and economizers (see Appendix A).

In the secondary system four pumps in parallel supply the coolant for one-half of the reactor (two quadrants) (see Fig. 21). Check valves and block valves make it possible to operate any or all of the pumps as desired and to remove any pump from the system for maintenance without interrupting the operation of the remaining pumps. Each pump in the secondary system is a single-stage, volute type with split case and single-suction, overhung impeller. The pump has end suction and top discharge and is arranged with axis horizontal. The pump has a frozen sodium seal similar to that in the primary system pumps. The pump is coupled directly to a horizontal motor mounted on a pad with the pump. The flow head characteristics of this pump are similar to the primary system main coolant pump. The pumps are mounted above ground and are easily accessible for repair.

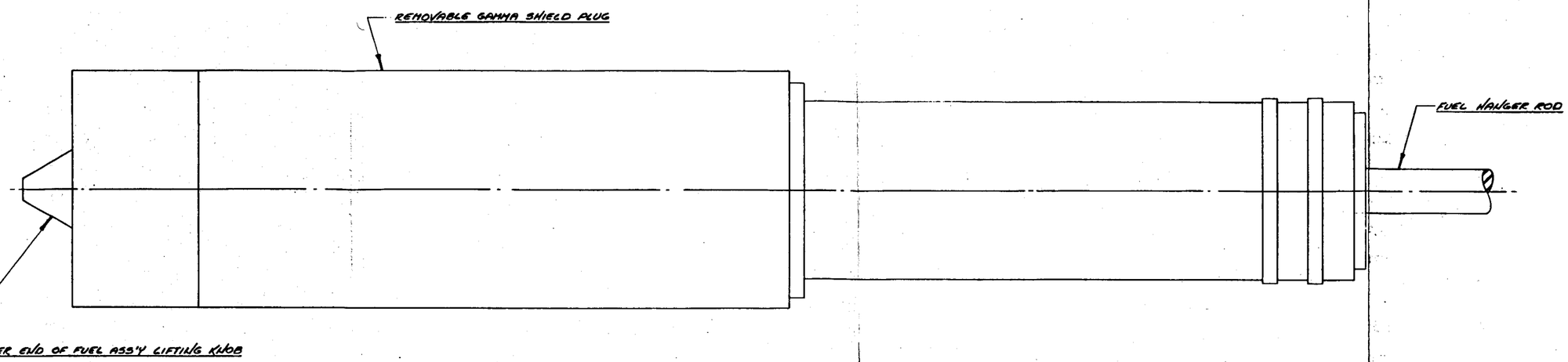


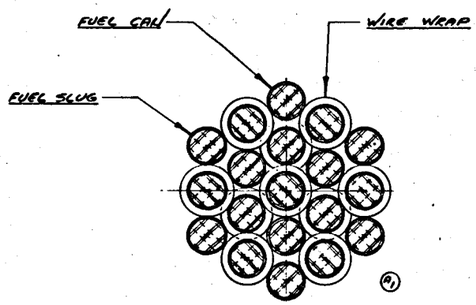
4. Pumps - The P-12 reactor has a total of 16 main coolant pumps, 8 in the primary system and 8 in the secondary system. In the primary system, as noted above, two pumps in parallel supply the coolant to one quadrant of the reactor. Both pumps are operated when the reactor is above 50 per cent rated power. With each pump there is associated a check valve and two block valves, so that at 50 per cent power or below, one pump may be shut down and, if necessary, taken out of the system for maintenance (see Figs. 22 and 23).

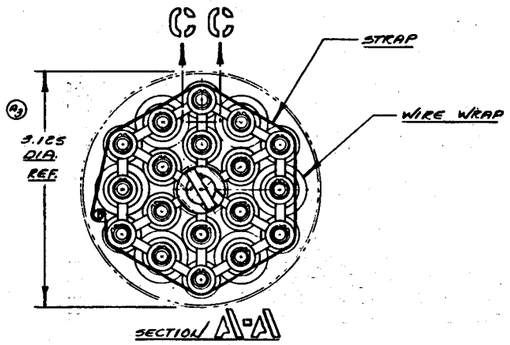
Each of the main coolant pumps in the primary system is a single-stage, volute type with split case and single-suction, overhung impeller. The pump has end suction and top discharge and is arranged with axis vertical. The pump includes a frozen sodium seal around the drive shaft between the impeller and the lower bearing. The frozen sodium annulus seals the sodium system from the atmosphere. Each pump is coupled directly to a vertical motor mounted outside the gamma shield. The pump is rated at 1450 gpm at a speed of 1600 rpm. Under these conditions the pump will develop a differential heat of 50 psi (170 ft at 0.84 specific gravity).

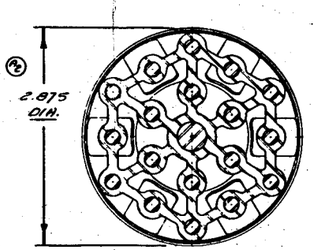
5. Piping - The piping system has been so designed that all dimensional changes resulting from temperature variations may be taken up by expansion without developing any undue stresses in the system. The primary system is anchored at the suctions and discharges of the pumps, at points near the inlet and outlet nozzles of the intermediate heat exchangers, and at the center of each discharge manifold header (see Fig. 18). In addition, each individual coolant tube is anchored to the top side of the upper neutron shield. At the lower neutron shield the coolant tubes are guided so as to permit longitudinal expansion only. To insure that the coolant tubes are never subjected to compressive stresses, they are loaded in tension by means of bellows. The bellows are not in the sodium circuit (see Fig. 6). The bellows also act as a gastight seal around the coolant tubes between the reactor cavity and the inlet manifold room.

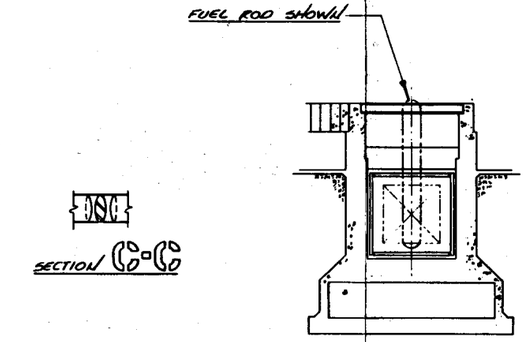
The weight of the inlet manifolds is distributed between pivot supports at the midpoints of the down-comers and rolling supports at the far ends of the inlet headers. The weight of the discharge manifolds and connecting piping is carried by the anchor points and intermediate pipe hangers.

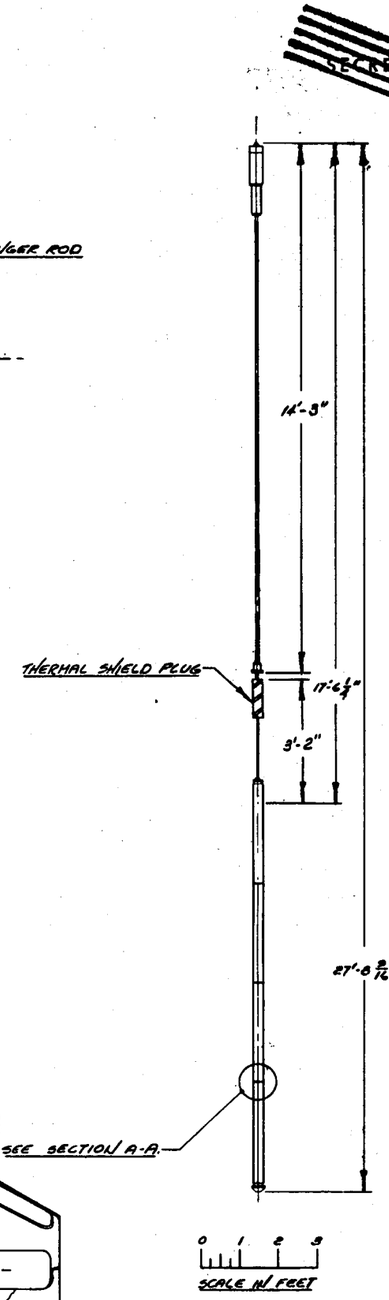












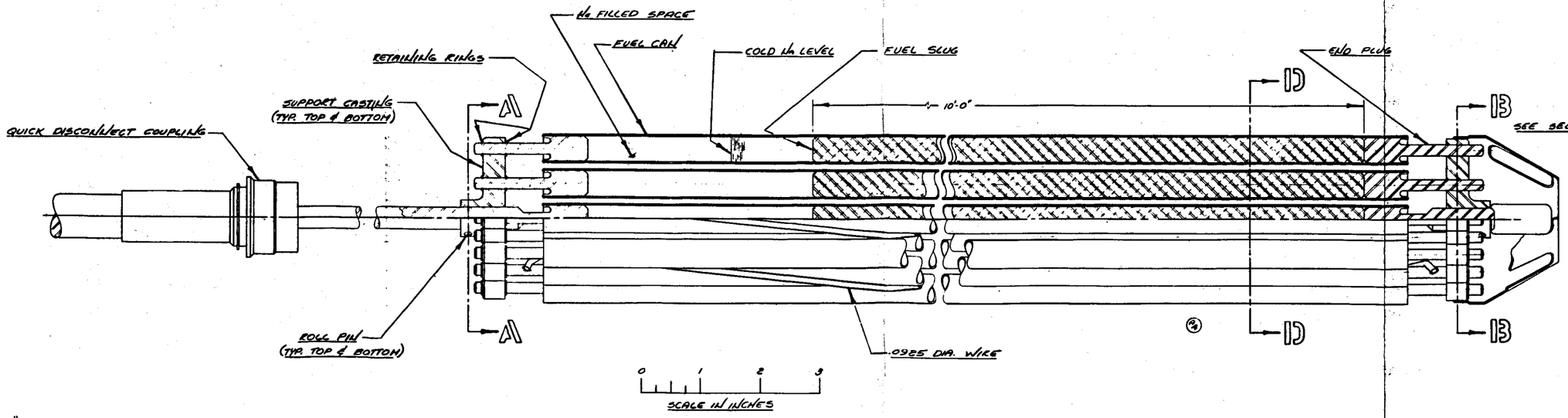


Fig. 21. Fuel Cluster 19 Rod

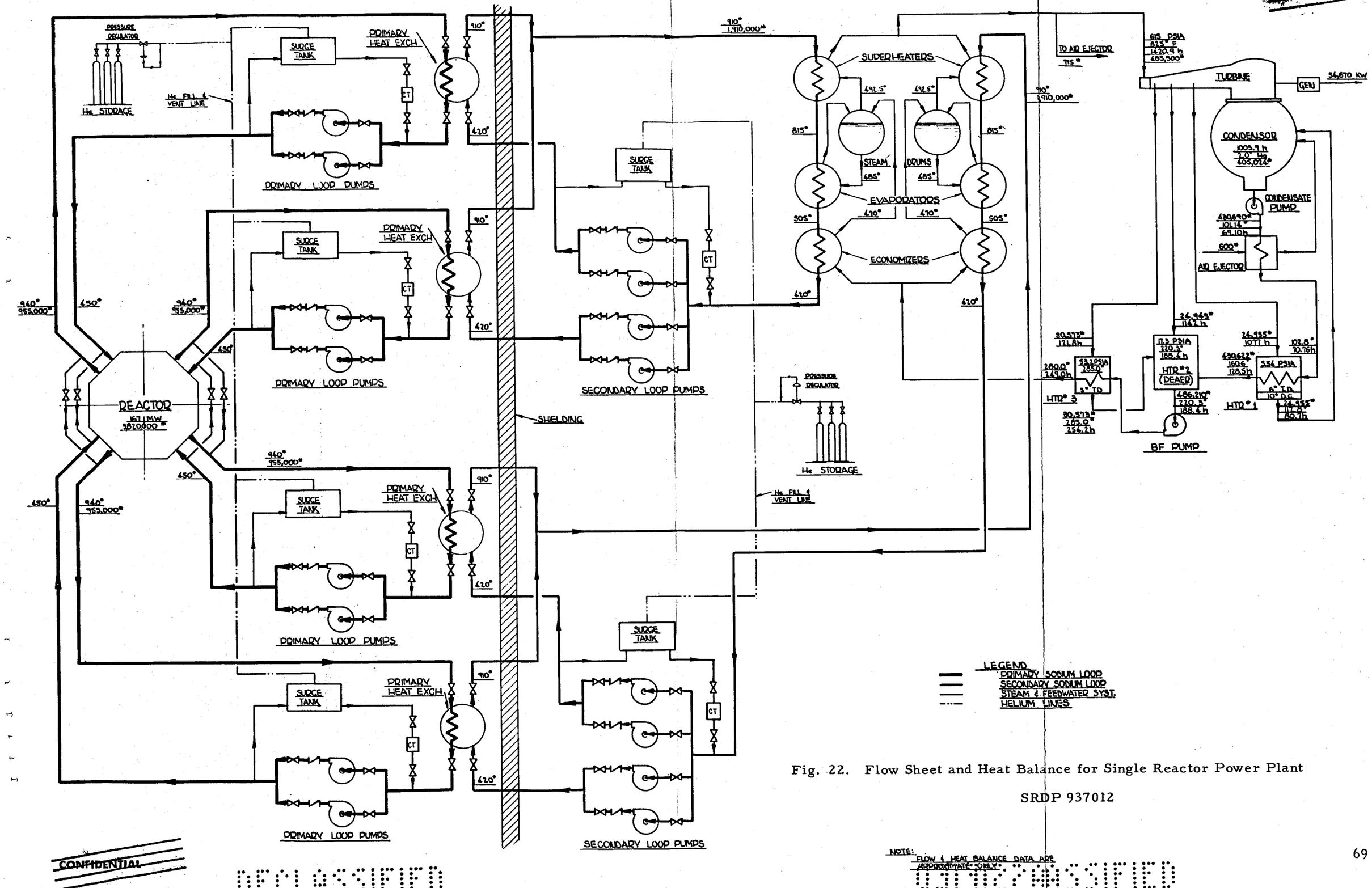


Fig. 22. Flow Sheet and Heat Balance for Single Reactor Power Plant

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NOTE: FLOW & HEAT BALANCE DATA ARE APPROXIMATE ONLY.

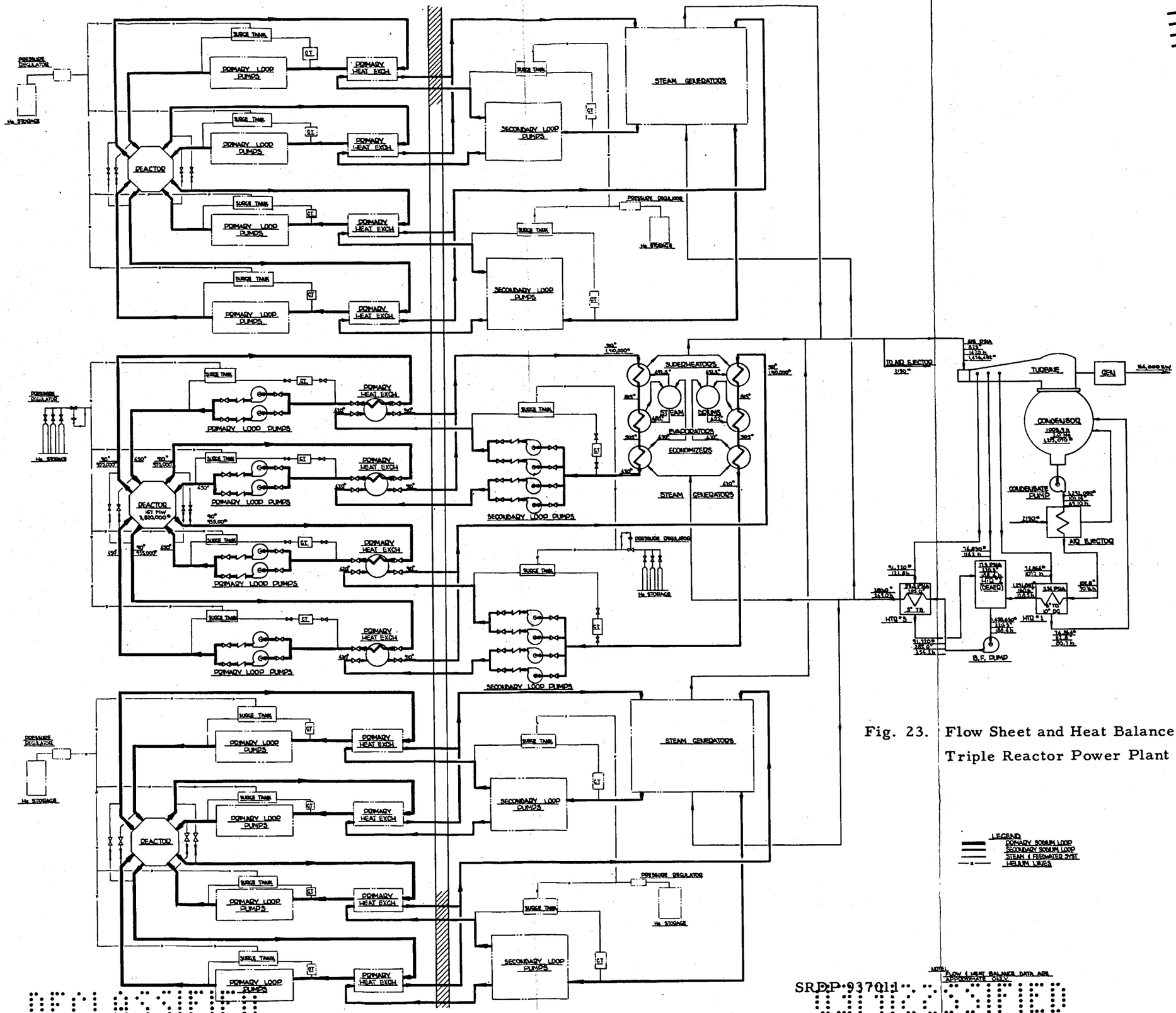


Fig. 23. Flow Sheet and Heat Balance for Triple Reactor Power Plant

The coolant tubes are described in detail above (see Section II-A-3, "Coolant Tubes"). A typical coolant tube assembly is shown in Fig. 20. This figure also illustrates the arrangement for terminating the assembly in the inlet and discharge manifolds. The outstanding feature of the arrangement shown is easy accessibility for removal and replacement of the individual coolant tubes. Another distinguishing characteristic of the manifold layout is the absence of bellows or other special devices for taking up dimensional variations due to changes in temperature.

C. Control and Instrumentation

1. Reactor Control

a. General - The basic method for establishing and controlling the neutron flux level in this reactor is by positioning a set of control and safety rods. These rods are constructed of a highly neutron-absorbing material (boron-steel). They achieve their control action by bringing about a balance between the number of neutrons being produced in the reactor and the number of neutrons being lost. When the control rods are in a position in which this balance is exactly maintained, the neutron flux level or power level is constant in time. An increase or decrease in this power can be obtained by withdrawing or inserting rods until the desired level is reached, then returning the rods to the equilibrium position for steady operation at the new level.

The entire system of rods consists of two groups designated as (1) safety rods, and (2) control rods. The safety rods are mechanically withdrawn and fall into the reactor by gravity (see Fig. 24). The control rods are mechanically driven in and out. All the safety rods will be fully withdrawn during normal operation and will be rapidly released whenever signals indicating a too fast period, an undesirably high neutron flux level, or excessive coolant temperature are received. Half the safety rods will be designated as a "safeguard bank" which will be withdrawn from the reactor immediately after shutdown.

The rod operating sequence will then be as follows: When the reactor is operating normally at power, all safety rods will be out and the control rods will be out as much as required. When a scram signal is received, all safety rods immediately are released and fall into the reactor. The control rods are immediately driven to their "in" position. When all the control rods are fully

inserted, half of the safety rods are withdrawn ready to fall into the reactor should any accident occur (i. e., inadvertent withdrawal of too many of the other safety and/or control rods). When it is desired to start the reactor again, the other half of the safety rods are withdrawn (either as a group or singly), and then the control rods are withdrawn as required to achieve criticality.

The control rods will be withdrawn to some intermediate position during normal operation and will perform both the shim and regulating functions. There are nine control rods, eight of which are designated shim rods and one of which is designated the regulating rod. The shim function refers to coarse adjustments of rod position to compensate for fairly large, slow reactivity changes due to such factors as moderator temperature, fuel temperature, fuel burnup, and poison buildup. The regulating function refers to small, rapidly-made, changes of rod position to counteract the effects of small reactivity changes which may constantly occur for a variety of reasons within the reactor. The single regulating rod is provided with a servomotor to drive it in this fashion. Normal changes in power level of the reactor will be made manually. The servomotor will be de-energized while the operator manipulates the shim rods to bring the reactor to the desired power level, after which the servomotor may be switched back in to resume its regulating function.

b. Design Features - A thimble-type of control rod, as shown in Fig. 25, will be used on this reactor. The neutron absorbing material consists of cylindrical rings of boron steel assembled on a pull-tube. This rod moves vertically in a thimble, and is driven by a ball-nut and lead screw arrangement. The rotation of the lead screw is translated to the linear motion of the boron-steel ring assembly through the action of the ball-nut. The drive motor is removable from the top, as it drives the lead screw through a simple clutch.

The thimble is placed in the core of the reactor in a sodium stream for removal of the heat generated in the neutron absorbent material. It constitutes a positive seal which excludes sodium vapor from the drive mechanism and permits easy removal of the control rod motor and rod itself.

c. Automatic Reactor Level Control Loop - A single regulating rod will be furnished with a servo control capable of positioning this rod to maintain a desired neutron flux level, or temperature drop across the reactor. The

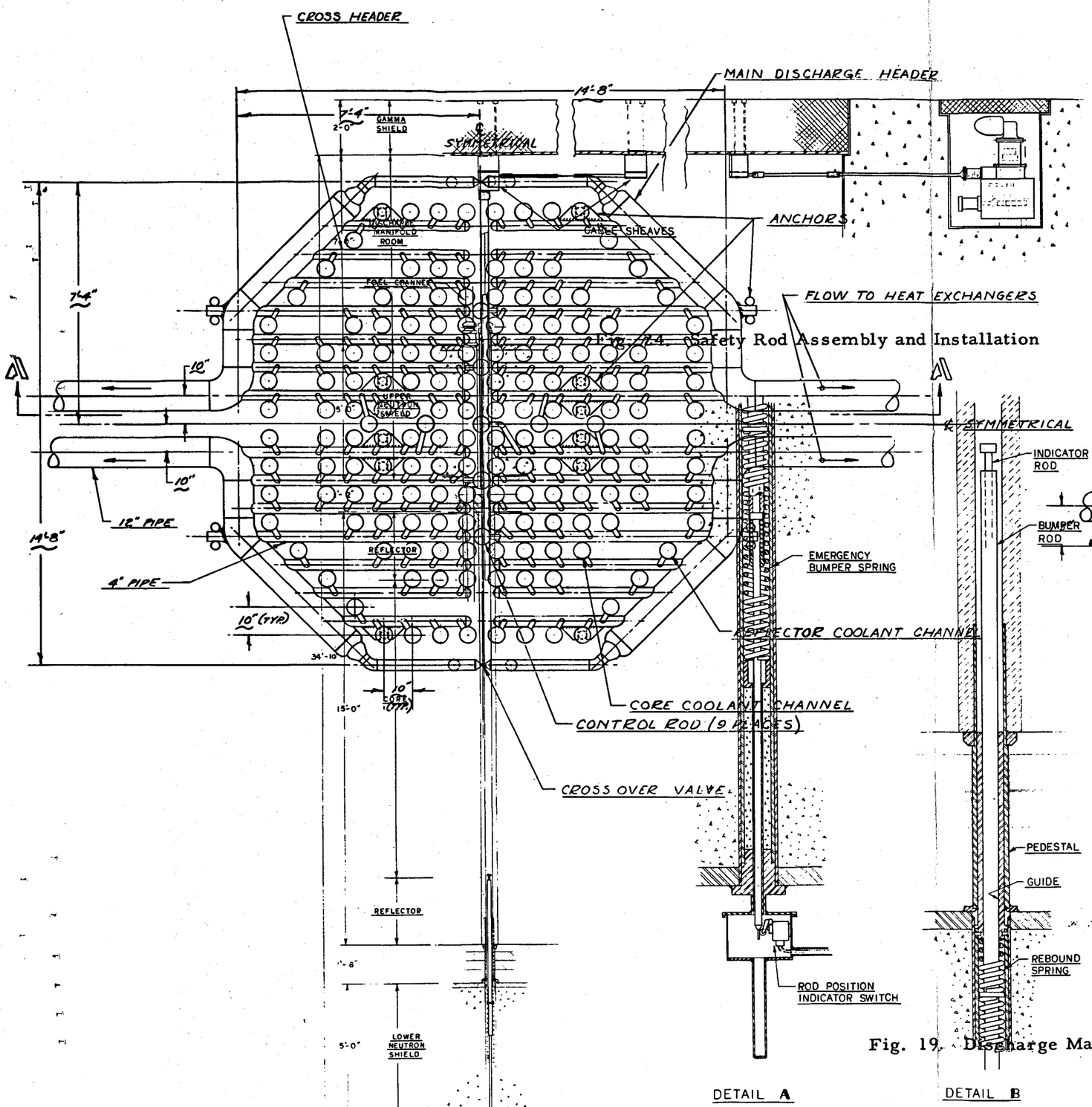


Fig. 18. Safety Rod Assembly and Installation

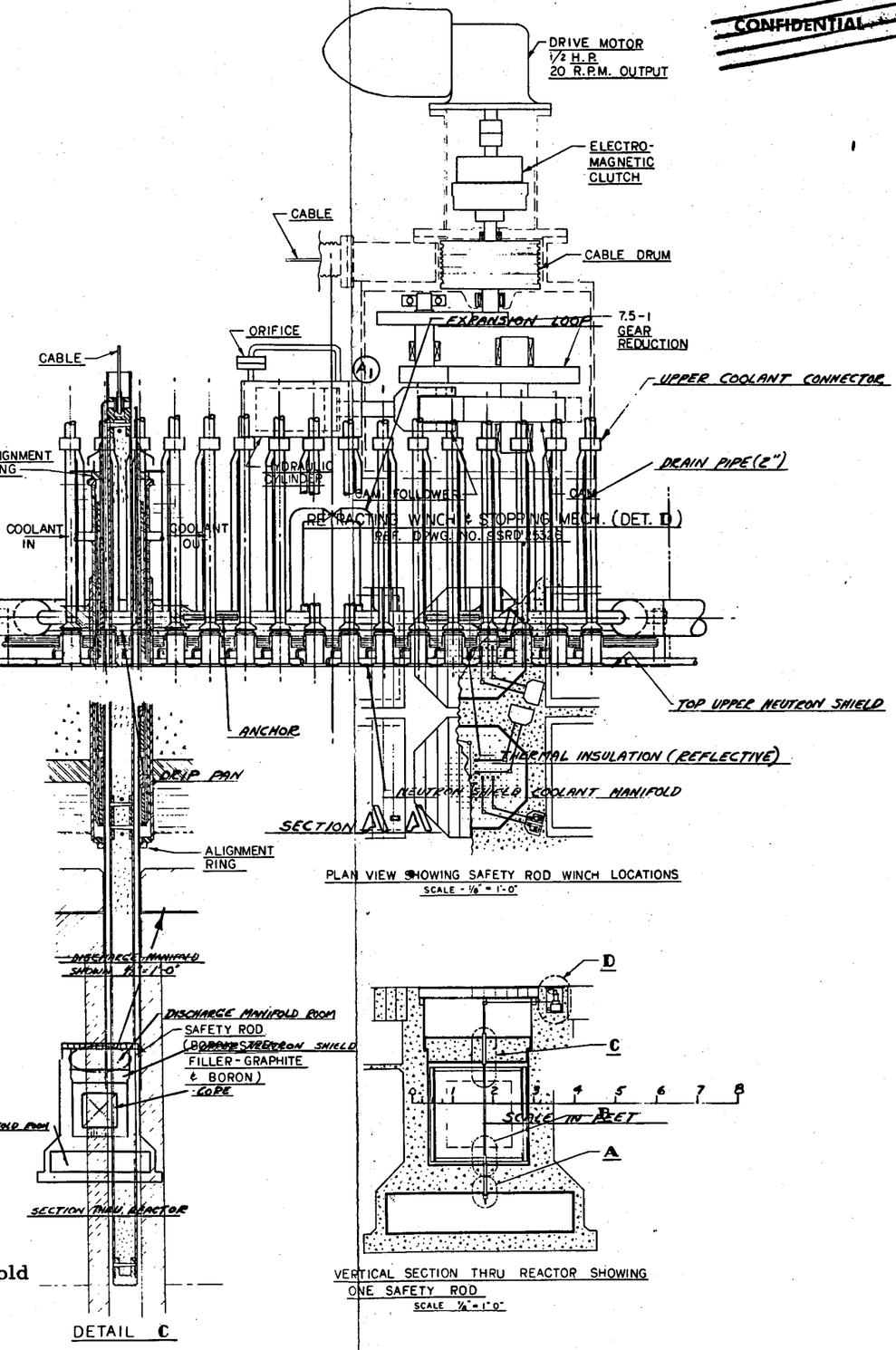


Fig. 19. Discharge Manifold

0 1 2 3 4 5 6 7 8 9 10 11 12
SCALE - QUARTER SIZE

input signals to this servo will depend on the desired mode of operation of the reactor.

2. Instrumentation

a. Reactor - Instruments will be provided to monitor the reactor neutron flux during startup and operation. Three types of chambers will be used to cover the wide range of neutron flux involved in going from shutdown condition to full operating power. These are fission chambers, gamma-compensated neutron ion chambers, and high-sensitivity neutron ion chambers. This selection has been made so as to provide continuous information over the range of approximately 10 decades which is easily attainable with a neutron source.

The fission chambers' pulses will be amplified and counted directly by a scaler, or be fed to a counting rate meter. This arrangement will provide neutron flux indication over a range of 4 decades. The location of these 4 decades of range with respect to the power level may be altered by remotely positioning the fission chambers. The gamma-compensated ion chambers will each cover a range of approximately 6 decades of neutron flux. They will be so arranged as to overlap each other and also overlap the coverage provided by the fission chambers. The signal from this type of chamber will be passed through a logarithmic amplifier and differentiated to obtain an output proportional to reactor period for control purposes. The amplifier output will also be displayed on a recording instrument with a 6 decade strip chart. The output from the neutron sensitive ion chambers, of the parallel circular plate type, will be used primarily for flux monitoring at the power level from about 1 decade below to 1/2 decade above the full power level. By the use of scale changing resistors in the amplifier input circuit, this range may be made to fall anywhere in about 7 decades of range from the power level downward. This feature is particularly useful in the initial startup of the reactor.

Because of the high temperature inside the 8-inch thick thermal shield of iron, it is desirable to locate the various chambers outside of this shield. Tapered slots in the outer face of the thermal shield permit the chambers to be exposed to a minimum thickness of 4 inches of shield. The fission chambers only must be located inside the thermal shield in the graphite reflector to provide the required sensitivity. The temperature in the reflector is about

800° F, and this requires the use of a chamber designed to operate at such an elevated temperature.

b. Cooling System - Instrumentation will be provided in each of the sodium circuits to measure the flow rate and temperature rise, so that the thermal power of the reactor may be determined. This information also enables the operator to establish the proper temperatures throughout the system to generate steam at the desired pressure and temperature, where the reactor is used as the heat source for an electrical generating station. In addition, measurements of pressure at different points in each sodium circuit will be made, as well as sodium level in the reactor and in the secondary system surge tanks. The outlet temperature of the sodium in each coolant channel will also be measured.

Temperature measurements will be made with chromel-alumel thermocouples inserted in suitable protecting wells in each sodium stream. Sodium flow will be measured with electromagnetic flow meters. Pressure measurements in sodium lines will be made with balances or null-force type diaphragm gauges. Sodium level will be measured by resistance type indicators, using the liquid sodium as the shorting bar of a resistance probe.

D. Auxiliary Systems

1. Helium System

a. General - Helium is to be used wherever an inert atmosphere is necessary to prevent oxidation of either graphite or sodium. Specifically, helium will be used in the graphite core, in the control rod thimbles, in the compartments containing the primary sodium system components, over free surfaces of sodium, and in the fuel changing system. No special building is required for the helium supply system.

b. Storage System - Helium will be stored primarily in large mobile high pressure storage tanks mounted on truck trailers. The high pressure units will be filled at a Bureau of Mines Helium Plant, transported to the reactor site, and then tied in to the reactor helium system. By using the high pressure storage tanks as the shipping containers, there is no need for a helium compressor at the reactor to transfer the gas from the shipping containers to the reactor storage system. Flow from the high pressure storage units will be to

low pressure storage tanks and will be governed by pressure reducing valves. Helium required by the various parts of the reactor will be supplied by the low pressure storage system.

c. Helium Purification - Extensive purification of the helium should not be necessary since the gas as received from the Bureau of Mines has a very high purity, and little contamination should result from transferring the helium to the low pressure storage system or from leakage in the reactor system. As precautionary measures, however, all helium going from the low pressure storage system to the graphite core will pass over a gas purifier of uranium or magnesium shavings to remove traces of oxygen. Also, all helium which is to blanket a free sodium surface will pass through a NaK bubbler.

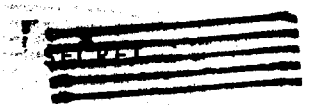
d. Helium Circulation - The helium systems for the graphite, primary sodium system compartments and free sodium surfaces will be non-circulating. Gas will be supplied from the low pressure storage tanks to maintain a pressure slightly higher than atmospheric and for purging as may be required. A purge line for each separate system will lead to the exhaust stack. These systems will all require pressure reducing valves to regulate flow, safety valves, flow meters, pressure indicators and gas samplers for occasional monitoring of the purity of the helium.

The helium system for the fuel changing system will not only provide an inert atmosphere, but can also be used to cool the spent fuel rods in the coffin if this should prove necessary. This system will therefore require a helium blower to circulate the gas and a helium-to-air heat exhcanger.

2. Sodium Handling and Purification System

a. General - When the reactor is first started, commerical sodium must be placed in the reactor system. During operation, a continuous purification system should be provided. Storage must also be provided for the sodium so the coolant can be drained from the system since major repairs on the cooling system would require removal of the sodium.

The design of the sodium handling and purification systems rests heavily on the experience gained by KAPL.



b. Sodium Charging System - The major components required to charge the reactor system with sodium are a melt tank and a transfer tank. Commercial sodium bricks cast in a dry mold will be charged manually in to the melt tank. Melting will proceed in an atmosphere of helium. The liquid sodium will then be sucked through filters to the transfer tank by means of a vacuum pump. Sodium is forced from the transfer tank through additional filters to the reactor system by displacing it with helium.

c. Sodium Storage - Separate storage systems will be provided for the sodium in the primary and secondary coolant loops. The tanks will be located at the lowest points in the system so that the sodium will drain from the coolant piping by gravity flow.

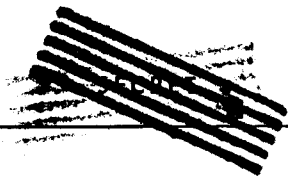
d. Purification of the Sodium - The sodium will be purified by filtering and cold trapping to prevent plugging and excessive corrosion. During operation continuous purification is obtained by the cold trap loop which parallels the main sodium circuit through the reactor. Before entering the cold trap, sodium is cooled in a heat exchanger or economizer, by giving up heat to sodium which has left the cold trap to return to the main sodium stream, The cold trap is essentially a holding chamber for cooled but liquid sodium, where impurities which become insoluble because of the decrease in temperature can condense on a mesh of steel wool.

3. Fuel Handling System

a. General - The fuel handling system will replace spent, irradiated, fuel elements with new, unirradiated fuel elements while the reactor is shut down. In addition, this system is designed to store, inspect, clean, and transport new and spent fuel elements. The design is intended to minimize reactor downtime, minimize the number of remote operations, and avoid operations necesssitating the removal of personnel from certain areas because of radiation hazards. The system must, of course, be consistent with the more basic objectives of low cost and safety.

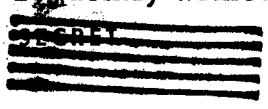
b. System components - The fuel handling system is comprised of the following major components:

1. New fuel storage area. New fuel elements will be stored in individual cells in an inert gas atmosphere.



2. Fuel and special cleaning cells. Fuel elements removed from the reactor will be cleaned of sodium by water spray in cleaning cells. Other cells will enable the cleaning of special equipment (such as control rod thimbles and process channels).
3. Fuel inspecting, decoupling and transfer tunnel. After the sodium has been cleaned from a fuel element which has been in the reactor, the fuel rods will be inspected and the hanger rod removed. Fuel elements will then be transferred to the storage pond.
4. Storage Pond. Spent fuel elements will normally be stored under water in the so-called storage pond before they are removed for chemical processing. Cells containing an organic liquid will be provided to store fuel elements with ruptured jackets.
5. Fuel Transfer Cask. While the reactor is shutdown the fuel transfer cask will be used to transfer new fuel elements to the reactor and to remove spent fuel elements from the reactor and transfer them to the cleaning cell. After reactor operation is resumed, the cask will transfer the fuel elements to the inspecting and decoupling tunnel. The design permits making and breaking gas-tight connections between the cask and the reactor, the new fuel storage cell or the cleaning cell.
6. Fuel Shipping Cask. Fuel elements will be transported from the storage area to chemical processing facilities in a fuel shipping cask.

c. Fuel Changing Procedure - New fuel elements will be placed in the storage cells, and helium atmosphere provided in the storage cells, transfer cask and the fuel cleaning cells. The reactor will then be shut down. A new fuel element will be loaded in the cask and moved to the reactor loading face. There a spent fuel element will be removed from the reactor and replaced with the new one. Then the spent fuel element will be moved to one of the fuel cleaning cells. This procedure will be repeated as many times as necessary to replace used fuel. The series of operations required to replace one fuel element is estimated to take slightly less than 1 hour. After the fuel is changed, the reactor can be started up. Cleaning, inspecting, transfer to the storage area and finally shipping to the chemical processing area can be made subsequently without affecting the reactor downtime.



4. Reactor Heating System - A problem faced by all sodium cooled reactors is how to get the reactor components heated sufficiently so that the sodium will not freeze during the initial filling of the system. Once the sodium is circulating it can be kept from freezing by heaters external to the core. All the external piping and heat exchangers will be wrapped in the usual manner with electric resistance type heaters, but it is not practical to wrap the coolant channels in the core in this manner. Two means have been considered as possible solutions to this problem: temporary electric heaters in the fuel element channels, and NaK or some organic fluid circulated in coils wrapped around the iron thermal shield.*

The thermal capacity of the core and thermal shield is quite high. It is expected that several days will be required to heat the unit to 250 or 300° F, but the time can be allowed since the heating is only required at initial startup and after a long shutdown for major repairs. The high heat capacity also means that the core will require a long time to cool making it unlikely that the sodium will freeze during a short shutdown.

E. Design Limitations

1. Power Rating of the Reactor - The basic limitation to the rate of heat removal that can be attained in a reactor using uranium metal as fuel is the maximum permissible fuel temperature. The maximum permissible fuel temperature in the case of uranium metal is generally taken as the α to β phase transition temperature, in the absence of sufficient experimental evidence of dimensional stability when the uranium alloy is operated with its interior portion in the β phase. The α to β phase transition temperature for 2 weight per cent zirconium uranium alloy is about 1220° F. The power rating of the present reactor is for a maximum fuel temperature of 1200° F. Due to a number of conservative assumptions used in the calculations, the actual nominal maximum fuel temperature will be somewhat less than 1200° F. These assumptions are:

*The method of utilizing heating coils wrapped around the thermal shield requires a permanent installation, but this gives it the advantage of being able to serve as an emergency cooling system should this be required. The installation cost will probably be more than for the electric heaters though no detailed analysis has been made to bear this out.

1. The rate of heat generation in the fuel rod is uniform in the radial direction. Actually, the rate of the heat generation increases with radial distance.
2. There is no heat flow in the axial direction within the fuel rods. Since the heat generation in the axial direction is a cosine distribution, there will be some axial heat flow from the point of maximum fuel temperature.
3. A fouling factor of 4 by 10^{-5} $(Btu/hr\ ft^2 \cdot F)^{-1}$ was assumed. No fouling is indicated by experimental data for sodium cooling a wetted surface. No deterioration in heat transfer coefficient as a result of fouling has been observed in one sodium loop in more than 2 1/2 years of operation (Ref. 9). For zero fouling factor, reactor power can be raised 5 per cent.
4. Allowance is made for a 3 per cent time variation in reactor power.

Other pertinent assumptions and conditions used in the heat transfer calculations are:

5. Ninety-four per cent of heat was assumed to be generated in the fuel and six per cent in the moderator.
6. The thermal conductivity of 2 weight per cent zirconium uranium alloy was assumed to be 15 Btu/hr ft °F; i. e., about 15 per cent less than for uranium. This is believed to be somewhat conservative on the basis of the thermal conductivity of 4 weight per cent columbium uranium alloy, a higher alloy, which is found to be about 15 per cent less than that of pure uranium (Ref. 10).
7. The heat transfer coefficient for sodium is given by the Martinelli-Lyon equation for flow in round pipes (Ref. 11).
8. Power generation in the reactor is a symmetrical chopped cosine distribution in the axial direction when control rods are out of the reactor core and a Bessel function distribution in the radial direction. The ratio of reflected to unreflected fuel rod length was taken as 0. 845 The ratio of reflected to unreflected core radius was taken as 0. 704. It was assumed that the effect of control rods, which are considered to be gang operated, when in nominal position in the core is to reduce the average to peak power ratio in the axial direction by

10 per cent for all fuel elements. Actually the reduction is not uniform, being greatest for fuel elements nearest the control rods and least for those farthest away. It is believed that the overall results based on uniform reduction will not differ appreciably from the overall results which would be obtained by consideration of the variation in reduction. The latter consideration is precluded at present by lack of quantitative information.

9. The difference in power between a fresh fuel element and a fuel element with a burnup of 15 per cent (the average in the reactor) was assumed to be 7 per cent.
10. The coolant flow is adjusted for maximum possible outlet temperature, but not in excess of 1050° F, in all coolant channels.
11. The coolant inlet temperature is 450° F.
12. Allowance is made for a 3 per cent variation (error in making orifice) in coolant flow rate in each coolant tube.
13. The temperature rise of the coolant through the reflector and thermal shield is from 450 to 800° F. The heat generated in the reflector and thermal shield was assumed to be 2 per cent of the heat generated in the core.

A summary of the cooling system data is given in Part B of the Table I Performance and Design Data. The coolant tube diameter has been somewhat oversized to permit the possible use of a higher power density fuel.

2. Selection of Fuel Element Configuration - While there are many possible ways of arranging a fixed amount of fuel material in a tubular coolant cell, the fuel shapes best suited, geometrically, for such a coolant channel are the rod and the tube. In the case of the rod shape, the given amount of fuel can be in one rod or in a number of smaller diameter rods distributed in the channel. Similarly, for the tube shape, a single tube or a number of concentric tubes can be used, the coolant flowing past both sides of all tubes.

The allowable power density for the fuel increases as the fuel material is divided up into a larger number of smaller diameter rods or thinner wall tubes. On the other hand, this results in an increase in cladding-to-fuel volume ratio which generally necessitates an increase in fuel enrichment. Furthermore,

increasing the power density decreases the size of a fixed output reactor which again necessitates an increase in enrichment due to the higher leakage. Fuel fabrication cost also increases with increase in subdivision. Hence, there is an optimum amount of subdivisi on of the fuel material within a coolant channel for any fixed output reactor. For a 500 thermal megawatt sodium-cooled, graphite-moderated reactor having zirconium-clad fuel rods, the number of fuel rods per coolant channel which results in minimum power cost has been found to be in the range of 7 to 19 (Ref. 9). The 19-rod fuel element is used in the present reactor. See Fig. 21.

Some investigation has been made of the hollow fuel element. Heat transfer studies (Ref. 12) show that the power density of a single tube fuel element will be somewhat greater than that of the 7-rod fuel element, but considerably less than that of the 19-rod fuel element, for the present ratio of fuel to cell volume. A drawback of the hollow fuel element is the added complication of balancing the inner and outer coolant flow rates. However, the single hollow fuel element may be cheaper to fabricate and assemble than the 7-rod or the 19-rod fuel element. Current lack of experience with hollow fuel elements dictates the use of rod type fuel elements in the present reactor (see Section III-E-4).

III. APPLICATION DATA

A. General

The 167 thermal megawatt P-12 reactor was designed primarily for the production of useful electric power. It is apparent, however, that it may also be used simply as a source of high temperature heat and as such might find wide application in industry.

The application mentioned above makes use of the nuclear fission energy only after it has been converted to thermal energy. Applications may be imagined where a portion of this energy is used in the form in which it was originally released. For example, the gamma rays which are absorbed in the shield material in the P-12 reactor might be found effective in inducing certain chemical reactions. A more obvious and, of course, well-proven use of a


nuclear reactor is as a neutron source. In the operating and cost analyses which follow it is assumed that the excess neutrons produced in the P-12 reactor are used to manufacture plutonium which is burned in the reactor as a fuel. When the value of the plutonium is greater, say as a weapons material than as a fuel, it may, of course, be extracted and sold. The plutonium then becomes a product along with electric power.

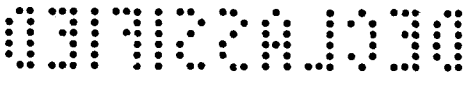
It is possible to alter the fuel in the P-12 reactor so that the excess neutrons are absorbed to produce artifically radioactive isotopes of commercial value. Once a use has been found for the fission fragments they may also become a useful and therefore valuable by-product.

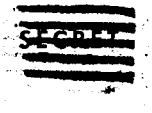
In the economic analyses which were made in support of the design of the P-12 reactor it was assumed that the reactor would be used with condensing turbines in a closed steam cycle and would be used to produce only electric power. Although many other arrangements would obviously be feasible, this seemed to have potentially the widest field of application and therefore would be of the greatest general interest.

A typical flow and heat balance diagram for a single P-12 reactor and associated steam-electric plant is shown in Fig. 22. This is a "unit" arrangement, one 50,000 kilowatt turbo-generator being associated with each P-12 reactor. Figure 23 is a similar diagram showing three P-12 reactors with a gross output of 500 megawatts associated with a single 150,000 kilowatt turbo-generator.

B. Site Considerations

The requirements for a site for this reactor and its auxiliary equipment are naturally affected by the use for which the reactor is intended. For use with a turbogenerator for the production of electrical power, the obvious requirements are (1) availability of sufficient water for make-up and cooling purposes, (2) proximity to the intended electrical load or an existing transmission line, (3) road or rail connections to the general vicinity, and (4) availability of housing for station personnel. And finally, some special considerations relevant to the location of a high power nuclear reactor must be satisfied. Past practice has been to define an exclusion area, whose radius is a function of the





thermal power of the reactor. * This practice is essentially a concession to the remote possibility of the release from the reactor of stored radioactive materials. There are, however, numerous features of this reactor system which so enhance its overall safety that the probability of its failure to contain its radioactive materials is vanishing small.

The specific features of this reactor system which contribute to its safety in operation are:

1. Core considerations

   a. A long neutron generation time resulting from the use of graphite as moderator in a thermal reactor.
   b. A negative temperature coefficient of reactivity.
   c. A low power density in fuel (because of the low enrichment) allowing for absorption of heat in case of a sudden rise in power level.
   d. A large heat capacity in the graphite moderator.

2. Shield Considerations - Attainment of an unusually affective radiation shield by locating the entire reactor core underground. This also makes the achievement of damage by sabotage more difficult.

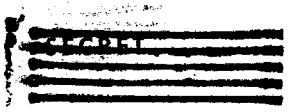
3. Coolant System Considerations

   a. Provision of a completely separate primary and secondary sodium coolant system. The primary sodium loop becomes radioactive in passing through the reactor and delivers its heat to a non-radioactive sodium loop in a shielded heat exchanger. Only the non-radioactive sodium loop emerges from an underground, concrete-shielded heat exchanger cell and enters the sodium-to-water heat exchangers.
   b. Location of sodium inlet manifold at low point of primary coolant system insures that sodium cannot drain away from this point.

*For this reactor, use of the conventional formula recommended by the ACRS yields an exclusion radius of

$0.01 \sqrt{16.7 \times 10^4} = 4.09$ miles for a single unit or

$0.01 \sqrt{50 \times 10^4 \text{ KW}} = 7.07$ miles for three units.



c. Provision of surge tanks in the primary coolant system to insure a reserve supply of sodium to make up any loss which might occur.

d. Adequate protection against coolant pump failure by provision of four cross-connected primary sodium loops, each of which includes two pumps operating in parallel. It is possible to operate or isolate these eight pumps in any combination.

e. Arrangement of primary coolant system to insure sodium flow by natural convection in case of failure of all primary coolant pumps (conceivable only in case of interruption of power to pumps). About 2 per cent of full power may be removed from the reactor in this manner, which is enough to avoid melting of fuel elements by afterglow heat.

f. Absence of water lines in the vicinity of the primary coolant system to make impossible a sodium-water reaction which could disperse radioactive sodium.

g. Design of auxiliary equipment for sodium withdrawal (or purification) with a rate of removal so low that accidental operation could not remove significant quantities in a time less than a few hours.

4. Instrumentation and Control System Considerations

a. Provision for both reactor period and flux level protection. Circuits are included to shut down automatically the reactor by releasing magnetically suspended safety rods in event of high flux level or short period.

b. Design of vital components on a fail-safe basis. This includes rod drives, safety ion chambers and their circuits, sodium flow and temperature measuring devices, etc.

c. Interlocking circuits to insure that the safety rods cannot be withdrawn unless the control rods are fully inserted and that the control rods cannot be withdrawn until the safety rods are fully withdrawn.

d. Presentation of essential information to the operator in concise form by means of graphic panel. Data displayed will include the control rod positions, flux data and flows and temperatures in the coolant system. An annunciator board will indicate the satisfactory condition of a large number of temperatures, pressures, etc.

C. Operation

1. General - The P-12 reactor is characterized by the following features:

a. Low enrichment uranium fuel
b. High fractional burnup of U^{235}
c. Moderately high heat generation per unit volume of reactor
d. Moderate conversion ratio
e. Low pressure, high temperature coolant (liquid metal)
f. Low pressure, high temperature moderator (graphite)
g. Periodic refueling by zones

The problems expected in the operation of a P-12 type reactor are numerous; several of the more obvious ones are discussed in the following paragraphs.

2. Xenon Effects - The production of xenon and other high cross section fission products causes a reduction in multiplication factor of 0.042 (3.7 per cent $\frac{\Delta k}{k}$) at full power after reaching equilibrium. Since this calculation is based on an effective flux of 4.6 by 10^{13}, the Xe^{135} and Sm^{149} poison the pile with 90 per cent of their maximum equilibrium effectiveness. The poison transient which occurs after shutdown from long-time operation at full power passes through a maximum about 9 hours after shutdown, and reaches the steady state value again after 29 hours of downtime. The peak transient poisoning amounts to an additional 3 per cent reduction in multiplication factor. This transient is shown in Fig. 26 for a shutdown after long-time operation at full power. It is not intended that the reactor be capable of start-up at all times during this transient period. Initially, it is planned to always have at least 1 per cent Δk available in the control rods. Generally, there will be more Δk available, but this depends upon the relative time or position in the reloading cycle as discussed later. One per cent Δk will allow start-up any time within the first hour following a scram from full power. The dead time will range around 20 to 24 hours. (It is current practice at Hanford to hold only a half-hour's worth of reactivity in the control rods.) After considerable operating experience is obtained with this reactor, it is hoped to be able to reduce the excess k to one-half per cent.

The reactor will be shut down to change fuel. The time involved to reload a zone of the reactor which may contain from 12 to 40 fuel elements, has been

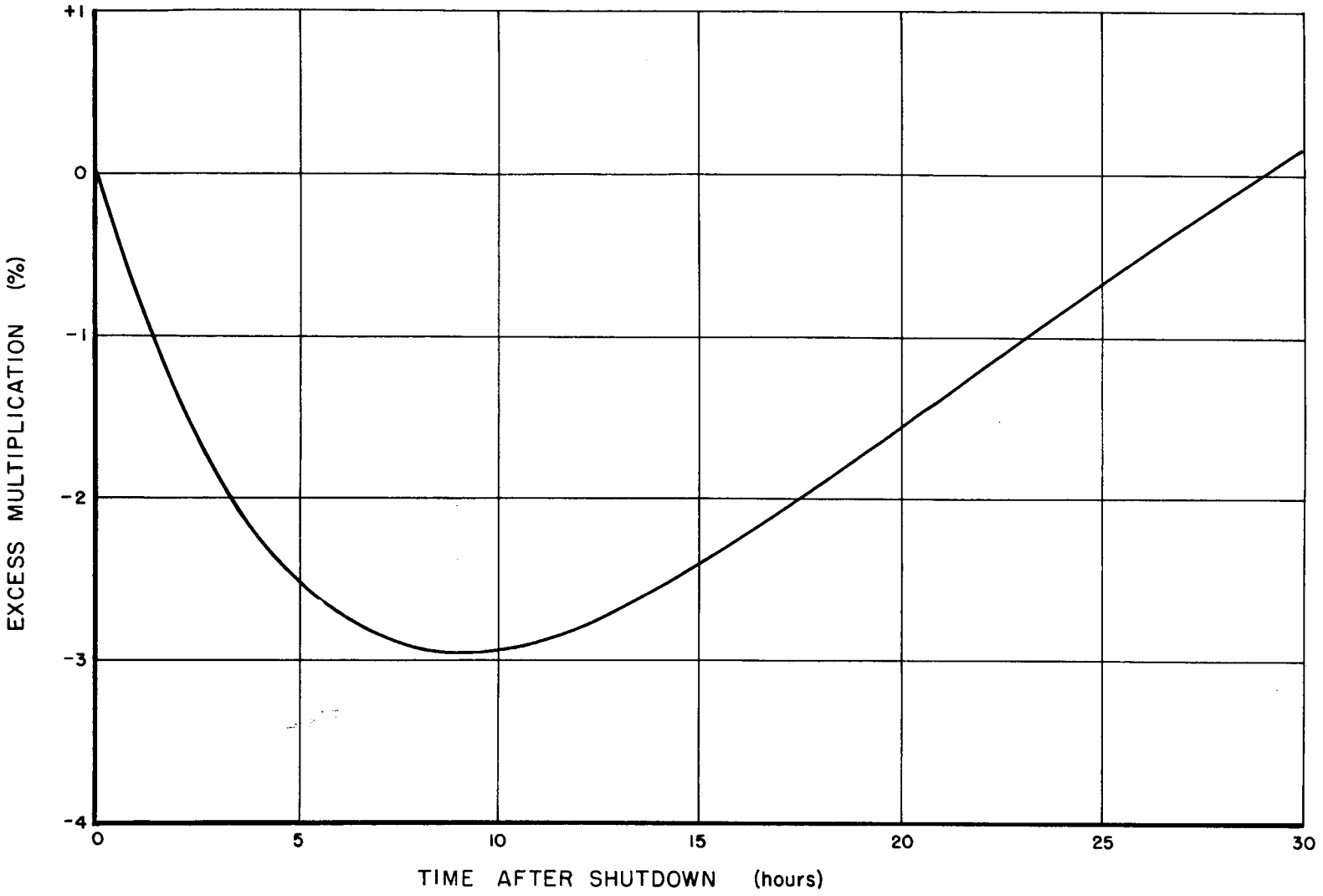


Fig. 26. Xenon Transient Expected in P-12

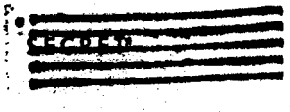
estimated to be at least 15 to 25 hours, depending on the speed with which a channel can be changed. There should be no problem of starting up after reloading due to the xenon transient.

3. Burnup and Refueling Effects - The reactor has tentatively been divided into 5 zones as shown in Fig. 27. The division between zones, as well as the number of zones involved, is by no means definite. The arrangement shown in Fig. 27 is mainly for illustrative purposes. The reactor will be shut down periodically and an entire zone refueled. When the central zone of elements has been irradiated to an average of 3800 mwd/t (30 per cent burnup of the original U^{235} present), the 12 central elements will be changed. Likewise, with the other zones, when their elements reach 30 per cent average burnup. The megawatt days of reactor operation and corresponding change in excess multiplication factor is shown in Fig. 28. It is seen that only about 2 per cent excess k need be built in to allow 30 per cent burnup to be achieved. Figure 28 has been drawn as though 30 per cent burnup was a limiting irradiation.

After the reactor has operated for about 50, 000 megawatt days, there appears to be excess reactivity available to carry the zones to more than 30 per cent burnup. This will, of course, be done if the fuel elements demonstrate sufficient radiation stability, but for the present a 4000 mwd/t limit seems wise.

The U^{235} content of the fuel as fed into the reactor has been tentatively estimated to be 0. 0155 in order to allow 2 per cent excess k. It will contain 0. 0109 U^{235} and 0. 0030 Pu^{239} when it is withdrawn after 3800 mwd/t exposure.

The radiation and thermal stability requirements of the fuel are rather severe. The central elements will be in the reactor for 250 days at continuous full power or 312 days at 80 per cent plant factor. The outer elements, however, will be in 525 days at continuous full load, or 660 days at 80 per cent load. The outer elements are exposed to the highest temperature sodium (1050° F) for the longest period of time. It will be necessary to develop a fuel element which will perform satisfactorily under these time-temperature-irradiation conditions. At present a 2 weight per cent zirconium-uranium alloy appears to offer the most promise.



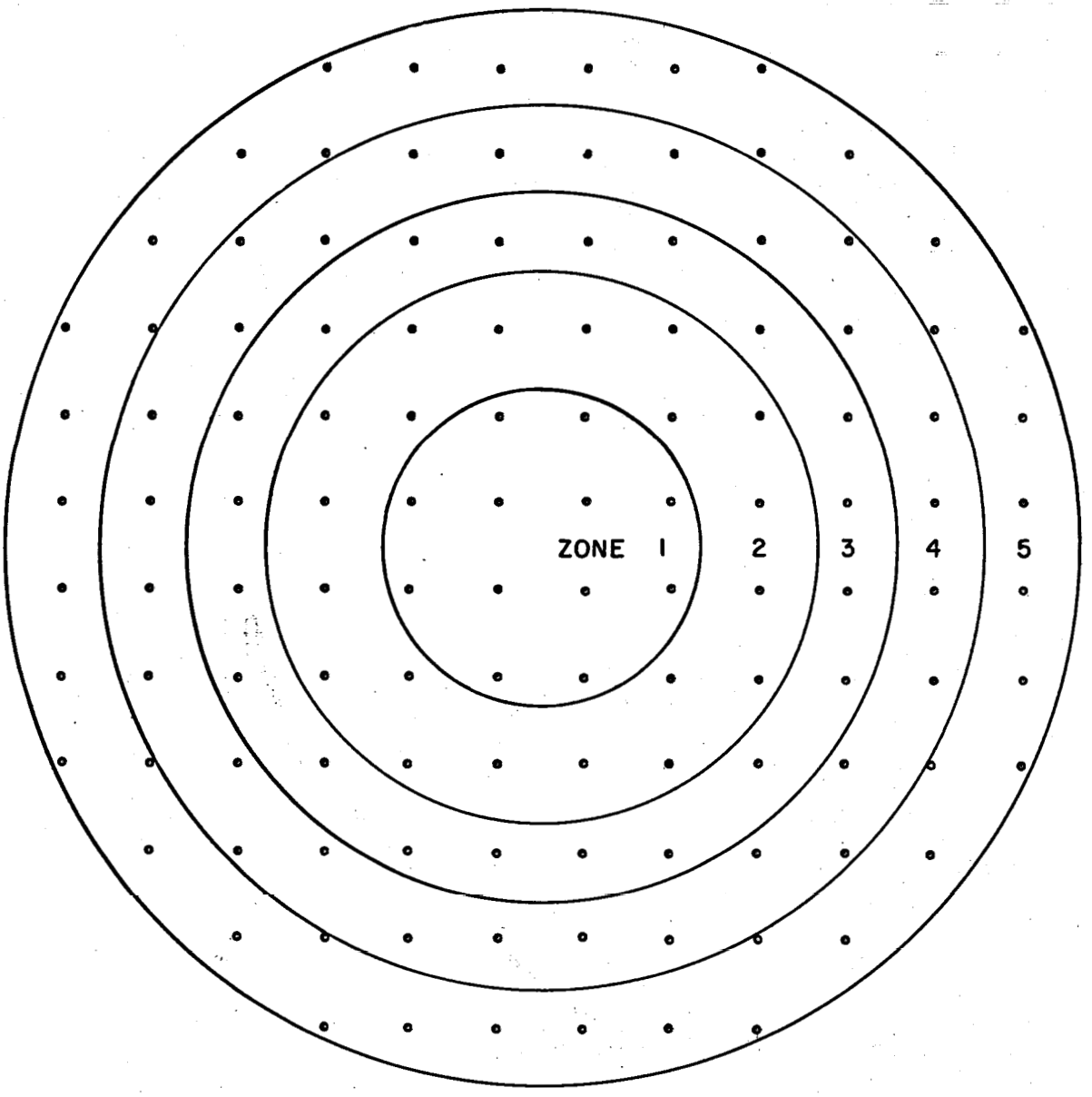


Fig. 27. Zonal Diagram of P-12

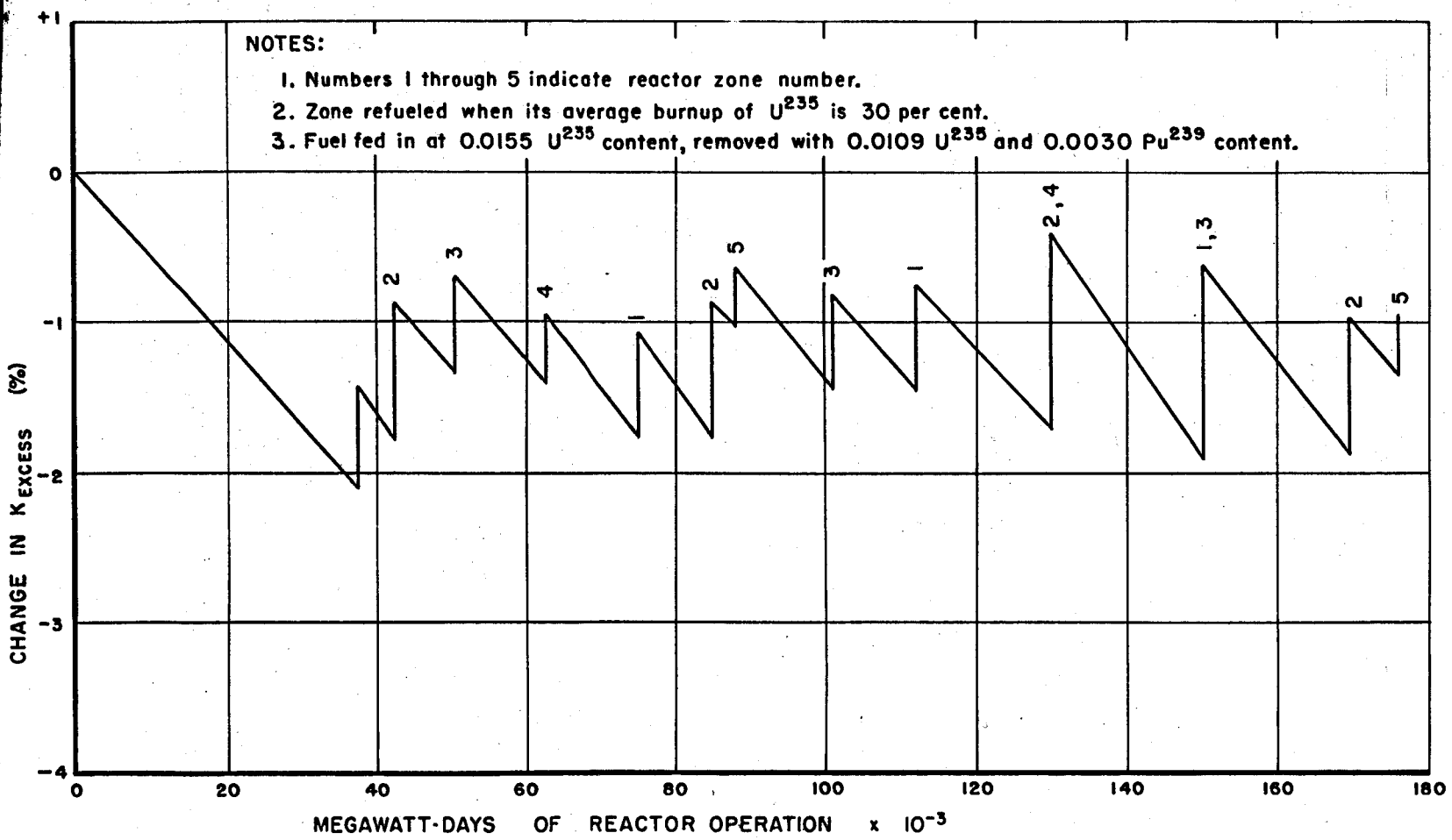


Fig. 28. Effect of Burnup and Refueling on K Excess

4. Temperature Effects - Since the reactor is designed to operate at elevated temperatures, the thermal neutron level will be greater than the usual 0.025 ev. The average neutron temperature for the P-12 has been assumed at 400° C (0.058 ev). Calculations, however, have been made for a range of neutron temperatures from 20 to 600° C to determine the importance of this parameter. It is realized that for a complete and accurate analysis considerable experimentation will be required. However, a fairly large uncertainty in the temperature effect manifests itself as a relatively small uncertainty in the required enrichment.

There is little doubt that the multiplication constant will increase as the temperature increases, and that the leakage will also increase. The net effect on pile reactivity can be either a positive or negative contribution. At present it appears that the overall temperature coefficient of reactivity will be positive for temperatures near 20° C, but will be negative for temperatures in the operating range. A rough estimate places the temperature coefficient between -3 and -4 by 10^{-5}/°C for a neutron temperature of 400° C.

5. Change in Fission Cross Section with Burnup - The reactor is intended to be operated at a constant power level -- nominally 167 mw. As the U^{235} is burned, it is replaced with Pu^{239} with an initial conversion rate of 0.7. (The conversion ratio may change slightly with burnup, but this is a small change within the exposure considered here). The total fission cross section at any time is the sum of the fission cross sections of each of the isotopes present. The effective cross section for each isotope in turn depends upon the neutron temperature, particularly the cross section of Pu^{239}. Using the f values as given in Ref. 13 for a neutron temperature of 400° C, the total fission cross section variation with burnup for P-12 is shown in Fig. 29. Also shown is the variation at 20° C. The power at which a rod operates is directly proportional to the fission cross section times the flux. When the fission cross section varies with degree of burnup and with neutron temperature, as well, one must be careful in rating a reactor so that the full power flux will not cause the center temperature of any fuel element to exceed 1200° F at any degree of burnup. An ample margin of safety with respect to this consideration has been allowed in the design of P-12.

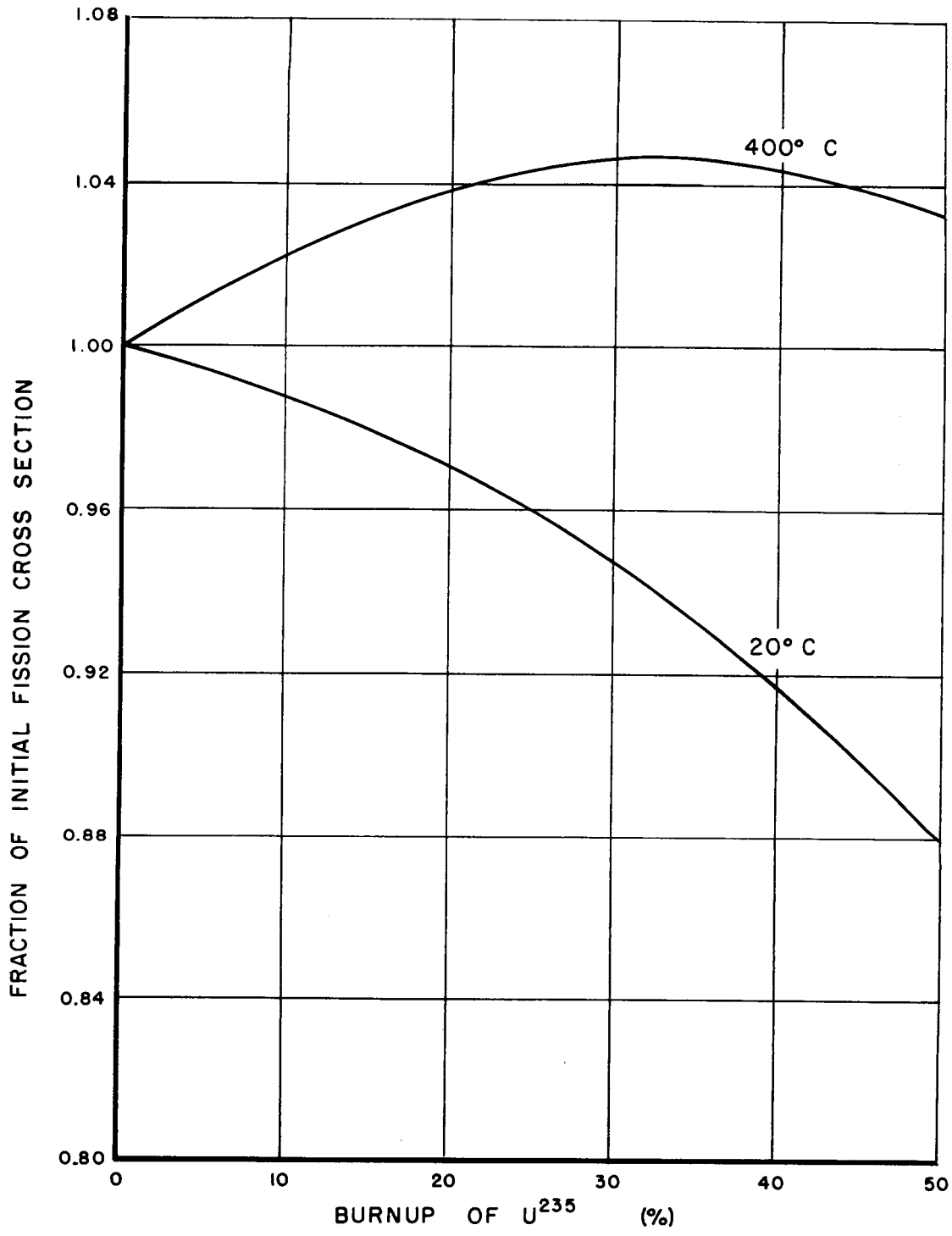


Fig. 29. Variation in Fission Cross Section with Burnup

6. U^{237} Content of the Fuel - After the irradiated fuel has been removed from the reactor, it will be sent through a chemical separations plant. The uranium will be recovered, converted into UF_6, and sent through a diffusion plant to restore the U^{235} fraction to its original value. There will always be some U^{237} present in irradiated uranium which means sufficient time must elapse between its removal from the reactor and its being fed into a diffusion plant for the U^{237} activity to decay to tolerance.

According to Ref. 14, K-25 has set the maximum U^{237} activity limit at 1 microcurie per gram of uranium.

The length of time required for the U^{237} activity to decay to 1 microcurie per gram of uranium depends upon the enrichment (U^{235} portion of the uranium) and upon the flux. The concentration of U^{237} atoms per cm^3, N(27) is given approximately as

$$N(27) \approx \frac{\phi \sigma_a(26)\, RN(u)\, \sigma_c(25)}{\lambda(27)\left[\sigma_a(25) - \sigma_a(26)\right]} \left\{ \exp\left[-\sigma_a(26)\phi t\right] - \exp\left[-\sigma_a(25)\phi t\right] \right\}$$

where

ϕ = thermal neutron flux
R = U^{235} fraction in uranium
$\sigma_a(25)$ = absorption cross section of U^{235}
$\sigma_a(26)$ = absorption cross section of U^{236}
$\sigma_c(25)$ = radiative capture cross section of U^{235}
N(u) = atoms uranium per cm^3
$\lambda(27)$ = decay constant of U^{237}

Delete
Do not delete

The time required for the U^{237} activity to decay to tolerance for P-12 fuel (R = 0.0155) is given in Fig. 30 vs burnup for several values of flux. In general, between 80 and 100 days must elapse between the end of irradiation and feed to the diffusion plant for 30 per cent burnup per pass through the reactor.

7. Chemical Processing Considerations - It is intented to mechanically decan the fuel elements before processing and clean them to remove any adherent sodium before they are dissolved. The purex process, as developed at ORNL (Ref. 15), appears to be the best large scale method to date for the separation

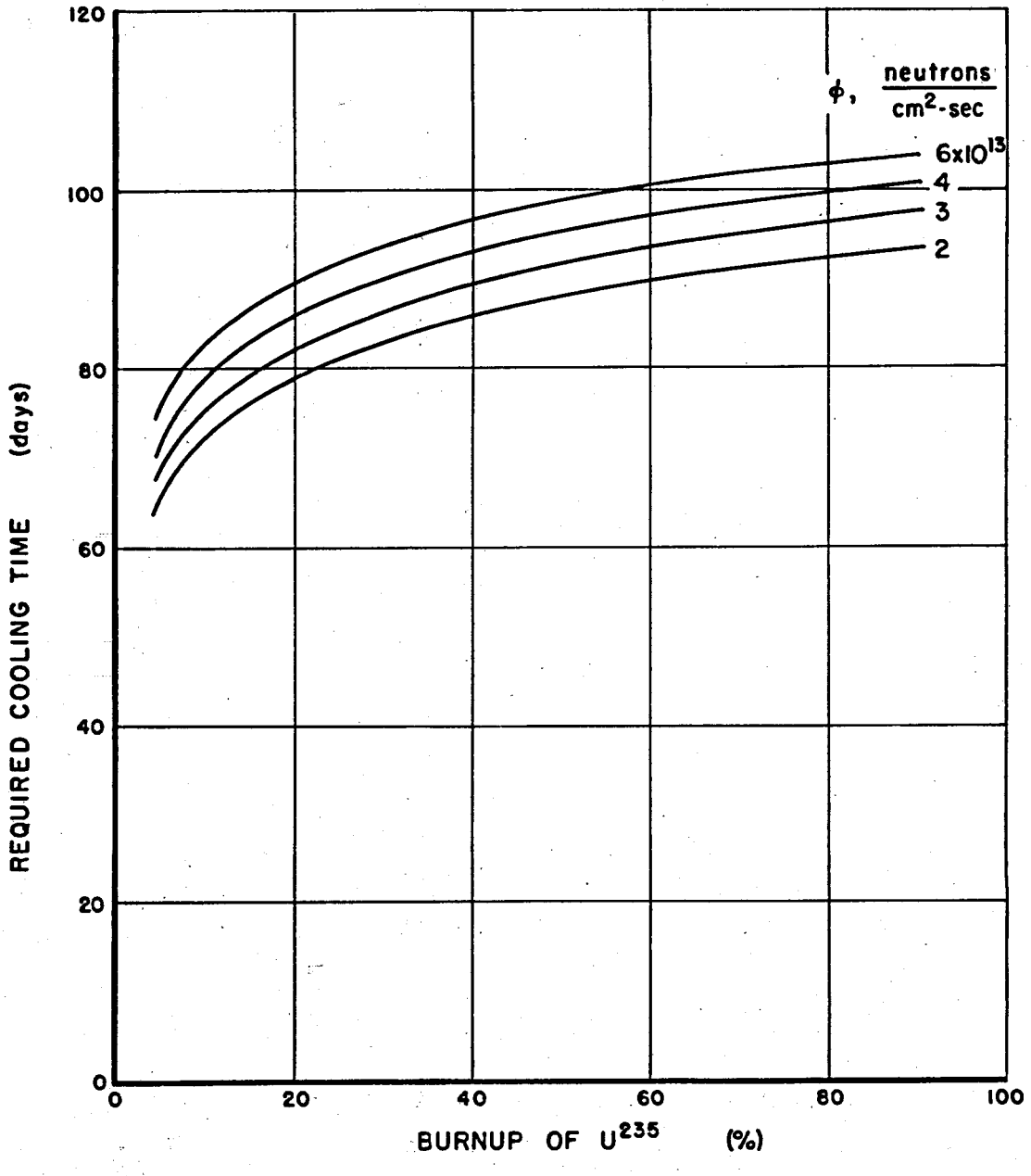


Fig. 30. Time Required after Shutdown for U^{237} Activity to Decay

of plutonium, uranium and fission products. It is felt that the use of small amounts of alloying elements will offer no basic difficulty in chemical processing. They will be removed in the first extraction cycle along with the fission products. The concentration of fission products and the concentration of plutonium in high burnup fuels is higher than the concentrations now being handled at production facilities. These factors should, however, offer no basic difficulty. After decontamination, the uranium will be converted to the hexa-fluoride so that it can be fed through a diffusion plant.

8. Plutonium Feedback - When an irradiated fuel element is removed after 3800 mwd/t exposure, it should contain 0.30 per cent Pu^{239} and 0.027 per cent Pu^{240} (9 per cent ratio of 40 to 49). This plutonium will, or course, be recovered during the chemical separation process. It would be of considerably greater value in a fast breeder reactor than in a thermal reactor and, accordingly, should be used in a fast reactor if one is then in operation.

In economic analyses to determine the cost of producing nucleoelectric power, it is standard practice to credit the plutonium only at its value as a fuel in the reactor which produces it. This assumes that the plutonium can be fed back into the reactor somehow and result in a reduction in the U^{235} feed requirement. There are, however, problems in fabricating fuel elements containing plutonium due its toxicity. It is estimated that the fabricating costs may increase by as much as a factor of 4 or 5 over the costs for uranium rods without Pu (using conventional fabricating procedures). If this should prove to be the case, it is a marginal consideration as to whether to use the Pu at all in this system.

9. Other Fuels - The P-12 reactor is designed to operate on slightly enriched uranium. It could, however, operate on uranium-plutonium alloy fuels. The uranium could be natural, depleted, or slightly enriched in U^{235}. Sufficient Pu would have to be added to cause the resulting mixture to have the same reactivity as the other uranium fuel. There would be no particular advantage to using this type of fuel except that it would allow a means of using the plutonium that results when low enrichment uranium is burned. The conversion ratio and also the reactivity-limited attainable burnup would not be appreciably different than those for slightly enriched uranium.

Thorium-uranium alloys can also be used a fuel. These would include $Th\text{-}U^{235}$ and $Th\text{-}U^{233}$. Also, $Th\text{-}Pu^{239}$ could be used as a fuel if the metallurgical problems can be solved. The conversion ratio with uranium-thorium alloys will be superior to that attainable with slightly enriched uranium only. Also the burnup that may be attained before loss of reactivity will be much more than 3800 mwd/t. It appears that thorium can be taken to 10,000 or 15,000 mwd/t irradiation before suffering radiation damage.

The P-12 reactor as described in this report, though, is not sufficiently large so that using all thorium-uranium elements a conversion ratio of 1:0 would result. The leakage is too great. Some thorium elements could be used to demonstrate their feasibility for use in a larger thorium power breeder reactor as described in Refs. 16 and 17.

10. Three Units vs One Unit - If one wishes to build a nucleoelectric plant of 150 electrical megawatts capacity, one can build it as one unit or several units. There are advantages and disadvantages either way. Since the sodium inlet and outlet temperatures are fairly independent of the number of units, the thermal efficiency will be the same for all the units, namely, 30 to 32 per cent. We have assumed 30 per cent net plant efficiency or a thermal power of 500 mw. The required thermal power of each of three reactors would then be 167 mw. There would, however, still be just one 150 mw (electrical) power plant. The three reactors could be located side by side to reduce shielding, but otherwise would operate independently of each other.

One single 500 mw unit would require a lower enrichment fuel but more of it than three units with the same thermal power. This results from less leakage from the larger unit, and a lower average-to-peak cell power. Each of the three units is reflected, which means that the outer rods operate in a higher flux than the outer rods of the single unit.

The major advantage of multiple units over a single unit is the possibility of improved plant factor. Every time a fuel cell needs to be changed the reactor must be shut down. With a single reactor the loss in kilowatt-hour output is three times the loss with three units where only one at a time is shutdown. It has been estimated that if a single unit operates at 80 per cent plant factor, three units could operate at 87 per cent plant factor. Also there would be more

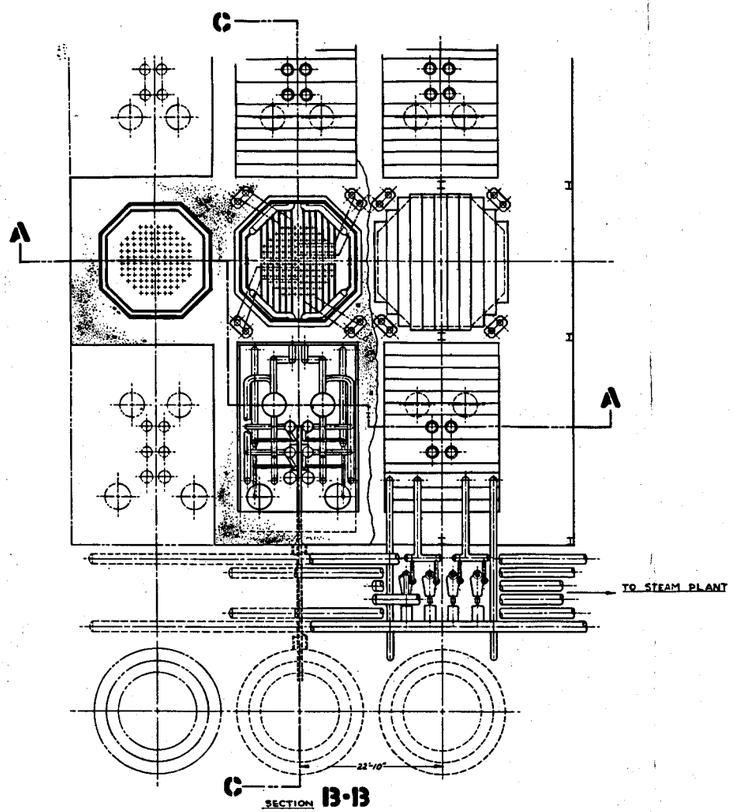
certainty that at least part of the plant (one or two units) could be kept operating 100 per cent of the time (turbogenerators allowing).

Cost estimates, however, indicate that the most economical power is produced by the single unit -- though the difference is small. The three units are considered here only because of their greater reliability in being able to supply at least part of their capacity all the time.

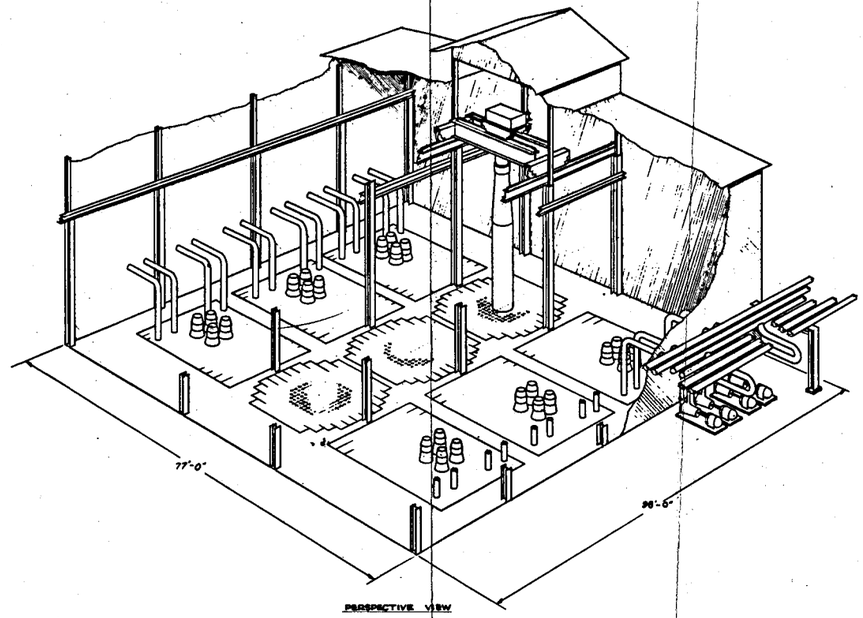
D. Construction Costs

A comprehensive study and detailed examination of the costs of each part of the reactor complex has been carried out and estimates have been made of the cost of each item allowing reasonable contingency factors. The estimates for the building, the more-or-less standard equipment, and labor costs were estimated as for normal firm-bid contracted construction work in a labor area such as that around Los Angeles. An itemized breakdown of the construction costs for a three-reactor complex (see Fig. 31) appears in Table II.

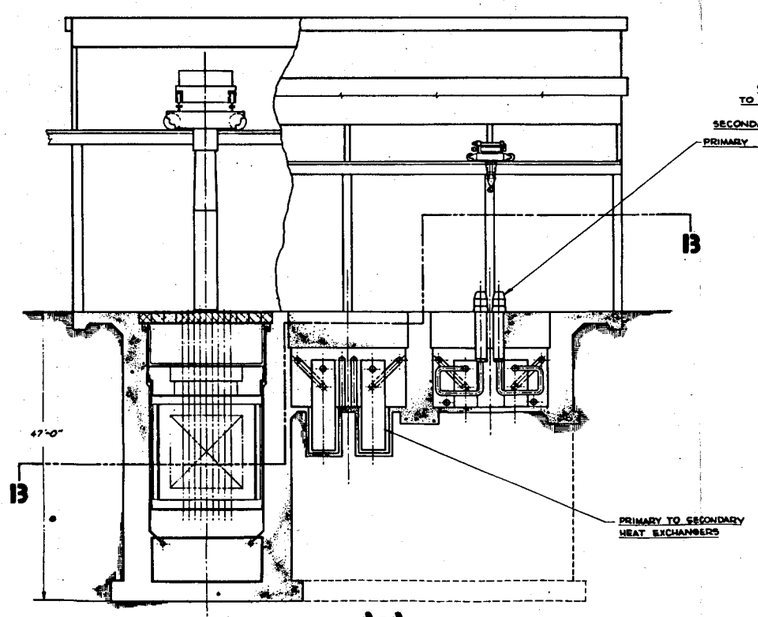
The power conversion equipment has been estimated at $100 per installed kilowatt of capacity; Ref. 5 uses $95 per kilowatt and Ref. 18 indicates $105 per kilowatt for the steam conditions here. This seems to be a reasonable range for this rather conventional equipment. The estimate of $100 per kilowatt includes a charge for land and improvements generally associated with a conventional power generating station. No estimate of land and improvement costs associated with the reactor has been included since this is strongly dependent upon site location. The boiler cost was taken directly from Ref. 5. No fuel has been included in the cost estimate nor the cost of fuel fabrication.



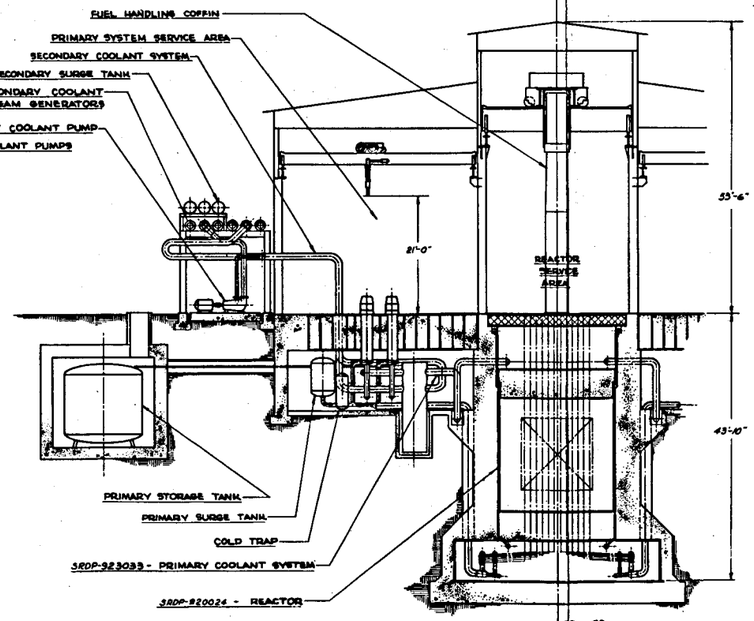
SECTION B-B



PERSPECTIVE VIEW



SECTION A-A



SECTION C-C

SCALE IN FEET

Fig. 31. Building Arrangement for Multiple Unit

TABLE II

CONSTRUCTION COSTS

500 mw Plant

1. Reactor Building		
A. Building Construction	$ 134,800	
B. Building Equipment	158,500	
C. Engineering, 15 per cent	44,000	
		$ 337,300
2. Reactor Structure		
A. Foundation	99,600	
B. Vertical Biological Shield	97,600	
C. Inner Gas-tight Liners	360,300	
D. Top Neutron Shield	187,400	
E. Lower Neutron Shield	22,200	
F. Alignment Washer Assemblies	400,000	
G. Upper Gamma Shield	820,000	
H. Heat Exchanger Vault Top Shield	585,000	
I. Thermal Shield	178,900	
		2,751,000
3. Reactor Core		
A. Graphite, 475 tons	1,422,000	
B. Coolant Channels	190,600	
C. Inlet Manifold System	332,500	
D. Outlet Manifold System	499,000	
E. Control Rods	565,000	
F. Safety Rods	264,000	
		3,273,100
4. Reactor Equipment		
A. Fuel Handling Coffin	151,900	
B. Helium Systems	81,000	
C. Shield Cooling System	242,000	
		474,900

TABLE II (Continued)

5. Instrumentation		$ 1,800,000
Subtotal for reactor		8,636,300
6. Fuel Handling and Storage Facilities		
A. New Fuel Storage	$ 78,100	
B. Fuel Handling Building	56,200	
C. Cleaning Cells	39,300	
D. Inspection Cell	14,500	
E. Storage Pond	32,400	
F. Shipping Coffins	510,000	
		730,500
7. Sodium Circuit		
A. Primary Cooling System	2,789,900	
B. Secondary Cooling System	1,750,400	
C. Steam Generator	2,720,000	
D. Sodium Purification System	500,000	
		7,760,300
Total materials and direct labor		17,127,100
Contingencies, 15 per cent		2,569,000
		19,696,100
Construction overhead, 30 per cent		5,908,800
		25,604,900
Engineering, Design and Development, 10 per cent		2,560,500
Subtotal		28,165,400
8. Power Conversion Equipment $100/kw(electrical)		15,000,000
Total		$43,165,400

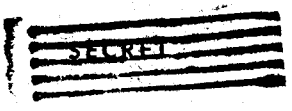
E. "Modernization" - Incorporation of New Developments

1. General - It is recognized that as time progresses, improvements on the existing technology will be made. It is not possible, of course, to foresee very many of these new developments at this time, but we can speculate about a few. This reactor design might incorporate many of these. The use of zirconium tubing as a cladding for the fuel rods and also as a coolant channel, the use of thorium-uranium alloys as a fuel, the use of multiple hollow fuel elements (concentric rings), the use of highly alloyed uranium fuel, and the use of other fluids in the intermediate heat exchanger loop are all possible new developments. These will be discussed in the succeeding paragraphs.

2. Use of Zirconium - Thin walled zirconium tubing for use as a cladding could probably be considered available at this time (Ref. 19). It was not specified as the cladding because at the time there was doubt as to its availability in the proper sizes. It has always been intended to use zirconium cladding as soon as it could be shown to be economically wise to do so. The use of zirconium as a coolant channel, however, presents some additional problems. It may be possible to draw seamless zirconium tubing of the required size, but this has not so far been demonstrated. It could be formed from welded tubing (Ref. 20), but this may not be satisfactory. Also the problems of joining the zirconium tube into the stainless steel system in a leak tight manner have not been completely solved. The welding of zirconium to stainless steel has been achieved successfully on an experimental basis (Ref. 21), but it remains to be proven that the joint will successfully operate under the conditions encountered in this reactor.

Using zirconium as a cladding or coolant tube or both will allow the U^{235} content of the fuel to be reduced slightly. This should allow higher burnups to be realized (radiation damage permitting), and will certainly reduce the fuel feed costs.

3. Use of Thorium-Uranium Alloy as a Fuel - As mentioned in Section III-C-9 the possibility of developing a fuel element of thorium and uranium (U^{233} and or U^{235}) has been considered. A rather thorough study of the merits of the thorium-U^{233} system has been presented in Ref. 16 so will not be repeated here. This reactor, however, might be too small (too much leakage) for the



thorium-U^{233} system to have a conversion ratio greater than unity, but if the conversion ratio is even close to unity, the fuel feed cost will be small. It is not known at this time how much of a saving over the uranium fuel feed cost this will be, but it could be substantial.

4. Use of Hollow Fuel Elements - Work is now under way to investigate the problems of manufacturing and cladding hollow fuel elements. According to Ref. 12 it appears that a single hollow rod element would not satisfactorily replace the 19-rod clump. Perhaps two or more concentric hollow elements would be required. This tends to complicate matters and to decrease any advantages that the hollow rods have in reduced fabrication costs. By use of powder metallurgy techniques, it may be possible to fabricate one or two large hollow rods less expensively than 19 small solid rods, but the magnitude of this saving (if any) is not known at this time. Also if several concentric rings are required, the problems of establishing and maintaining a balanced flow among the various annuli increase. If, however, the hollow rods should show any definite advantages over the 19-rod clump, they can very easily be used in the reactor.

5. Use of Highly Alloyed Uranium Fuel - It has been assumed throughout this study (and all others using uranium metal fuel, Refs. 1, 2, 3, 4 and 5) that the uranium metal fuel should not be operated above the $\alpha - \beta$ phase transition (see Section II-E-1). Reference 22 discusses the use of an alloy of uranium and zirconium, 20 atomic per cent zirconium (8.7 weight per cent), as a fuel. The uranium beta phase does not manifest itself in this alloy. On the assumption (as yet unproven) that the coexistance of the alpha and gamma phases is permissible in a fuel slug, considerably higher power density in the fuel can be attained. This could be utilized to advantage in this reactor in one of several ways: reduce number of rods per cluster, increase sodium temperature (and steam temperature), increase the power of the reactor or any combination of the three (Ref. 23).

6. Use of Other Intermediate Heat Exchanger Fluids - The current design specifies sodium (or NaK) as the intermediate heat exchanger fluid between the radioactive sodium and steam. J. R. Wetch, Ref. 24, points out the advantages that a mercury fog convection intermediate heat transfer system offers.

Some of these advantages are: the elimination of the sodium-water hazard; elimination of the need for pumps in the intermediate loop since thermal convection will suffice; elimination of the necessity to use relatively expensive stainless steel; the secondary system would be liquid at room temperature. Also considered are organic vapor and organic liquid intermediate heat transfer systems. It appears that the best system would use mercury fog for the high temperature sections and an organic vapor for the low temperature sections. The advantages lay in increased safety and reduction in investment of perhaps two million dollars. It would be necessary to develop such a system before final design since it would not be practical to "convert over" after designing for a sodium intermediate system.

IV. CONCLUSION

As a result of the studies presented in this report, no major development problems are foreseeable as a prerequisite to the construction of the reactor plant described. The plant has been considered as a power producer only, but could be operated as a plutonium producer as well be removing the fuel after 10 per cent burnup. The use of three 167 megawatt units should allow continuous operation at partial load (steam power plant permitting) and a rather high fractional innage at full load. Only one reactor need be shut down at a time for fuel replacement or routine maintenance.

The installed cost of this 500 thermal mw plant not including fuel or site is estimated to be $43,165,000, or $288 per electrical kilowatt. With an annual capital investment charge of 15 per cent and an average plant capacity factor of 87 per cent, the cost of electrical power generated is computed to be approximately 9.6 mills per kilowatt hour with uranium fuel. With thorium fuel this cost could be reduced somewhat.

APPENDIX A

HEAT TRANSFER AND POWER PLANT DATA

A. General

It is desired to convert as large a fraction of the reactor power into useful power as it is possible to do. The reactor heat is removed by the primary sodium coolant at a mixed mean temperature of 940° F. By means of an intermediate heat exchanger, the heat is transferred to a nonradioactive secondary coolant, again sodium. The use of an intermediate heat exchanger is considered necessary in order to obviate any possibility of a chemical reaction between the radioactive sodium and water in the event of serious damage to the steam generator. The secondary sodium transfers its heat in a sodium heated steam generator located a safe distance from the reactor. The steam is delivered to a standard turbo-generator unit where approximately 30 per cent of the reactor power is converted to electric power. The remainder of the power is disposed of in the condenser, which is cooled by water circulating through cooling towers. A schematic diagram of this system is shown in Fig. 23.

With regard to the technical feasibility of constructing such a plant, the following general statements can be made at the outset. The turbo-generators and associated equipment (condensers, ejectors, deaerators, cooling towers, etc.) are conventional and no major modification in the normal control or operation of these units is required. The radioactive sodium circuit, the intermediate heat exchangers, the steam generators and the sodium service system are, of course, not conventional in the same sense as the turbo-generator facilities. However, there does exist a backlog of some 5 years of system and component testing for the "SIR" project which is directly applicable. Examination of the information and experience thus far gained in sodium technology and the application of this technology to power conversion systems indicate that there exist no major problems for which adequate solutions have not been found. The components for such a system, piping, valves, heat exchangers, pumps and instrumentation are all presently available. These systems and components have not yet been proven as operationally reliable as the conventional steam plant; however, their technical feasibility has certainly been demonstrated.

The design of the sodium system discussed here is based on information and data obtained from KAPL and the various subcontractors associated with the SIR project. The design lies entirely within the framework of present technology.

B. Cooling Calculations

The general procedure for determining the heat transfer characteristics of liquid metal cooled rod type fuel elements is discussed in Ref. 25. In order to apply this method the following information must be known:

1. Dimensions of the fuel element and number of fuel elements per cluster.
2. Coolant flow area per cluster.
3. Maximum permissible temperature in the fuel element.
4. Axial and radial power distribution.
5. Number and spacing of clusters.

This information appears in various parts of this report and is as follows:

1. Seven rods per cluster. Uranium 0.750 inches in diameter, surrounded by 0.010-inch Na bond, clad in 0.010-inch type 304L stainless steel. Length of uranium is 10.0 feet.
2. 3.67 square inches or 0.0255 square feet.
3. 1200° F
4. Axial - chopped cosine with reflected length to unreflected length of 0.845. This implies a 28 centimeter reflector saving at each end. Radial - combination of real and imaginary zero order Bessel functions as obtained from two group theory (see Appendix B) with reflected to unreflected radius of 0.704 and peak to average clump power of 1.46.
5. 120 clusters on a 10.0 inch square lattice.

One may now proceed to calculate the central cluster power, Q, temperature rise of the coolant, $t_2 - t_1$, and pressure drop across the cluster, Δp, as a function of coolant velocity. This information is presented in Fig. A-1. All the assumptions previously discussed in Section II-E-1 have been included in these calculations making them quite conservative.

In order to attain 167 megawatts total power from 120 elements with a radial flux distribution calculated by two-group theory, one needs to extract 1950 kilowatts from the central elements. Reference to Fig. A-1 shows that this amount of power may be attained with a sodium velocity thru the central elements of only 10 feet per second. It is felt that the reactor is capable of operation at a much higher power level than 167 megawatts. Only actual operating experience will determine just how high the power may safely go. The total coolant flow rate is 3.82 by 10^6 pounds per hour resulting in a mixed mean outlet temperature of 940° F for 450° F inlet temperature.

On the assumption of a single Bessel function distribution (one-group theory), the central elements must deliver 2200 kilowatts. The total flow rate is then 3.91 by 10^6 pounds per hour and the mixed mean outlet temperature is 928° F. This would seem to indicate that the flow rate, mixed mean outlet temperature, and central element power are not particularly sensitive to the radial flux distribution so long as proper orificing is attained.

C. Power Plant Data

Primary sodium flows through the reactor from bottom to top at a rate of 3,820,000 pounds per hour. It is heated from 450° F to 940° F (mixed mean). The heat is transferred to a secondary (intermediate) sodium loop with about a 30° F approach. This intermediate loop operates between 420° F and 910° F, also with a flow rate of 3,820,000 pounds per hour giving up its heat to a steam cycle. The 910° F sodium enters a superheater from which it emerges at 815° F to enter an evaporator. It leaves the evaporator at 505° F and enters the economizer where it gives up the rest of its useful heat. There are four parallel circuits in the primary sodium system with two parallel loops in the secondary.

The entire piping system is of type 304L stainless steel with all joints and fittings of welded construction. Block valves are located on the inlet and outlet lines of the pumps and heat exchangers so as to permit isolation of any of these units should this be necessary. All pipes, fittings, and equipment in the sodium circuit are wrapped with electrical heaters for preheating prior to filling with sodium and as a heat source when the reactor is shut down.

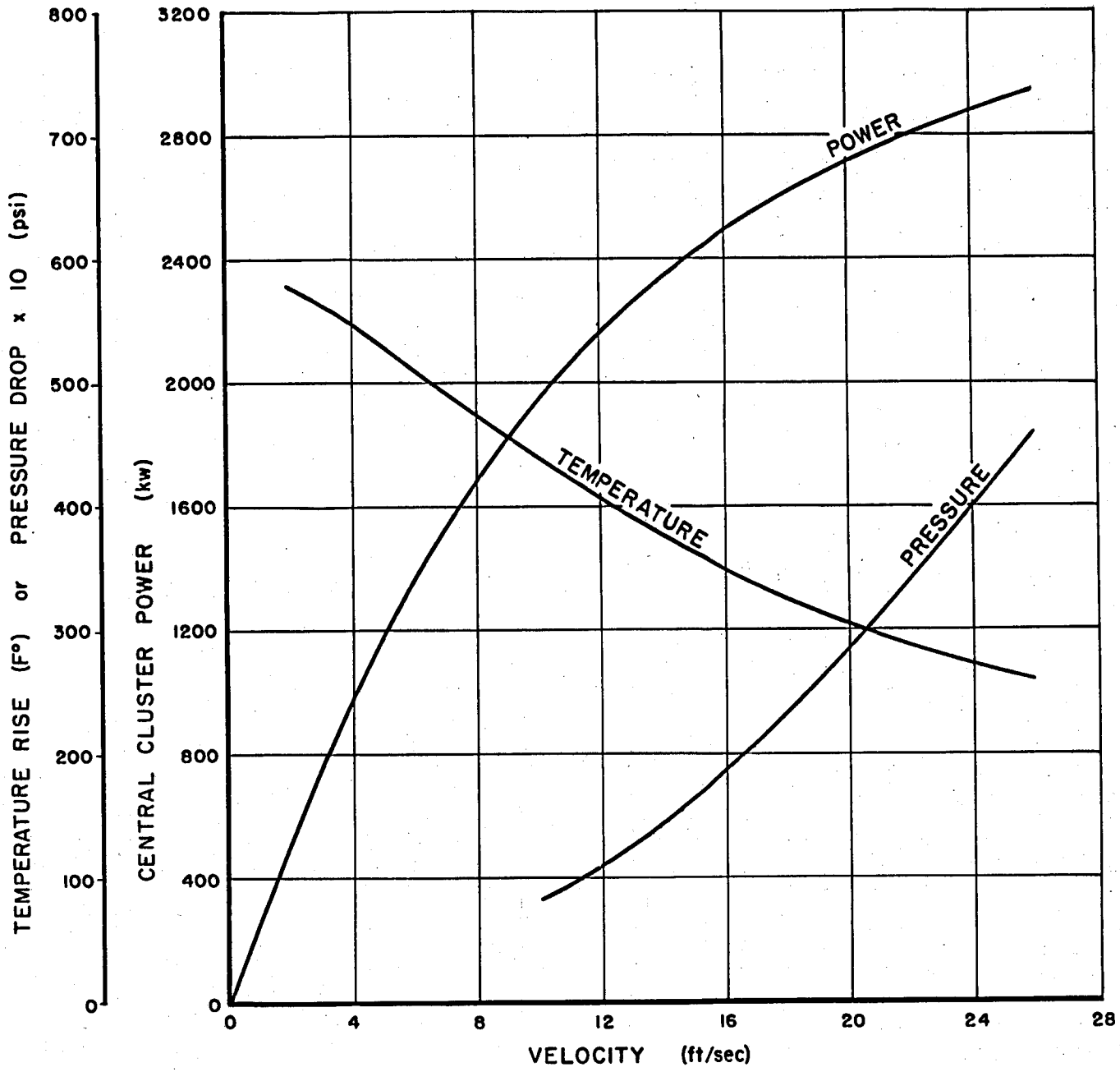


Fig. A-1. Central Cell Power, Pressure Drop, and Temperature Rise vs Coolant Velocity

The equilibrium gamma activity (2.75 Mev maximum) in the sodium circuit is 3 by 10^9 disintegrations per second per cubic centimeter of fluid. The primary coolant loop components then will require something like 6 feet of heavy concrete shielding to reduce the activity to one-tenth of tolerance (120 γ's per second per square centimeter).

Steam is produced at 615 psia and 825° F at a rate of 486,000 pounds per hour at full load. Deaerated feedwater for the boilers is returned at 280° F and 700 psia. The turbine operates condensing (2.0 inches Hg) and is provided with three extraction stages for feedwater heating. The cooling water system is assumed to be recirculating with heat removal to the atmosphere by means of a cooling tower. The electrical generator is a 50,000 kilowatt 0.8 phase factor, 13.8 kilowatt, 3 phase, 60 cycle unit.

A flowsheet and heat balance for a single reactor power plant (167 mw) is shown in Fig. 22. Figure 23 shows the same for a triple reactor power plant (500 mw).

APPENDIX B

NUCLEAR DATA

A. Reactor Dimensions

The basic fuel element of this reactor consists of a cluster of 19 rods (see Fig. 21) as described above in Section II-A-4. Each rod is made up of several slugs of uranium 0.455 inches in diameter contained within a type 304L stainless steel tube 0.495-inches OD by 0.010-inch wall. The 0.010 inch gap between the uranium and steel is filled with sodium (or NaK). The total length of uranium is 10.0 feet. The fuel cluster hangs within a steel coolant channel 3.125-inches OD by 0.035-inch wall. A thin zirconium guard tube (0.015-inch wall) surrounds the steel coolant channel. The nominal ID of the hole in the graphite is 3.175 inches. The elements are arranged on a square lattice with 10.0 inches between centers (see Fig. 4).

The active core of the reactor was specified as a right circular cylinder 10 feet in diameter and 10 feet high; it is approximately by a right octagonal prism, 10 feet across flats and 10 feet high. The core is surrounded by a 30-inch reflector of graphite (see Fig. 1). The reflector is on all sides of the core as well as the top and bottom. The top and bottom reflectors are not as effective as the side reflectors, as we shall see in the next section, due to the passage of coolant tubes and sodium through them. The reactor is designed to contain 120 fuel elements, though this is slightly more than is required for a 10-foot diameter core. Each fuel element will contain 113.8 kilogram of uranium for a total of 13,656 kilogram in the reactor. The total amount of graphite required, including the reflector, is 135 tons.

B. Nuclear Materials Constants

Using diffusion theory, the methods given in Ref. 24 for clumped-rod systems, and the latest constants given in Ref. 13, one may proceed to compute such quantities as multiplication factor, k_∞, material buckling, $\mathcal{B}^2$, thermal diffusion area, L^2, and slowing down area, τ, as functions of the enrichment of the uranium. The cross sections of all materials were averaged over a Maxwellian distribution about the assumed average neutron temperature of 400° C. The appropriate non-1/v factors were also taken from Ref. 13.

The multiplication factor vs enrichment is shown in Fig. B-1, materials buckling in Fig. B-2, and thermal diffusion area in Fig. B-3. Actually the slowing down area, or Fermi age, does not vary with enrichment, but it is necessary to redefine this quantity, as suggested in Ref. 26, so that the buckling used in two-group theory will agree with buckling calculated by the continuous theory. Thus,

$$\tau(\text{two-group theory}) = \frac{1}{\mathscr{l}^2}\left[\frac{k_\infty}{1+\mathscr{l}^2 L^2} - 1\right]$$

where the $\mathscr{l}^2$ is determined using τ(continuous theory). Proceeding then with the usual two-group theory, (Ref. 26), one may, by trial and error, determine what enrichment is required to make the assembly critical. The enrichment required for this reactor to just be critical is 0.0145.

The following constants exist for this enrichment:

$$k_\infty = 1.0992, \mathscr{l}^2 = 181 \times 10^{-6}\ \text{cm}^{-2} \qquad L^2 = 177\ \text{cm}^2,$$

$$\tau(\text{continuous}) = 343\ \text{cm}^2.$$

With a neutron accounting system one can determine where every neutron goes. This is shown in Table B-1 for a bare core with the above constants. The presence of a reflector alters this somewhat by increasing the fast leakage from the core and decreasing the thermal leakage. In fact, the thermal leakage can be negative, i.e. from the reflector into the core. Table B-I, however, is useful in shielding calculations where it is sufficient to know only approximately the capture gamma production spectrum and distribution within the core.

C. Flux Distributions

As a consequence of the two-group calculations performed above to determine the enrichment, one is able to calculate and plot the axial and radial flux distributions for each of the two groups of neutrons. For the thermal flux, the distribution can be represented by the following:

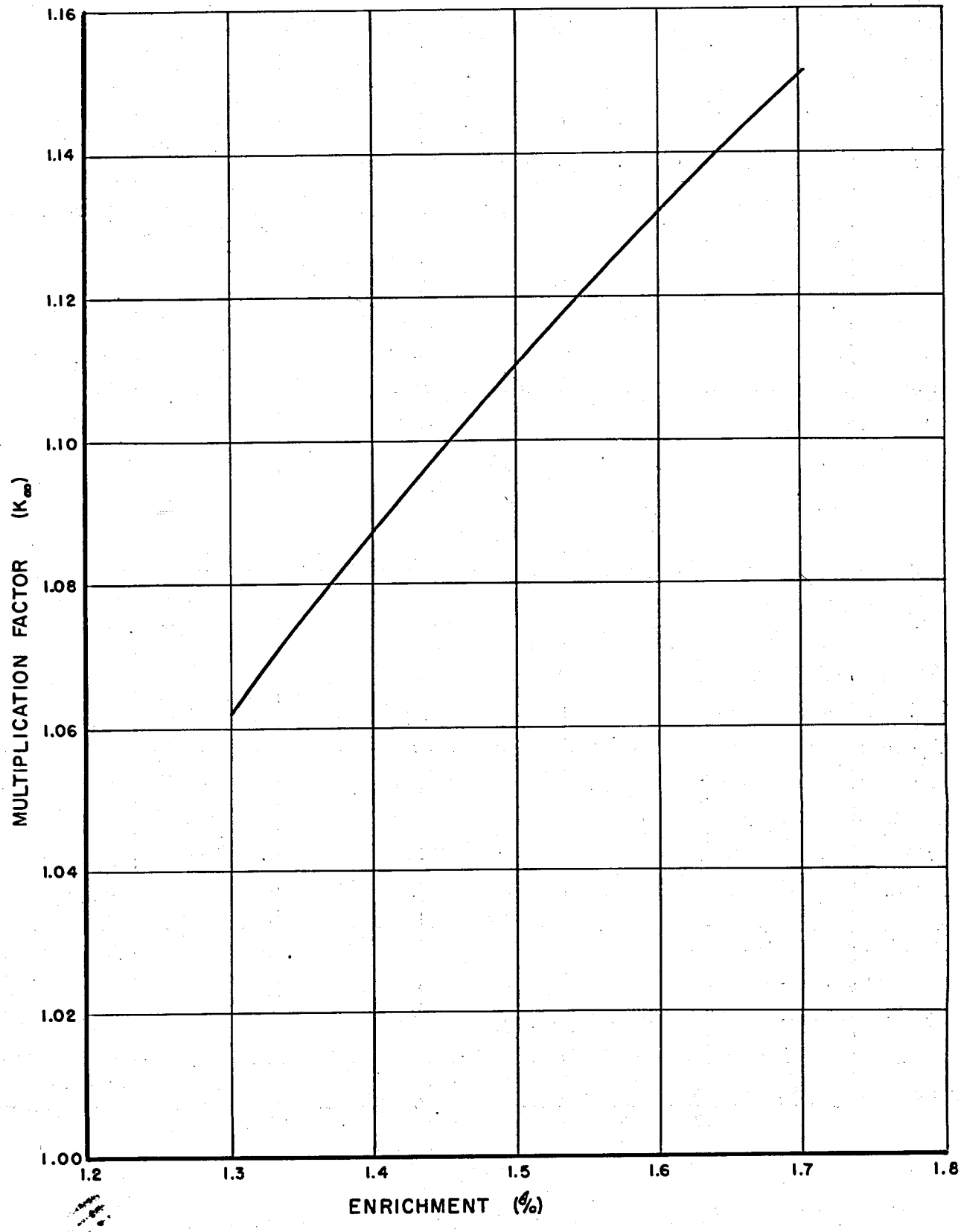


Fig. B-1. Multiplication Factor vs Enrichment

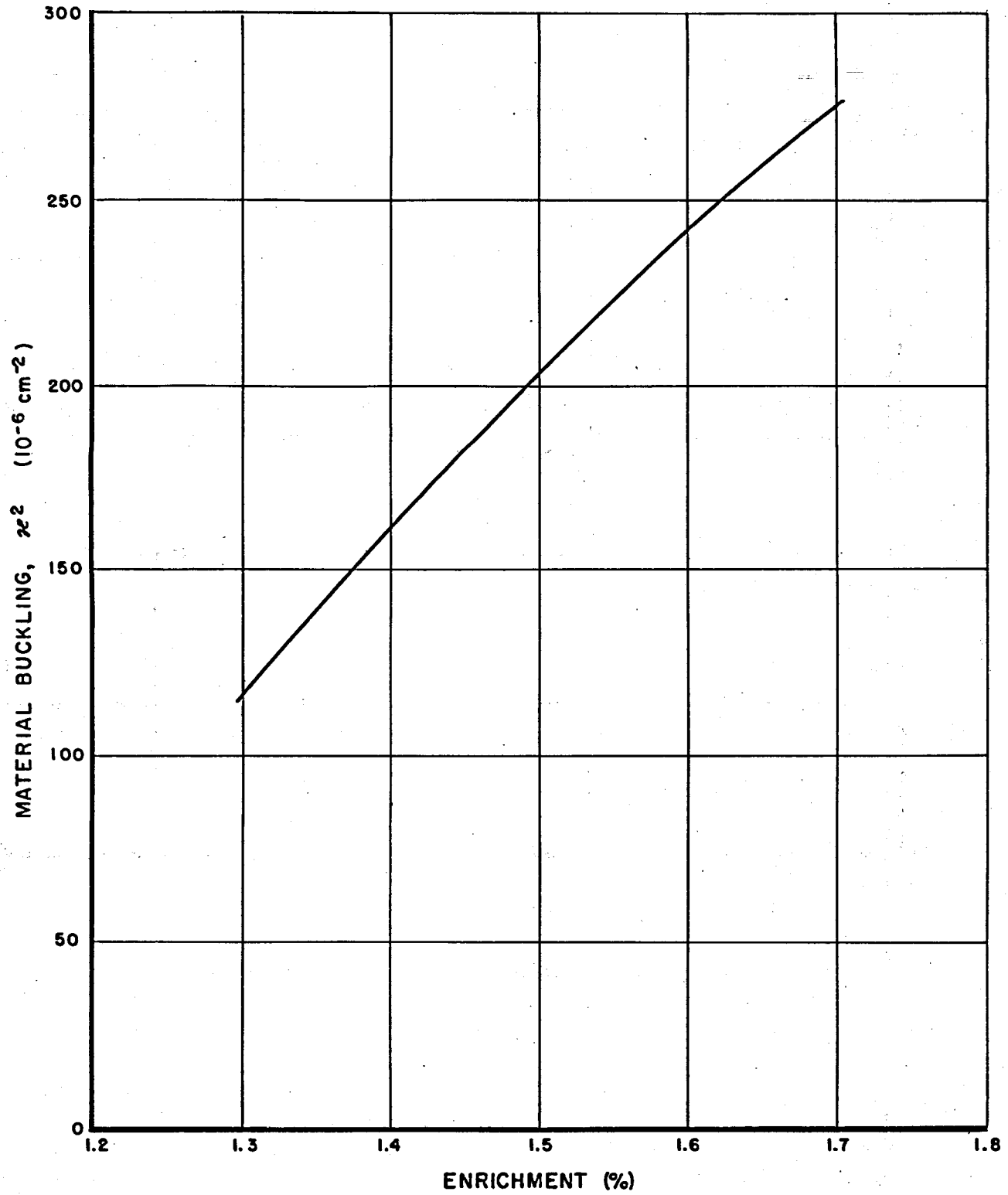


Fig. B-2. Material Buckling vs Enrichment

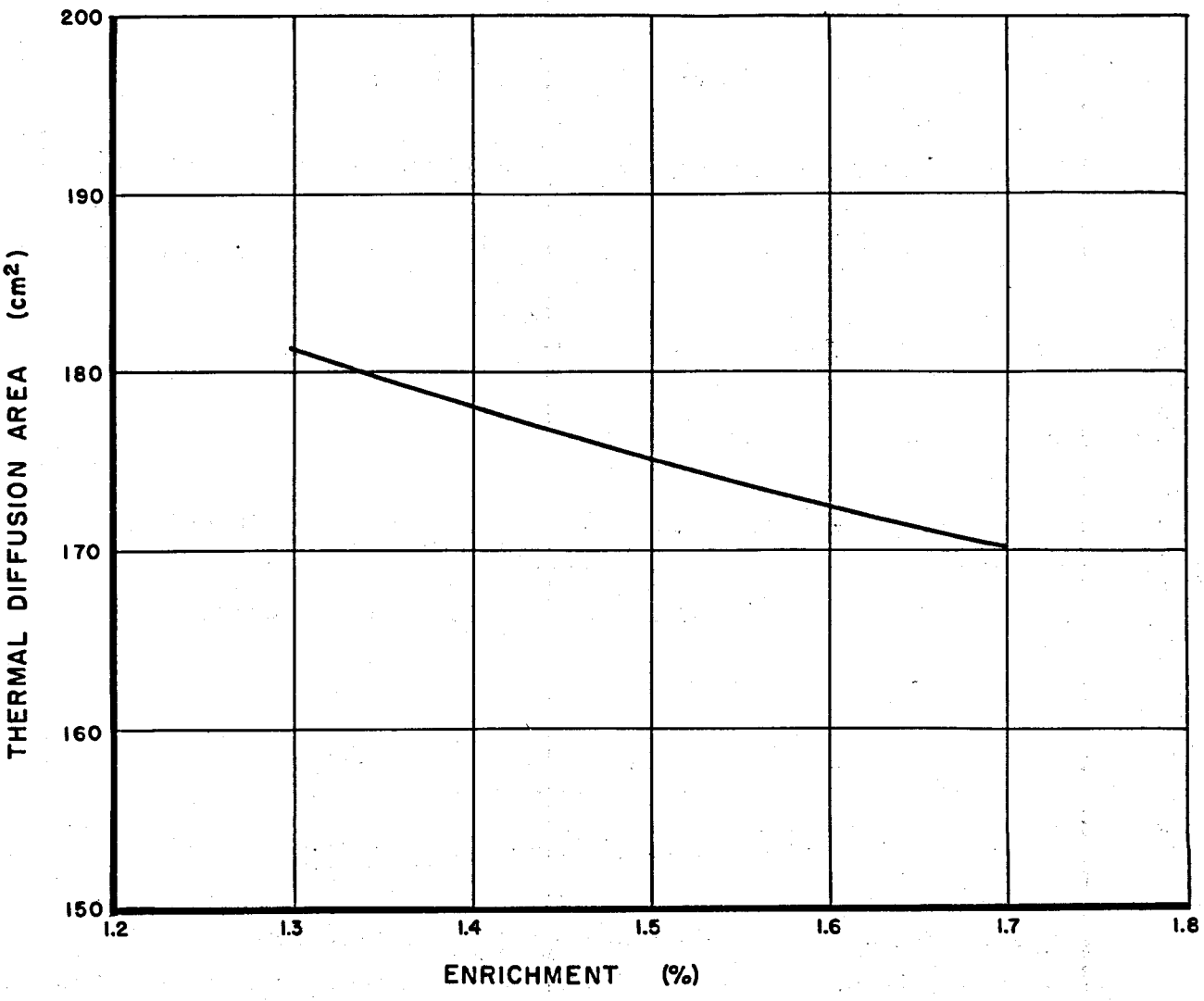


Fig. B-3. Thermal Diffusion Area vs Enrichment

In the core ($0 \leq r \leq 157$ cm, $0 \leq z \leq 152$ cm)

$$\phi_t = \left[\cos(0.00872z) + 7.793 \times 10^{-8} \cosh(0.009344z)\right] \times \left[J_o(0.01080r) + 9.377 \times 10^{-7} I_o(0.09320r)\right] \text{ A}$$

In the reflector ($157 \leq r \leq 233$ cm, $152 \leq z \leq 288$ cm

$$\phi_t = \left\{0.1048 \sinh\left[0.03768(228 - |z|)\right] - 0.0231 \sinh\left[0.0523(228 - |z|\right]\right\} \times \left\{0.331 \times 10^{-6} I_o(0.05404r) - 9.452 \times 10^{3} K_o(0.0540r) - 0.0291 I_o(0.0174r) + 33.030 K_o(0.0174r)\right\} \text{ A}$$

And for the fast flux:

In the core $$\phi_f = \left[2.1386 J_o(0.01080r) - 1.0170 \times 10^{-6} I_o(0.0932r)\right] \times \left[\cos(0.00872z) - 3.951 \times 10^{-8} \cosh(0.09344z)\right] \text{ A.}$$

In the reflector

$$\phi_f = \left[7.0105 \times 10^{3} K_o(0.05404r) - 0.2455 \times 10^{-6} I_o(0.05404r)\right] \times \left\{0.00801 \sinh\left[0.0523(228 - |z|)\right]\right\} \text{ A .}$$

A is the "peak" flux existing in the central region of the reactor. It will be determined in the following section. These flux distributions are plotted in Figs. B-4 and B-5. The radial reflector savings amount to about 66 centimeters while the axial reflectors are not nearly so effective, saving only 28 centimeters on a dimension. This is due to the presence of the coolant tubes and sodium passages thru the axial reflectors. The thermal shield coolant tubes are located in the side reflectors, but near the outer side and hence are not very detrimental to the reflector's effectiveness.

D. Leakage Currents

The net neutron leakage from the core may be determined from the above

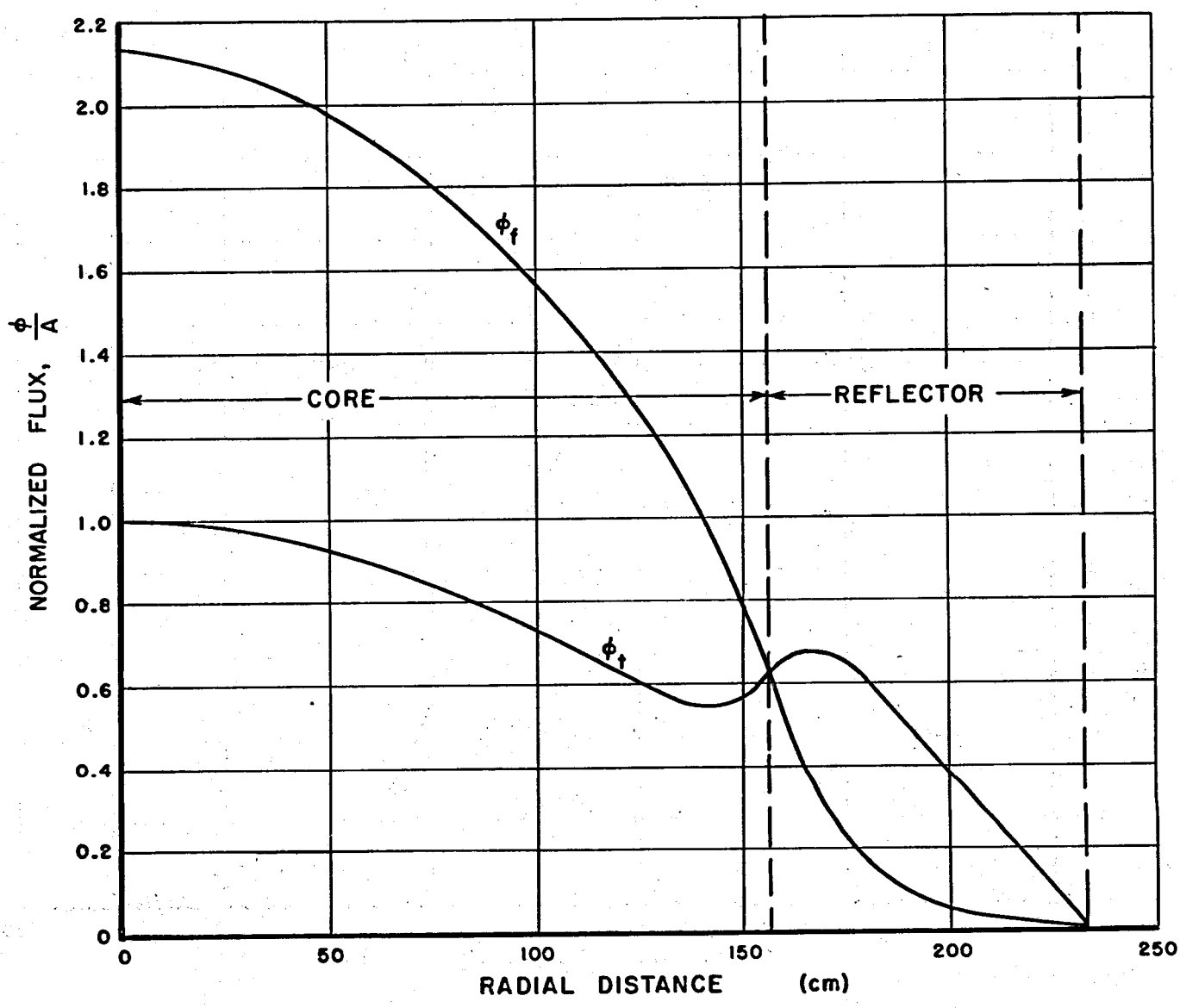


Fig. B-4. Radial Flux Distribution

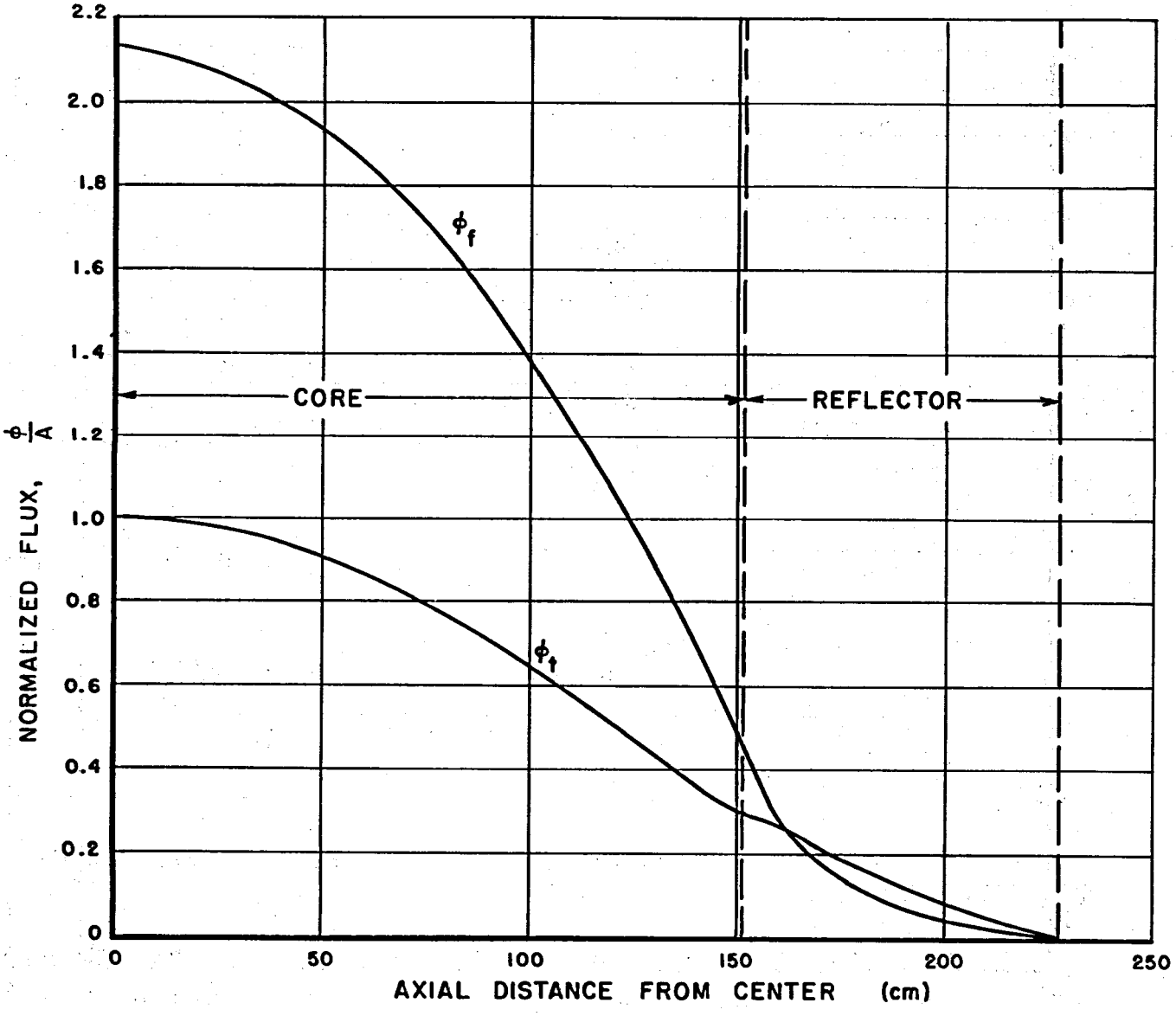


Fig. B-5. Axial Flux Distribution

flux distributions. The total fast leakage, T_f, is the sum of its radial and axial components.

$$T_f = 2\int_o^{L/2}\left[-\frac{\lambda f}{3}\frac{\partial \phi f}{\partial r}\Big|_{R_1} 2\pi R_1\right] dz + 2\int_o^{R_1}\left[-\frac{\lambda f}{3}\frac{\partial \phi f}{\partial z}\right]_{L/2} 2\pi r\, dr,$$

where

L = length of core, 304 centimeters

R_1 = radius of core, 157 centimeters

The total thermal leakage is found in a similar manner using the corresponding thermal values of λ and ϕ. In this manner the total leakage is found to be 8810 A neutrons per sec. The value of A can now be found. The number of neutrons produced per sec is equal to

$$3.1 \times 10^{10} \frac{\text{fissions}}{\text{watt-sec}} \times 167 \times 10^6 \text{ watts} \times (2.5 \times 1.047) \frac{\text{neutrons}}{\text{fission}} = 1.352 \times 10^{19}$$

Total leakage then is $\frac{k_\infty - 1}{k_\infty} \times 1.352 \times 10^{19} = 1.22 \times 10^{18} \frac{\text{neutrons}}{\text{sec}}$

$$A = \frac{1.22 \times 10^{18}}{8810} = 1.39 \times 10^{14} \frac{\text{neutrons}}{\text{cm}^2 \cdot \text{sec}}$$

We shall check this value in the following section using the intracell flux distribution for the central cell.

In a similar manner we can find the total leakage from the reflector and also the peak leakage current. These quantities are tabulated in Table B-II. The peak currents, of course, occurs at the reactor midplane z = 0, (radial peak) and along the vertical axis, r = 0, (axial peak).

E. Intracell Flux Distribution

An estimate of the thermal flux distribution within the central cell has been made using diffusion theory. The flux within the fuel channel is assumed to follow an I_o function and uniform thermal neutron production is assumed within the moderator. This yields the following distribution, which is plotted in Fig. B-6, for a power level of 167 mw:

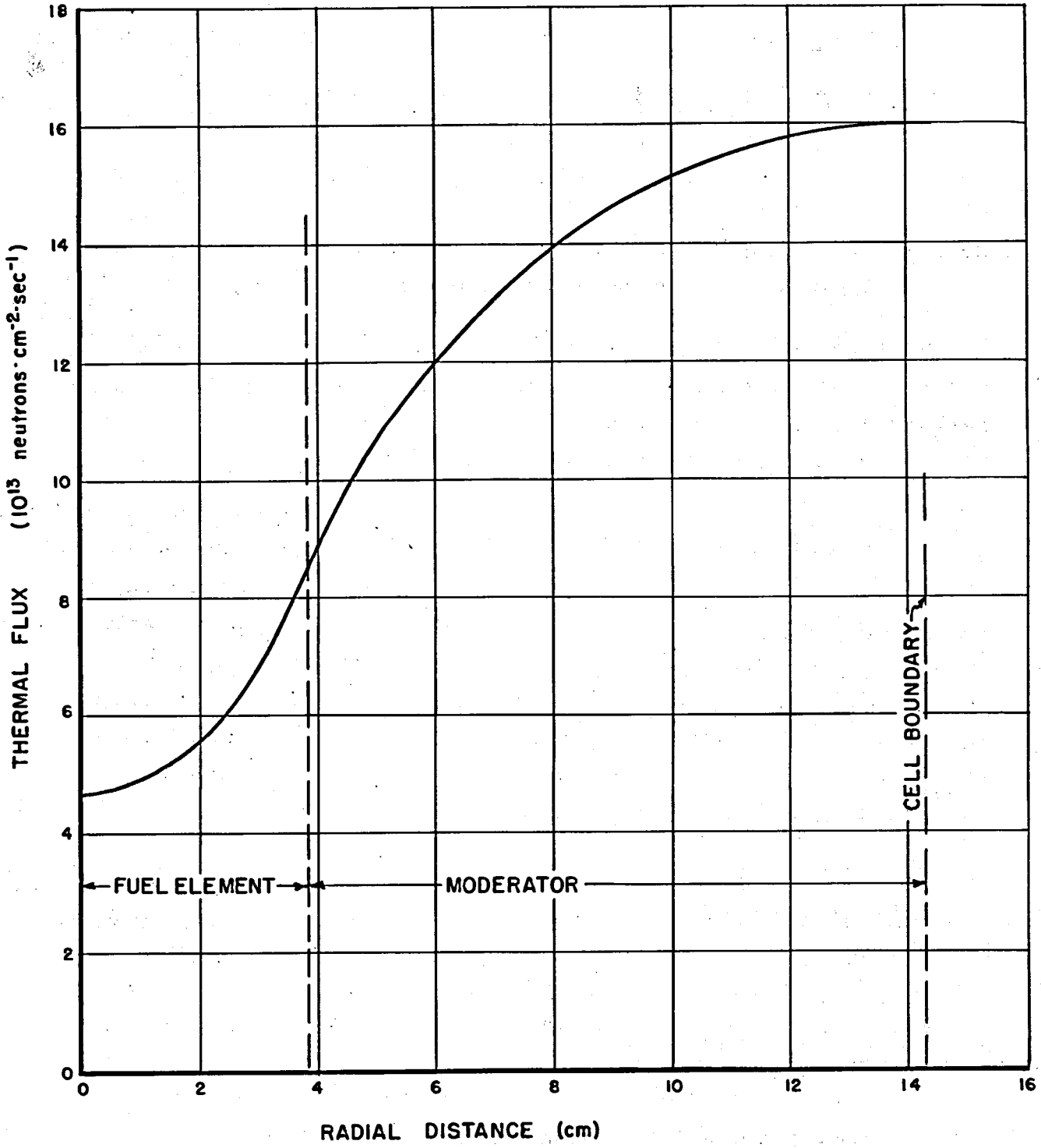


Fig. B-6. Central Cell Flux Distribution

$$\phi = 4.65 \times 10^{13} I_o (0.4362r) \qquad 0 \leq r \leq 3.88 \text{ cm}$$

$$\phi = 373.2 \times 10^{13} - 338.5 \times 10^{13} I_o (0.01547r) - 8.78 \times 10^{13} K_o (0.01547r)$$

$$\text{for } 3.88 \leq r \leq 14.33 \text{ cm}$$

By integrating over the area of the cell one can find the average flux at the center of the central cell; this quantity is the A used in the preceeding sections. The average thermal flux in the metal of the central cell (peak thermal flux in the metal) is 6.35×10^{13}. The average for the cell is $A = 1.405 \times 10^{14} \frac{\text{neutrons}}{\text{cm}^2 \text{ sec}}$. This value checks very well with the value of 1.39×10^{14} found in the previous section.

F. Temperature Coefficients

The excess multiplication factor has been determined at full power operation as a function of assumed average neutron temperature. This is plotted in Fig. B-7 normalized so that k excess at 400° C equals zero. By measuring the slope of the curve, one is able to estimate the temperature coefficient of reactivity. It is about -2×10^{-5} per degree centigrade at 400° C increasing with higher temperatures. Initially the coefficient is positive being about $+4.5 \times 10^{-5}$ at 100° C -- just above the melting point of sodium. This coefficient includes the effect of xenon poisoning and sodium density changes with temperature. At 20° C, without sodium and without xenon, the coefficient is probably positive, but it has not been estimated.

G. Control Requirements

The nine control rods will be called upon to shim for xenon, samarium, and fission product buildup as well as to compensate for the excess allowance for burnup. Also there is a slight temperature effect since the maximum multiplication occurs at a temperature lower than the operating temperature. The total control requirement is then:

allowance for Xe and fission products	0.042
allowance for U^{235} burnup	0.020
allowance for temperature	0.002
Total Δk	0.064

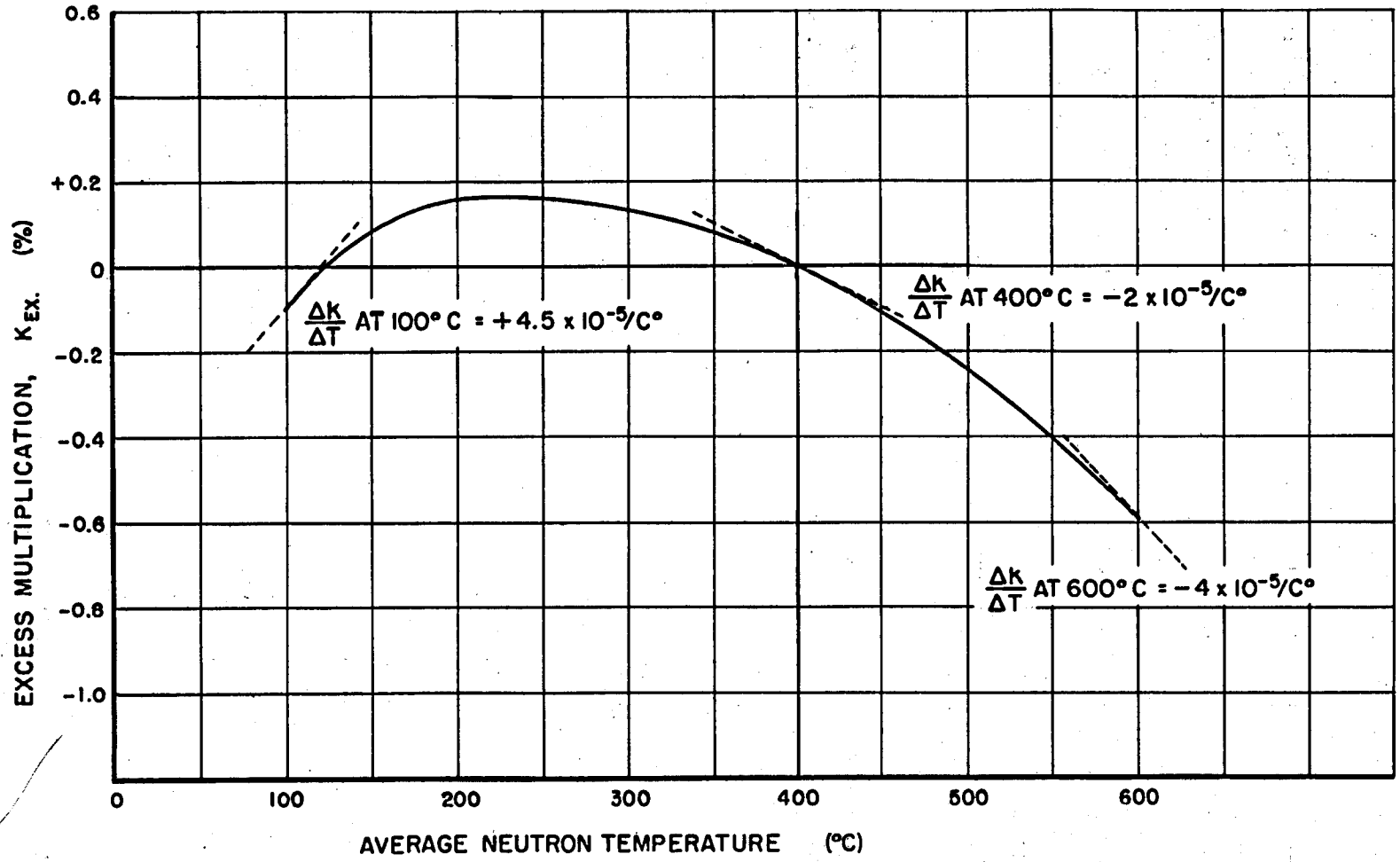


Fig. B-7. Excess Multiplication Factor vs Temperature

Calculations based upon control theory as given in Refs. 27 and 28 indicate that the nine rods will control about 0.075 Δk. The central rod is worth about 1.27 per cent, the inner ring of four rods 4.38 per cent acting along, and the outer four are worth 2.66 per cent. The total of 7.5 per cent allows for a reduction in effectiveness of 10 per cent due to shadowing. The variation in effectiveness of the central rod with amount of insertion is shown in Fig. B-8.

The requirements for the safety rod system are roughly as follows:

gain or loss of sodium	0.031
allowance for quick shutdown	0.020
Total Δk	0.051

The total Δk available in the safety rods is about 0.076. If the outer four rods are used as a safeguard bank, they will represent about 0.036 Δk.

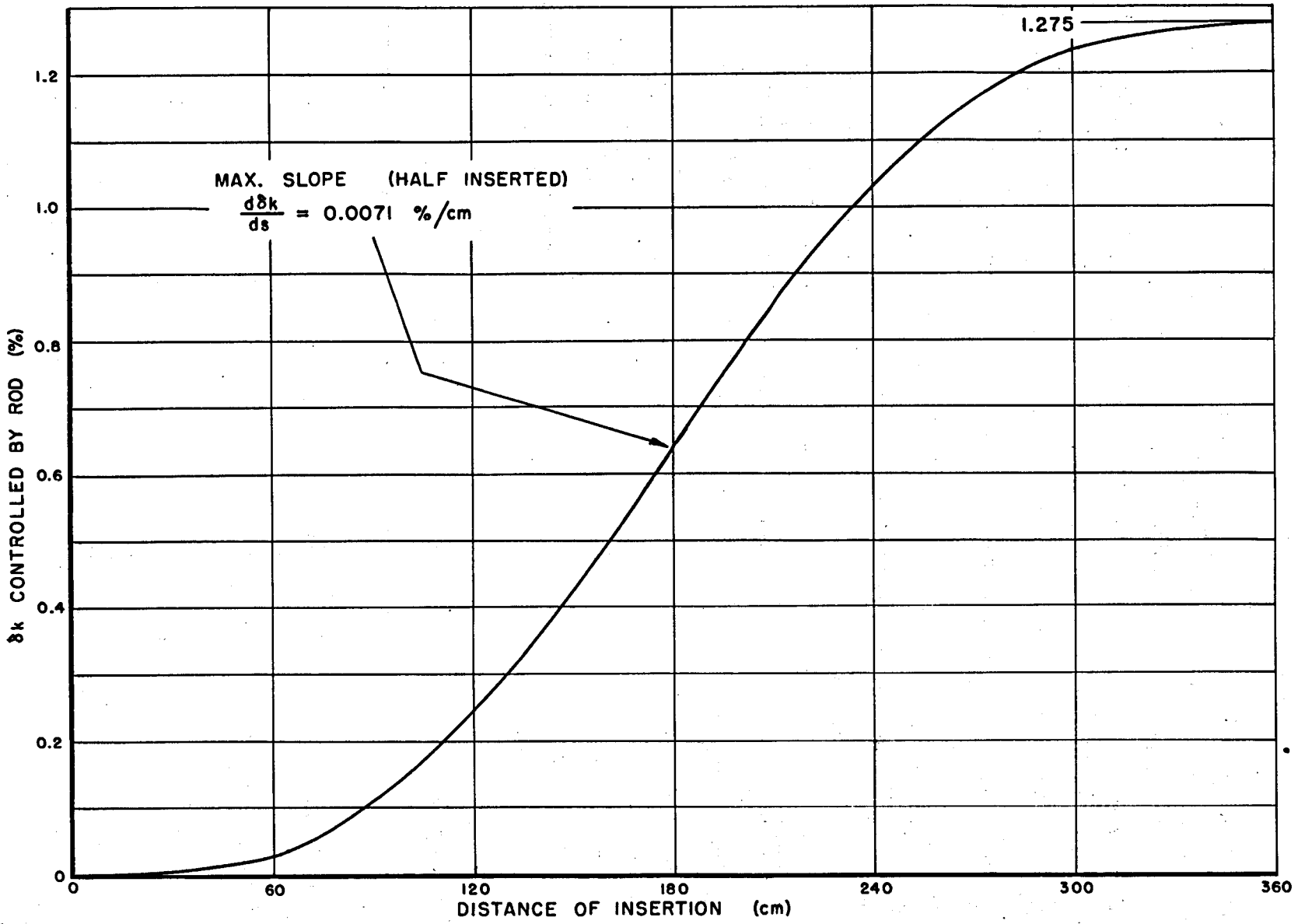


Fig. B-8. δ K vs Distance of Insertion for Central Control Rod

TABLE B-I

NEUTRON ECONOMY OF REACTOR CORE

Virgin Fast Neutrons		100.00
Fast leakage	6.03	
U^{238} resonance captures	19.16	
Thermal leakage	2.32	
Moderator captures	2.82	
Outer coolant tube captures	4.27	
Thermal captures in fuel rod	65.40	
	100.00	
Thermal Captures in Fuel Rod		65.40
U^{238} captures	13.28	
Xe and Sm captures	2.51	
Na coolant captures	1.61	
SS cladding captures	2.65	
U^{235} captures	45.35	
	65.40	
U^{235} Captures		45.35
Fissioning captures	38.30	
Radiative captures	7.05	
	45.35	

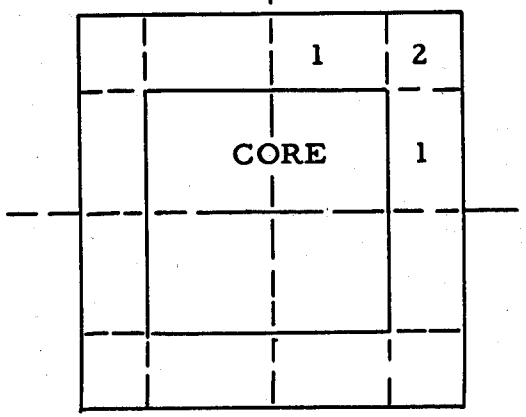
Initial conversion ratio $= \frac{19.16 + 13.28}{45.35} = 0.715$

Second generation neutrons $= 2.50 \times 1.047 \times 38.30 = 100.00$

TABLE B-II

FAST AND THERMAL LEAKAGES

	$A = 1.40 \times 10^{14}$	AXIAL			RADIAL		
		CORE	REFLECTOR		CORE	REFLECTOR	
			REGION 1	REGION 2		REGION 1	REGION 2
FAST	TOTAL LEAKAGE neutrons/sec	2740 A	103 A	10.7 A	8520 A	323 A	11.5 A
	PEAK CURRENT neutrons/sec-cm^2	0.0266 A	0.001 A		0.039 A	0.001 A	
THERMAL	TOTAL LEAKAGE neutrons/sec	295 A	262 A	181 A	-2740 A	3160 A	292 A
	PEAK CURRENT neutrons/sec-cm^2	0.0028 A	0.0025 A		-0.0125 A	0.0097 A	



APPENDIX C

SHIELDING CONSIDERATIONS

A. Gamma Radiation Level in Manifold Rooms after Shutdown

The gamma radiation level in the manifold rooms after shutdown should be low enough to allow access to these rooms in order to make repairs. It is desirable not to have to remove the fuel from the reactor to gain access to the manifold rooms. Also it is recognized that something like 2 weeks will be required for the sodium activity to decay sufficiently. The radiation level is actually composed of three distinct sources: (a) gammas due to fission products after shutdown, (b) those due to neutron activation of the steel manifolding itself, and (c) gammas due to mass transfer of steel from the active portion of the coolant channel.

Considering (a) above, calculations indicate that the radiation level 2 weeks after shutdown is about 0.28 Mev/cm^2-sec, or 8 by 10^{-3} of tolerance (7.5 mr/hr). This calculation considered the actual geomerty of the reactor and the finite size of the coolant channel. The 12-inch steel plug in the channel was also taken into account.

With regard to activation of the steel manifolding due to neutron capture, the calculation in this case is not so straight forward. The complication arises from the fact that the vast majority of neutrons which emerge from the sodium in the channel, and are subsequently absorbed in the steel manifolding, were not originally incident in that direction, but rather were probably scattered into the sodium from the concrete neutron shield and then scattered in the direction of the manifolding. This effect is more pronounced with fast neutrons than gammas since a collision in the case of the former results in larger angular deviations. A calculation of the fast flux incident on the manifolding gives about 5 by 10^9 neutrons/cm^2-sec. The calculation assumes that the neutrons penetrate the 12-inch steel thermal shield and 5 feet of sodium with negligible angular deviation. This gives what can be considered as an absolute upper limit to the leakage flux (and also to the induced activity of the manifolding). If the thermal flux is comparable to this, one might expect, with long irradiations (4500 hours) that the induced activity would be 5 by 10^5 γ's/sec per gram of

stainless steel (see Ref. 29). This corresponds to a gamma level of about 1.5 r/hr at 6 inches. It is expected that this result is high by at least one, and possibly two or more orders of magnitude. If the 12-inch steel plug were removed, the induced radiation level would be at least several orders of magnitude higher. Although this calculation is very crude, the magnitude of the result indicates that more accurate considerations are not warranted.

The third contribution to the radiation level is from mass transfer. It is only recently that the possible hazards from this phenomenon have been recognized. At present NRL, KAPL and NAA are devoting time to this problem. Several dynamic "mini-harp" tests have been completed at NAA, (Ref. 30), however, these dynamic tests serve only as a qualitative indication of the extent to which mass transfer occurs. The rather idealized flow patterns which are established in the mini-harp capsules certainly defy extrapolation to the conditions present in an operating reactor. The following conclusions however may be inferred from the data in Refs. 30 and 31.

1. Ta^{182} and Co^{60} are the principal contributors to the gamma activity.
2. Deposition of activity on initially inactive surfaces does not seem to be a function of the surface temperature alone.
3. The presence of oxygen in the sodium greatly accelerates the transfer of activity.

The data presently available are not applicable to the case of a full scale reactor system. Several additional reasons for this are: (a) the effect of turbulent flow has not yet been established, (b) the saturation properties of this phenomenon have not yet been determined, (c) the relationship between oxide content of the sodium and the enhancement of mass transfer has not been determined, and (d) the relation between temperature of the initial inactive surface and mass transfer appears to be more complicated than has been anticipated, and again, no correlation has been established. Nothing short of an in-pile test, it appears, will yield enough data to answer adequately the question of accessibility to the manifolding system.

B. Shield Requirements

The problem of the shield requirements for the sides and bottom of the reactor appears to center about the activation of the earth due to neutron leakage

from the concrete shielding around the reactor. Since it is proposed to place the reactor underground, it is obviously no longer necessary to base the shielding requirements on human tolerance levels. With this in mind, the problem resolves itself into one of a determination of the thickness of concrete necessary to reduce neutron leakage and hence activiation of the earth around the concrete to a permissible value.

Using the data in Ref. 32, but changing the constants to those for this reactor, the maximum fast leakage current out of the thermal shield is 7 by 10^8 neutrons/cm^2-sec. The thermal leakage current would be about 3 by 10^7. In this analysis we shall consider the attenuation of the fast flux and assume that the thermal flux is equal to the fast component at any point in the concrete shielding. (There is experimental evidence to verify this). Assuming 3 feet of Portland concrete (there is no reason for considering heavy concretes for this shield since gamma attenuation is not a problem), the attenuation would be roughly 10^{-3}, so that the peak emergent thermal current would be 7 by 10^5 neutrons/cm^2-sec. Using a peak-to-average ratio of 3.5, the average leakage current is 2 by 10^5 neutrons/cm^2-sec. The equilibrium induced activity corresponding to this flux is 6 by 10^6 microcuries. The volume of water required to dilute this activity to drinking water tolerance (assuming all the activity to be concentrated in the form of a single isotope) is listed below.

Isotopes	AEC Tolerance (Ref. 33)	Required Dilution Volume (values have been rounded off)
Ca^{45}	5×10^{-4} μc/ml	1×10^{10} ml
Fe^{55}	4×10^{-3} μc/ml	1×10^{9} ml
Sr^{89}	7×10^{-5} μc/ml	1×10^{11} ml

It is well to note that the provisional level of permissible water contamination has been set at 10^{-7} μc/ml for β - γ emitters. However, this applies when only the gross activity is known. After essentially all the activity has been identified, it is possible to base the maximum permissible concentrations (MPC) on the values quoted for the individual isotopes. Examining the individual values one finds only one β - γ emitting isotope with an MPC less than

10^{-5} μc/ml listed, namely, Sr^{90} - Y^{90}. It is not possible to produce this directly from natural strontium in large enough quantities to pose any problem. If we utilize an overall MPC of 10^{-5} μc/ml, then about 6×10^{11} ml are required for adequate dilution of the activity.

If surface drainage is properly controlled and the water table is at a sufficient depth, the permissible induced activity in the earth can certainly be increased at least several orders of magnitude. A simple expedient in this respect is to pave the area around the reactor, grading the pavement so that surface drainage is away from the reactor.

Summarizing in regard to the radiation level in the manifold rooms: fission product gammas do not present any hazard. Activation of the manifolding does not appear to be serious; however, the work period might have to be limited. The factor which will decide the accessibility to the system is mass transfer. At present there are not sufficient experimental data available to extrapolate to a full scale reactor system. Until more information concerning mass transport in sodium-347 (or 304L) stainless steel systems is available, the accessibility problem cannot possibly be evaluated, and hence, the question cannot be answered.

With regard to the shield requirements on the sides and bottom of the reactor, if surface drainage is properly controlled and the water table is at a sufficient depth, it appears that the shielding offered by the concrete needed for structural purposes is sufficient.

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