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DESIGN STUDY OF SMALL BOILING REACTORS FOR POWER AND HEAT PRODUCTION

by

M. Treshow

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by

M. Treshow

ABSTRACT

A design study has been made of a small "Package" nuclear power plant for the production of electric power and heat in remotely located, inaccessible areas devoid of natural fuels. The design utilizes a horizontal boiling reactor as a steam generator consistent with safe and simple equipment and a minimum building height.

A reactor design of 5-1/2 mw capacity, with a combined net electric power output of 750 kw and a heat plant output of 4500 kw, was studied in detail. Tentative cost estimates are presented on the basis of this combination. General comparisons have been made between different systems designed for either independent or combined production of 425 kw net electric power and 2500 kw available heat.

I. INTRODUCTION

The demand for electric power in remotely located, inaccessible areas devoid of natural fuels, offers the best possible background for small nuclear power plant applications.

In far northern regions, where heating as well as electric power is in demand most of the year round, the fuel supply problem is even more critical. In addition, the total height of the power plant is strictly limited to lower building structures capable of withstanding severe wind and snow storms. The so-called "Permafrost" region precludes excavations and structures below ground level.

A boiling reactor is a type of steam generator which lends itself particularly well to the combined production of heat and electric power. The reactor design presented herein is a horizontal modification of a boiling reactor. The power plant is 18 ft high and is enclosed in a Quonset-type building (Fig. 1).

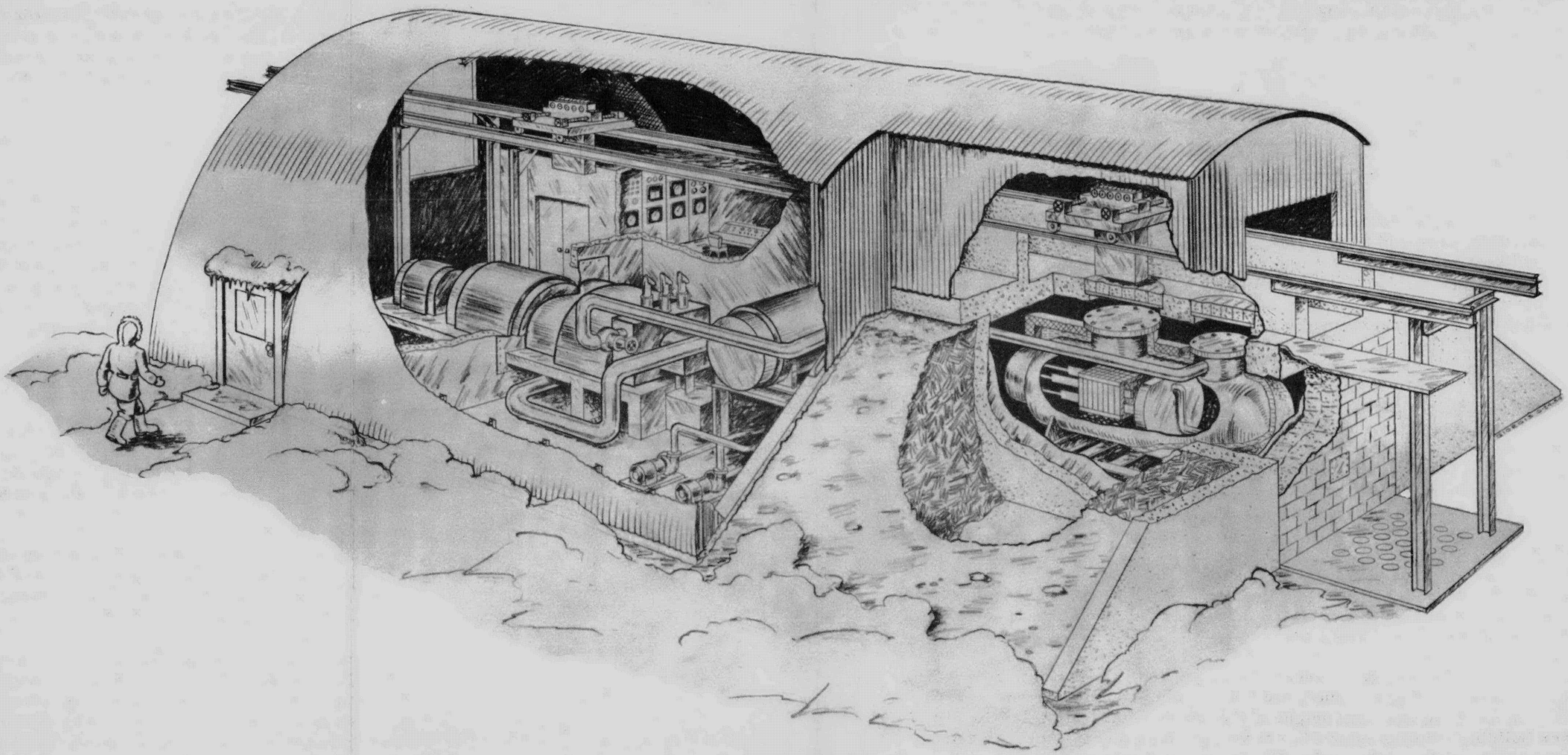

FIG. I

SMALL BOILING REACTOR
POWER PLANT

Argonne has been engaged in a series of experiments[1] concerning small, natural circulation boiling reactors. The experiments have demonstrated the feasibility and the inherent high degree of safety of such reactors. The experiments are still in progress and their final results will greatly influence the direction of further developments.

II. DESCRIPTION OF HORIZONTAL REACTOR

A. General

The reactor which is shown in Figs. 2 and 3 is designed to operate at a total power level of 5-1/2 mw. It is horizontal as far as pressure vessel and control rods are concerned. The vertical fuel plates are cooled by natural convection of the boiling water. Zirconium is considered as the base metal in the fuel elements instead of aluminum in order to permit operation with a pressure of 600 psia and a steam temperature of 486F.

B. Pressure Vessel

The pressure vessel, of USS Carilloy T-1 and clad on the inside with 1/8-inch stainless steel, is 17 ft long with an inside diameter of 5 ft. It has a net weight of not over 10 tons and is divided into two compartments - reactor compartment and steam-separating compartment - by means of a partition which forms a weir.

The weir consists of a double wall of two dished heads 6 in. apart. The interspace is filled with lead or other heavy material after the vessel is installed. The weir will, therefore, augment the shielding at the feed-water inlet end of the pressure vessel.

The reactor compartment contains the core and has a 3-1/2 ft diameter access opening through which the fuel elements can be exchanged. The height of the weir determines the maximum water level in the reactor compartment. The minimum level is determined by the position of the two water return pipes which enter the compartment at a level close to the top of the fuel plates. The changing operating conditions can, as will be explained later, influence the actual water level and thereby regulate the natural circulating rate in a manner which tends to stabilize the operation. Nevertheless, the level of the return pipe is such that it cannot drain the reactor compartment to a point where the fuel plates would be uncovered. The normal circulating rate is about 60 lb of total flow for each lb of steam leaving the core.

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[1] J. R. Dietrich, et al., "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled Water-Moderated Reactor," ANL-5323, (1954).

Under normal conditions the steam developed in the core will have ample space to separate from the water before it leaves the reactor compartment, which has a free water surface of 28 sq ft. The circulating water in the downcomer area around the core moves with an average velocity of less than 0.5 fps. Very few bubbles will be carried down with this slow stream. The steam passes over the weir with a normal velocity of 1.5 fps, together with a certain amount of moisture and overflow water.

In case of a power surge a large amount of water may be expelled from the reactor compartment according to the established safety feature typical for boiling reactors.

The steam-separating compartment serves to remove moisture and impurities from the steam and also helps to equalize fluctuations of the steam pressure. Furthermore, this compartment serves as a water surge tank for the system; the water level can be allowed to surge to a reasonable extent since this will not directly influence the operation of the reactor. The turbine condenser will therefore be operated "dry" with only a small amount of water in the hot well. The steam moves through the separating compartment with a low velocity (~0.5 fps), and the bulk of the moisture particles will settle here. The final "drying" of the steam takes place in a moisture separator which may be installed in the outlet steam dome.

The feed water is introduced through an injector nozzle located below the water level in the steam compartment from which it picks up an additional volume of circulating system water before it enters the reactor compartment and the core. The water carried in with the feed water may equal several times as much as the feed-water flow; any excess water flows back over the weir.

C. Reactor Core and Fuel Assembly

The core and the core frame for the 5-1/2 mw reactor are shown in Figs. 4 and 5. The frame structure and the bottom grate bars are made of stainless steel. The frame can accommodate 76 fuel elements, but normally it is expected to contain only 72 elements, namely, four rows of 10 and four rows of 8 elements.

The active section of the core is 60 cm or 23-5/8 in. high, 27-1/4 in. wide, and 32-3/8 in. long. The individual cell area is 3-1/8 in. x 3-1/4 in.. Each cell, or element, has 10 fuel plates, 0.070 in. thick and 2.8 in. wide including cladding and "picture frame."

The fuel bearing material is 0.040 in. thick and contains about 5-1/2% of enriched uranium. The water channels between fuel plates are 0.254 in. thick.

The possibility of using a small content of boron-10 in the fuel plates has been considered. This absorber will burn out simultaneously with the fuel and thereby help to maintain a constant reactivity.

The amount of fuel in the original reactor loading is great enough to allow xenon override in case of any period of shutdown during the specified one-year operation or more.

Figure 6 shows an individual fuel element which, geometrically, is similar to the Borax II element. The main difference is the use of zirconium in place of aluminum; only aluminum has been used for the Borax experiment which is expected to be satisfactory for steam pressures up to 300 psi. The reactor design under study is considered to operate at a higher and more economic pressure (600 psia), which makes it necessary to use materials with higher resistance to corrosion. If pressures below 300 psi are planned, aluminum may still be preferred.

The reactor core is divided into four "slabs," each consisting of two rows of elements. The three division plates consist of 1/2-in. thick zirconium spacer bars with a perforated 1/8-in. zirconium plate on each side. Between the bars are channels through which the control plates can move through the core in horizontal direction.

The total volume of the core is 308 liters. A power level of 5500 kw will therefore correspond to a power density of 17.8 kw per liter of core volume. Considered with reference to the volume of the water space in the reactor, a power density of 26.4 kw per liter is found. The continued Borax experiments, supplemented by pilot plant tests, would show whether this or even higher power densities can be achieved safely with the proposed reactor core operating with natural circulation at 600-psi pressure.

D. Control Rods

The reactor is controlled by six or more flat plates which slide on suitable liners in channels between every other row of fuel elements.

Figures 2 and 3 show the installation of control plates in connection with the control end shield plug. Each plate and its horizontal drive rod are installed and guided in a rectangular housing which can be removed and re-installed individually without disturbing the main shield plug or the other control rod guides. This handling of the control rod assemblies requires, of course, that the pressure-tight steel cover plate be first unbolted and removed.

Essentially, the control material when inserted in the core forms three continuous "curtains." When cadmium is used, practically all thermal neutrons can be captured in these curtains. It is possible that other material, such as hafnium, will have to be used which will absorb a certain amount of epithermal neutrons.

The physics calculations (Appendix A) indicate that three "curtains" using cadmium only may not be quite sufficient in a clean, new reactor core filled to its maximum level with cold water. In case criticality tests with the proposed reactor core should indicate the advisability of more control plates, the design can be easily modified to use five or more of these curtains, instead of three, without increasing the number of drive mechanisms. The present design uses one drive for each of the six control plates. It is expected that the inherent control characteristics of the power plant can be utilized to such extent that the rods will hardly have to be used for anything other than as "shim rods" or shutdown safety rods. Boron in solution may be used for extra shutdown safety when the tank is open for fuel exchange.

Three different types of drive mechanisms can be used to operate the control rods: mechanical, electromagnetic or hydraulic. The latter two have the advantage of being totally under pressure and therefore require no packing glands. All of these systems are undergoing development at Argonne.

Ordinarily, the control rods, or plates, are moved at a low speed, but in case of "scram" they are released and compression springs or counter weights will quickly pull the absorber ends of the rods into the reactor core. The arrangement is such that the reactor steam pressure would tend to move the plates in the safe direction. The springs are released by disconnecting a magnetic holding circuit.

E. Inherent Controls

1. Self-Limiting Power

The safety feature in connection with water expulsion from the core may be expected to operate particularly well on the horizontal reactor due to the fact that expelled water will not fall right back into the core. It will have to be returned gradually from the steam compartment by means of the water injector. Physics calculations have shown that if 50% of the water is expelled from the channels, this will be more than sufficient to shut the reactor down under all circumstances.

2. Self-Limiting Pressure

The normal operating density of the moderator is calculated to be about 14% lower than the non-boiling, saturated water around the reactor core. The capacity of the injector to return the overflowing water and keep the reactor compartment filled is related to the jet velocity or the amount of feed water being pumped through the injector nozzle. Inherently, if for any reason the feed-water flow is reduced, the flow of return water from the steam compartment to the reactor compartment will also be reduced.

The water level around the core will gradually recede and the natural circulating rate will be reduced, causing an increase of steam voids and a lower reactivity until the power has adjusted itself downwards and re-established the normal moderator density. This feature, among other advantages, provides the reactor with a safeguard against excessive steam pressures. If the pressure in the reactor should rise above the predetermined maximum pressure which the feed-water pump can produce, the flow through the jet nozzle will cease and no water can be returned to the reactor compartment. In the meantime, some water will be carried over from this compartment with the steam and some will drain back through the injector, all resulting in a lowering of the water level and a decreased steam output, such as described above, until the safe range of pressure has been reached.

F. Automatic Regulating

1. Power Demand

The steam by-pass valve shown in the flow diagram (Fig. 7) is operated automatically by the turbine governor in case of a change in power demand from the turbine. The steam which is by-passed will be condensed in a feed-water preheater. The purpose of this arrangement is not only to preserve heat energy, but also to produce a regulating effect on the reactor power level. A drop in steam demand for the turbine will result in an increase in the amount of steam by-passed through the feed-water heater, thereby increasing the temperature of the inlet feed water. Consequently, boiling will start at a lower level in the core and the average volume of steam voids in the moderator will be increased. This will effect reductions in reactivity and power level (steam production) until the normal moderator density has been restored and equality established between steam production and demand.

2. Pressure

It is the intention to operate the reactor with a constant steam pressure of 600 psia. If the pressure should have a tendency to rise gradually, this can be taken care of automatically as shown on the flow diagram (Fig. 7). The increased pressure will open the feed-water by-pass or proportioning valve. Less water will then enter the injector nozzle and the circulation will decrease, resulting in a reduced power output such as described above and tending to re-establish the correct pressure. To obtain the correct rate of influence, the valve opening and its spring can be adjusted during initial operations or at any other time. A certain fraction of the water will be by-passed even at the normal operating pressure.

G. Shielding

The shielding materials and arrangement are shown in Figs. 8, 9, and 10. The pressure vessel will be insulated thermally with a 3-in. layer of stainless steel wool enclosed in a casing of steel (Figs. 9 and 10). Around this casing will be placed "Boral" and 6 to 8-in. layers of lead or other heavy material which will serve as the primary shield for the reactor end of the horizontal pressure vessel. The access openings to the reactor compartment are closed with steel cover plates and inside shield plugs of 1-1/2 to 2 ft thickness.

The vessel and primary shielding are designed for installation in a rectangular open steel tank filled with cooling water containing boric acid in solution. This primary "Shielding Tank" is shown in Figs. 9 and 10. The water may be circulated to keep its temperature down during operation. During a shutdown no circulation will be necessary. On the other hand, it is assumed that the reactor decay heat will be sufficient to keep the cooling water from freezing.

The dimensions of the shielding tank are such that, if desired, it would be possible to transport the tank, with the reactor installed, on a heavy railroad flat-car without obstructing the clearances in standard railroad tunnels. Actually, sufficient shielding materials could be carried in the tank to allow the reactor to operate for brief trial runs before shipment.

In case of air transport, everything must be dismantled; no activity would be allowed in the parts.

The building space above the primary shielding tank is further shielded by a 2-ft thick slab of heavy concrete with plugs in line with the plugged access openings on top of the pressure vessel. The space above the reactor is to be closed off during normal operation. The shielding must be sufficient for loading and unloading operations after a shutdown.

In case the reactor steam is used directly in the turbine, as is assumed in the building drawings, a concrete wall about 12 in. thick is built between the turbine and condenser room and the remaining part of the building. Any prolonged access to the turbine can be permitted only when it is shut down.

The outer shielding consists of a concrete wall plus earth slopes (6 to 12 ft thick) along each side of the shield tank. The earth slopes augment the concrete shielding and also serve as an anchor for the rest of the building. The ends of the tank are shielded with walls and plugs of heavy concrete 3-1/2 ft thick.

III. POWER AND HEATING CYCLE

The energy supplied by the reactor is considered to be used for two different purposes: power for the electrical system and heat for the heating system. The required power and heat, the ratio between these values, and the optimum type of heat-and-steam cycle will depend on the circumstances.

In general, it can be said that the power steam for the turbine could be either primary reactor steam or it could be secondary steam produced indirectly in a heat exchanger by heat from the reactor steam. The heating plant, whether it uses steam or hot water, must always operate as a secondary cycle with indirect heat since no radioactivity can be permitted here.

If the two systems are combined in such a manner that the exhaust steam from the turbine furnishes the heat a very high total efficiency can be obtained, particularly so in cases where circulated hot water instead of steam can be accepted as the heating medium.

For the purposes of comparison, four different systems have been studied briefly; they are identified as Cases I, II, III, and IV and are discussed in Appendix C. All four systems were designed for a net electric power demand of 425 kw and a useful heat demand equivalent to 2500 kw. In each case, it was considered that pump and other auxiliary power required for plant operation would need 75 kw so that the total electric power demand would be 500 kw.

The primary steam pressure was taken to be 600 psia. This pressure is considered in connection with the use of zirconium or Zircaloy-2 in the fuel plates and cladding. If aluminum is used, such as in the present BORAX experiments, the pressure must be kept below 300 psi. The reactor costs will be reduced, but the steam cycle efficiency will be less favorable. Individual circumstances will determine whether aluminum or zirconium should be preferred.

The engineering data, plant layout, and tentative cost estimates are based on a combined power and heating system similar to Case IV, but operating at higher power levels, namely, 5-1/2 mw reactor power producing 850 kw generated electric power (750 kw net available power) with the equivalent of 4-1/2 mw heat to be utilized in the heating system. The flow diagram is shown on Fig. 7.

The over-all plant efficiency for this system is:

$$\frac{750 + 4,500}{5,500} \times 100 = 95\%$$

Most of the features shown on the flow diagram have been described previously in connection with the control and safety aspects of the system. The following details, also shown in Fig. 7, pertain to the secondary heating system:

The auxiliary radiator serves to dissipate heat from the circulating water in the event the heating system is not using sufficient heat to keep the condenser pressure down to 5 psig.

The auxiliary burner serves to keep the heating system warmed up whenever the reactor is shut down. It can also serve for indirect preheating of reactor water before the startup.

IV. PLANT LAYOUT

A. Description of Plant

The reactor building and equipment for the combined 5-1/2 mw reactor system are shown in Figs. 8, 9, and 10. The turbine, condenser, and feed-water heater, as well as radioactive piping, are enclosed in an isolated room which is well ventilated and may only be entered intermittently. The room walls are either poured concrete (12 in. thick) or concrete blocks to shield against gamma radiation from the equipment.

The heating system pumps and tanks are located in an accessible room. Sufficient water is stored in the tanks to keep the system operating for a few hours in case of emergency or until an auxiliary oil burner (Fig. 7) can be started to keep the water hot.

The long open space in front of the reactor is used for control rod inspection and maintenance.

Figure 10 (b) is a sectional view of the turbine building. The building is installed on a continuous concrete slab and steel frame which, in turn, is supported on piers drilled down in accordance with the type of soil upon which the plant is constructed.

Figure 10 (a) is a cross sectional view of the reactor. The rectangular "shielding tank" contains the primary shielding which is water cooled. The thickness of the shielding is equivalent to 8 in. of lead, although iron, or even iron ore may be used. The cooling water will contain boric acid in solution.

The gallery above the reactor is used only during shutdown periods while the fuel is being exchanged. The concrete slab and shielding above the reactor give sufficient protection for this purpose.

close

B. Unloading and Loading of Fuel

delete

Exchange of fuel is to be made once a year. The reactor must then be shut down. With reference to the encircled numbers on Fig. 11, the ~~unloading~~ procedure is as follows:

The gasketed, pressure-tight cover plate over the reactor core is removed, while the heavy eccentric shielding plugs are left in place. It would hardly be possible to turn these plugs while they rest on the shoulders;

friction forces would be too big and the surfaces would gall. A special tool, in form of an eccentric turn table (1) with ball bearings, is therefore brought in by the crane and placed over the two plugs (2). Six lifting screws (3) are tightened between this turn table and the plugs, whereby these are individually lifted 1/2 in. so as to be suspended from the ball bearing plates. They can now easily be turned into any position or combination of positions so that the 4-1/2-in. unloading hole (4) can line up with any one of the fuel elements or with any one of the compartments of the coffin (5). The coffin is mounted on a movable slide on the unloading trolley and moves in a rectangular coordinate system.

The coffin is aligned with the 4-1/2-in. shield plug (6) which is lifted up into one of the compartments; then, successively, four of the fuel elements (7) are lifted into the other four compartments. During this operation the opening in the shield plug is kept covered by the lower surface of the lead coffin itself so that little radiation can escape. The coffin is then moved back until the plug can be set down again on its place in the main plugs (2) before the trolley is taken away and unloaded. Each compartment is equipped with a bottom gate (8) which will be kept closed during the transportation.

The two screw spindles (9) are interconnected with chain and sprockets so that they can be driven from the same wheel. Each of the individual hoists (10) are driven from a worm drive (11). The hoisting medium (12) consists of two separate steel bands, side by side. Each winds up on its own half of the hoisting drum. The catching tool or gripper (13) consists of two parts, the coupling (14) and the locking sleeve (15). The coupling lifts the tool. A pull on the sleeve will release the fuel element provided it is in place where its weight is supported. The weight of gripper and fuel element can be shifted from the coupling to the sleeve while the two bands are operated from the same drum. This shifting of the load is done by means of the small roller (16) and the guide roller (17). The small roller is mounted on a straight shaft (18). When the shaft is pulled to the left (on the drawing) the band that is attached to the sleeve (15) will be tightened and the fuel element released. This is done when new fuel is being loaded into the reactor.

The gripper and the fuel element hang inside a "bell" or sleeve guide (19) which has a square cross section to match the fuel element. This guide must be kept oriented in the correct direction while the element is being lowered into its place in the core. This is taken care of by the coffin extension sleeve (20) which has a round cross section with four grooves so that the square sleeve (19) will be guided while sliding down through it. The square flange at the top end of the coffin extension sleeve (20) will be turned into a position square with the coffin and the reactor core. This will insure the correct orientation of the fuel element.

The above described system can be built equally well to handle blocks of several fuel elements built together instead of handling just one at a time.

V. ENGINEERING DATA

A. Reactor

1. Performance

Power level, mw	5.5
Power density in core volume, kw/liter	17.8
Steam pressure, psia	600
Steam temperature, F	486
Steam production, lb/hr	20,000
Recirculation rate (estimate)	60:1
Average density reduction due to boiling, %	14.2

2. Core

Base metal, fuel plates	$Zr + U^{235}$
Length, in.	32-3/8
Width, in.	27-1/4
Active height, in.	23-5/8
Number of elements (3-1/8 in. x 3-1/4 in. x 23-5/8 in.)	72
Number of plates per element	10
Total thickness of plates, in.	0.070
Thickness of zirconium clad, in.	0.015
Water channel gap, in.	0.254
Cooling surfaces, sq ft	620
Fuel per element, gm U^{235}	139
Average heat flux, Btu/(hr)(sq ft)	31,000
Average thermal neutron flux in fuel plates at middle of fuel cycle	2.1×10^{13}
Metal to water volume ratio (total core)	0.48

	New Charge	At end of cycle
Total fuel content, kg	10.0	8.0
k_{eff}, cold (no xenon)	1.18	1.15
k_{eff}, operating temperature, no boiling (equilibrium xenon at full power)	1.09	1.06
k_{eff}, boiling at full power, 600 psi (no xenon)	1.094	1.064
k_{eff}, boiling at full power (equilibrium xenon)	1.06	1.03
Fuel burnout in 12 months at 70% operating rate (leaving 0.25 kg excess burnout fuel), kg		1.75

3. Pressure Vessel

Material: USS Carilloy T-1 with 1/8 in. stainless steel clad

Tank diameter, ID, ft	5
Overall length, ft	17
Weight, tons	10

B. Power System

Total steam flow, lb/hr	20,000
Turbine power, kw	900
Generator output, kw	850
Net electric power, kw	750
Throttle pressure, psia	600
Turbine exhaust pressure, psia	20
Condensate temperature, F	228

C. Heating System

Utilized heat (4-1/2 mw), Btu/hr	15,400,000
Hot water temperature, F	215
Return water temperature, F	170
Circulating water, gpm	700
Condenser - heater surface, sq ft	1,000

VI. POWER COST ESTIMATE (TENTATIVE)

A. Background for Estimate

The following cost estimate is based on several arbitrary assumptions; the actual costs will depend on circumstances and policies which may vary in individual cases.

1. Development Work

It is recognized that a substantial sum is required for development of these power units before they are ready to be produced. The estimate does not include this expenditure; rather, it is assumed that all development work has been completed and that a "frozen design" is on hand before the contract is let.

2. Plant Location

The plant is designed for a minimum of construction work on the site. It is also specified that no single part can weigh more than 10 tons in case air transport should be required.

It is considered that the plant will be remotely located and no allowance is made for the cost of the building site.

3. Instrumentation

It is assumed that the reactor is inherently safe against power surges and excessive steam pressure and also that the steam system is substantially self-regulating. These circumstances simplify the operation and help to lower the expenditure for elaborate and delicate instrumentation.

4. Operating Data

Net electric power output, kw	750
Available energy for heating system, mw	4-1/2
Ratio of heat to power	6:1
Reactor power, mw	5-1/2
Fuel burnup per mwd, gm	1.25
Plant operating rate or utilization factor, %	70
Average time between reloading, months	12

5. Net Power Production

750 kw at 70% operating rate, average kw	525
Net power output per year, kwh	4.6×10^6

6. Fuel Burnup

Fission power produced in 1 year at a 70% operating rate 5.5 x 365 x 0.7, mwd/yr	1,400
Fuel consumption, 1,400 x 1.25 gm/yr	1,750

It is expected that not all fuel elements will have to be exchanged each year, as some will be irradiated considerably less than others. Neither the final loading schedule nor the physics constants for subsequent years of operation have been calculated.

In order to obtain a fair idea of the average fuel cost the following arbitrary assumption has been made:

One-third, or 24, of the fuel elements can be left in the reactor core to be used for a second years' operation provided that the fuel charge in the 48 new elements will be increased sufficiently to replace the total 1.75 kg which has been burnt out before shutdown. The charge in each of the replacement elements is considered to be 149 gm instead of the 139 gm in the original elements.

The cost of new fuel is taken to be $20 per gm. Chemical re-processing of fuel is assumed to cost $4 per gm. Zirconium and fabricating charges are assumed to be $1,000 per fuel element.

6/7. Credit for Heating System

Heating plant capacity	4.5 mw
Average usage when operated at rate of 70%	3.15 mw
Useful heat output per year	2.76×10^7 kwh/yr = 9.4×10^{10} Btu/yr

This heat might have been produced by 5.4×10^6 lbs of fuel oil at 17,500 Btu/lb useful heat value in the boiler. Assuming the cost of oil is 20 cents per gal, or 2.6 cents per lb (or $8.40 per barrel at 42 gal), this would be a relatively high price in the United States, but a very low price in Artic areas. It corresponds to 5 mills per kwh heat.

8. Investment and Capital Charges

The plant investment includes building and all power and heat producing equipment with switch gear for the power and circulating pumps for the heating system. It does not include pipe lines and radiators for the buildings which use the produced heat or power.

The investment for plant and equipment ($530,000) is charged at the rate of 15% per year whereas the inventory of fuel elements and fuel is charged at the rate of 6% per year. Depreciation of the fuel is, of course, taken care of by the cost of yearly replacement of fuel elements.

9. Plant Investment, Summarized

Reactor Component		
Reactor, including erection (fuel elements not included)	$65,000	
Reactor shielding	20,000	
Fuel loading and unloading equipment, including storage	30,000	
Reactor instrumentation	20,000	
Miscellaneous	10,000	
		$145,000
Power and Heating Component		
850-kw turbo-generator set (erected), including condenser and switch gear	$115,000	
Steam and water system	50,000	
Turbine room shielding	6,000	
Power and heat system instrumentation	8,000	
Miscellaneous	6,000	
		$185,000
Building Component		
Quonset building and foundation slabs	$38,000	
Facilities	10,000	
Tools and miscellaneous	6,000	
		$ 54,000
Subtotal		$384,000
Contingencies (15%)		58,000
Contractors' fees (20%)		88,000
Total plant investment		$530,000

B. Yearly Costs

1. Capital Charges

15% of plant investment (0.15 x 530,000)	$ 79,500	
6% of zirconium fuel element cost (complete core, 72 elements at $ 1,000)	4,300	
6% of new reactor fuel inventory (10,000 gm at $20 per gm)	12,000	
Total investment charges		$ 95,800

2. Fuel Charges

48 new fuel elements, fabrication and cost of zirconium	$ 48,000	
New fuel replacement, 1,750 gm at $20 per gm	35,000	
Chemical re-processing of remaining fuel from 48 elements, 48 x 149 - 1750 = 5,400 gm at $4 per gm	21,600	
Total fuel cost		$104,600

3. Operating and Maintenance Costs

Man Power	$ 32,000	
Materials	25,000	
Total operating costs		$ 57,000
Total yearly expenses		$257,400
Credit for heating system (2.76×10^7 kwh/yr at 5 mills)		$138,000
Chargeable to electric power, $/yr		$119,400

C. Cost of Power

1. Cost of Electric Power ($119{,}400/4.6 \times 10^6$ kwh)	26 mills/kwh
2. Cost of Energy for Heating System	5 mills/kwh
Ratio of electric power cost to assumed value of heat	5.2:1

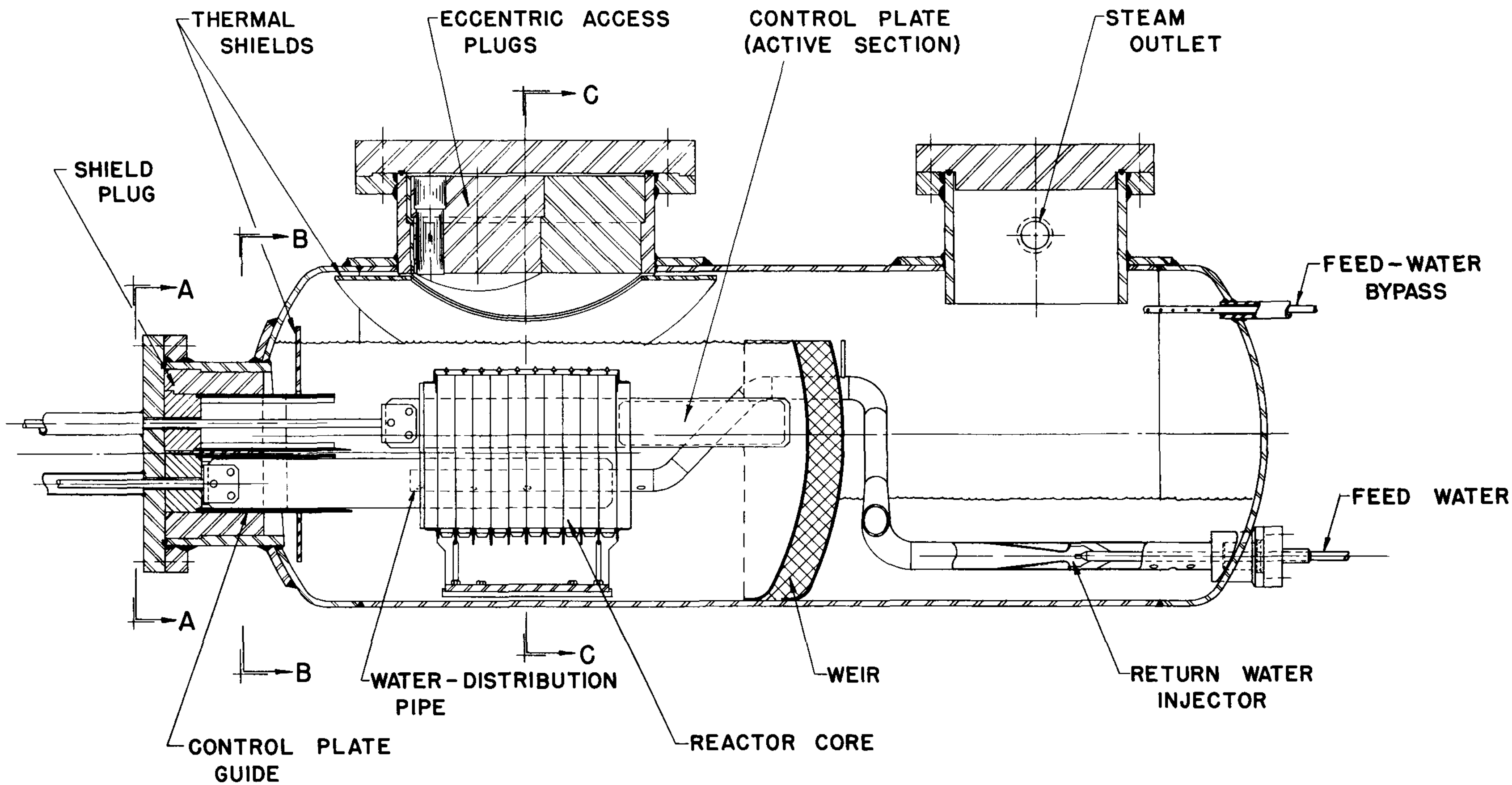
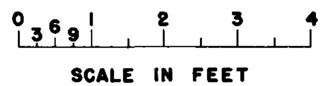


FIG. 2
HORIZONTAL BOILING REACTOR



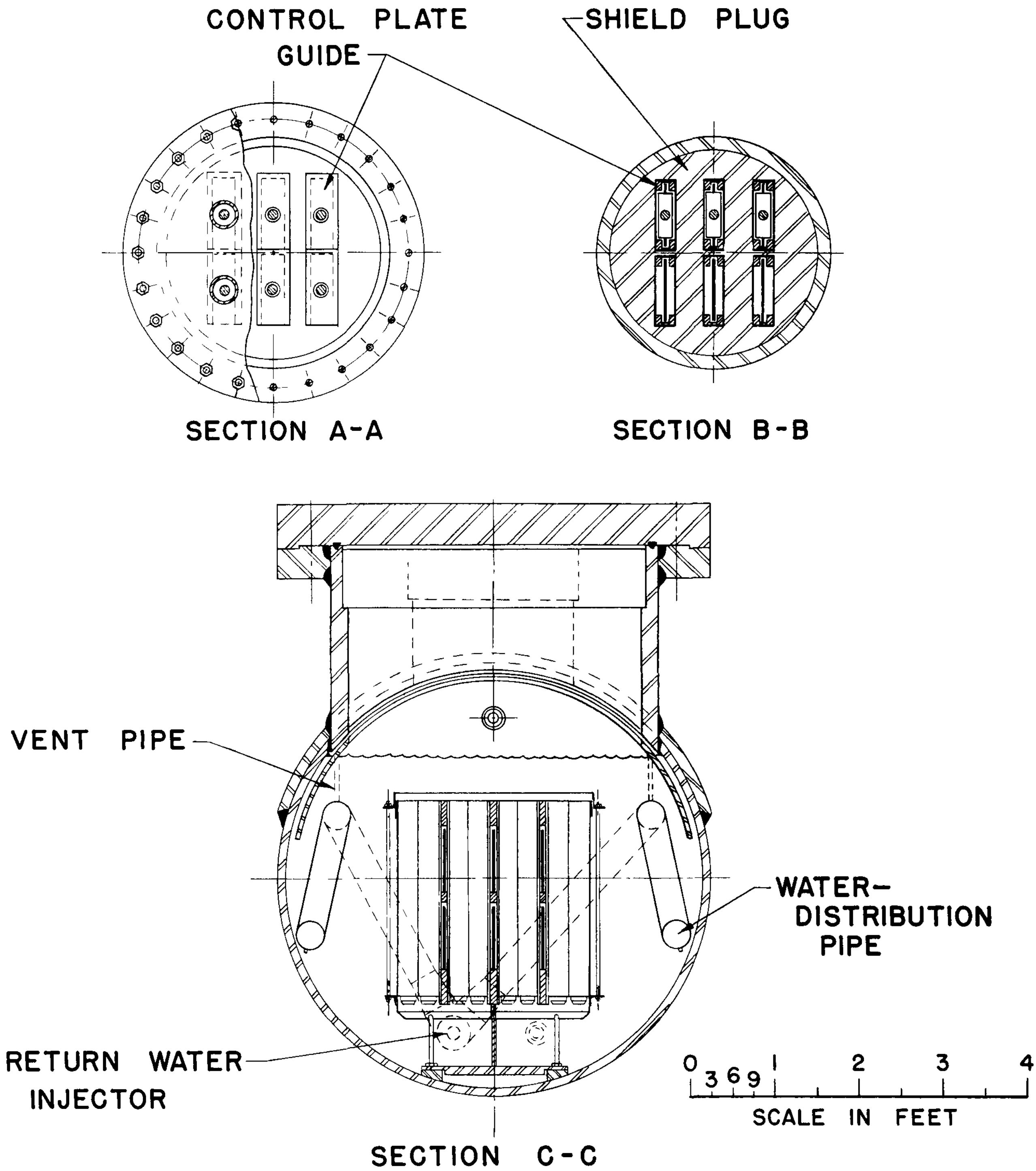


FIG. 3

SECTIONAL VIEWS OF HORIZONTAL BOILING REACTOR

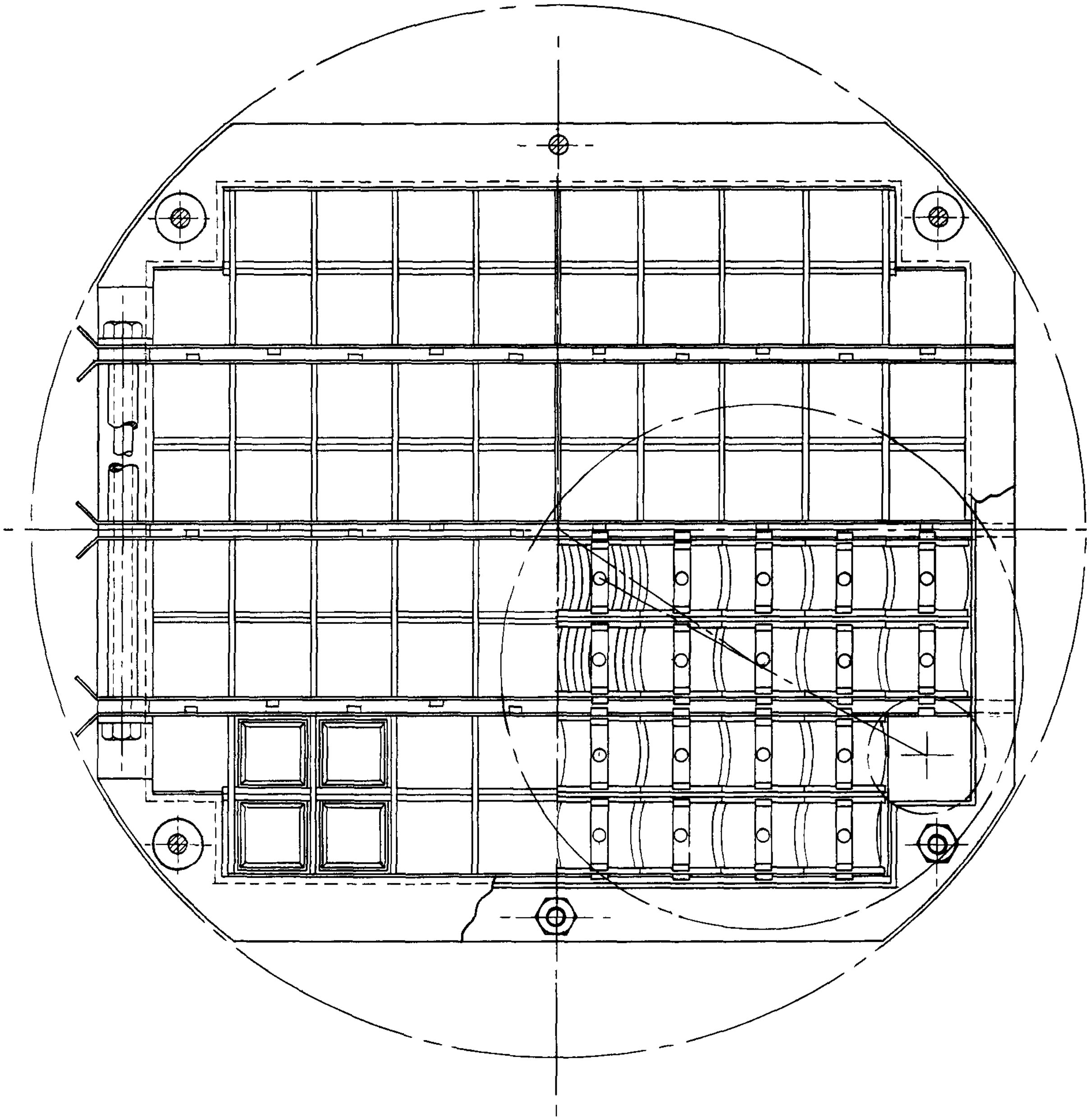


FIG. 4

REACTOR CORE
PLAN VIEW

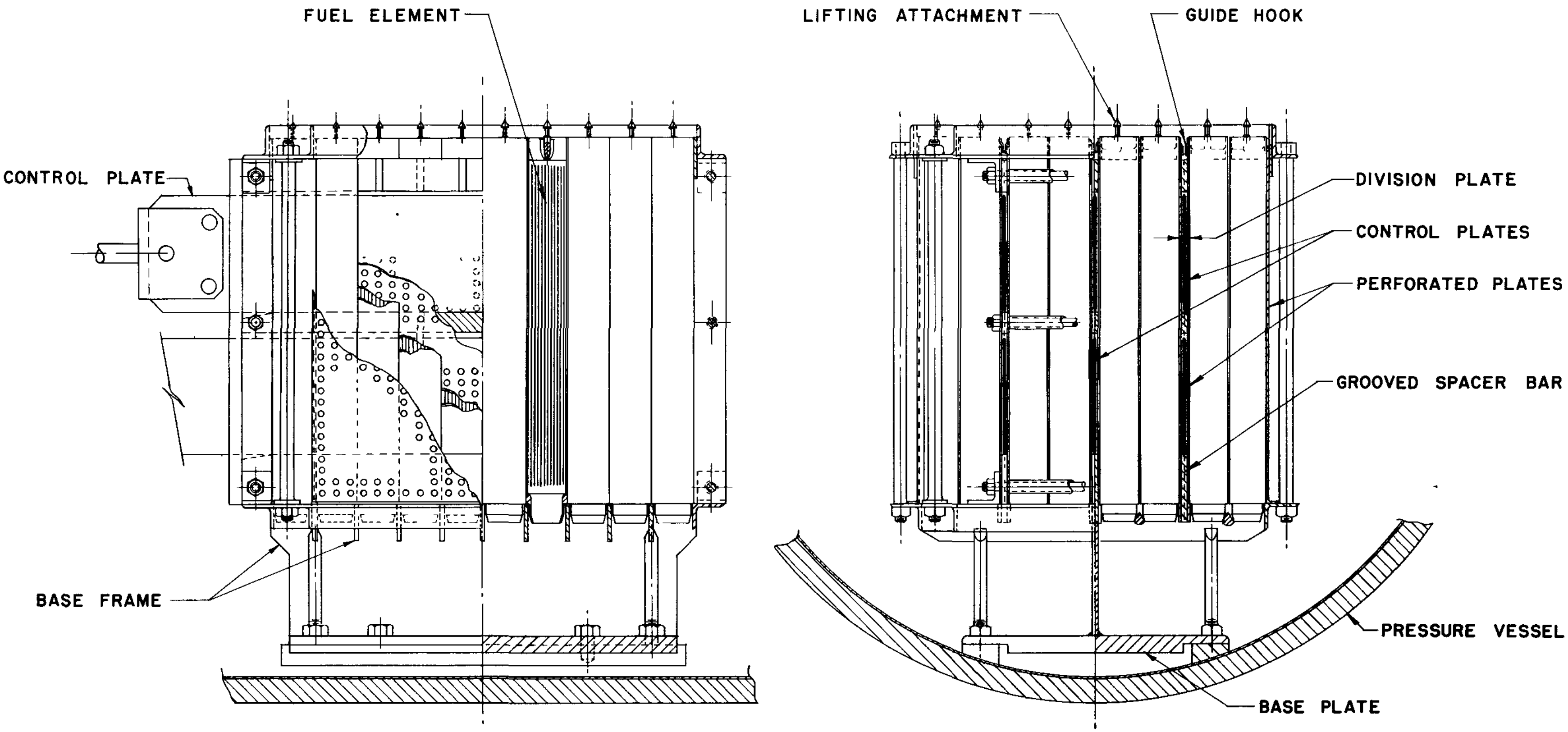


FIG. 5
REACTOR CORE (ELEVATION)

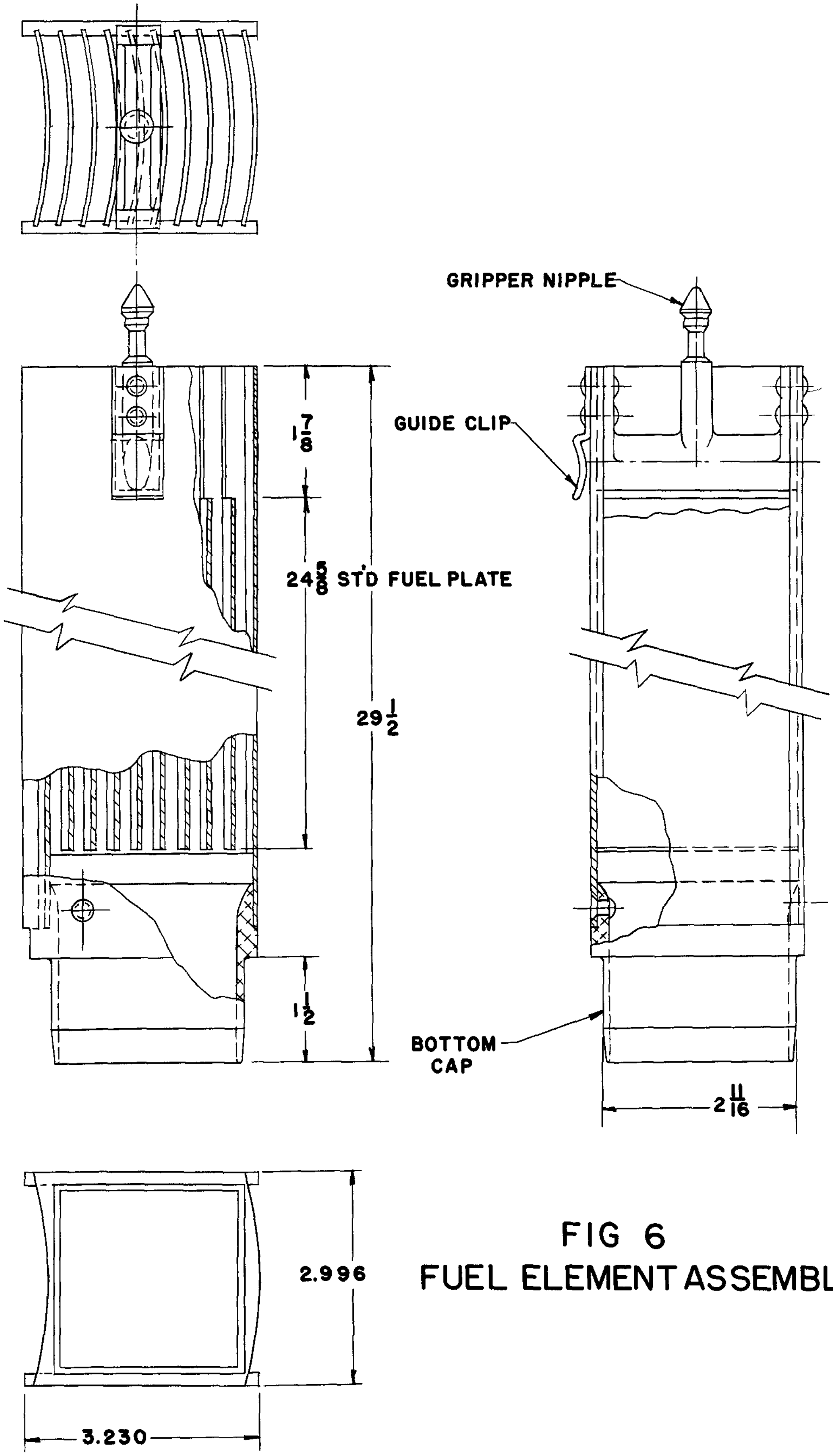


FIG 6
FUEL ELEMENT ASSEMBLY

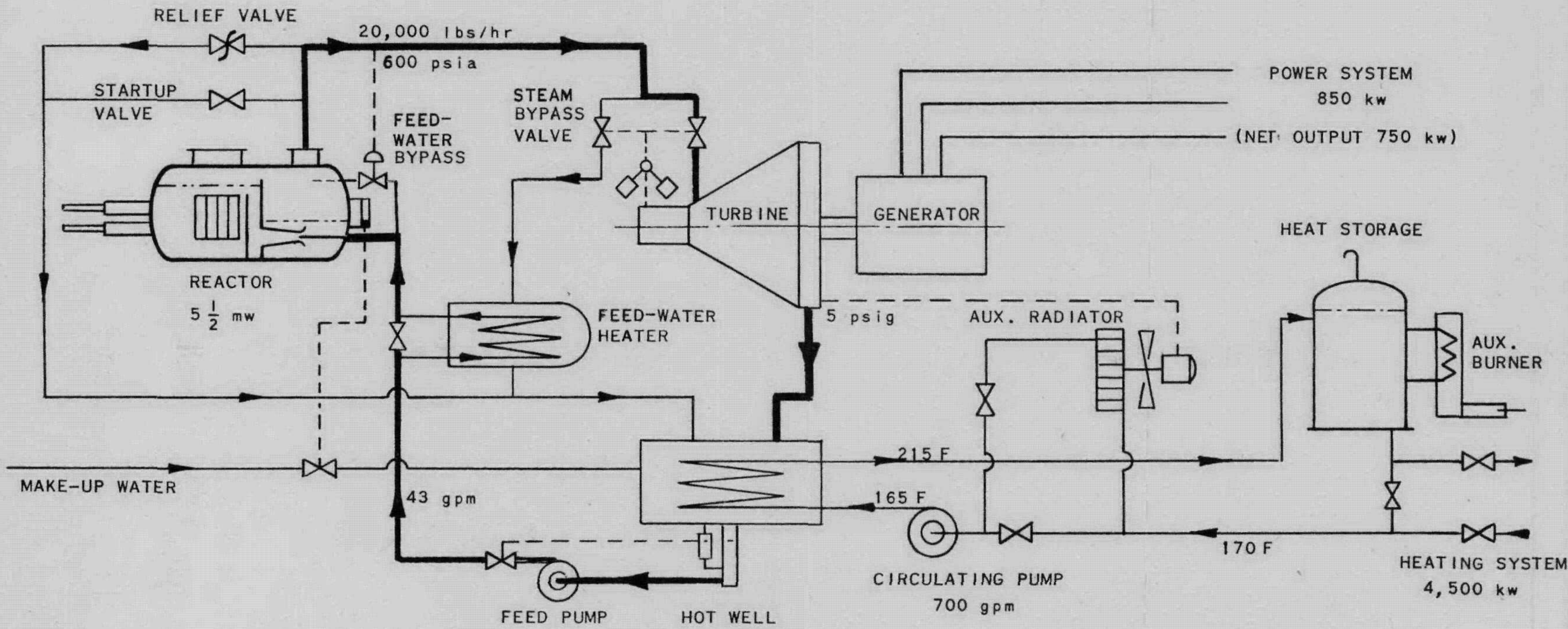


FIG. 7
FLOW DIAGRAM FOR
COMBINED POWER AND HEATING PLANT

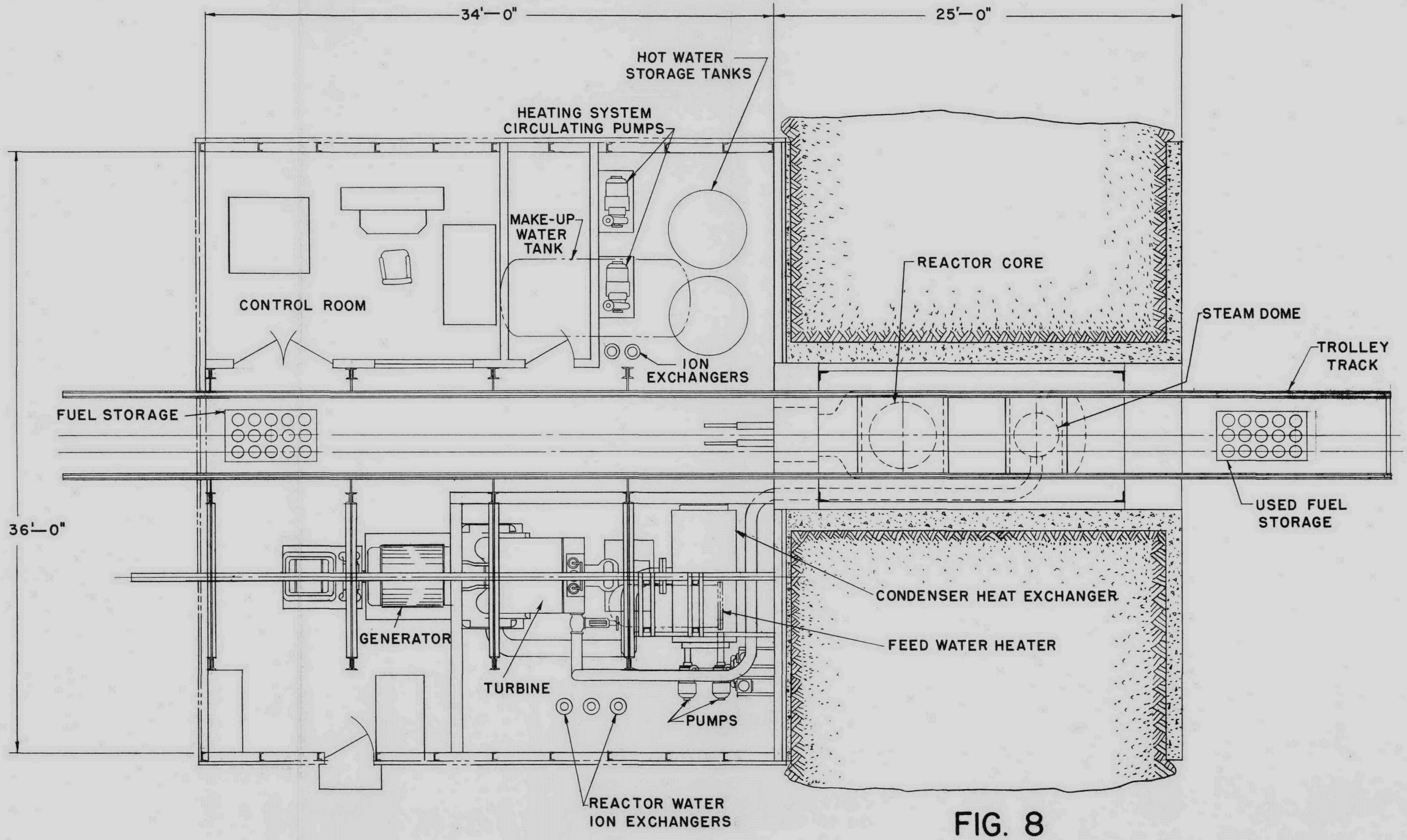


FIG. 8
BUILDING & EQUIPMENT
(PLAN VIEW)

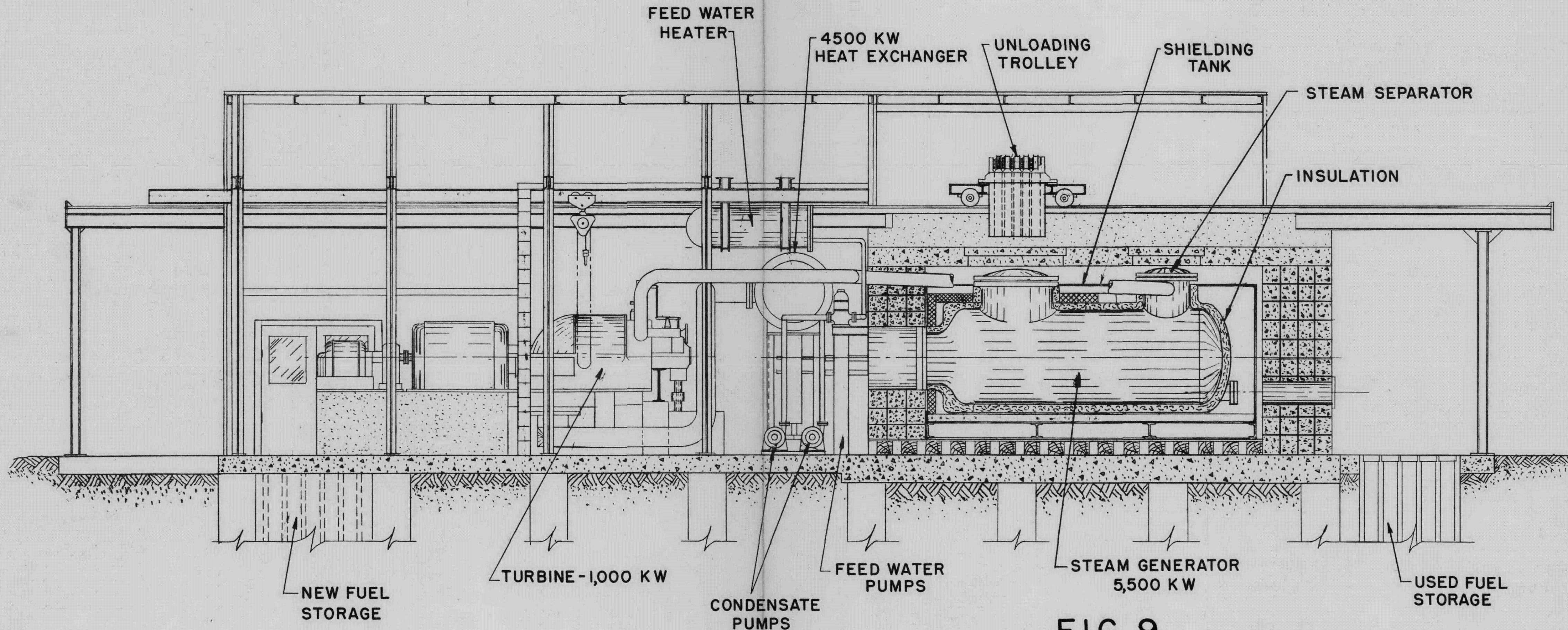
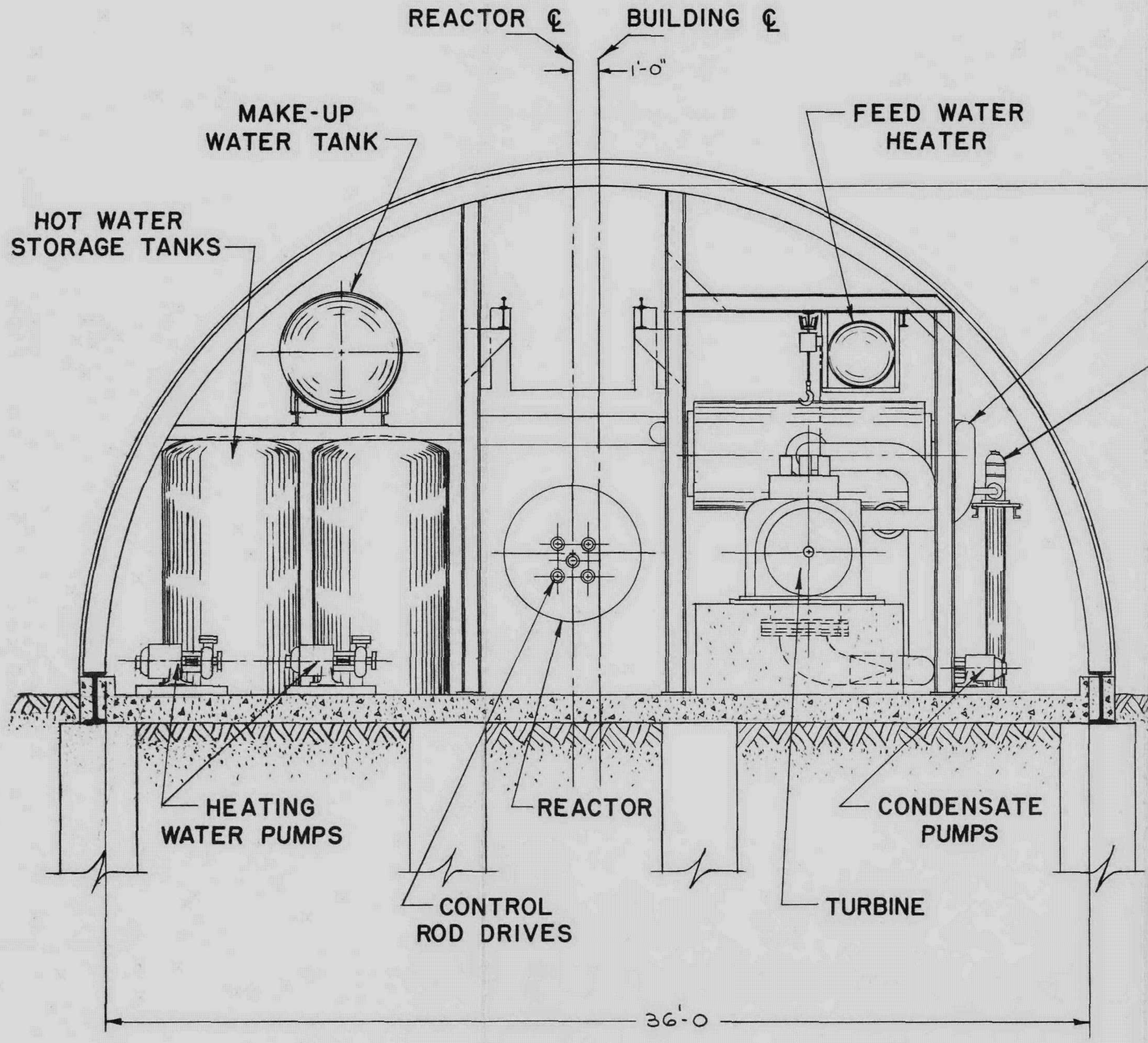


FIG. 9
BUILDING & EQUIPMENT
(ELEVATION)



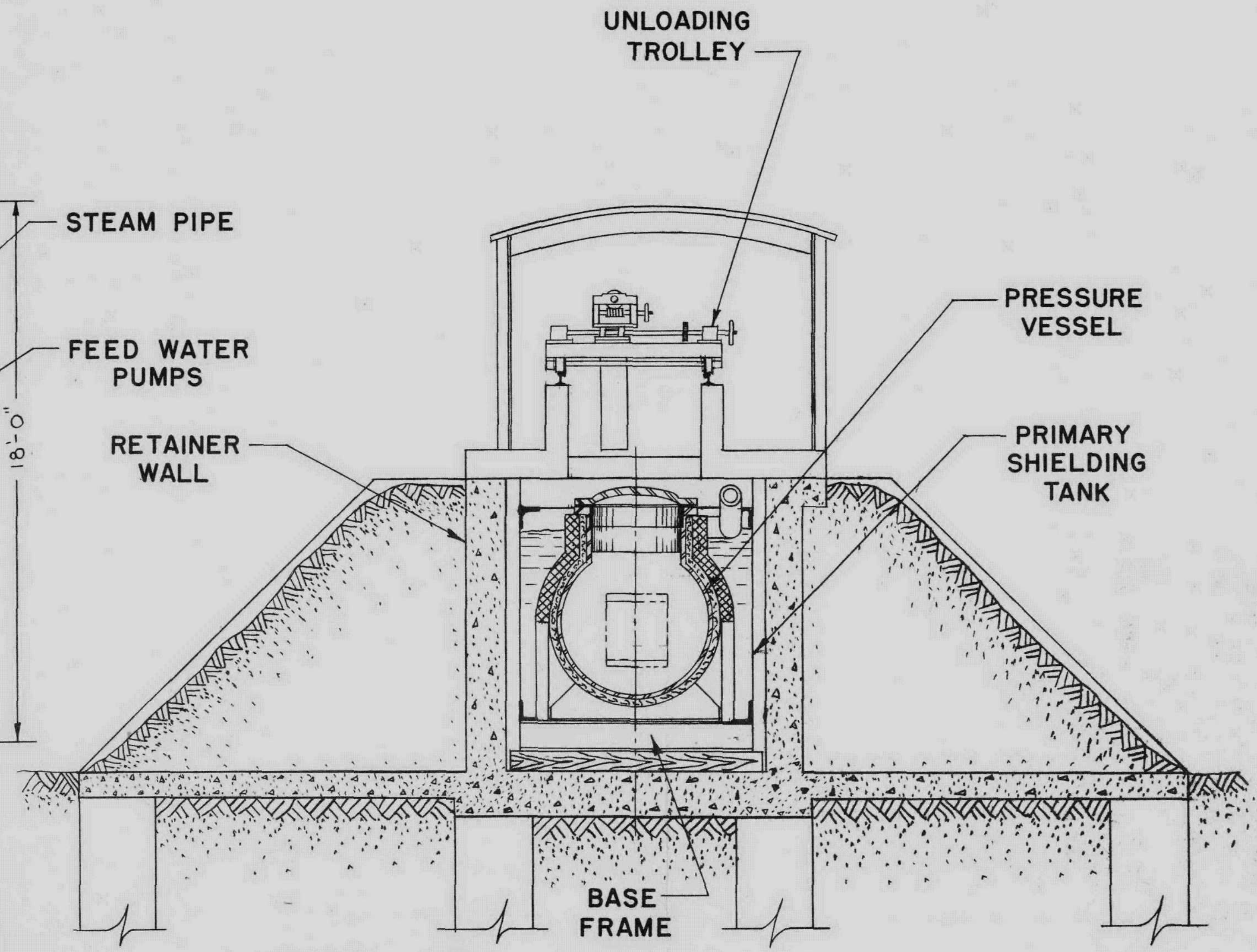


FIG. 10
BUILDING AND EQUIPMENT
(CROSS SECTION)

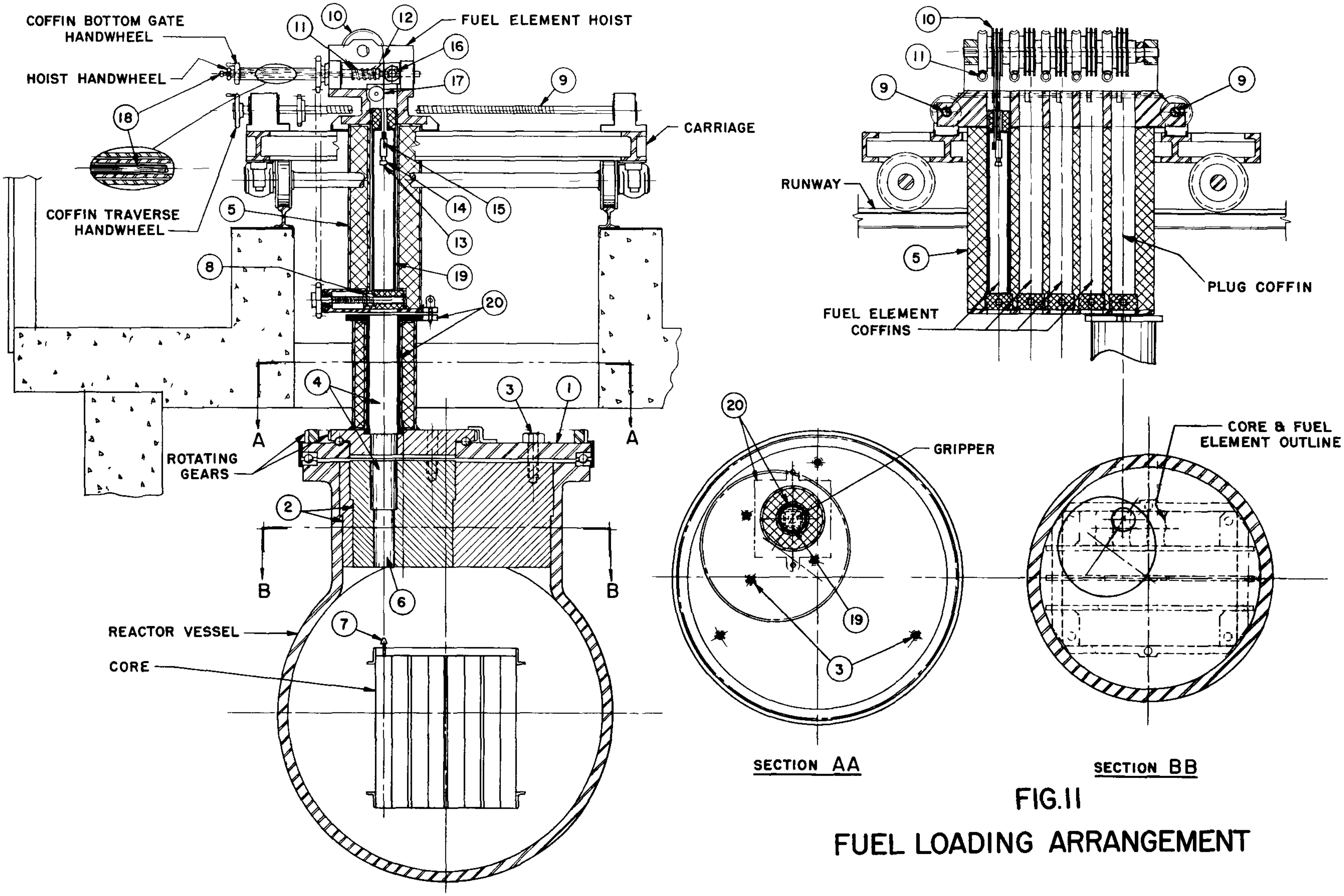


FIG.11
FUEL LOADING ARRANGEMENT

ACKNOWLEDGEMENT

The author gratefully acknowledges the constructive comments and suggestions by members of the Reactor Engineering Division and Laboratory Director's Staff; in particular the assistance of E. E. Hamer in connection with cost estimates; J. W. Butler and M. Grotenhuis for their cooperation with regard to shielding problems; and D. H. Shaftman, who performed the Physics Calculations which appear as Appendix A.

APPENDIX A

PHYSICS CALCULATIONS

by

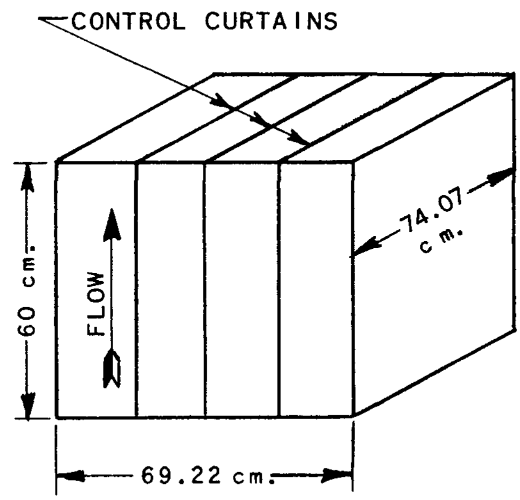
D. H. Shaftman

The core structure of the reactor described in this report is similar to that of the BORAX assembly at Arco, Idaho. The analysis presented in this appendix was directed toward an investigation of the nuclear properties of this modified BORAX reactor design with the intention of proving its feasibility for the proposed application; further alterations in design were considered only when necessary to ensure this feasibility. The core design differs from that of BORAX with respect to the meat and clad thicknesses for the fuel plates, the percentage of enrichment in the meat, and the dispersion of a burnable poison in the fuel alloy of this reactor. The structural and cladding material is zirconium. Adjustable control is obtained through the motion of absorbing curtains in the reactor. These are not major modifications; no attempt was made to optimize design with regard to fuel requirements or to flat flux distribution.

The analysis is highly simplified. It has been assumed that sufficiently indicative information may be obtained by using averaged nuclear properties, the averaging being of the most naive type, based on the volume weighted homogenized mass of core material. A final design should be accompanied by a more detailed analysis. Brief discussions of some of these details are presented.

(PHYSICS) DESCRIPTION OF THE REACTOR

The reactor shape approximation adopted for facilitation of the physics calculations is illustrated below.



This approximation is adequate for the large part of the analysis. In the boiling reactor the actual top reflector fluid contains, perhaps, 35% steam (by volume); therefore even the nine to eleven inches of fluid present is not effectively infinite. A partial compensation for this error has been made by assuming a somewhat reduced effectiveness for the vertical reflectors.

In the present design there are three control sections parallel to the direction of fluid flow (depicted as three vertical lines on the front face of the illustration) which contain movable plates of a material, such as cadmium, to present black barriers to thermal neutrons. (Use of some hafnium may be desirable since there are epithermal resonance regions in the absorption curve of hafnium. Thus some of the fast neutrons may be parasitically absorbed while the thermal blackness is preserved.) When the absorbing plates are removed there remains a framework of structural material and fluid. The content of these sections has been included in the volume weighted core mass. In these calculations the cold core is described as a mixture of metal and water in a volume ratio of 0.48. For other reactor core conditions it is assumed that the fluid volume remains 0.67568 (or 1/1.48) of the core volume.

The total core volume is 3.0763×10^5 cm^3.

METHOD OF ANALYSIS; PARAMETERS

Two-group diffusion theory was employed, involving a thermal or slow group and an epithermal, or fast group, of neutrons. The coupled differential equations of the theory are:

$$D_f \nabla^2 \phi_s - \frac{D_f}{\tau} \phi_f + k_{eff} \Sigma_{a_s} \phi_s = 0$$

$$D_s \nabla^2 \phi_s - \Sigma_{a_s} \phi_s + \frac{D_f}{\tau} \phi_f = 0$$

where

ϕ_f = epithermal, or fast neutron, flux

ϕ_s = thermal neutron flux

D_f = fast diffusion coefficient

D_s = thermal diffusion coefficient

τ = two-group age of fast neutrons

$k_\infty = \dfrac{\eta^{fuel} \bar{\Sigma}^{fuel}_{a_s}}{\bar{\Sigma}_{a_s}}$ = infinite multiplication constant ($p\epsilon = 1$)

$\overline{\Sigma}^{fuel}_{a_s}$ = average thermal macroscopic absorption cross section of the fuel (U^{235})

$\overline{\Sigma}_{a_s}$ = total average thermal macroscopic absorption cross section in the region

η^{fuel} = average number of fission neutrons produced per thermal neutron absorption in the fuel

The coupled equations were solved by assuming that both the thermal and fast neutron fluxes satisfied the equation

$$\nabla^2\phi + B^2\phi = 0,$$

and also that

$$\phi(x, y, z) = X(x)\, Y(y)\, Z(z).$$

However, it is not possible to satisfy the usual core-reflector flux and current continuity conditions when we assume separability of the solutions. This method of approximation is standard and the techniques and form sheets for the solution have been published.[2]

The two-group constants were based on the volume weighted homogenized mass of materials. Actually this represented a good description of the core for the cold reactor and for the hot (but not boiling) reactor, since the lattice of fuel, zirconium, and fluid is close-packed. The fuel density is not large and the thermal flux is expected to be only slightly depressed in the fuel. Thus the lattice thermal disadvantage factor is close to unity and homogenization alone does not induce an error of more than a few per cent in the (under) estimation of the fuel requirements. In the case of the boiling reactor, the vertical flow pattern of the moderating fluid led to a distribution of voids which may be a rather rapidly varying function of vertical position. For this case simple volume weighting of mass was somewhat questionable. A 14.2% "average" steam void in the fluid was assigned and in the homogenized core calculations this number was postulated to be a statistical average. Deviation from the 14.2% average in the actual operation of the reactor should be taken into account in more elaborate schemes of calculation.

The UNIVAC at New York University was utilized in solving several ten region two-group diffusion theory slab problems, successively refining the partitioning of the core in the direction of flow (and varying steam void). The fluid density (ρ) variation in the core was approximated by (Fig. 12):

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[2] D. Kurath, B. I. Spinrad, "Computation Forms for Solution of Critical Problems by Two-Group Diffusion Theory," ANL-4352, March 1, 1952.

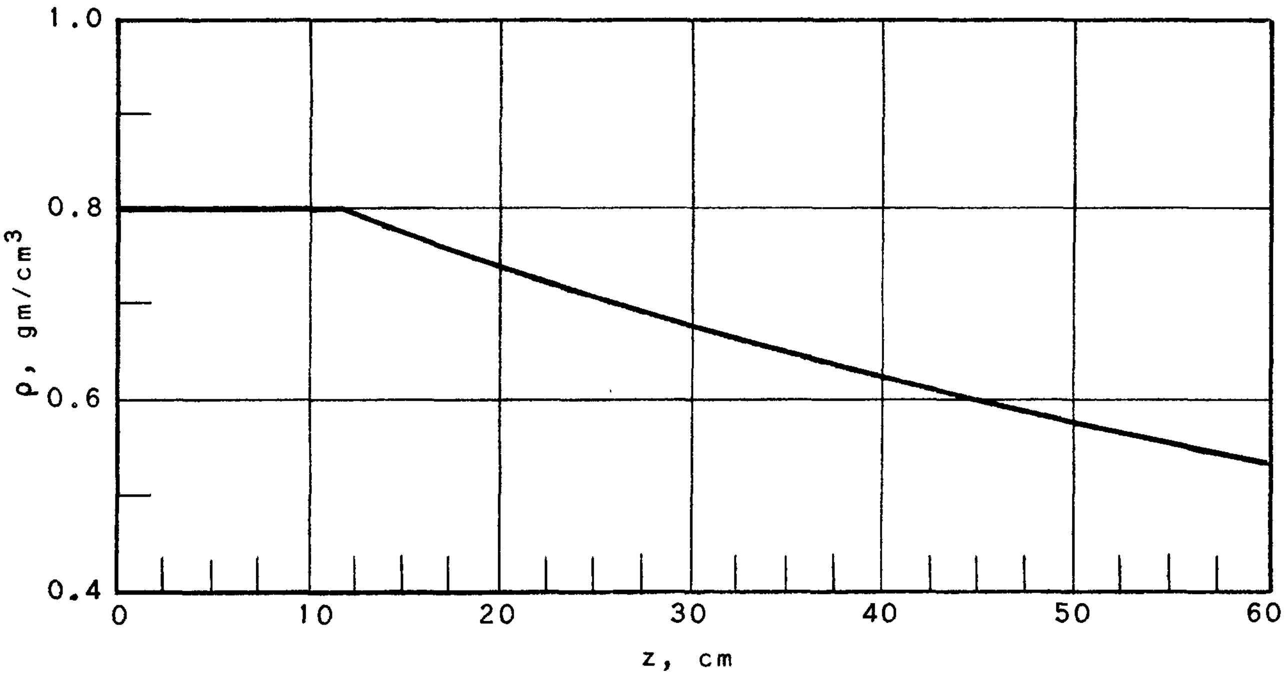

FIG. 12
DENSITY (ρ) OF STEAM-WATER MIXTURE AS A FUNCTION OF VERTICAL DISTANCE (z) FROM BOTTOM OF CORE

$$\rho(z) = \begin{cases} 0.80 \text{ gm/cm}^3, & 0 \leqq z = \leqq 12 \\ \frac{77}{84+z} \text{ gm/cm}^3, & 12 \leqq z = \leqq 60 \end{cases}$$

where z is measured (in cm) from the bottom of the core (Appendix B, page 56). In problem No. 1 the core was treated as one homogenized medium. In problem No. 2, three different core sections were used, again using volume weighting to homogenize. Problem No. 3 involved the further subdivision of the central core region of problem No. 2 into four equal sections. For problem No. 4 the bottom two core regions of problem No. 2 were combined. The resulting k_{eff}'s differed by less than one half of one per cent.

It should be emphasized that the fluid density curve was calculated on the basis of a flat thermal neutron flux in the direction of flow; hence the thermal neutron flux distributions obtained from the UNIVAC problems represent only the beginning of the second iteration for the solution of the flux-fluid density relationship in the operating reactor. In Fig. 13 there are presented the thermal neutron flux (ϕ_s) distributions calculated by the UNIVAC for the four core refinements described above and for the somewhat artificial case of the boiling virgin reactor without xenon or samarium. These fluxes were normalized to give the same total (i.e., space-integrated) flux. (It is expected that analogous curves would be obtained in the case of equilibrium xenon and samarium.) The two-group constants for these problems appear in Table I.

It should be noted (Fig. 13) that the successive refinements of the core induced successively greater shifts in the peak flux position in the direction of higher fluid density.

Tables II and III list the two-group parameters used in the hand calculations on the cold reactor, the hot reactor (operating temperature, no steam void), and the boiling, or operating, reactor. It was assumed that in the cold and hot reactors the reflector savings were the same in each direction. For the operating reactor it was felt that more conservative statements could be made by taking, for the vertical reflector savings, a number slightly smaller than the lateral reflector savings. (This is a slight modification of the previously expressed attitude that the vertical void variation is implicit in the assumption of a (statistical) average 14.2% steam void.) The two-group constants listed for the operating reflector apply only to the lateral reflector. The top reflector worth was estimated by a comparison of the two-group constants. (This estimate was later verified by the UNIVAC results.) The constants in Table II were assumed to apply during the lifetime of the reactor.

The thermal macroscopic absorption cross sections for the various materials in the reactor are included in Table III. Three cases were examined: cold, hot, and operating. Three subcases were detailed: virgin,

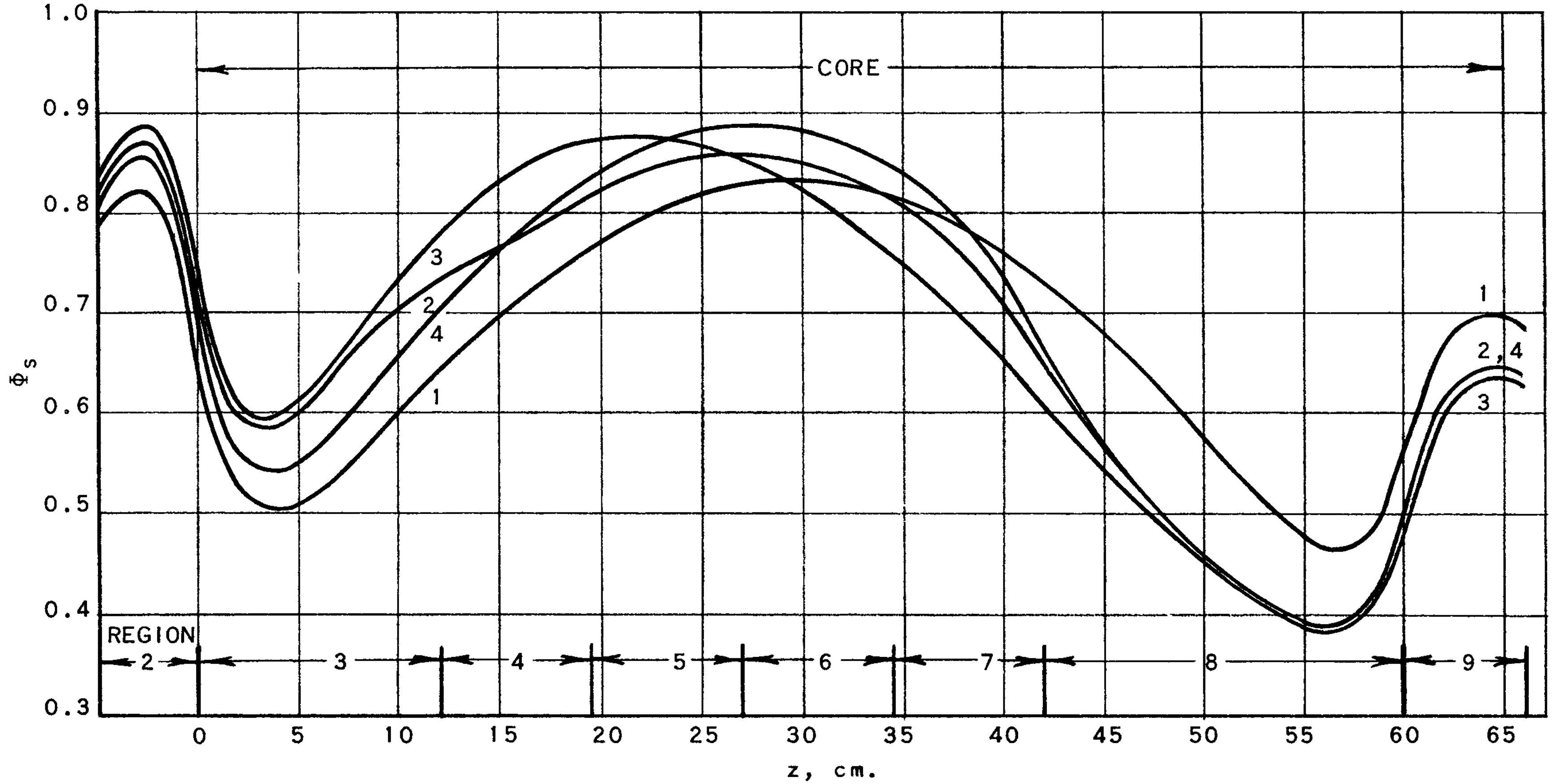

FIG. 13
THERMAL NEUTRON FLUX (Φ_S) AS A FUNCTION OF VERTICAL DISTANCE (z) FROM BOTTOM OF CORE

Table I

TWO-GROUP CONSTANTS FOR THE UNIVAC COMPUTATIONS

Problem No.		Region Number 1 & 2	3	4	5	6	7	8	9 & 10
13001 2 3 4	τ	51.6	89.5 69.5 69.5 80.8	89.5 86.1 73.4 80.8	89.5 86.1 82.5 80.8	89.5 86.1 91.5 80.8	89.5 86.1 101.4 80.8	89.5 118.1 118.1 118.1	122
1 2 3 4	D_f	1.429	1.370 1.245 1.245 1.319	1.370 1.350 1.272 1.319	1.370 1.350 1.329 1.319	1.370 1.350 1.381 1.319	1.370 1.350 1.431 1.319	1.370 1.510 1.510 1.510	2.198
1 2 3 4	L^2	19.5086	7.3091 6.1713 6.1713 6.8266	7.3091 7.1243 6.4100 6.8266	7.3091 7.1243 6.9184 6.8266	7.3091 7.1243 7.4133 6.8266	7.3091 7.1243 7.9080 6.8266	7.3091 8.7170 8.7170 8.7170	46.17
1 2 3 4	D_s	0.2263	0.3485 0.3015 0.3015 0.3286	0.3485 0.3409 0.3114 0.3286	0.3485 0.3409 0.3324 0.3286	0.3485 0.3409 0.3528 0.3286	0.3485 0.3409 0.3731 0.3286	0.3485 0.4063 0.4063 0.4063	0.3482
1 2 3 4	k_∞	0	1.5250 1.4883 1.4883 1.5105	1.5250 1.5195 1.4967 1.5105	1.5250 1.5195 1.5133 1.5105	1.5250 1.5195 1.5278 1.5105	1.5250 1.5195 1.5411 1.5105	1.5250 1.5600 1.5600 1.5600	0
1 2 3 4	Avg. % void	20	21.6 13.5 13.5 18.5	21.6 20.5 15.4 18.5	21.6 20.5 19.1 18.5	21.6 20.5 22.3 18.5	21.6 20.5 25.0 18.5	21.6 29.0 29.0 29.0	48
Region thickness, cm		23;5	12	7.5	7.5	7.5	7.5	18	6;22
$\nu\Sigma_f^{U^{235}}$		0	←			0.07271		→	0
No. of Intervals		5;6	8	3	3	3	3	15	8;6

Table II

TIME - INDEPENDENT TWO-GROUP CONSTANTS

	CORE			REFLECTOR		
	Cold	Hot	Operating	Cold	Hot	Operating
D_f(cm)	1.082	1.245	1.363	1.143	1.429	1.429
D_s(cm)	0.2181	0.3015	0.3459	0.1588	0.2263	0.2263
τ (cm^2)	48	70	89	33	51.6	51.6
ηU^{235}	2.09	2.09	2.09			
Average Fluid Steam Void (%)	0	0	14.2	0	0	0
Region Net Void (%)	0	13.5	21.2	0	20	20
Total Buckling B^2 (cm^{-2})	0.004456	0.004132	0.003987			

half-depleted (one kilogram of fuel destroyed), and fully depleted. It has been assumed that the reactor operates at an average rate equal to 70% full output (5.5 mw) until two kilograms of fuel have been destroyed, or continual operation for slightly more than one year (based on 200 mev per fission). The ratio

$$\frac{\bar{\Sigma}_{a_s}^{Xe + Sm}}{\bar{\Sigma}_{a_s}^{U^{235}}}$$

is the ratio of equilibrium xenon and samarium absorption to fuel absorption for full power reactor operation at that time. Finally, it was assumed that the average fission product absorption, $\bar{\Sigma}_{a_s}^{f.p.}$ was directly proportional to the time-integrated number of fissions per cm^3 in the reactor, and the pessimistic value of 100 barns per fission was assigned; this microscopic cross section was postulated to hold for all reactor temperatures. Xenon and samarium are not included in the term $\bar{\Sigma}_{a_s}^{f.p.}$.

No attempt has been made to take into account the epithermal absorption and fission events. In a "thermal" reactor of this type, without a very high macroscopic absorption cross section of fuel and of other

Table III

TIME - DEPENDENT AVERAGE THERMAL MACROSCOPIC ABSORPTION CROSS SECTIONS

	COLD			HOT			OPERATING		
	Virgin	Half-Depleted	Depleted	Virgin	Half-Depleted	Depleted	Virgin	Half-Depleted	Depleted
					CORE				
Total fuel mass (kg)	10	9	8	10	9	8	10	9	8
$\bar{\Sigma}_{a_s}^{U^{235}}$ (cm^{-1})	0.04982	0.04484	0.03986	0.03524	0.03172	0.02820	0.03524	0.03172	0.02820
$\bar{\Sigma}_{a_s}^{B^{10}}$ (cm^{-1})	0.00597	0.00311	0.00150	0.00443	0.00231	0.00111	0.00443	0.00231	0.00111
$\dfrac{\bar{\Sigma}_{a_s}^{Xe + Sm}}{\bar{\Sigma}_{a_s}^{U^{235}}}$ (equilibrium)	0	0	0	0.048	0.049	0.050	0.048	0.049	0.050
$\bar{\Sigma}_{a_s}^{f.p.}$ (cm^{-1})	0	0.00071	0.00141	0	0.00071	0.00141	0	0.00071	0.00141
$\bar{\Sigma}_{a_s}^{mod}$ (cm^{-1})	0.01593	0.01593	0.01593	0.00985	0.00985	0.00985	0.00985	0.00873	0.00873
					REFLECTOR				
$\bar{\Sigma}_{a_s}^{mod}$ (cm^{-1})	0.01955	0.01955	0.01955	0.01564	0.01564	0.01564	0.01564	0.01564	0.01564

$\sigma_a^{U^{235}}$ (0.0253) = 685 barns σ_a^{Zr} (0.0253) = 0.22 barns σ_a^{H} (0.0253) = 0.33 barns absorbers, such events do not have a profound effect on criticality of the reactor; their effects are of a second order of importance. Of greater importance are the proper selection of the spectrum of the thermal neutron flux, the correct averaging for the fast diffusion coefficient (D_f) in the core and reflector, and the correct averaging of the two-group age (τ) for fast neutrons in the core. In this preliminary analysis the approximation was made that the thermal neutron flux is distributed in a Maxwellian with mode kT corresponding to the kT of the moderator atoms. The fast core constants, D_f and τ, were computed on the basis of void corrections applied to the results of rather ancient calculations.[3]

Specifically,

$$D_f = \frac{D_f\ (\text{metal: } H_2O \text{ of density } 1)}{1 - (\text{void fraction})} \quad ,$$

and

$$\tau = \frac{\tau(\text{metal: } H_2O \text{ of density } 1)}{(1 - \text{void fraction})^2} \quad .$$

TIME-DEPENDENT TWO-GROUP PARAMETERS

Having made the approximations discussed above, the problem was to determine a fuel mass that would permit operation of the reactor two hours (or more, if possible) after instantaneous shutdown from full-power operation; that is, there must be enough reactivity to permit full-power override of the xenon and samarium formed within (at least) two hours after such a shutdown. This fuel mass could not be so long as to require a prohibitively cumbersome control system; that is, in the most reactive condition during its lifetime, the reactor must be controllable by a reasonable system. Since two kilograms of fuel are present in the virgin core as a burnup allowance - perhaps 20% of the total virgin fuel content - such control would not be feasible unless additional variable control is employed, such as a parasitic neutron absorber which would be largely destroyed during the reactor operation. (If no burnable poison were used, the requirement of enough reactivity to permit full-power override after a two-hour fast shutdown would imply a k_{eff} (cold virgin) larger than 1.25.) Boron-10 is a relatively inexpensive material for this purpose. It is estimated that less than one extra kilogram of U^{235} would suffice to permit insertion of enough B^{10} to reduce k_{eff} (cold virgin pile) to 1.18, which can be controlled without great difficulty.

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[3] "Naval Reactor Division Quarterly Report," ANL-4337, September 15, 1949.

The space-averaged thermal flux was used to compute fuel and B^{10} burnout and the formation of fission products. In place of time, one may use the ratio, x, of the time-integrated reactor heat output to the total time-integrated output as a "time" variable. In the actual operating reactor, fuel and B^{10} burnout are related by:

$$N^{U^{235}}(\underline{r},x) = N^{U^{235}}(\underline{r},o) \exp\left[-\int_0^x \overline{\phi_s(\underline{r},u)\,\sigma_{a_s}^{U^{235}}}\; du\right]$$

$$N^{B^{10}}(\underline{r},x) = N^{B^{10}}(\underline{r},o) \exp\left[-\int_0^x \overline{\phi_s(\underline{r},u)\,\sigma_{a_s}^{B^{10}}}\; du\right]$$

where

$$\overline{\phi_s(\underline{r},u)\,\sigma_{a_s}^{M}} = \int_0^{E_{th}} \phi(\underline{r},u,E)\,\sigma_{a_s}^{M}(E)\; dE$$

Assuming that only thermal absorptions occur,

$$\overline{\phi_s(\underline{r},u)\,\sigma_{a_s}^{M}}$$

is approximated by

$$\phi_s(\underline{r},u)\,\bar{\sigma}_a^{M}$$

where $\phi_s(\underline{r},u)$ is the two-group thermal flux at the position $\underline{r}$ in the reactor and at "time" u, and $\bar{\sigma}_a^M$ is the average for material M with respect to a Maxwellian distribution of thermal neutron flux. It is then apparent that

$$N^{B^{10}}(\underline{r},x) = N^{B^{10}}(\underline{r},o)\left[\frac{N^{U^{235}}(\underline{r},x)}{N^{U^{235}}(\underline{r},o)}\right]^{\frac{\bar{\sigma}_{a_s}^{B^{10}}}{\bar{\sigma}_a^{U^{235}}}}$$

$$\left(\text{Here } \frac{\bar{\sigma}_{a_s}^{B^{10}}}{\bar{\sigma}_{a_s}^{U^{235}}} = 6.1994\right)$$

The space-averaged fuel atomic density at time x is approximated by

$$N^{U^{235}}(x) = N^{U^{235}}(o)\left[\frac{\text{Virgin fuel mass} - 2x}{\text{Virgin fuel mass}}\right] ,$$

whence it is easy to determine

$$\bar{\Sigma}_{a_s}^{U^{235}}(x) \text{ and } \bar{\Sigma}_{a_s}^{B^{10}} .$$

Also,

$$\bar{\Sigma}_{a_s}^{f.p.}(x) = \left[N^{U^{235}}(o) - N^{U^{235}}(x)\right]\left(\frac{1}{1.183}\right)\left(100 \times 10^{-24}\right)$$

or

$$\bar{\Sigma}_{a_s}^{f.p.}(x) = \left[N^{U^{235}}(o)\left(\frac{100 \times 10^{-24}}{1.183}\right)\left(\frac{2}{\text{Virgin fuel mass}}\right)\right]x .$$

The average thermal neutron core flux, $\bar{\phi}_s(x)$, based on full power operation at x, is given by (Fig. 14):

$$\bar{\phi}_s(x) = \frac{1.719 \times 10^{17}}{\bar{\Sigma}_{fs}^{U^{235}}(x)\,(\text{volume of core})} ,$$

where

$$\bar{\Sigma}_{fs}^{U^{235}}(x) = \frac{\bar{\Sigma}_{a_s}^{U^{235}}}{1.183} .$$

Xenon and samarium equilibrium absorption, based on full power reactor operation at time x, is then given by

$$\frac{\bar{\Sigma}_{a_s}^{Xe + Sm}}{\bar{\Sigma}_{a_s}^{U^{235}}}(\text{equilibrium, } x) = 0.011 + 0.064\,\frac{\bar{\phi}_s(x)\,\bar{\sigma}_{a_s}^{Xe}}{\lambda^{Xe} + \bar{\phi}_s(x)\,\bar{\sigma}_{a_s}^{Xe}}$$

where

$$\lambda^{Xe} = 2.09 \times 10^{-5}/\text{sec}$$

and

$\bar{\sigma}_{a_s}^{Xe} = 2.37 \times 10^{-18}$ for the operating reactor.

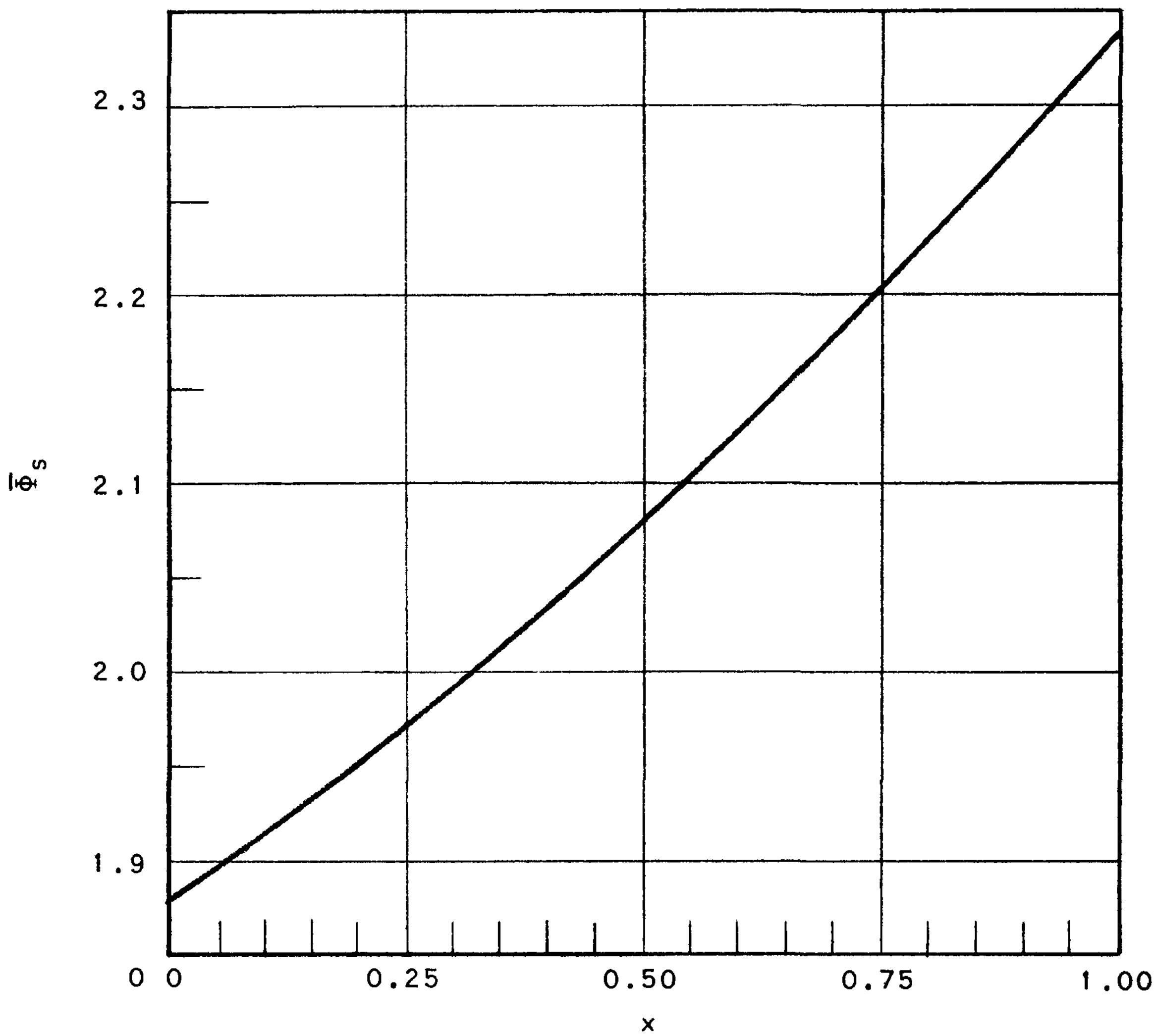


FIG. 14
AVERAGE THERMAL NEUTRON FLUX ($\overline{\Phi}_s$) AS A FUNCTION OF THE RATIO (x) OF TIME-INTEGRATED HEAT OUTPUT TO TOTAL TIME-INTEGRATED HEAT OUTPUT

The changing atomic densities of fuel and of the various parasitic absorbers must be compensated by changes in control settings. The latter, in turn, induce further changes in flux distribution. Thus the application of the volume averaged flux is a questionable procedure; here again more sophisticated analysis is suggested for a final design.

If we define the effective multiplication factor, k_{eff}, as

$$k_{eff}(x) = \eta^{U^{235}} \frac{\bar{\Sigma}_{a_s}^{U^{235}}(x)}{\bar{\Sigma}_{a_s}(x)} \quad \frac{1}{\left(1 + \frac{D_s}{\bar{\Sigma}_{a_s}(x)} B^2\right)(1 + \tau B^2)}$$

we may examine (Fig. 15) the time variation of pile reactivity for the several core conditions described earlier. It should be emphasized that the cold core is undoubtedly an artificial condition once the reactor has been operated. For the cold core it is assumed that the xenon and samarium are not present.

On the basis of the approximations discussed previously, it appeared that there is little change in k_{eff} during the life of the reactor. (If much larger fuel burnouts were required, it is to be expected that k_{eff} would exhibit a much greater time variation unless more complicated variable control schemes were admitted. It is also indicated that override of maximum xenon is easily attained with little extra fuel required. The fuel mass listed appears sufficient to permit reactor startup at any time and, with the possible exception of the final 10% of core life, to permit override of maximum xenon and samarium by full-power operation.)

CONTROL

In going from the cold reactor to the operating reactor condition with two-hour xenon, the fast non-leakage probability, $1/(1 + \tau B^2)$, decreases by approximately 10% and k_∞ increases an average of 1% in these approximations. Also, there is slightly more than 1% decrease in the thermal non-leakage probability,

$$\frac{1}{1 + \frac{D_s}{\bar{\Sigma}_{a_s}} B^2}$$

Considering also the time dependence of these parameters, k_{eff} (cold) must be at least 1.13 in order to meet the requirements of criticality and to permit override of xenon and samarium two hours (at least) after a fast shutdown. As may be observed from Fig. 15, k_{eff} (cold) decreases from initial pile operation to final pile operation. Stipulating k_{eff} (cold virgin) = 1.18 then permits k_{eff} (cold) to be at least 1.13 at all times.

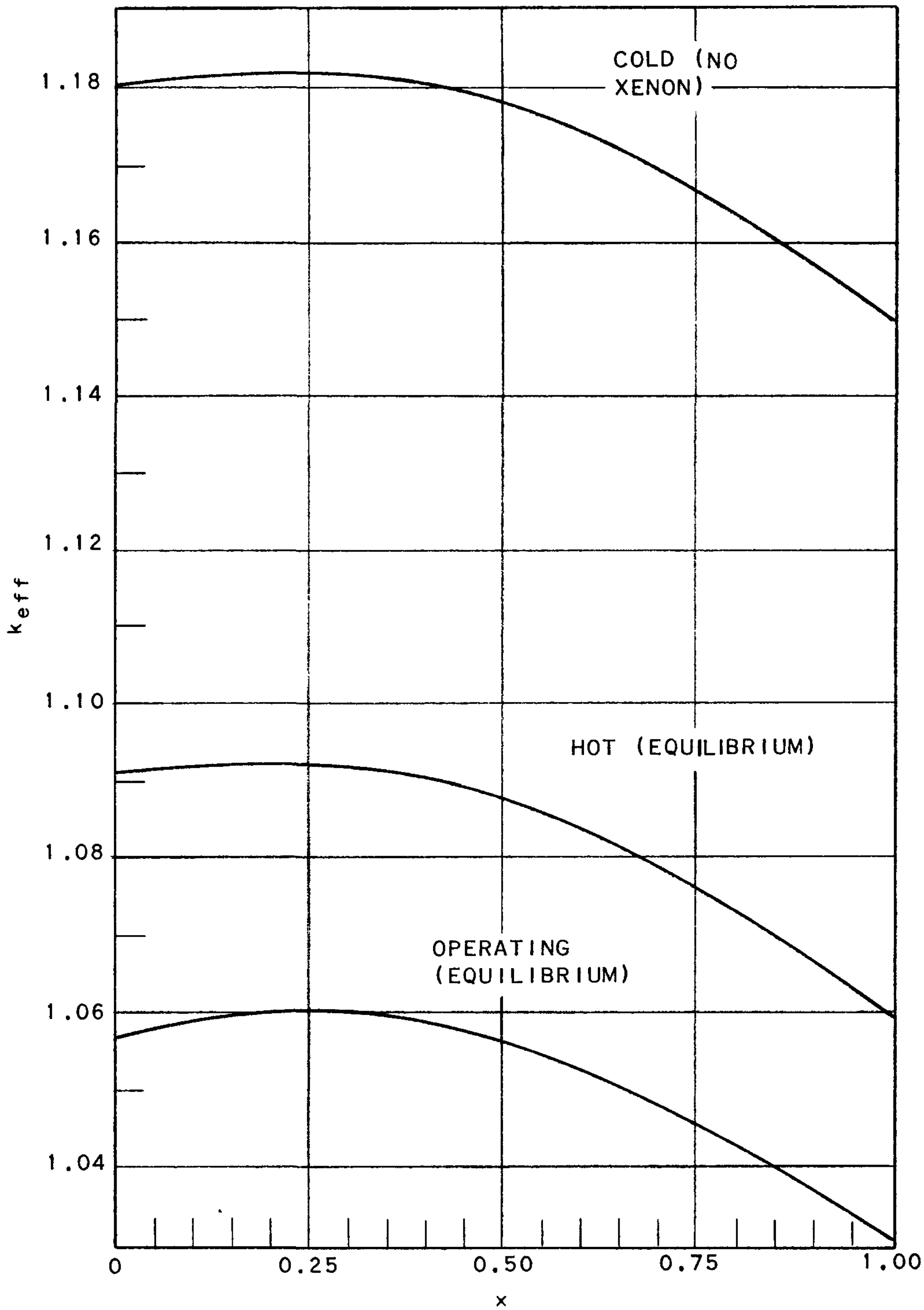
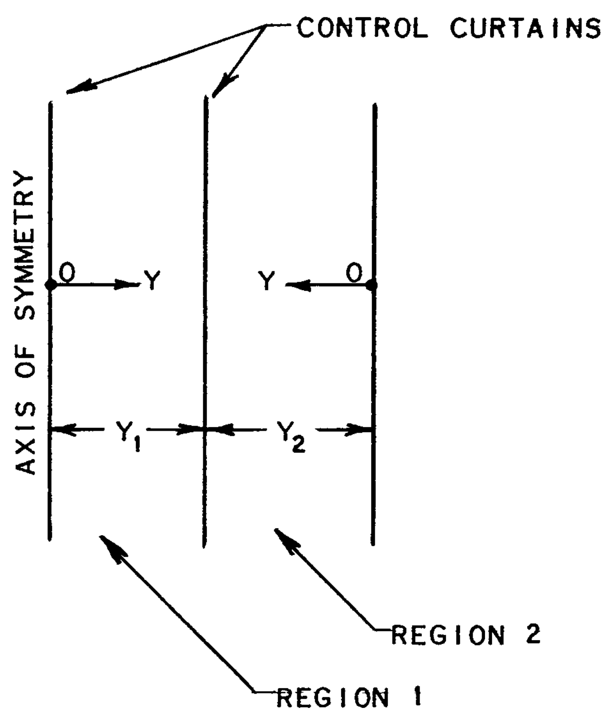


FIG. 15
EFFECTIVE REACTOR MULTIPLICATION FACTOR (k_{eff}) AS A FUNCTION OF THE RATIO (x) OF TIME-INTEGRATED HEAT OUTPUT TO TOTAL TIME-INTEGRATED HEAT OUTPUT

Sufficient adjustable control must be provided to enable the operator to shut down the reactor at any time. It is contemplated that only for the virgin reactor would the core temperature be so low as to permit a k_{eff} as large as 1.18. As a possible control mechanism, an arrangement of control blades has been chosen which provides essentially three absorbing curtains in the core. It is expected that these curtains, plus a possible variation of the water level in the reactor, would be sufficient to control the reactor after initial operation. For the virgin reactor, additional control could be obtained by heating the water before pumping it into the reactor. In the event that critical experiments indicate that still more permanently available control is to be preferred, it is anticipated that two more curtains could be supplied with a relatively minor increase in fuel content (less than 10%). For extreme emergency control a soluble-poison injection system is provided.

In evaluating the effectiveness of the curtains for the cold virgin reactor, the transport theory boundary condition was applied for a thermally black curtain in a non-absorbing medium. The correction for such weak absorption as is present is not important. The curtains were considered as absorbing planes (zero thickness). The outer core regions were augmented by the savings of the reflector (7.2 cm) to reduce the number of regions, thus facilitating computation. The approximation introduces only a small error for such large regions.

The problem may be illustrated as follows:



The flux solutions are those of two-group theory:

In region j

$$\phi_{fj}(y) = R_{fj}[A_{j_1} \cos \mu_{fj}y + A_{j_2} \sin \mu_{fj}y] + R_{sj}[A_{j_3} \cosh \mu_{sj}y + A_{j_4} \sinh \mu_{sj}y]$$

$$\phi_{sj}(y) = [A_{j_1} \cos \mu_{fj}y + A_{j_2} \sin \mu_{fj}y] + [A_{j_3} \cosh \mu_{sj}y + A_{j_4} \sinh \mu_{sj}y],$$

$j = 1, 2.$

The curtain conditions are:

$$\frac{\phi'_{s_1}(0)}{\phi_{s_1}(0)} = \frac{1}{0.7104\ \lambda_{total}} = a; \frac{\phi'_{s_1}(y_1)}{\phi_{s_1}(y_1)} = -a$$

$$\frac{\phi'_{s_2}(y_2)}{\phi_{s_2}(y_2)} = -\frac{1}{0.7104\ \lambda_{total}}$$

The other boundary conditions are:

$$\phi_{f_1}(0) = 0,$$

$$\phi_{f_1}(y_1) = \phi_{f_2}(y_2)$$

$$\phi_{f_2}(0) = 0 = \phi_{s_2}(0)$$

and

$$-D_{f_1}\ \phi'_{f_1}(y_1) = D_{f_2}\ \phi'_{f_2}(y_1)\ .$$

A good approximation for such a large region is to replace $\sinh \mu_{s_1} y_1$ by $\cosh \mu_{s_1} y_1$. Since $\cosh \mu_{s_1} y$ exhibits a tremendous variation over region one, it is clear that $A_{13} \approx - A_{14}$.

We may then write:

$$\phi_{f_1}(y_1) \approx R_{f_1}\ [A_{11} \cos \mu_{f_1} y_1 + A_{12} \sin \mu_{f_1} y_1] + R_{s_1} \cosh \mu_{s_1} y_1\ [A_{13} + A_{14}]$$

and eventually we may solve for the terms A_{11}, A_{13}, and $(A_{13} + A_{14}) \cosh \mu_{s_1} y_1$. This technique simplifies the evaluation of the determinant $|A|$ of the matrix A of coefficients in the resulting homogeneous linear system $AX = 0$, where X may be written as

$$X = \begin{bmatrix} A_{11} \\ A_{13} \\ [A_{13} + A_{14}] \cosh \mu_{s_1} y_1 \\ -A_{22} \\ -A_{24} \sinh \mu_{s_2} y_2 \end{bmatrix}$$

The equation $|A| = 0$ is the consistency condition for solvability of the system obtained after application of the various curtain and boundary conditions. For this problem the two-group constants for regions one and two are identical and we may write the equation:

$$\Delta \equiv \begin{vmatrix} a \cos \mu_f y_1 - \mu_f \sin \mu_f y_1 & \alpha[a \sin \mu_f y_1 + \mu_f \cos \mu_f y_1] & a + \mu_s & 0 & 0 \\ -a & \mu_s \left[\frac{R_s}{R_f} - 1\right] - a & 0 & 0 & 0 \\ \cos \mu_f y_1 & \alpha \sin \mu_f y_1 & \frac{R_s}{R_f} & \sin \mu_f y_2 & \frac{R_s}{R_f} \\ \mu_f \sin \mu_f y_1 & -\frac{R_s}{R_f} \mu_s \cos \mu_f y_1 & -\frac{R_s}{R_f} \mu_s & \mu_f \cos \mu_f y_2 & \frac{R_s}{R_f} \mu_s \\ 0 & 0 & 0 & a \sin \mu_f y_2 + \mu_f \cos \mu_f y_2 & a + \mu_s \end{vmatrix} = 0$$

(where $\Delta = \dfrac{1}{D_f R_f} |A|$)

where

$$\mu_f = \sqrt{B^2_{total} - B^2_{OD}}$$

B^2_{OD} = sum of the geometric bucklings in the other directions

$$\mu_s = \sqrt{\left[B^2_{total} + \frac{1}{L^2} + \frac{1}{\tau}\right]^2 + B^2_{OD}}\,, \quad L^2 = \frac{D_s}{\overline{\Sigma}_{a_s}}$$

$$\frac{R_s}{R_f} = \frac{L^2 B^2_{total} + \frac{L^2}{\tau}}{1 + L^2 B^2_{total}}\,, \quad R_f = \left(\overline{\Sigma}_{a_s}\right)\left(\frac{\tau}{D_f}\right)(1 + L^2 B^2_{total})$$

$$\alpha = \frac{\mu_s R_s}{\mu_f R_f}$$

$$a = \frac{1}{0.7104\ \lambda_{total}}$$

For this cold virgin reactor, $k_{eff} = 1.18$ without curtains. With curtains in, we artifically vary η to satisfy the condition $\Delta = 0$, getting $\eta^{criticality}$. Then

$$\frac{\eta^{U^{235}} - \eta^{criticality}}{\eta^{U^{235}}} = \frac{\Delta k_{eff}}{k_{eff}}$$

for the transition to the system with curtains fully inserted.

RESULTS

$\Delta k_{eff}/k_{eff} \approx 0.015$ when the curtains are 7 in. apart; that is, the curtains alone do not quite control the cold, virgin reactor. However, as discussed earlier, if at least this much control is inherent in three curtains it is not difficult to provide additional control capacity if necessary. For example, some additional control would be achieved by using a material (possibly hafnium) which absorbs epithermal neutrons or by using preheated water in the virgin reactor at startup.

(If the two outer curtains were moved to a position 3-1/2 in. form the central curtain $\Delta k_{eff}/k_{eff}$ would be ≈ 0.01.)

WORTH OF ONE PERIPHERAL FUEL SUBASSEMBLY

To get some notion of how much reactivity is implicit in a peripheral fuel subassembly, four subassemblies were removed from the core along one side, giving an effective depth of 69.95 cm instead of 74.07 cm. Assuming that the reflector savings are unchanged, it was found that k_{eff} for the cold virgin reactor drops from 1.180 to 1.174 and k_{eff} for the operating virgin reactor drops from 1.057 to 1.049.

APPENDIX B

RELATIONS BETWEEN CIRCULATING RATIOS, STEAM VOIDS AND MODERATOR DENSITIES

Appendix B is a summary of certain heat engineering equations which have been developed because they are considered useful for the determination of the moderator density and its variation in relation to such parameters as flow rate, steam pressure, subcooling, and slip ratio. The equations are based on the approximation that heat is generated at a uniform rate along the whole height of the core.

In addition, density differentials are determined as functions of variations in each of the parameters.

Application of these general equations yields data pertinent to the operating and physics characteristics of the reactor.

NOMENCLATURE

G_s	Flow rate of steam, lb/sec
G_t	Total flow rate, lb/sec
Y	Steam quality G_s/G_t
X	Recirculation factor, lb liquid/lb steam at channel discharge point
X_ℓ	Recirculation factor at intermittent point of channel
p	Steam pressure, psia
v_s	Steam velocity, fps
v_w	Liquid water velocity, fps
r	Slip ratio, v_s/v_w
A	Channel flow area, sq ft
ρ	Density of two-phase fluid, lb/cu ft
ρ_s	Steam density at existing conditions, lb/cu ft
ρ_w	Saturated liquid density at existing conditions, lb/cu ft
$\bar{\rho}_b$	Average fluid density in boiling section of channel, lb/cu ft
$\bar{\rho}_t$	Average density of all fluid in both subcooled and boiling channel sections, lb/cu ft
V	Specific volume (same subscripts as above), cu ft/lb
R_V	Steam void ratio, steam flow area/ total channel flow area
P	Power generated in channel or combination of channels, Btu/sec

L or L_t Length of channel, ft

L_1 Length of subcooled channel section, ft

L_b Length of net boiling channel section, ft

ℓ Length of channel from point of initial boiling to intermittent point, ft

e Heat transferred per linear foot of channel, Btu/ft (this value is assumed constant over whole length)

H Enthalpy of steam, Btu/lb

ΔH Heat of vaporization for water, Btu/lb

h Enthalpy of liquid, Btu/lb

h_{sl} Enthalpy of saturated liquid, Btu/lb

h_o Enthalpy of total liquid entering channel, Btu/lb

Δh Subcooling of liquid, $(h_{sl}-h_o)$, Btu/lb

h_f Enthalpy of feed water alone entering reactor, Btu/lb

m Ratio: $(\rho_w - r\rho_s)/r\rho_s$

m* Ratio: $m^2 r\rho_s/(\rho_w - \rho_s)$

F Feed water entering reactor, lb/sec

Z Recirculated water entering channel together with each pound of feed water, lb/lb

SUMMARY OF EQUATIONS

$$v_s = (V_s + XrV_w)\, G_s/A \qquad (1)$$

$$v_w = v_s/r = (V_s/r + XV_w)\, G_s/A \qquad (2)$$

$$\rho = \frac{1 + Xr}{V_s + XrV_w} \qquad (3)$$

$$R_V = \frac{1}{(1 + Xr)\,\rho_s/\rho_w}\ ;\ \text{or } X = \left(\frac{1}{R_V} - 1\right)\rho_w/r\rho_s \qquad (4)$$

Variation of X along the boiling channel:

$$X_\ell = (G_t\,\Delta H/e\ell) - 1 \text{ where } G_t = (X + 1)\, G_s \qquad (5)$$

$$\rho_\ell = \frac{\ell e\,(1 - r) + rG_t\,\Delta H}{\ell e\,(V_s - rV_w) + r\,G_t\,\Delta H V_w} \qquad (5a)$$

$$G_s = e\ell/\Delta H \qquad (6)$$

$$Y = 1/(X + 1) = \text{Steam quality} \qquad (7)$$

Average fluid density in net boiling section L_b is given by:

$$\bar{\rho}_b = \frac{\rho_w}{m^* Y} \log (1 + mY) \tag{8}$$

Ratio of subcooled distance to boiling length of channel:

$$L_l/L_b = \Delta h/(Y\Delta H) \tag{9}$$

$$L_l/L_t = \Delta h/(\Delta h + Y\Delta H) \tag{9a}$$

Average fluid density in total length of channel:

$$\bar{\rho}_t = \frac{\bar{\rho}_b\, Y\Delta H + \rho_w\, \Delta h}{\Delta h + Y\Delta H} \tag{10}$$

or

$$\bar{\rho}_t = \left(\frac{\rho_w}{\Delta h + Y\Delta H}\right)\left(\frac{\Delta H}{m^*} \log (1 + mY) + \Delta h\right) \tag{11}$$

Subcooling $\Delta h = h_{sl} - h_o$; $\Delta h = \dfrac{h_{sl} - h_f}{Z + 1}$ (12)

or, when recirculation factor X at outlet is equal to recirculation factor Z at inlet (steady state)

$$\Delta h = Y\,(h_{sl} - h_f) \tag{13}$$

DENSITY DIFFERENTIALS

I. Variation of total average fluid density as a result of changing steam quality alone:

$$\frac{d\bar{\rho}_t}{dY} = \left(\frac{\Delta H}{\Delta h + Y\Delta H}\right)\left[\frac{m\rho_w}{m^*(1 + mY)} - \bar{\rho}_t\right] \tag{14}$$

Expressed in terms of changes of the recirculating factor X the equation is

$$\frac{d\bar{\rho}_t}{dX} = -Y^2 \cdot \frac{d\bar{\rho}_t}{dY} \tag{15}$$

II. Variation of total average fluid density as a result of changing the sub-cooling alone:

$$\frac{d\bar{\rho}_t}{d\Delta h} = \frac{\rho_w - \bar{\rho}_t}{\Delta h + Y\Delta H} \tag{16}$$

2

III. Variation of total average fluid density as a result of changing steam pressure alone. There are two influences: (1) result of changing steam density; and (2) result of changing subcooling.

(1) Due to changed steam density

$$\frac{d\bar{\rho}_t}{dp} = \left(\frac{d\bar{\rho}_t}{dm}\right)\left(\frac{dm}{d\rho_s}\right)\left(\frac{d\rho_s}{dp}\right) \qquad (17)$$

where

$$\frac{d\bar{\rho}_t}{dm} = \left(\frac{\rho_w \Delta H}{\Delta h + Y\Delta H}\right)\left(\frac{Y}{(1 + mY)m^*} - \frac{\log(1 + mY)}{(m^*)^2}\right) \qquad (18)$$

and:

$$\frac{dm}{d\rho_s} = \frac{-\rho_w}{r\rho_s^2}$$

$\frac{d\rho_s}{dp}$ = steam density change per 1 psi pressure change as found by steam tables. (20)

(2) Due to the changed subcooling [see Eq. (12)]

$$\frac{d\bar{\rho}_t}{dp} = \left(\frac{d\bar{\rho}_t}{d\Delta h}\right)\left(\frac{d\Delta h}{dp}\right) = \left(\frac{d\bar{\rho}_t}{d\Delta h}\right)\left(\frac{dh_{sl}}{dp}\right)\left(\frac{1}{Z + 1}\right) \qquad (21)$$

where

$\frac{d\bar{\rho}_t}{d\Delta h}$ is found by Eq. (16) and $\frac{dh_{sl}}{dp}$ from steam tables.

EQUATIONS APPLIED TO 5-1/2 mw HORIZONTAL REACTOR

The fluid density equation used in the Physics Appendix for a 5-1/2 mw reactor and shown on page 37 was derived from Eq. (5a). In the present case, we have for the total fluid flow through the core:

Total Flow Area A = 3.42 sq ft

$X + 1 = 60$

G_s = 20,000 lb/hr = 5.55 lb steam/sec

$G_t = (60)(G_s)$ = 333 lb/sec

ΔH at 600 psia = 732 Btu/lb

e = 5-1/2 mw per 2 ft channel length = 2,600 Btu/sec-ft

V_s = 0.77 cu ft/lb, V_m = 0.02, r = 1.25 (at 600 psi)

These values were applied in Eq. (5a) after which the engineering units were converted, density into gm/cc and distances into cm. The equation then takes the approximate form

$$\rho = 77/(96 + \ell)$$

For average operating conditions it is assumed that the subcooled part of the cooling channels is 20% of the total core height (60 cm) or 12 cm.

The equation can then be written in the form used for the physics calculations:

$$\rho = 77/(84 + z)$$

where z is the vertical distance (cm) from the bottom of the core and ρ is measured in gm/cc. This applies for the boiling region where $z > 12$ cm, such as shown graphically on Fig. 12 in Appendix A.

The required fluid velocities through the core are found by Eqs. (1) and (2).

Average steam and water velocities at top ends of the channel are:

$v_s = 3.6$ fps and $v_w = 3.6/1.25 = 2.9$ fps.

The average water inlet velocity at the bottom end of the channels is 1.9 fps. The downcomer velocity is less than 0.5 fps.

By application of Eqs. (8) and (10) it is found that the average fluid density loss in the core, due to boiling, is 14.2% of the hot, non-boiling liquid density.

Eq. (4) shows that the steam volume at the top end of the channels will be $R_V = 34\%$ of the moderator space.

APPENDIX C

ALTERNATE POWER AND HEATING CYCLES FOR 425 kw ELECTRIC POWER PLUS 2,500 kw HEAT

Case I

Separate power plant and steam heating plant, each using indirect steam cycles. Turbine operates with full expansion.

This system is illustrated by the simplified flow diagram Fig. 16.

Power Cycle

500 kw generator output (net demand 425 kw).

Medium: Secondary steam, 8,000 lb/hr at 300 psig, condensed at 3 in. Hg.

Water rate: 16 lb/kwh.

Required from reactor: 2.6 mw power, 9,600 lb/hr primary steam condensed and cooled in heat exchanger to 300F.

Heating Cycle

Heat demand equivalent to 2,500 kw.

Medium: Secondary steam, 8,500 lb/hr at 50 psig, condensed and cooled to 210F.

Required from reactor: 2.5 mw power, 9,100 lb/hr primary steam condensed and cooled in heat exchanger to 300F.

Reactor

5.1 mw total power

18,700 lb/hr steam at 600 psia returned as feed water at 300F. Operating period for 2 kg fuel burn out = 15 months at 70% of capacity.

Over-all plant efficiency $\frac{(425 + 2,500)}{5,100} \times 100 = 57\text{-}1/2\%$.

Advantage: 1) Detached power and heat operations.
2) No radioactivity in power unit or condenser.

Disadvantage: Investment in evaporator equipment.

The flow diagram, Fig. 16, indicates separate heat exchangers or evaporators for the power plant and the heating system. Obviously, these two steam supplies can be generated in one single evaporator if a combination is preferred.

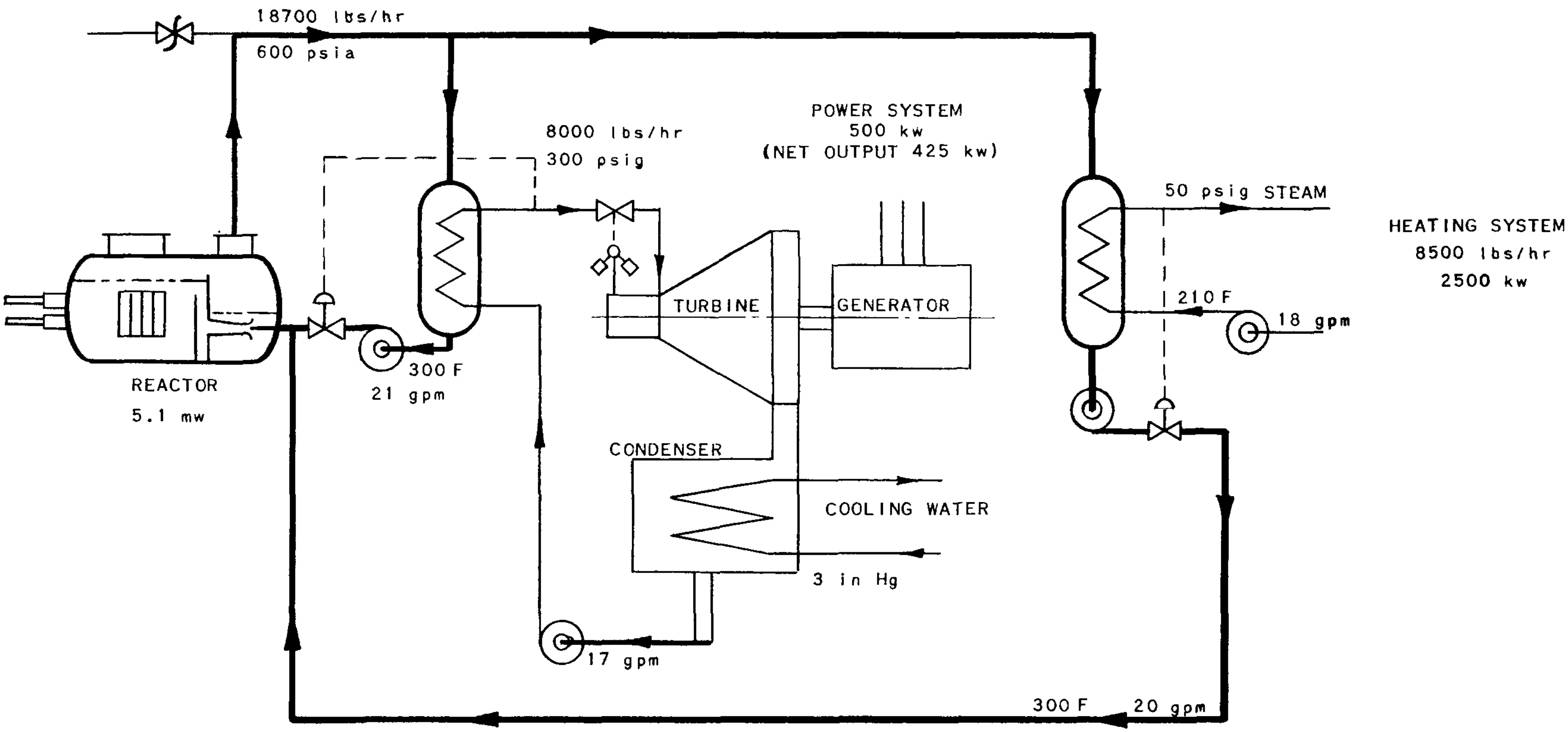


FIG. 16
POWER AND HEATING PLANT -
INDIRECT STEAM CYCLES

The evaporator, in order to be accessible, can be installed in a room separate from the reactor. However, in case it is essential to keep the space requirement down to a minimum, the secondary steam coils can actually be built as a unit inside the steam compartment of the reactor pressure vessel without increasing its size. The end of the vessel through which the coils are installed may be built with a large flanged opening for a bolted lid, or the coil system could be installed directly in place before the closure is welded.

The heat exchanger could conveniently be designed as a "Once-Through" boiler with 300-psi secondary steam being generated in the coils. The primary steam would be condensed within the pressure vessel, thus avoiding radioactivity in the outside steam cycles.

Case II

Separate plants.

Power plant using primary steam, full expansion.

Heating plant using separate indirect steam cycle.

Power Cycle

500 kw generator output (net demand 425 kw).

Medium: Primary steam, 7,000 lb/hr, 600 psia, condensed at 3 in. Hg.

Water rate: 14 lb/kwh.

Required from reactor: 2.3 mw power.

Heating Cycle

Heat demand: 2,500 kw.

Medium: Secondary steam, 8,500 lb/hr at 50 psig, condensed and cooled to 210F.

Required from reactor: 2.5 mw power, 9,100 lb/hr, primary steam condensed and cooled in heat exchanger to 300F.

Reactor

4.8 mw total power.

1/2 16,100 lb/hr steam at 600 psia returned as feed water at 220F (average).

Operating period for 2 kg fuel burnout = 16 months at 70% of capacity.

Over-all plant efficiency $= \frac{(425 + 2{,}500)}{4{,}800} \times 100 = 61\%$

<u>Advantage:</u> Detached power and heat operations.

<u>Disadvantage:</u> 1) Radioactive power steam.
2) Maintenance of vacuum in active condenser.

Case III

Power: Topping turbine with evaporative condenser.

Heat: Secondary steam produced in turbine condenser.

Power Cycle

500 kw generator output (net demand 425 kw).

Medium: Primary steam, 17,000 lb/hr, 600 psia, through "topping turbine," condensed at 80 psig.

Water rate: 34 lb/kwh.

Required from reactor: 4.5 mw power.

Heating Cycle

Heat demand: 2,500 kw.

Medium: Exhaust heat transferred in turbine condenser to secondary steam cycle. 8,500 lb/hr at 50 psig utilized out of a total of 13,600 lb/hr secondary steam. The balance of the secondary steam is condensed in an auxiliary radiator.

(In case power demand goes down to 60% normal all exhaust heat may be utilized).

Reactor

4.5 mw total power.

17,000 lb/hr steam at 600 psia returned as feed water at 324F.

Operating period for 2 kg fuel burnout = 17 months at 70% of capacity.

Over-all plant efficiency: $\dfrac{(425 + 2,500)}{4,500} \times 100 = 65\%$

<u>Advantage:</u> 1) No condenser vacuum.
2) Small turbine and condenser size.

<u>Disadvantage:</u> Active power steam.

Case IV

Topping turbine with circulated water heating system. A similar system is illustrated by the simplified flow diagram Fig. 7.

Power Cycle

500-kw generator (net demand = 425 kw).

Medium: Primary steam: 11,000 lb/hr at 600 psia through turbine, condensed at 5 psig.

Water rate: 22 lb/kwh.

Required from reactor: 3.25 mw power.

Heating Cycle

Heat demand: 2,500 kw.

Medium: Exhaust heat transferred in turbine condenser to 400 gpm circulated water heating cycle. 2,700 kw heat available out of which 2,500 kw utilized with temperature drop 212-170F.

Reactor

3.25 mw total power.

11,000 lb/hr steam at 600 psia returned as feed water at 228F. Operating period for 2 kg fuel burnout = 2 years at 70% of maximum capacity. Lute

$$\text{Over-all plant efficiency} = \frac{(425 + 2,500)}{3,250} \times 100 = 90\%$$

Advantage:
1) No condenser vacuum.
2) Small power unit size.
3) High fuel efficiency.

Disadvantage: Active power steam.

APPENDIX D

FORCED CIRCULATION IN HORIZONTAL BOILING REACTORS

Appendix D describes how the principle of forced circulation might be applied to the horizontal boiling reactor in case development should show the need for greater power densities than can be obtained by means of natural convection alone.

The theory behind the use of forced circulation or induced drive pressure is as follows:

Any given boiling reactor with a given position of its control rods, a given pressure and a negative "void coefficient" will have a reactivity which is inherently regulated by the steam volume entrained in the moderator at any time. With a very simplified expression we can state that:

Negative void coefficient maintains constant void volume or a constant ratio steam production/velocity. Therefore, steam production, or reactor power, is proportionate to the steam velocity.

The steam and water velocities in the core (circulating velocities) depend on equality between pressure losses in the channel and the drive pressure available for the circulation.

In case of natural circulation both the friction losses and the drive pressure are functions of the entrained steam volume (void %); both are of a relatively small magnitude and not too predictable.

If, on the other hand, a liberal drive pressure were to be provided from an outside source the situation would be simpler because the drive-pressure could then be regulated at will so as to produce and hold any desired fluid velocity (and power level) regardless of the nature of the two-phase friction characteristics and substantially without aid from the control rods. The distribution of water could be secured by an orifice at each element.

The introduction of circulating pumps would certainly add to the investment as well as the maintenance cost, but it should add considerably to the potential capacity of any given size of reactor and to the stability of operation. It would also provide an excellent medium for control of the power output. It should be noted that the capacity of the small boiling reactors is limited by the steam voids, or the rate of steam removal rather than by the surface heat flux which could probably stand to be doubled after the natural convection of steam has reached its limit.

Figure 17 is the same reactor which was shown on Fig. 2 with the difference that forced circulation has been added. Figure 18 shows the reactor and circulation system installed in connection with the previously described power plant.

The injector tubes are placed at such angle that, with the pumps shut down, the water is prevented from syphoning from the reactor compartment through the pumps back to the steam compartment. For this reason a "vent pipe" with check valve connects the high point of the loop to the steam space in the reactor vessel.

Since 5-1/2 mw reactor power corresponds to an average heat flux of only 31,000 Btu/(hr)(sq ft), it should be fairly safe to expect that the power could be increased to 11 mw in case the induced circulating velocity were increased to twice the velocity expected from natural circulation. This would mean a liquid velocity of 5.8 fps instead of 2.9 at the top end of the cooling channels.

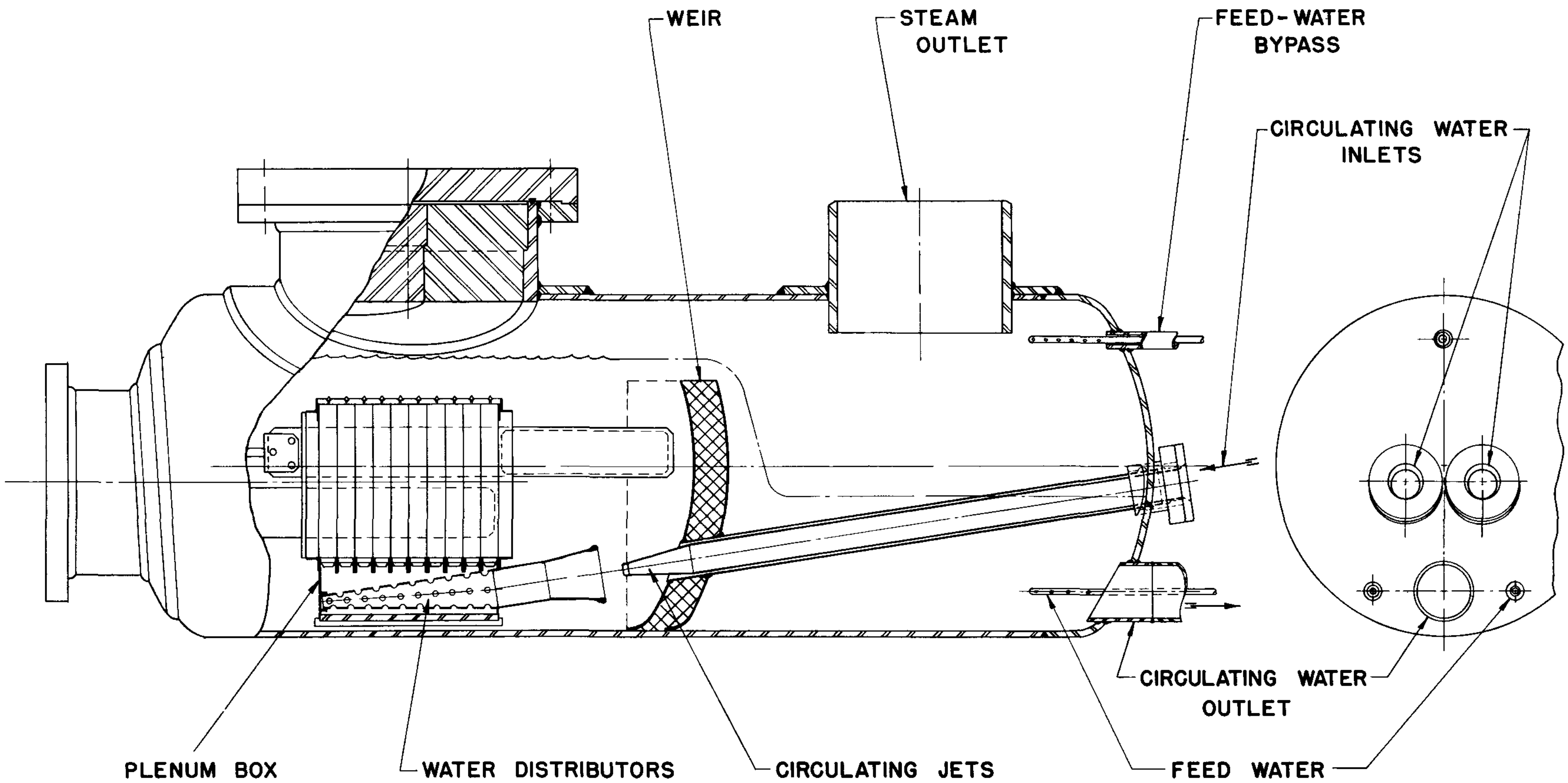


FIG. 17
FORCED CIRCULATION
BOILING REACTOR

0 3 6 9 1 2 3 4
SCALE IN FEET

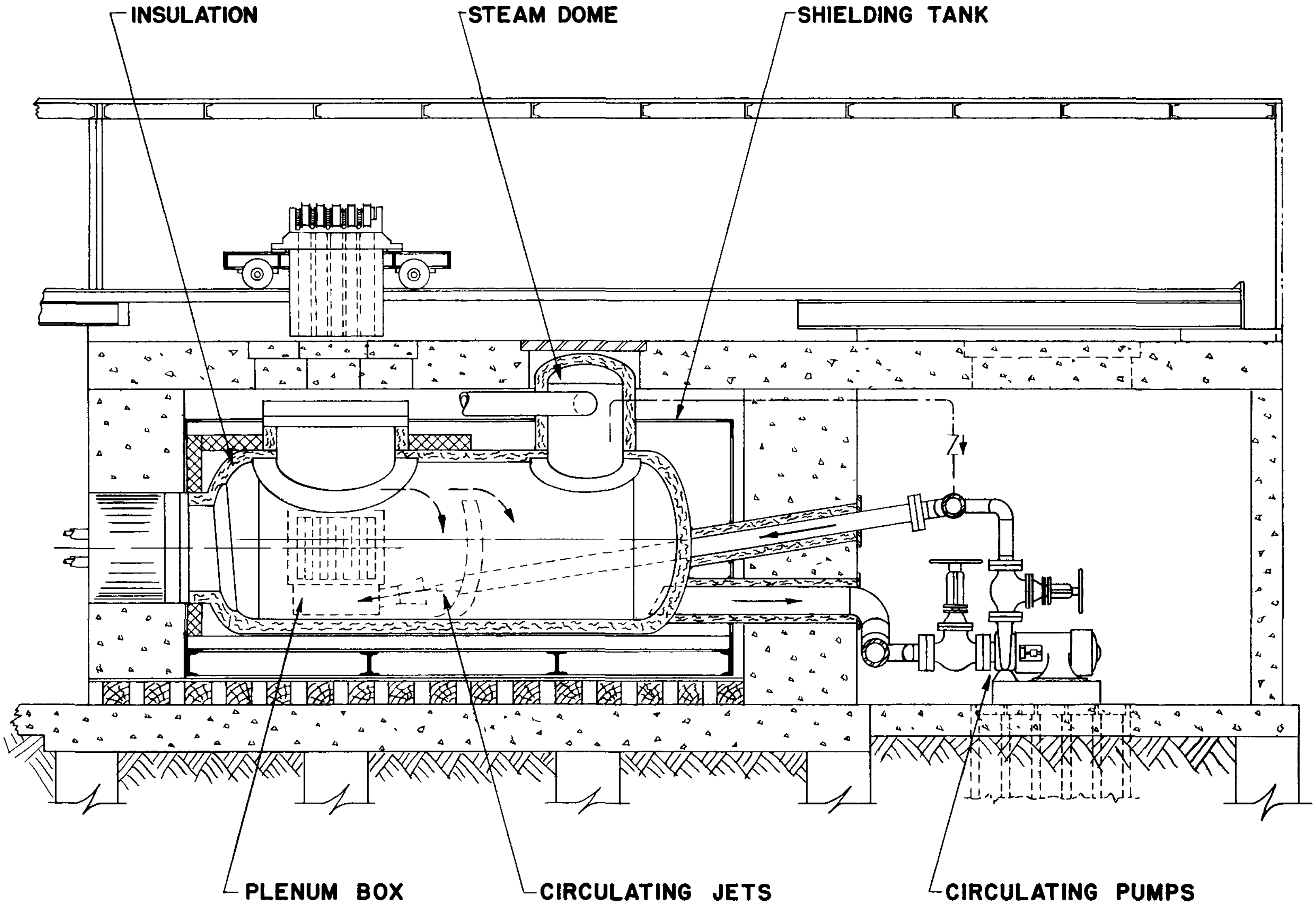


FIG. 18
FORCED CIRCULATION
INSTALLATION

0 1 2 3 4 5 6
SCALE IN FEET