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ARGONNE NATIONAL LABORATORY
P. O. Box 299
Lemont, Illinois

A STUDY OF HEAVY WATER CENTRAL STATION BOILING REACTORS (CSBR)

by

M. Treshow, D. Shaftman, L. Templin,
M. Petrick, B. Hoglund, and L. Link

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I. INTRODUCTION: BACKGROUND AND DESIGN OBJECTIVES

The reactors which are described and studied in this report are of the boiling heavy water type. They are large reactors designed for economic operation in stationary central power plants.

From the outset of this study certain objectives presented themselves as being of particular importance if an attractive economy were to be obtained. If such objectives could all be incorporated in one design, so much the better. Actually, we are confronted with the usual problem of balancing between special requirements which vary with policies of different countries.

The principal objectives referred to above are as follows:

1. use of ceramic (UO_2) rather than metallic fuel;
2. use of natural uranium;
3. high conversion ratio;
4. long fuel life;
5. natural convection for heat transfer;
6. production of superheated steam;
7. a high power output;
8. a limited size of pressure tank;
9. fuel handling without use of large pressure lid; and
10. moderator control.

Two different types of boiling reactors have been considered. One is the pressure tank type and the other is the pressure tube type. Both are feasible and both have their special advantages and their special problems to be considered. At the time of the present writing, only the pressure tank reactors have been analyzed in sufficient details to be reported here.

The following is a qualitative discussion of the above listed design objectives, their interdependence and the conflicts between them.

Item 1, ceramic fuel, is considered a necessary requirement for any water-cooled power reactor of the size contemplated. A metal-water reaction would appear to be too great a risk. It is realized that this restriction means a limitation on reactivity and on several of the other objectives which are stated.

Once the UO_2 is adopted, the main "conflict" is between items 2 and 3 (natural uranium and high conversion) on the one side and items 4, 5 and 6 (fuel life, natural convection and superheated steam) on the other. Natural convection requires large channel spacing in the fuel elements, which increases resonance capture, thereby jeopardizing the use of natural uranium. Likewise, superheating of steam requires a certain amount of stainless steel in the reactor core, also jeopardizing the use of natural uranium.

Superimposed on this situation, it is found that items 7 and 8, (that is, the desire for a large power production within a limited size of reactor) will influence the combination of the above items. A lower power output would permit greater lattice spacing which would improve the resonance escape probability and the reactivity - up to a point - where the "void coefficient" and the operating stability would be endangered.

The largest pressure tank to be considered was, somewhat arbitrarily, taken to have a 16-ft inside diameter.

The long fuel life (item 4) is, of course, very important in the economic picture. It is aided by natural uranium with a high conversion ratio, provided that a sufficient amount of excess reactivity is available. Otherwise, it can only be bought by the use of more or less enrichment of the fuel.

Objectives 9 and 10 (fuel handling and moderator) do not conflict with any of the other.

The objectives which should be given the highest priority are not likely to be the same in different parts of the world. In some cases, the use of natural uranium fuel is considered to be the most important quality. Even so, this demand is sometimes modified to apply only to purchased fuel, whereas it is recognized that plutonium cycling will later be able to provide useful enriched material originated in the reactor proper. In Europe, for instance, it might be visualized that a boiling heavy water reactor could be built which, for a few years, would operate at reduced power with natural uranium and a large lattice spacing. Later on, a different reactor core containing re-cycled enriched fuel could be installed for a maximum power production.

In other places, particularly in U.S.A., the use of natural uranium in central station power reactors is not in itself important. The ultimate power cost is the determining factor.

In view of the above situation, a total of four different types of power reactors have been presented in this report, one placing particular emphasis on the use of natural uranium and the other three with more emphasis on longer fuel life, high thermal efficiency and other items which help bring the total power cost down. These reactors are, briefly:

1. a natural uranium, boiling heavy water reactor with forced circulation: "Type 500 FC";

2. a boiling reactor with natural convection and slightly enriched fuel: "Type 500 NC";

3. and 4. heavy water boiling and superheating reactors using slightly enriched fuel and natural convection, "Type 800, A and B."

The proposed loading and unloading arrangement is one which allows a number of fuel elements to be handled from each of the relatively small unloading openings in the top of the reactor vessel. The control system referred to is one which essentially consists of a number of stationary, although removable, "Control Cylinders" with heavy water levels regulated individually or in gangs in accordance with desired changes in reactivity. This system will do away with moving parts within the reactor vessel.

The following chapter on summarized operating data indicates the degree of success achieved in meeting the ten above-named objectives.

II. SUMMARIZED ENGINEERING DATA

A. Operating Data

Table I contains such data which are considered significant for evaluation and comparison of the four reactor types considered here for production of saturated or superheated steam.

Table I

SUMMARIZED OPERATING DATA

Reactor Type	500 FC	500 NC	800 A	800 B
Type of fuel (ceramic UO_2)	Natural	Enriched	Enriched	Enriched
Steam quality	Saturated	Saturated	340° Superheat	300° Superheat
Coolant circulation method	40-psi pump	Natural Circulation	Natural Circulation	Natural Circulation
Total reactor power, tmw	985	1100	1000	1150
Generated electric power, mw	280	310	325	370
D_2O in reactor, short tons	125	125	125	125
Operating factor	0.80	0.80	0.80	0.80
Average D_2O temperature outside shrouds, °F	155	155	165	165
Exit steam pressure, psia	750	750	725	725
Exit steam temperature, °F	510	510	850	810
Power used on superheating, % of total	0	0	16.8	15.5
Steam cycle efficiency, %	28.2	28.2	32.6	32.2
Maximum exit steam void in shrouds, %	75	75	72	75
Maximum boiling heat flux (unflattened flux pattern), BTU/ft^2 hr	400,000	373,000	242,000	390,000
Maximum surface temperature (superheater), °F	none	none	1200	1200
Steam production, lb/hr	2,980,000	3,300,000	2,500,000	2,900,000

Special physics operating data are summarized in Table II for convenient reference.

Table II

SUMMARIZED PHYSICS DATA FOR INITIAL OPERATION

Reactor Type	500 FC	500 NC	800 A Super-heating	800 A Boiling	800 B Boiling	800 B Super-heating
ϵ	1.025	1.015	1.01	1.018	1.015	1.015
p (control tubes filled with D_2O)	0.860	0.769	0.834	0.731	0.769	0.764
η_f (including equilibrium xenon)	1.200	1.398	1.240	1.480	1.398	1.369
Assumed η U^{235}	2.06	2.06	2.06	2.06	2.06	2.06
κ_∞	1.058	1.091	1.044	1.101	1.091	1.062
L^2 (cm^2)	165	95	95	70	95	70
Fermi Age τ (cm^2)	170	205	250	220	205	255
Reactor B^2 (cm^{-2})	0.000134	0.000134	0.000134		0.000134	
Local excess reactivity in fresh zone, %k	1.2	4.6	(flat flux zone) 3.0	5.9	4.6	1.7
Initial conversion ratio (excess reactivity controlled by p)	0.887	0.904	0.528	0.911	0.904	0.723

B. Design Data

The following tabulation (Table III) contains information regarding engineering and design of the four types of reactors. The data should be seen in connection with Figs. 1 through 21.

Table III

SUMMARIZED DESIGN DATA

Reactor Type	500 FC	500 NC	800 A	800 B
Reactor power level, tmw	985	1100	1000	1150
Generated electric power, emw	280	310	325	370
Coolant circulation method	Forced	Natural	Natural	Natural
Pressure vessel ID, ft	16	16	16	16
Pressure vessel inside height, ft	40	40	40	40
Average core diameter, ft	13.50	13.75	13.75	13.75
Core height (active), ft	14	14	14	14
Hexagonal lattice spacing of fuel columns, in.	9.5	8	8	8
Boiling Zone				
Number of fuel columns	228	330	276	270
Inside diameter of inner shroud, in.	4.85 (avg.)	6.00	6.20	6.00
Thickness (Zr) of inner shroud, in.	0.02	0.02	0.02	0.02
Thickness (Zr) of outer shroud, in.	0.04	0.04	0.04	0.04
Outside diameter of outer shroud, in.	5.05 (avg.)	6.20	6.40	6.20
Number of fuel pins per assembly	85	69	85	69
Lattice spacing of fuel pins, in.	0.46 avg. (hex)	0.60 (square)	0.60 (hex)	0.60 (square)
Outside diameter of fuel pins, in.	0.344	0.344	0.344	0.344
UO_2 fuel pin diameter (net), in.	0.31	0.31	0.31	0.31
Zr clad thickness, in.	0.016	0.016	0.016	0.016
Number of control cylinders	31	55	48	37 + 18 partials
Control cylinder, ID, in.	9.75	8.50	8.50	8.50
Control cylinder Zr wall thickness, in.	0.125	0.125	0.125	0.125
Superheating Zone				
Location of superheating zone	none	none	center of core	periphery of core
Number of fuel columns	-	-	54	60
Inside diameter of inner shroud, in.	-	-	5.15	5.15
Thickness (SS) of inner shroud, in.	-	-	0.020	0.020
Thickness (Zr) of outer shroud, in.	-	-	0.040	0.040
Outside diameter of outer shroud, in.	-	-	5.35	5.35
Number of fuel pins per assembly			85	85
Outside diameter of fuel pin, in.	-	-	0.344	0.344
Lattice spacing of fuel pins, in.	-	-	0.50 (hex)	0.50 (hex)
(UO_2 + MgO) fuel pin diameter (net), in.	-	-	0.322	0.322
SS clad thickness, in.	-	-	0.010	0.010
Number of control cylinders			7	18 partials
Control cylinder ID, in.	-	-	8.50	8.50
Control cylinder Zr wall thickness, in.	-	-	0.125	0.125

Table IV shows the initial fuel loading in the boiling and superheating zones of the reactors.

Table IV

FUEL LOADING

Reactor Type	500 FC	500 NC	800 A	800 B
Boiling Zone				
Number of fuel columns	228	330	276	270
Number of pins per assembly	85	69	85	69
Uranium content in zone, kg	34,530	40,590	41,810	33,210
Uranium per fuel column, kg	151.5	123	151.5	123
Enrichment, U^{235}/U (atomic ratio)	0.00714 (natural)	0.0115	0.013	0.0115
U^{235} content in zone, kg	244	461	537	377
U^{235} per fuel column, kg	1.07	1.40	1.95	1.40
Superheating Zone				
Number of fuel columns	0	0	54	60
Number of pins per assembly	-	-	85	85
Uranium volume ratio, $UO_2/(MgO + UO_2)$	-	-	0.30	1.0
Uranium content in zone, kg	-	-	2650	9780
Uranium per fuel column, kg	-	-	49.0	163
Enrichment, U^{235}/U	-	-	0.030	0.019
U^{235} content in zone, kg	-	-	78	184
U^{235} per fuel column, kg	-	-	1.44	3.07
Total Fuel in Reactor				
Uranium content, kg	34,530	40,590	44,460	42,990
U^{235} content, kg	244	461	615	561

III. PLANT DESCRIPTION

A. Reactor Component

1. Pressure Tank

a. Design and Fabrication

The pressure tanks for the reactors herein described vary only in regard to the arrangement of nozzles for steam and circulating water. They are shown in elevation in Figs. 1, 3, 5 and 7, while Figs. 2, 4, 6 and 8 show the horizontal cross sections and plan views.

The main tank dimensions are identical in all four cases. The inside tank diameter is 16 ft and the total inside height is 40 ft. They are all designed for the same operating pressure, 750 psi, and wall temperature, 600°F. The reactor power is approximately 1000 thermal mw.

In the types 500 FC and 500 NC reactors the saturated steam outlets are placed near the top of the reactor tanks. The outlet nozzles for superheated steam in reactor types 800 are placed in the bottom head. The forced circulation used for the Type 500 FC reactor involves 6 outlet and 6 intake openings for recirculated D_2O coolant.

The design and construction of these pressure vessels amount to another step forward in a development already in progress, although perhaps not as big a step as one might think. Larger pressure tanks, such as the 60-ft Calder Hall gas-cooled reactor vessel, have been built. Also, tanks for much greater pressures have been built. The Shippingport reactor is built not only for a greater pressure, but the pressure multiplied by the diameter (2000 psi x 9 ft) is also higher than required for the four reactors presented (750 psi x 16 ft); this means that the hub stress per lineal inch will be less in case of the CSBR 16 ft tank.

However, the total weight and size of the 16-ft tank will make it necessary to ship it in sections A considerable part of the welding will have to be done at the place of erection, as in the Calder Hall tank or as similar to the construction of a ship or submarine pressure hull. This is actually no great disadvantage. It is felt that whenever a 1000-mw reactor would be built, there would be available a substantial building structure including as much crane and other construction facilities as found in any first-class welding shop. Expert welders with expert equipment could easily be installed in the new power plant building for use while this work was going on. Any required heating for stress relieving of the tank could be done by building a temporary fire brick oven around it in the same way as if the work were done on the home grounds.

It is intended that all separate sections of the tank be shaped and machined as required in the home factory before shipment, although some reboring of nozzles could be performed locally. Fortunately, these reactors do not have any of the usual large diameter, gasketed pressure lids. They have only the 9-in. or 10-in. gasketed fuel-handling thimbles. Water and steam connections would, of course, be welded locally as in some other reactors.

A high grade carbon steel with a tensile strength of 76,000 psi and elongation 31% (in 2 in.) should be used for the tank shell and heads. This corresponds to the SA-212 Grade B carbon steel used for the EBWR. On this basis the tank wall would be approximately 5 in. thick. A low alloy steel with higher strength could be considered.

The carbon steel is clad on the inside surface with stainless steel Type 304 in accordance with present technique. In cases where nozzles are used for water or steam of a temperature different from the pressure shell, thermal sleeves will be used to equalize the expansions or contraction. This applies particularly to superheated steam outlets and to feedwater inlet.

b. Tank Support Brackets

Twelve support brackets are welded to the knuckle section of the lower head, continuing upward along the cylindrical shell surface.

c. Internal Thermal Shield

The pressure shell is protected against overheating due to gamma absorption by means of a 2-in. thick steel shell installed inside the tank. This shield will be fabricated and installed in sections with a clearance of about 1 inch from the tank wall. The flow of coolant is such that a small amount of water or steam will flow through this clearance, thereby preventing overheating of the vessel.

d. Core Supporting Structures

The design of the core support depends on whether the reactor is to operate with natural convection or forced circulation. The difference appear in Figs. 1 through 8. In case of natural convection (Types 500-NC and 800), the core must be supported in a manner which offers little or no restriction to the recirculation of the coolant from the downcomer area to the space under the core. Such supporting structures are shown in Figs. 3 through 8.

In the case of forced circulation (Type 500-FC, Figs. 1 and 2), the situation is the opposite. The core must be supported continuously along its outer edge in such a way that only a small fraction of the recirculated coolant will leak around the fuel core. The supporting ledge, in this case, is integral with the thermal shielding shell which, in turn, is fastened to the pressure shell. Provisions are made to allow relative movement due to thermal expansions.

2. Core Design and Structure

a. General Design

The reactor core is composed of 228 or 330 "fuel columns" placed in a "core structure" designed to secure the fuel columns in their correct place with the predetermined volume of moderator between them. The fuel columns will be described in Section A-3. Essentially, they are tubes or shrouds which contain the fuel elements and provide proper flow channels for the coolant.

The core structure consists of three main items: a lower grid, an upper grid, and a downcomer shroud. The fuel columns are guided in the grid openings. The grids are made of stainless steel and installed at elevations 18 in. above and below the active fuel core. The lower grid forms a continuous horizontal platform with machined apertures in which the bottom ends of each fuel column are supported so as to be relatively leak tight. The individual apertures are exchangeable fixtures. The upper grid is for lateral support only. It is supported by the thermal shield, allowing for relative expansions, and it is so designed that the fuel columns can be taken in and out of position without first being lifted up above this grid (see Section A-5).

The grids will be fabricated in sections and assembled in the tank during erection.

The "downcomer shroud" is a 3/16-in. thick zircaloy sheet which forms a wall between the reactor core and the downcomer area. The bottom edge of the shroud is fastened to the outer edge of the bottom grid plate. The upper edge of the shroud reaches a level 6 in. below the top of the active core. It is supported against the thermal shield wall in a manner which allows for thermal expansion.

The downcomer shroud, in connection with the lower grid plate, forms a container for "cold moderator." The moderator is normally cold because it consists of feed water with a temperature not much above that of the turbine condensate. This feed water enters the reactor core inside of the shroud just above the bottom grid (see Figs. 1, 3, 5 and 7).

As the moderator feedwater gradually rises through the core, its temperature increases by approximately 100°F. At the top level of the core the water spills over the edge of the downcomer shroud and, together with recirculated water separated from the steam, will descend through the downcomer area to the bottom level of the core. There it enters the fuel columns either as a result of natural (gravity) circulation or through the use of circulating pumps.

The downcomer shroud, therefore, serves two purposes. It maintains a low moderator temperature within the core, and it provides a balanced non-turbulent flow of moderator with evenly distributed temperature throughout any horizontal cross section of the core.

In case of the natural circulation types, reactors 500 NC and 800 B, a special arrangement (Figs. 3 and 7) makes it possible to operate with extra subcooling in the central, high-flux zone of the core. This feature is accomplished by diverting a regulated portion of the feedwater to two ring pipes shown just below the central section of the core. The cold water is sprayed upwards and mixed with the hot downcomer water, resulting in greater subcooling for the central zone.

The present reactor calculations have not taken advantage of this feature. It is included in the design only to make available an extra safety margin in regard to permissible exit steam void where natural circulation is depended on.

b. Reactors Designed for Boiling Only

This case is shown in connection with the forced circulation boiling reactor Type 500 FC and the natural circulation reactor, 500 NC.

In the case of reactor 500 FC (Figs. 1 and 2), the lower grid is supported along its edge in contact with the tank wall and it is built with strong, girder-like ribs so as to withstand the load on the long free span. The circulating pump pressure under the grid is about 30 psi above the steam pressure. The lattice spacing for the 228 fuel columns in this reactor is $9\frac{1}{2}$ in. Orifice rings in grid apertures secure proper flow distribution.

In the case of Type 500 NC (Figs. 3 and 4) the grid plate is supported near its center and is free at the edges. It holds 330 fuel columns spaced 8 in. center to center.

c. Boiling and Superheating

The reactors designated as Types 800 A and B (shown in Figs. 5 through 8) operate with natural circulation. Both have 330 fuel columns spaced 8 in. center to center. It will be noticed that, also in these cases, the lower grid is kept clear along the edge, allowing the downcomer water to freely enter the space under the core.

The superheating of the steam is done in fuel columns which are placed together in separate zones of the core. The saturated steam from the upper steam dome flows down through these columns at a high velocity.

The difference between Types 800 A and 800 B is essentially that the former has the superheating zone located in the center of the core, while the superheating zone in the latter is a narrow annulus in the outer region of the core. The exit steam manifold in Type A is, therefore, a cone-shaped funnel, concentric with the core (see Fig. 5). Fifty-four superheating fuel columns connect to this manifold, and the exit steam pipe extends through a thermal sleeve in the center of the bottom head of the tank. The grid is supported on the central structure.

In case of Type 800 B, the exit steam manifolds consist of six horizontal cylinders placed under the peripheral superheating zone (see Figs. 7 and 8). Each manifold takes care of ten fuel columns. The exits through the tank head consist of six individual 12-in. steam pipes, joined at some point outside of the shielding.

Manifolds and pipe connections are insulated from the steam by means of inside stainless steel linings, placed with a gap toward the actual pipe wall.

The reactor core in this design is supported on the manifold walls which transmit the load to the tank head (see Figs. 7, 8 and 10). The design provides for individual thermal expansion of all parts, such as grid plate versus tank head. Self-aligning seal plates placed around the fuel columns and resting on the grid plate prevent leakage of cold moderator from the reactor core to the space under the grid. (Such leakage would, in any case, be slight due to the low pressure difference from natural convection.)

In the central superheater, Type 800 A, the intention is to use a flat flux distribution across this zone. All the fuel columns work under the same conditions. In case of Type 800 B, the superheating zone near the periphery of the core cannot have a uniform flux distribution. For this reason, the steam flow through individual columns will be adjusted in proportion to their power output. This adjustment is done by means of orifice openings placed as restrictions in some of the individual spouts on the steam manifolds (see Fig. 10). This method leaves the superheating fuel columns interchangeable and it secures equal temperature conditions in all of them.

3. Fuel Columns (Shrouds and Fuel Element Structures)

In the present description a "fuel column" is taken to mean one of the long cylindrical tubes or shrouds loaded with a number of fuel elements, which are stacked on top of each other to a height of 14 ft, the height of the active core. The actual shrouds reach up above the active core, particularly those for the superheating columns, which are designed to reach 3 ft above the maximum water level or $6\frac{1}{2}$ ft above the core.

In case of the fuel columns in the natural circulation boiling reactors, the shrouds continue 3 ft above the core height. This is done to obtain the extra circulating drive pressure from these 3-ft "chimneys" with high steam voids and very low friction losses. Of course, the forced circulation reactor, 500 FC, needs no chimney effect. The top ends of the shrouds are perforated to allow a gradual exit of the steam into the D_2O top reflector. This is done to increase the negative "power coefficient," which in this particular case is rather close to zero.

The shrouds have double walls with a narrow stagnant water or steam gap between. In these reactors all the shrouds are made of zircaloy-2 except the inner shroud on fuel columns used for superheating, in which case stainless steel is used. Figs. 9, 10 and 11 show examples of fuel columns and shrouds used respectively for boiling and for superheating. The lower ends of the shrouds are extended into conical seats which support the units and seal the water or steam flow.

The individual fuel elements are made in heights of 28 in. each. Six elements are connected over each other into one 14-ft fuel column. All of the elements are built according to the same general pattern. They consist of either 69 or 85 fuel pins arranged in square or triangular lattice and connected at the ends by grids designed in a special manner, which will offer the least possible obstruction to the flow. Details of a fuel pin pattern and the connecting grid structures are shown in Figs. 12 and 13. The grids will be made by spotwelding of pressed metal strips and bushings of the same metal as the pin cladding (zircaloy or stainless steel). Each grid is spotwelded to an inner-shroud section.

The pin spacing and subsequent shroud diameter varies with the special application of the element. In case of natural circulation the pin spacing in the boiling zone must be large, (0.60 in. center to center). This requires a shroud with an inside diameter of 6 or 6.2 in. In the case of forced circulation, say Type 500 FC, a narrower fuel element can be used with shrouds of, for instance, 4.85 in. diameter. The advantage of this is that the neutron resonance absorption is decreased and natural uranium can be used.

Likewise, in case the fuel column is designed for superheating of steam, the pin spacing will be comparatively small in order to obtain a high velocity of the steam and optimum heat transfer conditions. This is the situation in the superheating zones of Types 800.

4. Fuel and Fuel Pins

The fuel pins consist of uranium oxide (UO_2) in the form of ceramic pellets enclosed in 0.344-in. o.d. tubing, which is made of zircaloy for boiling fuel elements and of stainless steel in case of superheating elements.

Reactor type 500 FC utilizes natural uranium. The fuel in the other types is more or less enriched (see Table IV). The individual fuel pins are approximately $27\frac{1}{2}$ in. long and are closed at each end by a welded end cap from which a tap is protruding 1/4 inch in the axial direction to be used for positioning and support of the pin in the fuel element grid bushings.

The fuel material used for superheating differs from that used in fuel elements for boiling. For example, stainless steel cladding results in a need for greater enrichment of the fuel. This would increase the heat flux as well as the heat generated per fuel pin in the superheaters. However, the dry steam coolant will not permit as high a heat flux as can be tolerated where the coolant is boiling water and, for this reason, it is necessary to decrease the heat generation per fuel pin. This is done by diluting the enriched uranium with a certain portion of an inert material, for example magnesium oxide.

In case of Type 800 A, where the superheating takes place in the central high flux zone, a considerable ratio of MgO must be used: 70 volume per cent.

Type 800 B requires no diluent for the superheating fuel, because this zone is located along the outside of the core where the neutron flux level is low. In this case, the allowable heat flux in the superheater will be greater than the heat flux in the adjoining area of the boiling zone.

The relation between heat generation in boiling zones and superheating zones can be seen from the two charts in Fig. 27, which show the variation of power with the distance from the center lines of the reactor.

The principal difference between the two superheat locations used on Types A and B appears clearly from the chart. The comparison indicates that Type B utilizes the potential power of the core more fully.

5. Fuel-Handling Arrangement

An important feature of the large tank-type reactors herein described is the special means for loading and unloading fuel without the use of large gasketed and bolted pressure lids such as are usually found in light water reactors. In the CSBR reactors, all the fuel is handled through a number of fuel exchange thimbles not larger than ten inches in diameter. The fuel-handling tool is shown in Fig. 14. It is so designed that, after it is lowered through one of the fuel exchange thimbles, it can reach out in six different directions for orientation above any one of six fuel columns at a distance of one lattice spacing from the center line. Therefore, a reactor core with 330 fuel columns can be handled from a total of 55 of these fuel exchange thimbles. The space in the core which is directly in line under a thimble could, of course, be used for another fuel column; but in the present reactors nearly all of these spaces are occupied by control devices in form of moderator control cylinders such as described in the following chapter. These cylinders are removable and the one in line with a certain fuel exchange thimble will be removed through this thimble to make the six adjoining fuel columns accessible for the tool.

For a core lattice spacing of 8 in., the center distance between each of the 9-in. fuel exchange thimbles will be 21.17 in. A $9\frac{1}{2}$-in. lattice spacing corresponds to a 25.14-in. center distance between openings.

Operation of the Fuel-Handling Tool

The fuel-handling tool, when ready to use, will be placed inside the fuel transfer coffin which, in turn, is mounted on a trolley. When the cover plate and shield plug have been removed, the control cylinder will first be taken out through the thimble by means of a special tool and placed in a lead coffin. Thereafter, the fuel-transfer coffin is brought into position over the thimble and the fuel-handling tool is lowered to its operating level within the tank. The levers on the tool are still in a folded position along the vertical axis. The tool has two lifting rings by which it is suspended from a dual hoist mounted on the coffin. By shifting the weight of the tool from one of these rings to the other, the levers will unfold and place the gripping device exactly over the particular fuel column which is to be removed. The tool is then lowered to make contact with the column. The tool with fuel column attached is then raised a distance of 12 to 24 in. Next the load is shifted back to the first ring, which will cause the levers to fold back again and place the fuel column in the center line of the thimble opening, at a level 4 in. below the unfolded position.

For a more detailed description of this system, Fig. 15 (in connection with Fig. 16) shows a cut-out of the reactor top-grid. It is so designed that it will allow any one of the fuel shrouds to be moved radially to or from the center line of the handling tool provided that the shroud is first raised 5 in. or more from its normal position in the core.

As shown on the drawings, each shroud is equipped with a short sleeve or belt of relatively large diameter located at the grid level and is so dimensioned that it will keep the shroud in position whenever it is all the way down in the core.

When the shroud is lifted, the sleeve will disengage from the grid bars and the shroud can then be moved freely in to the center line of handling tool and fuel exchange thimble.

The tool and the fuel column can thereafter be hoisted straight up into the coffin through a bottom gate valve. The transport of the coffin between reactor and storage canal is done by the trolley or a crane.

The unloading arrangement includes pneumatic operation of the gripper and also connections to provide for cooling of the hot fuel column while it is being moved. These connections are indicated in Fig. 14. To secure dependable operation, it is assumed that a periscope or, perhaps, a television camera will be inserted in the tank through another thimble.

It is obvious that the individual fuel columns must be steadied in their position in the top grid in a manner which will not allow them to move or shake; yet, any tight clearance must be avoided to prevent binding. Figures 15 and 16 show the holding device used for this purpose. An expandable "spider" is installed on the vent pipe of each of the control cylinders. When the cylinder is in place the six "feet" of the spider will press against the upper stainless steel ends of the adjoining six fuel columns due to the weight of the loose sleeve on the vent pipe. The upper end of the sleeve forms the knob by which the cylinder can be lifted; in case it is lifted to be removed, the spider feet will automatically withdraw from the fuel columns which are thereafter only loosely guided in the horse shoe-shaped cut-outs of the grid.

6. Reactor Control Devices

Neutron-absorbing control rods or plates could be employed in these reactors. The logical locations of the rods would be in the centers of each group of six fuel columns or in line with the fuel exchange thimbles indicated on the drawings. The drive mechanisms would be bolted to each of the tube flanges. A ball-screw drive system with canned motor is one of the types that might be used.

A good mechanical system is expensive, not only in regard to the investment and maintenance, but also due to the absorption of neutrons. For this reason the following system is recommended in the case of these reactors where conversion ratio and long time use of the uranium fuel are given a high priority.

a. D_2O Control Cylinders

This system is based on the idea of regulating the reactivity and flux pattern by adjustment of the moderator level in individual "control cylinders" in the reactor in a manner similar to the adjustments performed with control rods in other large thermal reactors. The advantage in using the moderator for control instead of poison is that most of the excess neutrons, far from being lost, are captured in uranium to yield additional plutonium. Fuel is therefore conserved.

In the case of reactor types with 330 fuel columns and an 8-in. lattice spacing, this system is considered to employ 55 D_2O control cylinders, each with an $8\frac{1}{2}$-in. inside diameter. If these cylinders are all emptied, approximately 1/3 of the cold moderator outside of the fuel columns will be removed. In case of the Type 500 FC reactor with its $9\frac{1}{2}$-in. lattice and 31 control cylinders ($9\frac{3}{4}$in. i.d.), the control capacity of the cylinders is smaller.

The control cylinders are held in place by means of their lower ends which are threaded and fit in apertures in the lower core grid (see Fig. 16). The apertures are installed with permanent pipe connections extending out through the tank wall to be connected to level indicators and to the control system. Figure 17 indicates the basic principles of the system.

The top ends of the cylinders are vented to the steam dome, $3\frac{1}{2}$ ft above maximum water level. The top ends of the level indicators are also connected to the dome.

Changes in the liquid levels can be accomplished either by means of small reversible positive displacement pumps or by quickly opening valves in case fast discharging is required. It is also intended to use pressure safety valves on the water lines; these will open immediately and discharge liquid moderator in case the reactor pressure should exceed a predetermined limit.

Finally, connections to a constant pressurizer tank can be utilized for automatic maintenance of a desired reactor steam pressure. This is possible because a decrease in reactor pressure would allow additional moderator to enter from the pressurizer, whereas an increase in pressure would force moderator out of the control cylinders into the pressurizer tank.

The natural variations in reactivity which result from the gradual burnout of fissionable material will be compensated by changes in moderator level inside the control cylinders. Ordinarily, it will be advantageous to maintain a fairly even distribution of power across the reactor (flattening). This condition can be approached if the control cylinders are regulated in such manner that the levels farthest away from the axis of the core are maintained as high as possible while the levels in the centrally located control cylinders are depressed as much as allowed by control of criticality.

Level indicators will at all times show the moderator level in each cylinder. Some of these levels will be regulated or maintained by automatic devices, but in most cases the cylinders will be used for shim control only. Each shim control cylinder will be either full or empty. The steps between shims will be taken care of by the first-mentioned control cylinders with continuous regulation of the moderator level.

b. Moderator Level

The regulation which can be effectuated by means of the moderator control cylinders can be supplemented by control of the general water level in the reactor. At normal operating level, the D_2O is several feet above the active core. By adjustment of the regulator mechanism which maintains the water level, the top reflector can be raised or lowered. In the 500 FC reactor, where forced circulation is used, the water level can safely be lowered even beyond the core level and an additional reduction in reactivity can be obtained.

c. Moderator Temperature

The temperature of the "cold" moderator has a significant influence on the reactivity. A valve arrangement on the feedwater system makes it possible to bypass more or less of the feedwater directly to the boiling water or steam dome and thereby regulate the moderator temperature over a range practically from the condensate temperature (about 100°F) up to 510 degree saturation temperature.

d. Poison Injection

A final emergency measure for control or runaway conditions is proposed in the form of a boron injection system, which can rapidly produce a large negative reactivity effect.

7. Shielding

The 2-in. interior thermal shield for protection of the pressure tank shell was described in Section A-1. Outside the shell are, first, 4 in. of stainless steel wool for thermal insulation. This is covered with airtight sheeting of stainless steel.

The reactor is placed in a concrete pit with 9-ft thick walls as its main shielding. This pit is, furthermore, lined with a 5-in. thick water-cooled thermal shield made of steel and lead. Cooling coils are also placed to a certain depth in the concrete walls and in the bottom of the pit.

The shielding design is not included in the drawings.

B. Steam System

1. General Description

a. Saturated Steam Cycle

Reactor Types 500 FC and 500 NC (see flow diagram in Fig. 18) supply saturated steam at 750 psia (511°F) to the combination steam dryer and shutdown condenser. The dry steam flows to the high-pressure turbine where it will expand from a throttle pressure of approximately 725 psia, to a pressure of about 55 psia. At this point the steam will be extracted with a moisture content estimated to be 14 per cent and passed through an intermediate mechanical steam separator. Steam with a quality of 99 per cent will leave this separator and enter the low-pressure turbine, where it will expand to the condenser back pressure, assumed to be $1\frac{1}{2}$ in. Hg. The final exhaust moisture is expected to be approximately 14 per cent.

Condensate from the condenser hotwell, along with deaerated makeup water, is pumped by the condensate pump through the steam air ejector condensers, generator hydrogen coolers, etc., to the reactor feed pumps. The condensate from the moisture separator between the high and low pressure turbines is pumped by an additional feed pump directly into the reactor vessel, where it mixes with the recirculating water. In the 500 FC reactor this condensate is pumped into the suction lines of the six reactor circulating pumps. This has the advantage of increasing the sub-cooling of the water and reducing the possibility of cavitation.

The thermodynamic cycle efficiency for the saturated steam systems was calculated to be 28.2 per cent. Assuming that normal steam plant auxiliaries consume 6 per cent of the gross electrical power, and the reactor circulating pumps on the Type 500 FC reactor consume 0.5 per cent, the over-all plant efficiencies were calculated to be 26.7 per cent and 26.8 per cent for the 500 FC and 500 NC types, respectively.

b. Superheated Steam Cycle

A schematic diagram of the superheated steam cycles associated with reactor Types 800 A and 800 B is shown in Fig. 19. Saturated steam generated in the boiling zone at a pressure of 750 psia (511°F) passes through the superheating shrouds where it is heated to 850°F in the "800-A" reactor and to 810°F in the "800 B" reactor. After leaving the superheater and manifolds the steam pressure will be approximately 725 psia. Allowing for additional losses in the external piping, the turbine throttle pressure is assumed to be 700 psia. The steam expands in the turbine to a condenser pressure of $1\frac{1}{2}$ in. of mercury with an exit moisture estimated to be 13 per cent for the "800 A" reactor and a 14 per cent for the "800 B" reactor. The cycle from the condenser hotwell to the reactor is the same as for the saturated steam cycle.

The cycle efficiency for the 850°F steam was found to be 32.6 per cent, while that for the 810°F steam was found to be 32.2 per cent. Assuming once again that the plant auxiliaries consume 6 per cent of the gross electrical power, the over-all plant efficiency will be 30.8 per cent for the 800-A reactor and 30.5 per cent for the 800-B reactor.

It may be noted that the proposed heat cycles make no provisions for feedwater heating. While regeneration will improve the cycle efficiency, the adverse effects of the warmer feedwater, such as reduced moderating ability and a reduction in the reactor power output, may more than offset this gain.

The external systems of both the saturated and superheated steam cycles will employ conventional power plant equipment modified to obtain "complete" recovery of the D_2O water and vapor leakage. The equipment in the steam and water systems will be constructed of relatively corrosion-resistant materials.

2. Turbo-Generators

It is expected that in each case the reactor will be connected to two turbine-generator units ranging in size between 140 and 180 megawatts. It is assumed that these units will be built as conventional equipment with the following exceptions: (1) modification of the seals for the shaft, housing flange, and throttle valve stem to prevent leakage of the D_2O (reference is made to ANL-5607 for details of turbine sealing methods used on EBWR); (2) plating of the internal turbine parts to prevent corrosion due to oxygen liberated by radiolytic decomposition; and (3) incorporation of a mechanical steam dryer between the high- and low-pressure turbines for the saturated steam cycle.

The main condensers will be of the deaerating type. No external deaerating heaters are contemplated because of the desire to maintain the moderator at the lowest possible temperature. A spray-type desuperheater will be incorporated to protect the condenser from thermal shock in the event that a sudden loss of load trips the turbine throttle and bypasses the steam to the condenser. In addition to this, the various reactor relief valves will be discharged into the desuperheater to prevent loss of D_2O into the atmosphere.

The possibility of contamination of the D_2O condensate in the main condenser by the condenser circulating water is held to a minimum by employing double tube sheets and a divided water box. In the event a failure of a condenser tube, valves in the circulating water lines will isolate the half of the condenser which contains the faulty tube. The contaminated water will be pumped to a storage tank for future reclamation.

The condenser system includes, in addition to the conventional type of steam-air ejectors, a mechanical air pump to be used for air and gas removal upon occasions where the steam pressure would be too low for the ejectors. The air pump is connected to the D_2O recovery system. In the case of plants operating with superheated steam, the system also incorporates a 30,000-cfm circulating air blower intended for the final stage of fuel cooling, as described in connection with shutdown procedure, Chapter IV-B. The blower is not included in the diagram.

3. D_2O Maintenance

a. Leakage Recovery

The main reactor feed pumps will operate at low water temperatures, approximately 100°F, and will be supplied with a drip recovery system to return any D_2O leakage to the storage tank. The feed pump associated with the turbine moisture separator (in the saturated steam cycle) will handle water at approximately 270°F. This temperature will necessitate a dry air seal to recover the D_2O leakage after it flashes to steam. Recovery of D_2O from valve steams, flanges, pump packing glands, etc., will be accomplished with bellow-type seals and air-tight housings which drain to the drip recovery system.

The condenser vacuum will be maintained with the usual two-stage, steam jet air ejectors as indicated in Fig. 20. The gases vented from the air ejectors will consist largely of deuterium and oxygen with possibly some radioactive, non-condensable gases. These gases will be vented through a catalytic D_2-O_2 recombiner (see Fig. 20). Leaving the recombiner, the D_2O vapor and noncondensable gases pass into a freon-cooled condenser and are cooled to 40°F to remove the major portion of the D_2O. The mixture from the cooler is estimated to contain 20 grains of D_2O per pound of dry gas, which will be reduced to 0.5 grain per pound of dry gas by passing through chemical dessicators. If sufficient radioactivity is detected in the remaining gases, they will be passed through charcoal absorber beds and allowed to decay before being released into the atmosphere. Since air inleakage to the system is expected to be quite low, the quantity of gases expelled to the atmosphere will be small; hence, the D_2O loss should be negligible.

The loss of D_2O vapor and the inleakage of air containing light water vapor at large valves, pump shafts, turbine shafts, etc., operating at sub-atmospheric pressures is prevented by sealing glands consisting of three annuli. The outer annulus is provided with dry air at a slight positive gauge pressure, while the inner annulus is supplied with D_2O steam, also at a slight positive pressure. The central annulus, evacuated to a partial vacuum by the D_2O recovery system, collects the escaping D_2O vapor and inleaking dry air.

The D_2O recovery system for the gland-sealing air is shown schematically in Fig. 21. The air and vapor from the various sealing glands enters the vent condenser where the vapor is condensed. Saturated air is drawn from the vent condenser and dried by first passing through a freon cooler, and then through chemical dessicators. This dry air is supplemented with dry make-up air before being readmitted to the sealing glands. Light water vapor is removed from the make-up air by a system identical to that used to recover the D_2O vapor. It is estimated that it should be possible to limit the inleakage of light water to less than one pound per day and the loss of D_2O to less than two pounds per day.

b. Purifications

Make-up water to the reactor is filtered and demineralized in a mixed bed ion exchanger. The water is added to the cycle through the condenser, where the necessary deaeration takes place. Because inleakage of light water is practically negligible, the extensive feedwater treatment facilities associated with conventional steam plants are unnecessary.

The reactor water is maintained at a constant purity by recirculating a small percentage of it through a clean-up system composed of appropriate filters and mixed bed ion exchangers. A regenerative heat exchanger and a secondary cooler, shown in Figs. 18 and 19, will cool the reactor water to approximately 120°F before it passes through the ion exchanger. Leaving the ion exchanger it will be reheated in the regenerative heat exchanger to approximately 350°F before being readmitted to the reactor. The secondary cooler will also serve as a shutdown cooler and is sized to dissipate one per cent of full reactor power. For operation in this capacity, the regenerative heat exchanger is bypassed.

The primary steam dryer used in the saturated steam system is designed to serve also as a normal shutdown condenser or an emergency condenser. It is rated to dissipate up to 8 per cent of full reactor power. Once the reactor decay heat has dropped to less than one per cent, the reactor cooling duty may be taken over by the secondary cooler in the water purification system. A similar shutdown condenser, without the steam drying feature, is incorporated into the superheater steam system.

4. Start-Up Heater

An oil-fired start-up heater, designed to heat the system at a rate of 15°F per hour, is employed whenever preheating of the water is desirable. The preheating is essential in the case of the superheating reactors to secure the initial cooling steam.

5. Local Activity and Shielding

Local activity in the steam lines, turbine, condenser, etc., has been shown by the BORAX reactors and EBWR to be an insignificant problem under normal operating conditions. Probability of major damage to the plant in the event of a ruptured fuel element will be considerably reduced for the CSBR reactors as a result of the use of oxide fuels. Local shielding problems were, likewise, found to be of small consequence. The ion-exchange systems appear to be the only component outside of the reactor where a substantial amount of shielding is required.

IV. PLANT OPERATION

A. Start-up Procedure

1. Start-up with Fresh Fuel Charge

Before the reactor is started with a fresh fuel charge, the water (D_2O) is preheated to approximately 200°F by means of an auxiliary start-up heater (see diagrams in Figs. 18 and 19). The moderator control cylinders described in Section III-A.6 will be kept drained during this period, and the general bulk moderator in the tank will also be held at a low level. This level will be raised only gradually as the water temperature increases.

Before criticality is reached a condenser air pump, which produces a suction in the whole steam system including the reactor tank, will be started. Removal of air from the system will reduce the subsequent contamination problem. The low pressure will, furthermore, make it possible for the water to start boiling in the fuel columns at a relatively low temperature and at a very low power level. This is important, not only for a smooth start of the boiling, but, even more, for the cooling of the superheating fuel columns which require a certain steam flow as soon as power is generated. Any liquid water should be drained from the superheating columns at this point, allowing the steam flow to pass through the channels.

As the reactor power is gradually raised the steam flow must be increased, which means that the pressure must not be allowed to rise to the point of saturation. During this first period of the start-up the steam will be bypassed directly to the condenser. Flow and temperature recorders will be used to secure safe conditions.

During initial operations the moderator control cylinders adjacent to the superheating zone will be so adjusted as to keep the local flux at a relatively low level until substantial power and steam flow rates have been reached. The turbines will then be started and the bypass line closed and final control adjustments can be made.

2. Start-up with Active Fuel

Starting the superheating reactor with a used fuel charge which is still highly active may be a problem. Yet, the reactor may have been cooled off and reduced to atmospheric pressure. For instance, after a total shutdown time of 20 hours the residual power is still 3/4 of a per cent of the operating power, or 8.6 mw of heat in case of the reactor Type 800-B, and the share of this to be credited to the superheater alone is 15.5% or 1.34 mw of decay heat.

During the shutdown the superheater, like the main section of the core, may be kept full of cooled water which keeps the temperature low. Before restarting the reactor the decay heat can be allowed to heat the reactor water to atmospheric boiling temperature and, when the superheater is connected to the condenser and drained, the reactor steam will flow through the superheater channels and maintain a permissible temperature of the fuel while criticality is being established and the power gradually built up.

It should be mentioned here that a 30,000-cfm air blower is incorporated in the cooling system for superheating reactors. The main function of this is connected with the final shutdown cooling of the superheating fuel before refueling, such as described under Section B.3 of this chapter. Nevertheless, the same blower would, of course, also be available to assist with the cooling during the first stages of the start-up procedure if this should be found a desirable precaution.

B. Shutdown Procedure

1. General Procedure

Whenever the reactor is to be refueled it must be shut down and cooled to a point safely below atmospheric boiling temperature. The general procedure is along the following lines. (See Figs. 18 and 19.)

The reactor will be shut down by dropping the moderator from all control cylinders. As the steam pressure drops, the turbines will be shut down and the remaining steam output bypassed to the de-superheaters and turbine condensers. Adequate time will be allowed for a slow decrease of pressure and temperature, thereby avoiding danger from thermal stresses.

When the pressure is too low for the air ejectors to operate, an air pump will be used to remove gases and to maintain a slight vacuum in the condenser.

This first stage of the shutdown operation will be extended over a period of about six hours, during which time the decay power level will have dropped to one per cent of the full operating power. By proper regulation of the steam by-pass valve the dome pressure will have dropped gradually to zero psig and the water temperature will be approximately 212°F.

2. Saturated Steam Reactors

In the case of a reactor without superheating, the continued cooling of the water can now be accomplished by means of the recirculating shutdown cooler, while the steam generated in the reactor is condensed, either in the turbine condenser or in the standby shutdown condenser in connection with the D_2O recovery system. A certain amount of dry air will be circulated through the reactor steam dome and condenser to improve the removal of gases and excess moisture. When the steam remains steady at atmospheric pressure it is safe to begin the refueling operation as described in Section A.5 of Chapter III. The fuel element temperature will only be slightly above the 212° cooling water temperature.

3. Superheating Reactors

The first stage of the shutdown and cooling of these reactors is the same as described above for saturated boiling reactors. The superheating reactors have the same advantage of being able to provide their own means of cooling (boiling water and steam) inasmuch as a sufficient flow of steam can be maintained from the boiling zone through the superheater as long as desired, even when the steam is gradually reduced to atmospheric pressure. The temperature of the superheating fuel elements will drop in spite of the generated decay heat, because the steam flow from the boiling zone will decrease only at the same rate as the decay heat in the superheater. In the case of reactor Type 800 B the following data have been estimated. When the shutdown power has decayed to one per cent of operating power and atmospheric pressure is reached, approximately 6 hours after the shutdown, the maximum fuel temperature in the superheating elements will have dropped to below 800°F, while the average exit temperature of the steam will be close to 650°F. At this time, the steam flow will amount to 30,000 lb/hr.

Figure 22 shows the pressure and temperature variations during this cooling period. The diagram is based, primarily, on the decay curve for uranium reactor fuel. This is plotted as shutdown power vs time. The superheating shutdown power is considered a constant ratio (15.5%) of the total power during the whole shutdown period. Furthermore, the pressure drop is considered to be regulated by the exit steam valve in such manner that the temperature of saturated steam in the reactor dome will drop linearly over a 6-hour period. This will even out the temperature drop in the pressure vessel shell and minimize thermal stresses.

Reactor power levels as related to temperatures and other qualities of the steam were used for calculating and plotting the maximum temperatures of the superheating surface at the different elapsed times.

Although safe shutdown cooling is secured by the steam flow, the superheating columns cannot be removed from the core before they have been filled with reactor water; this cannot be done safely unless the maximum fuel temperature is first brought down to 300°F or less.

Cooling of the superheating fuel elements from 800°F to a temperature not higher than 300°F at the hottest point is done by means of induced circulation of air in addition to the reactor steam flow. It is estimated that an air blower operating at atmospheric pressure and a flow rate of 30,000 cu ft/min will provide the necessary mass flow for the transfer of heat between reactor and condenser.

The steam and air flow coming from the reactor is cooled in the condenser to about 100° F. When the wet air re-enters the reactor dome it will be heated by the saturated steam which is still being formed in the boiling channels. A portion of this steam will thereby be condensed in form of a "fog" which will pass through the superheater columns together with the steam and air coolant. While the fog is drying on its way through the core the flow temperature remains close to 212° F. Thereafter the coolant temperature will increase until it reaches close to 275° at the exit. The maximum surface temperature is estimated not to exceed 300° F after equilibrium has been reached.

In order to extend the time used for cooling of the fuel pins, the air flow will be kept at a reduced rate to begin with and only gradually increased to 30,000 cfm.

When the minimum air and fuel temperatures have been reached, the draining of the control cylinders should be checked and the "bulk moderator" level dropped sufficiently to allow the superheating columns to be filled with liquid D_2O.

The safest method of entering the reactor water into the superheater shrouds is to first reverse the flow of air so that it enters the reactor through the superheater manifolds and leaves through the steam dome exits. This will decrease the maximum fuel temperature further and the water can safely be introduced with the air which can finally be turned off entirely.

The continued cooling of the reactor will now take place in the same manner as explained for the saturated steam reactors by means of the shutdown cooler for water and the shutdown condenser for vapor and gas. The reactor can then be opened and all fuel columns handled as described under A.5 in Chapter III.

V. THERMAL-HYDRODYNAMIC ANALYSIS

A basic goal of hydrodynamic and thermal design for boiling water reactors is to procure the maximum power level while maintaining a permissible amount of steam volume in the core. A high steam volume within the core generally tends to restrict the reactor power level because of its adverse effect on hydrodynamic stability (especially for a natural circulation system), burnout, and nuclear kinetics.

The optimization of the ratio of power to unit steam void is strongly dependent upon the recirculation flow rate of the coolant. Circulation of the coolant can be accomplished by forced flow (pumps) or by natural convection. In stationary power plant reactors, it is generally desirable to obtain the core coolant flow by natural circulation because of the large coolant flow capacity required (see Tables V through VIII). Also, the use of forced circulation necessitates a substantial initial capital investment for pumps and associated piping and incurs the additional expense and inconvenience of pump maintenance. In most instances a reactor can be designed so that the desired coolant flow can be obtained by natural circulation, particularly if risers are used. In case of heavy water boiling reactors the use of risers is greatly limited because of the additional cost of the higher D_2O inventory. As a result, the recirculation flow rate that can be achieved by natural circulation is limited and indirectly restricts the reactor power through its relation to the steam volume fraction. As the coolant flow rate increases, the steam volume in the core decreases.

In a two-phase fluid, such as a steam-water mixture, the steam phase possesses a higher velocity than the liquid phase due to the buoyancy forces resulting from their density difference. The ratio of the steam velocity to liquid velocity is called the slip ratio, C_s/C_l, and is related to mixture quality through the continuity equation as follows:

$$\frac{C_s}{C_l} = \frac{X}{1 - X} \frac{1 - \alpha}{\alpha} \frac{V_g}{V_l} \quad ,$$

where X is the mixture quality [lb steam/lb (steam & water)], α is the steam volume fraction, and V_g and V_l are the specific volumes of steam and water, respectively.

Table V

REACTOR TYPE 500 FC

Thermal Power, mw	985
Electrical Power, mw	280
Maximum Heat Flux, BTU/hr-ft^2	400,000
Average Heat Flux, BTU/hr-ft^2	137,500
Subcooling (t_{sat}-t) at core inlet, °F	50
Coolant Flow Rate (by circulating pump), lb/hr	21,600,000
Steam Produced, lb/hr	2,980,000
Average Exit Steam Quality, lb steam/lb mixture	0.138
Ratio of Boiling Channel Length to Total Length	0.615

Table VI

REACTOR TYPE 500 NC

Thermal Power, mw	1100
Electrical Power, mw	310
Maximum Heat Flux, BTU/hr-ft^2	373,000
Average Heat Flux, BTU/hr-ft^2	130,000
Subcooling (t_{sat}-t) at core inlet, °F	22.5
Coolant Flow Rate (natural circulation), lb/hr	39,000,000
Steam Produced, lb/hr	3,300,000
Average Exit Steam Quality	0.0853

Table VII

REACTOR TYPE 800 A

Thermal Power, mw	1000
Electrical Power, mw	325
Boiling Zone	
Maximum Heat Flux, BTU/hr-ft^2	242,000
Average Heat Flux, BTU/hr-ft^2	96,000
Subcooling (t_{sat}-t) at core inlet, °F	21
Coolant Flow Rate (natural circulation), lb/hr	32,000,000
Steam Produced, lb/hr	2,500,000
Average Exit Steam Quality	0.078
Superheating Zone	
Average Heat Flux, BTU/hr-ft^2	99,100
Power Produced by Superheater, %	16.8
Average Exit Steam Temperature, °F	850
Average Steam Velocity, ft/sec	115
Estimated Maximum Temperature of Fuel Surface, °F	1200

Table VIII

REACTOR TYPE 800 B

Thermal Power, mw	1150
Electrical Power, mw	370
Boiling Zone	
Maximum Heat Flux, BTU/hr-ft^2	390,000
Average Heat Flux, BTU/hr-ft^2	141,000
Subcooling (t_{sat}-t) at core inlet, °F	24
Coolant Flow Rate (natural circulation), lb/hr	33,500,000
Steam Produced, lb/hr	2,900,000
Average Exit Steam Quality	0.0865
Superheating Zone	
Average Heat Flux, BTU/hr-ft^2	94,500
Power Produced by Superheater, %	15.5
Average Exit Steam Temperature, °F	810
Average Velocity in Maximum Power Shroud, ft/sec	130
Estimated Maximum Temperature of Fuel Surface, °F	1200

The steam volume in the core is determined by the slip ratio and the mixture quality. Therefore, the thermal and hydrodynamic computations are essentially only as accurate as the values of the slip ratio used in the analysis. Due to the complexity of two-phase flow, the slip ratio cannot be calculated, but must be obtained from experimental studies. However, no empirical correlations for calculating the slip ratio are known at present that adequately account for all factors that are known to affect it. Therefore, recourse was made to extrapolation of existing experimental data obtained from a natural circulation system operating between 150 psi and 600 psi at ANL. It is felt that the extrapolations are realistic and that the calculated results, as presented, are not substantially in error due to the slip ratios used.

Another uncertainty encountered in the thermal-hydrodynamic analysis of a boiling water reactor is the prediction of inception of reactor instability. Experimental studies have indicated that the inception of hydrodynamic instability is strongly related to the exit steam volume fraction. It is felt that in a boiling reactor, nuclear instability, and hence reactor instability, is intimately related to, and in most instances will be preceded by, hydrodynamic instability. It is also possible that the stability problem is less acute in a heavy water boiling reactor when compared to a light water reactor due to the much smaller void reactivity coefficient. In the present analysis the maximum reactor power was determined by arbitrarily setting an upper limit of 0.75 on the exit steam volume fraction in the maximum power channel. This value is based on preliminary data obtained from experimental studies of the inception of hydrodynamic instability in a natural circulation system.

The prediction of burnout is an equally major uncertainty that must be evaluated in order to arrive at the permissible reactor power level. The value of the maximum heat flux within the core was arbitrarily set at 400,000 BTU/hr-ft^2 and is based on experimental burnout data available at present. The limits of the experimental burnout data obtained at ANL at a pressure of 1000 psia show a low value of approximately 700,000 BTU/hr-ft^2 and a high value of 1,000,000 BTU/hr-ft^2 for similar flow conditions. However, data in the Russian literature show the burnout heat flux to be substantially higher and of the same magnitude as pool boiling. It is interesting to note that EBWR has operated at a peak flux of approximately 350,000 without adverse effects. Also, it is generally felt that the burnout heat flux in a reactor may be substantially higher than values obtained in the laboratory from electrically heated tubes.

Based on the afore-mentioned limitations a thermal-hydrodynamic analysis was made for each reactor type at maximum design conditions. The method of analysis used is basically the same as suggested in "Method of Analysis of Natural Circulation Boiling Systems" by P. A. Lottes and W. S. Flinn, Nuclear Science and Engineering, Vol. 1, No. 6, and in

ANL 5720, "Performance and Potential of Natural Circulation Boiling Reactors" by W. S. Flinn and M. Petrick. The results shown represent performance characteristics during initial reactor operation and will necessarily change with continued reactor operation due to reactivity changes, fuel burnup, etc.

The properties of light water and an axial linear power input were two approximations used for all computations. The results differ slightly from the results that would be obtained by the use of an axial cosine power input as well as the properties of heavy water. In all reactor types the feed water acting as moderator enters the reactor at 100°F and is preheated before overflowing to the downcomer. The temperature of the moderator was calculated by estimating the heat transfer from the individual assemblies to the moderator due to natural convection and by gamma attenuation and neutron heating. The results of the analysis of the four reactor types follow.

A. Reactor Type 500 FC (Fig. 1. Natural uranium, forced circulation, saturated steam)

The core design of this reactor was dictated primarily by physics requirements concerning lattice spacings and diameters of fuel assemblies and pins to achieve a natural uranium-fueled reactor. As a result, the total number of fuel columns that could be used in the given tank size was 228, while 330 fuel columns were used in the other three reactor types. This, in connection with the above limitation of maximum heat flux to 400,000 restricted the reactor power to 985 mw, which yields 280 mw of electricity.

Due to the high power density in the central region of the core, forced circulation of the coolant is necessary for this reactor. In the hydrodynamic analysis, an attempt was made to reduce pump capacity and pump head requirements by using two fuel assembly designs with different pin lattice spacing and by orificing the individual inlet apertures.

The fuel assemblies located in the central portion of the reactor (within a radius of 3.85 ft) have a pin lattice spacing of 0.48 in., while the remaining fuel assemblies have a spacing of 0.44 in. The core pressure drop was determined by conditions in the "hot" fuel assembly. Orificing of the individual fuel inlets was specified in order to compensate for the radial power distribution (essentially a J_0 distribution) and was designed to maintain a constant boiling length and constant steam quality in all fuel assemblies.

Although it is believed that a higher steam volume fraction can be tolerated in forced flow than in case of natural circulation, the exit steam volume fraction was set at $\alpha e = 0.75$ to maintain the same upper limit for all reactors.

The results of the hydrodynamic analysis are presented graphically in Fig. 23, which shows the radial variation of the water velocity, steam volume fraction, channel pressure drop and orifice pressure drop requirements. The sharp discontinuities shown in the curves are due to the change to the smaller pin lattice spacing (0.44 inch) at this point. Additional pertinent data are summarized in Table V.

B. Reactor Type 500 NC (Fig. 3. Natural circulation, saturated steam)

The permissible reactor power level was determined on the basis of a detailed performance analysis including the radial variations of the steam volume fraction, boiling length, mixture quality and power density. These variations occur in a natural circulation system on account of the constant inlet temperature across the core in connection with the variable radial power distribution. The results of the analysis are shown in Fig. 24 and in Table VI.

The thermal power generated in the core is 1100 mw, which yields 310 mw of electricity. Approximately ten per cent of the total power is transferred to the feedwater while it acts as moderator. The power density shown in Fig. 24 represents the net power that is transferred to the coolant flowing through the fuel assembly.

The inlet velocity is fairly constant across the core up to a radius of 5 ft. It then drops sharply toward the periphery. Such an occurrence is characteristic of a natural circulation system and is due to the compensating effect of the steam volume fraction on the "net driving head" and the two-phase frictional resistance.

In the event that tests should indicate that hydrodynamic instability is likely to occur due to excessive steam voids in the "hot" channels, a provision has been incorporated into the design to allow for diversion of a portion of the cold feedwater from the turbine condenser to a point in the reactor directly below the "hot" channels (see Fig. 3). This would have the effect of increasing the subcooling in this region and thus reducing the steam volume fraction in the fuel assemblies.

C. Reactor Type 800 A (Fig. 5. Natural circulation, central superheater)

The performance characteristics of the 800 A reactor are shown graphically in Figs. 25, 26 and 27 and are listed in Table VII.

The location of the superheating zone in the center of the reactor imposes somewhat of a limitation on the total reactor power due to an incompatibility that exists between the desired power distribution in the reactor and the nuclear considerations regarding fuel enrichment and burnup time. The heat flux in the superheating zone must be maintained at a lower level than that which could be tolerated in the boiling zone on account of the poorer heat transfer characteristics. However, the neutron flux is maximum in the superheating zone and the enrichment of the fuel is also greater, due to stainless steel cladding. To compensate for this, a filler of magnesium oxide is used in the superheating zone to reduce the total mass of U^{235} and hence the thermal flux. The use of a filler, in turn, reduces the fuel lifetime considerably. The final reactor power of 1000 mw represents a compromise value between thermodynamic and physics considerations. As mentioned previously, the performance characteristics shown represent conditions that exist during initial reactor operations. Power distribution and reactivity will change during continued operation.

A detailed heat transfer analysis was made for a portion of the superheating fuel assembly, which was considered to be a "hot channel" (see Fig. 26), in order to estimate the highest temperature that could exist in the fuel pins during continuous operation. The analysis was based on the following assumptions, which are felt to be very pessimistic: (1) each of the seven pins in the "hot channel" operates a maximum heat flux; (2) there is no transverse mixing of the steam across the fuel assembly. A finite difference-type solution was employed to account for the varying physical properties of the steam as its temperature increased. The results of this analysis for a total reactor power level of 1000 mw are given in Fig. 26, which shows the cladding and centerline fuel pin temperatures as a function of core height. The maximum temperature of the fuel cladding (1200° F) is the major factor in determining the power that can be removed in the superheating zone, since the maximum centerline temperature in the UO_2 (1600° F) is considered quite moderate. The fact that the steam temperature shown in Fig. 26 is much higher than the average outlet temperature of 850° F is due to the assumptions stated above. The actual temperatures that could be tolerated during continuous operation would have to be obtained from experimental studies, but the temperatures shown in Fig. 26 are not considered excessive.

The natural circulation performance characteristics of the boiling zone are given in Fig. 25, which shows the radial variation of the coolant velocity, exit and mean steam volume fraction, steam quality and net power density. The exit steam volume fraction in the maximum power channel (boiling zone) is 0.72, which is below the maximum value of 0.75 that occurs in the other reactors. However, the performance characteristics in the boiling zone of the 800-A reactor is essentially the same as the identical region in the 500-NC reactor.

A comparison of the radial variation of the thermal power generation of the two reactors is given in Fig. 27. The power distributions are practically identical in the boiling zones, but a sharp dip occurs in the superheating zone of the 800-A reactor. This dip is due to the use of magnesia filler in the superheating region for the reasons stated previously. Thus, the difference in thermal power of the two reactors is primarily due to the reduction of power in the superheating zone of the 800-A reactor.

In spite of the lower thermal power, the superheated steam from the 1000-mw 800-A reactor will produce more electric power than the saturated steam from the 1100-mw 500-NC reactor (see Tables VI and VII).

D. Reactor Type 800-B (Fig. 7. Natural circulation, peripheral superheater)

The core of the reactor Type 800-B is essentially the same as Type 500-NC with the exception that 60 of the outermost fuel columns, along the core periphery, comprise a superheating zone in the 800-B reactor. As expected, the performance characteristics in the boiling zone of the two reactors are practically the same. This is shown in Figs. 24 and 28. Figure 27 shows a comparison of the power distribution within the two reactors. The slightly higher power generation in the boiling zone of the 800-B reactor is due to the somewhat greater inlet subcooling.

Figure 28 shows the radial variation of the power densities, exit quality, exit steam volume fraction and inlet water velocity across the boiling zone. Additional information pertaining to the boiling zone is given in Table VIII.

As a result of the peripheral location of the superheating zone and the slope of the flux curve in that area, the maximum heat fluxes compared with averages are higher in the Type 800-B reactor in contrast to the Type 800-A superheater in which the radial flux distribution is essentially flat. Thus, the peripheral superheater offers the advantage of longer fuel life, since the fuel is not diluted, but it has the disadvantage of a somewhat higher ratio of maximum to average heat flux. For this reason and due to the greater steam production in the 800-B, an exit steam temperature of 810°F was decided on as compared with 850°F for the 800-A reactor. In addition to this, orificing of the fuel exits is used to proportion the steam flow in accordance with the radial power distribution. Temperature conditions are thereby equalized for the different columns. A heat transfer analysis on the superheater "hot channel" similar to the one performed for 800-A reactor showed the temperatures to be very similar for both reactors. It should be re-emphasized that these results are considered pessimistic as a result of the assumptions made.

VI. PROPOSED RESEARCH WORK AND EXPERIMENTS

The following is a list of some items which would require thorough study prior to a final decision for a full-scale power plant:

1. Reactor tank. Evaluation of materials, fabrication and cost.

2. Fuel columns. Evaluation of fabrication and cost.

3. In-pile tests of fuel pins.

4. Boiling experiments with mock-up fuel to test heat transfer qualities and stability for boiling in long channels.

5. Superheating experiments with mock-up fuel assemblies to to test mechanical stability and cladding erosion at high steam velocities and temperatures.

6. Physics. Zero power experiments including reactor constants at different core patterns and different "steam voids." Worth of control devices.

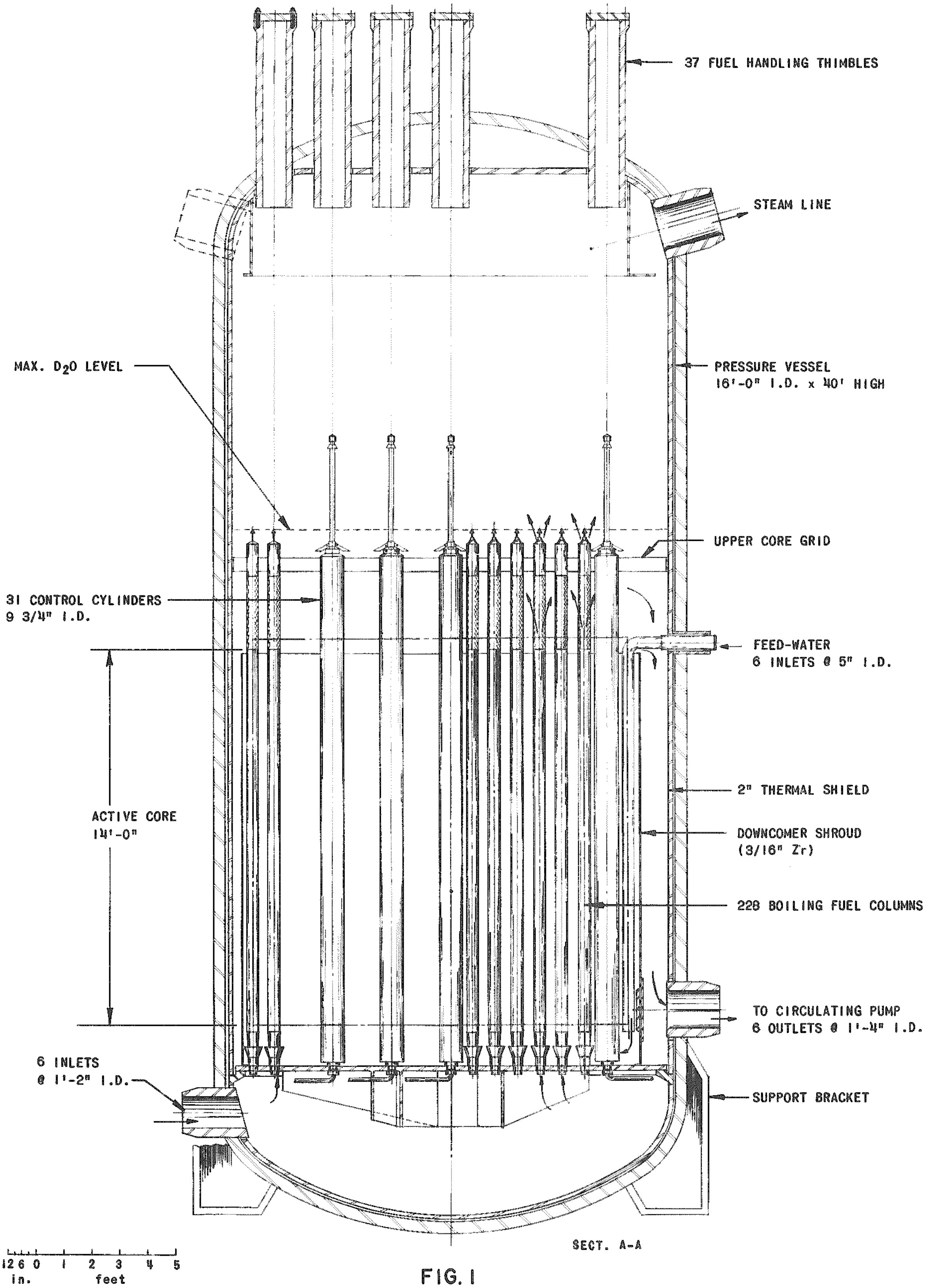


FIG. 1
REACTOR TYPE 500 FC, ELEV.

NATURAL U. FORCED CIRCULATION, SATURATED STEAM

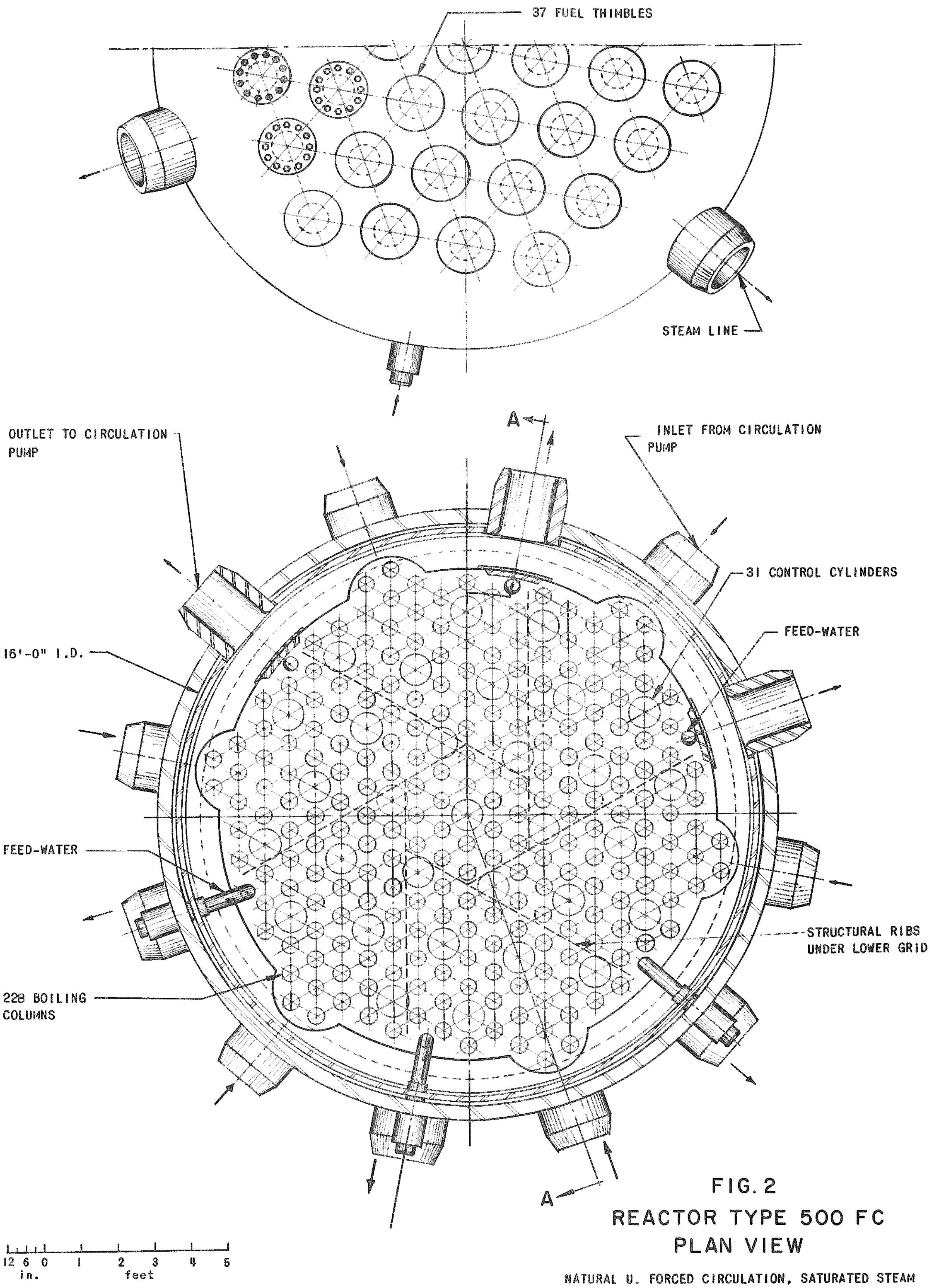


FIG. 2
REACTOR TYPE 500 FC
PLAN VIEW

NATURAL U. FORCED CIRCULATION, SATURATED STEAM

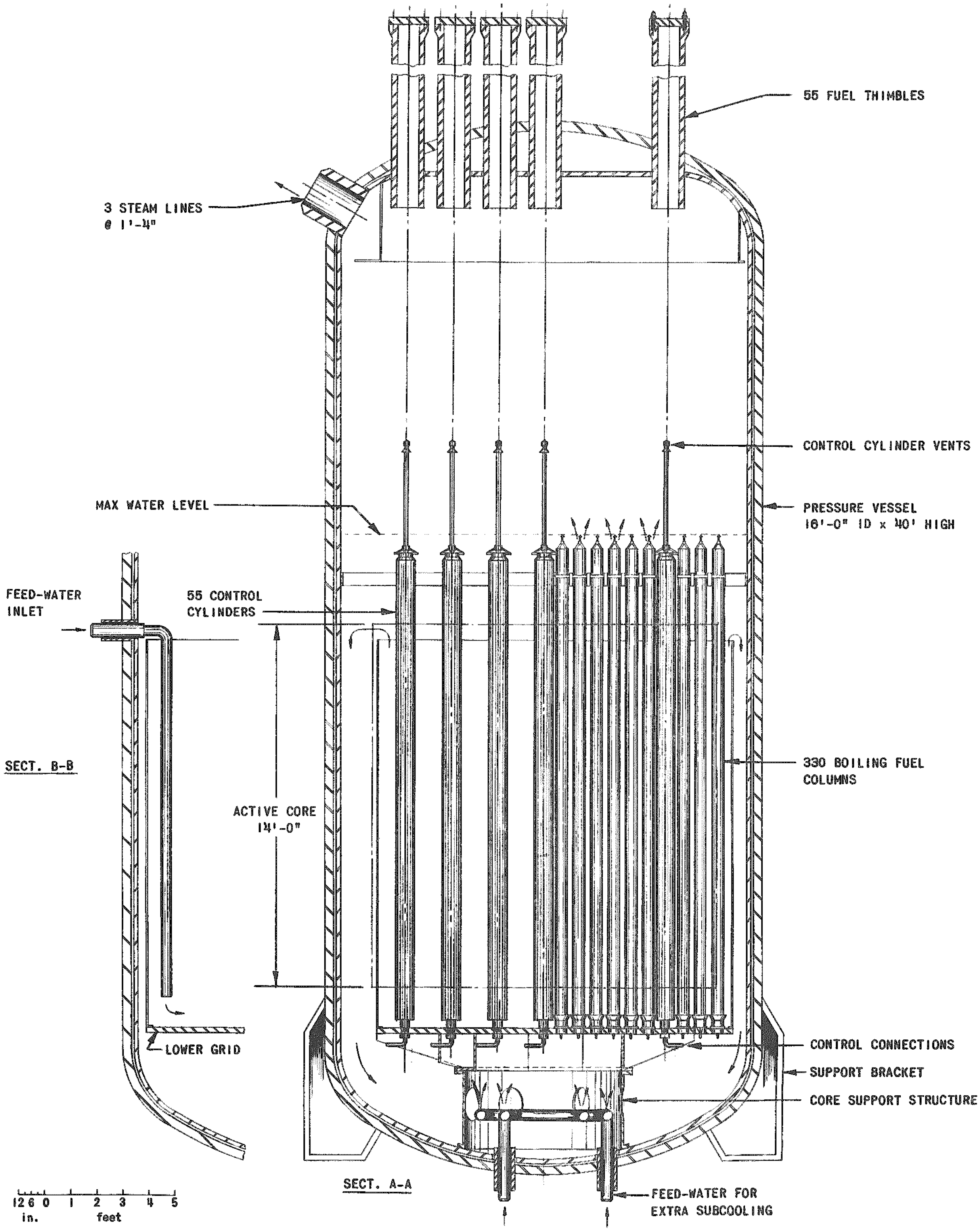
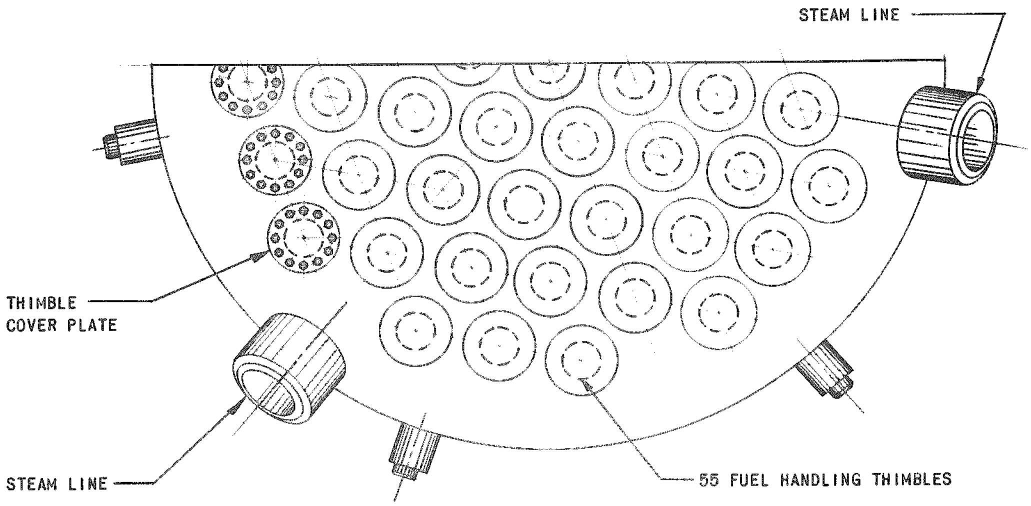


FIG. 3

REACTOR TYPE 500 NC, ELEV.

NATURAL CIRCULATION, SATURATED STEAM



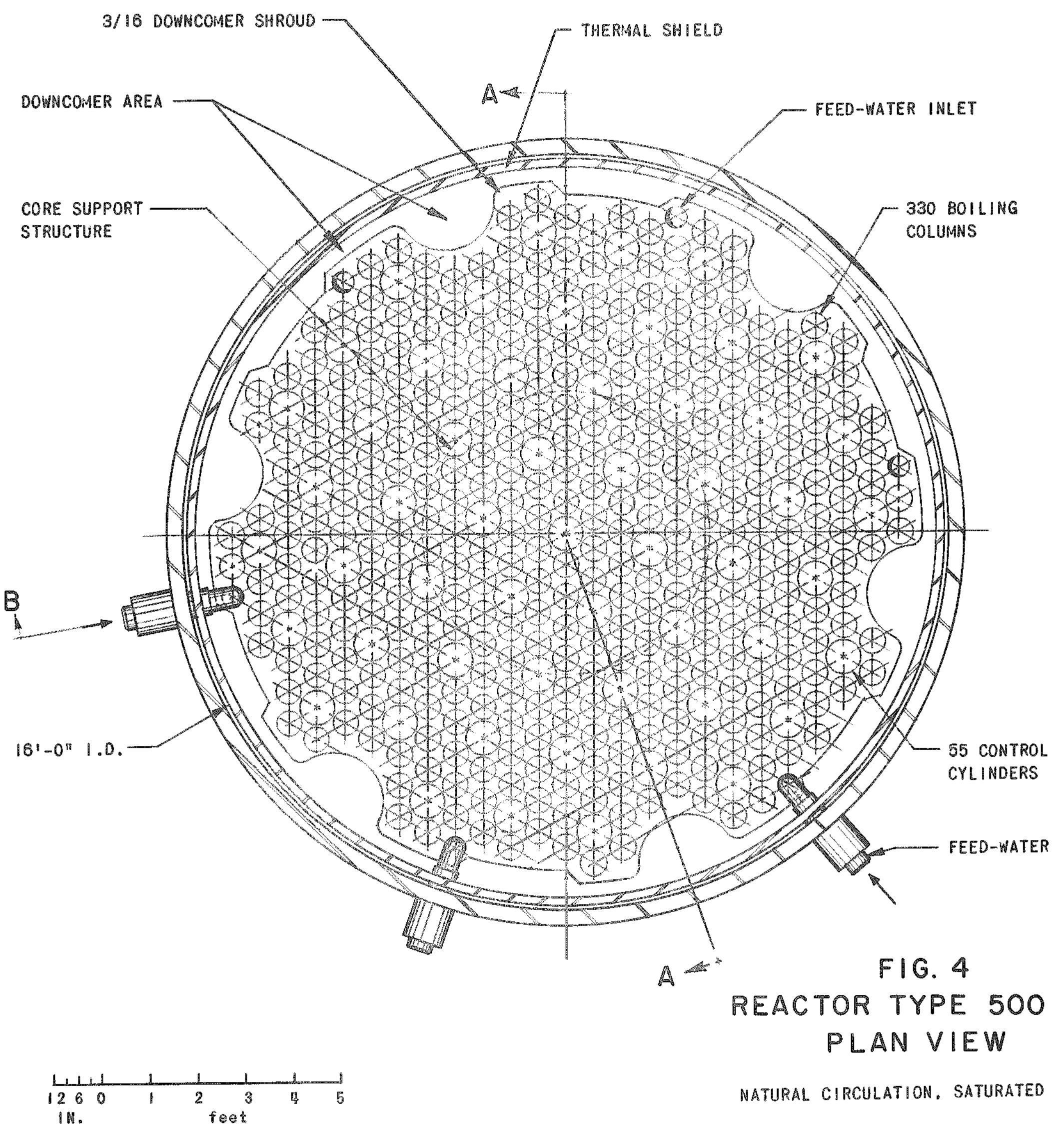


FIG. 4
REACTOR TYPE 500 NC
PLAN VIEW

NATURAL CIRCULATION, SATURATED STEAM

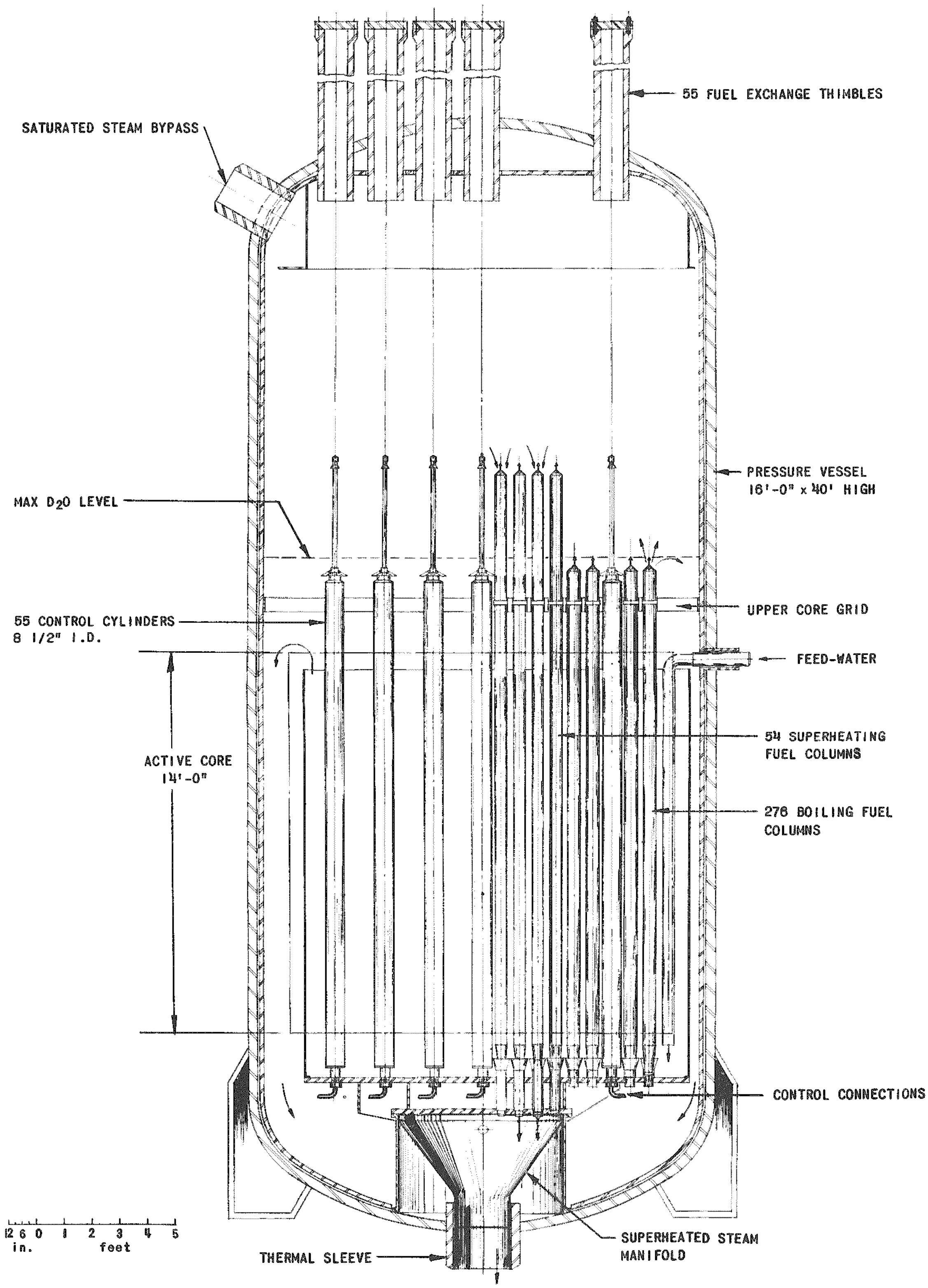


FIG. 5

REACTOR TYPE 800-A, ELEV.

NATURAL CIRCULATION, CENTRAL SUPERHEATER

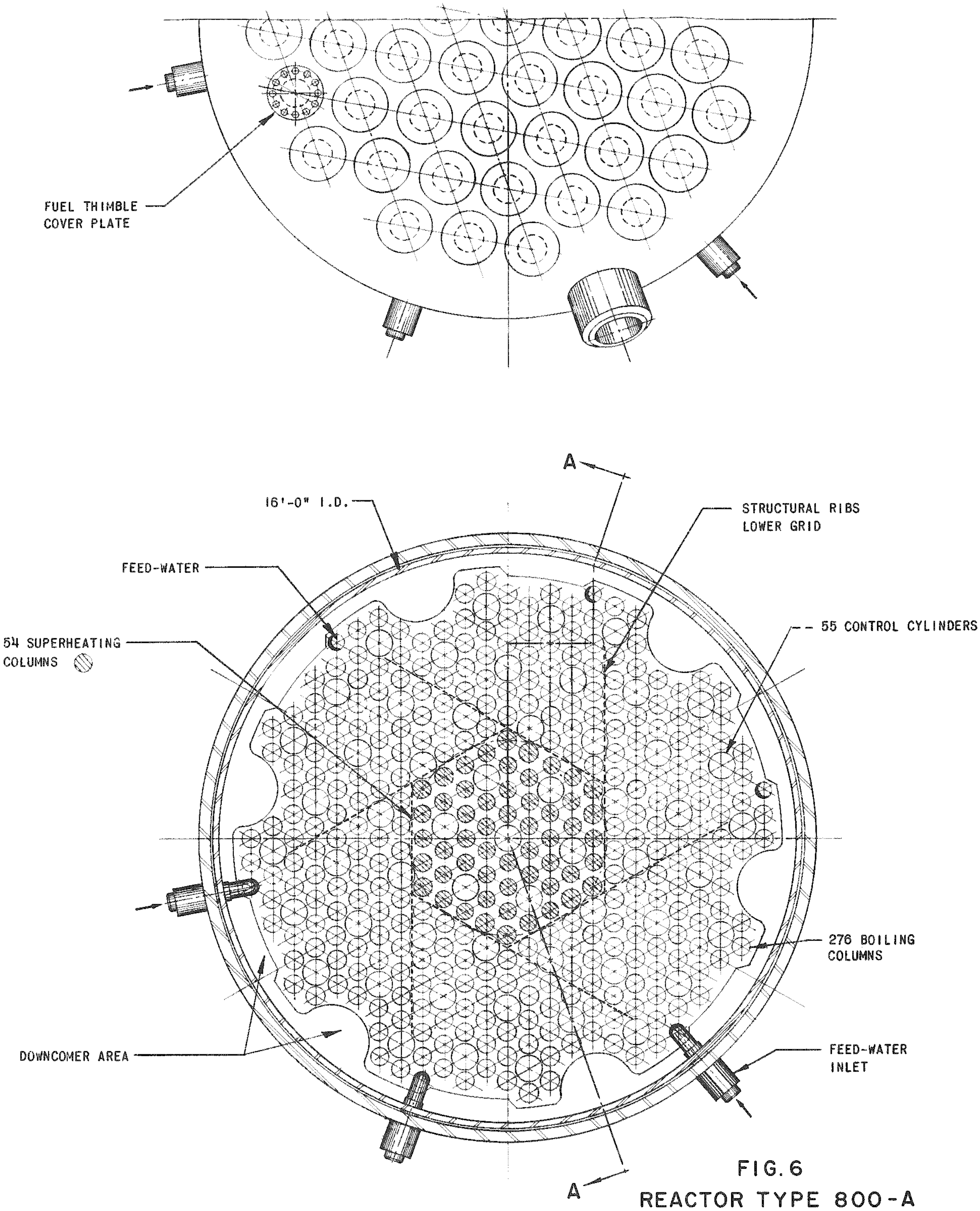


FIG. 6
REACTOR TYPE 800-A
PLAN VIEW

NATURAL CIRCULATION, CENTRAL SUPERHEATER

12 6 0 1 2 3 4 5
in. feet

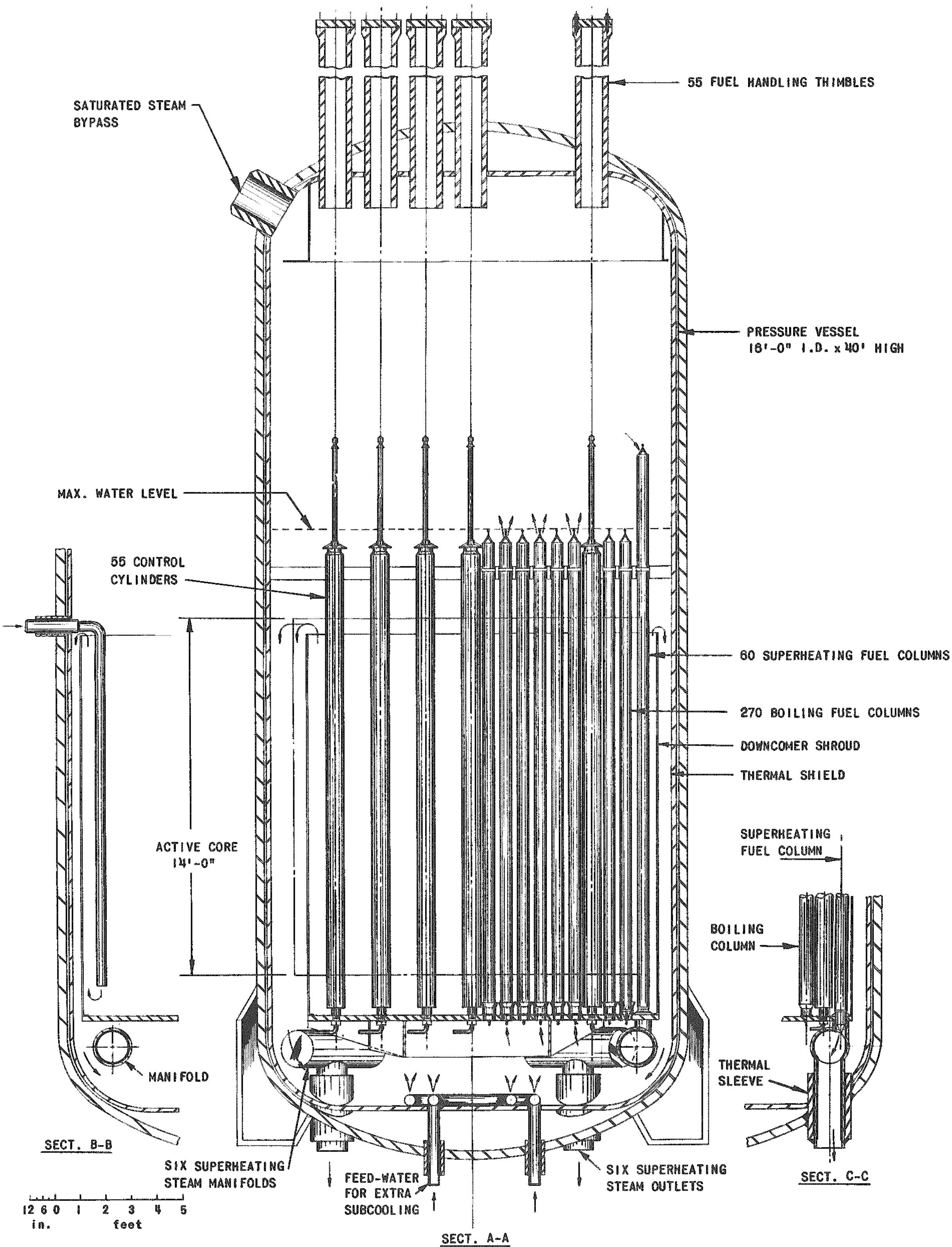
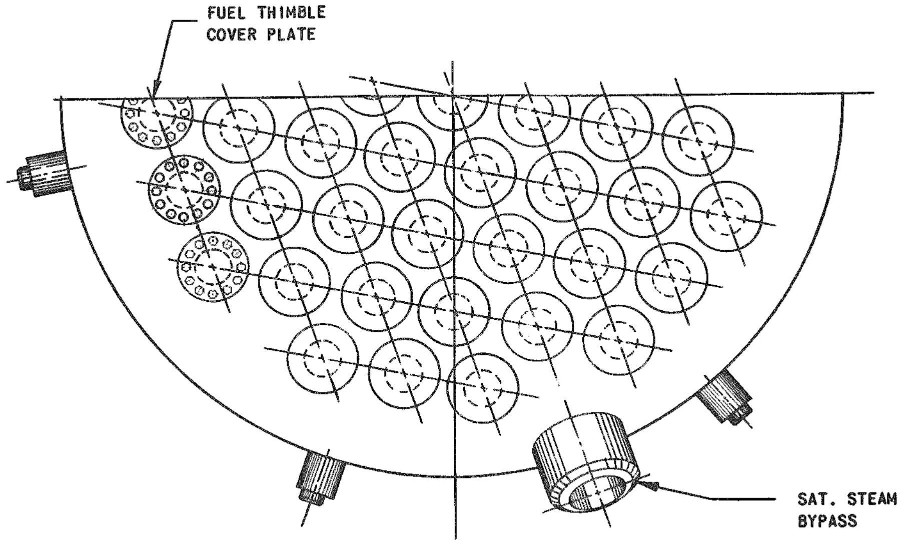


FIG. 7

REACTOR TYPE 800-B, ELEV.

NATURAL CIRCULATION, PERIPHERAL SUPERHEATER



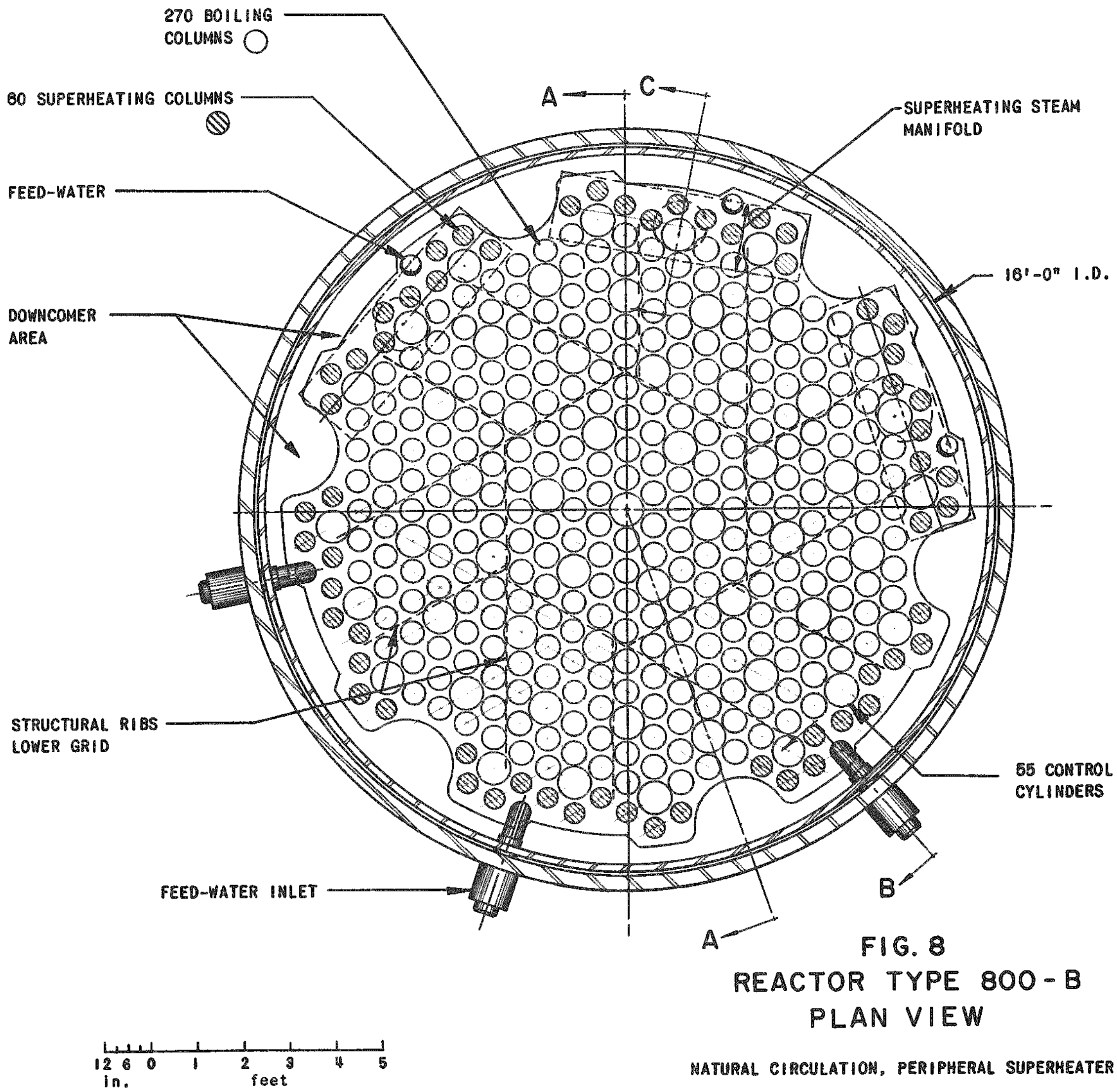


FIG. 8
REACTOR TYPE 800-B
PLAN VIEW

NATURAL CIRCULATION, PERIPHERAL SUPERHEATER

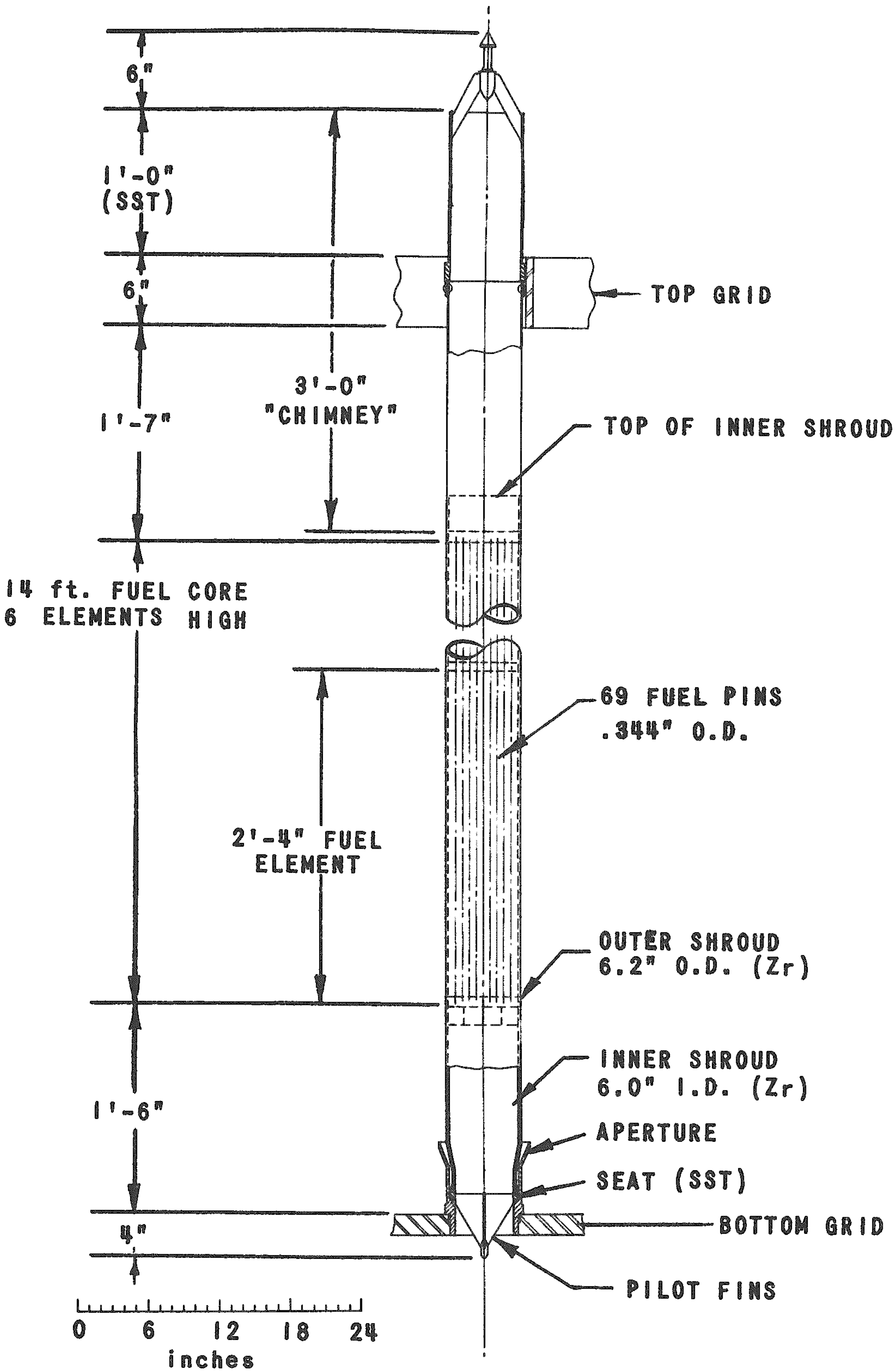


FIG. 9
BOILING FUEL COLUMN, REACTOR TYPE 800-B

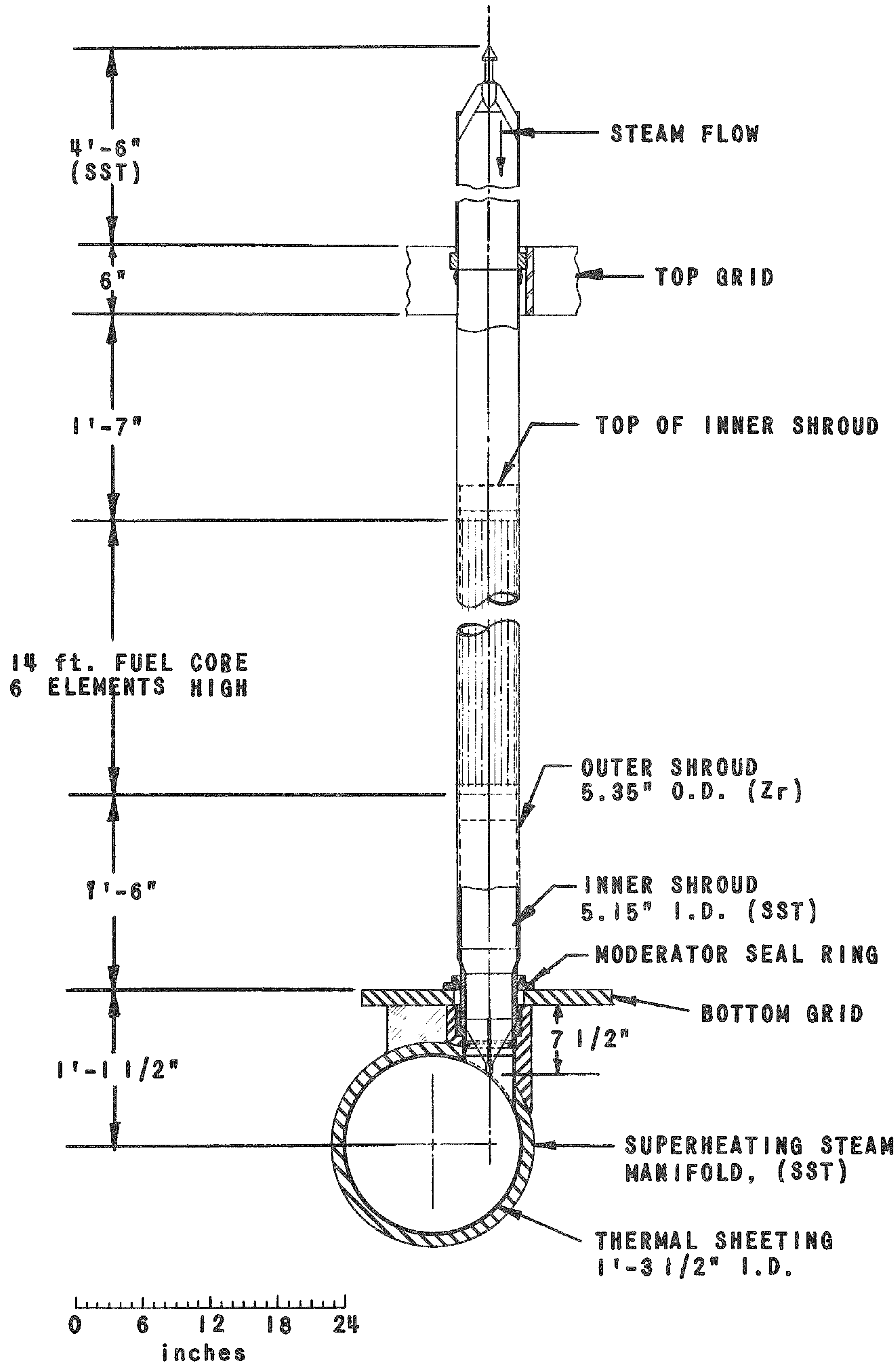


FIG. 10
SUPERHEATING FUEL COLUMN, REACTOR TYPE 800-B

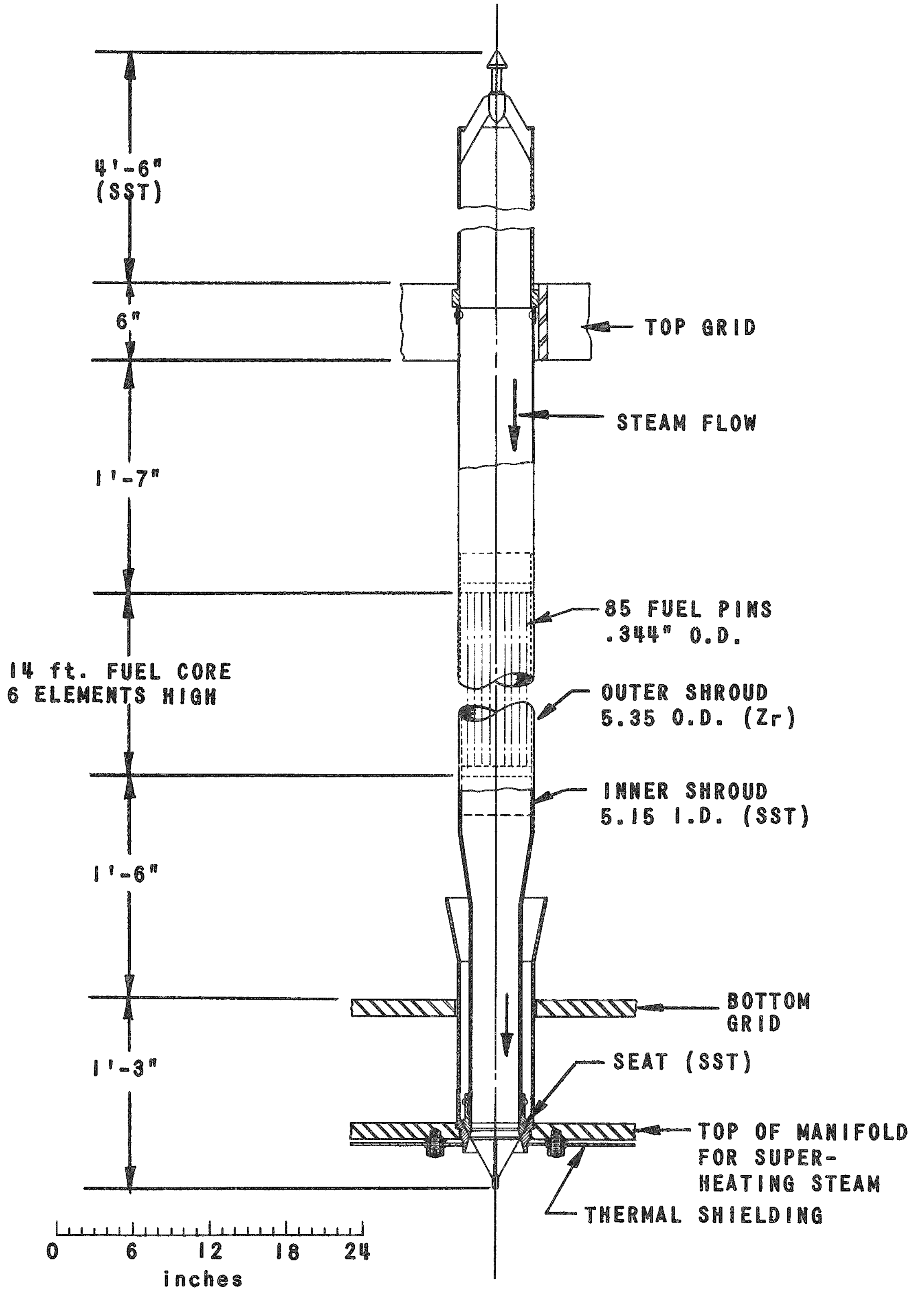


FIG. 11
SUPERHEATING FUEL COLUMN REACTOR TYPE 800-A

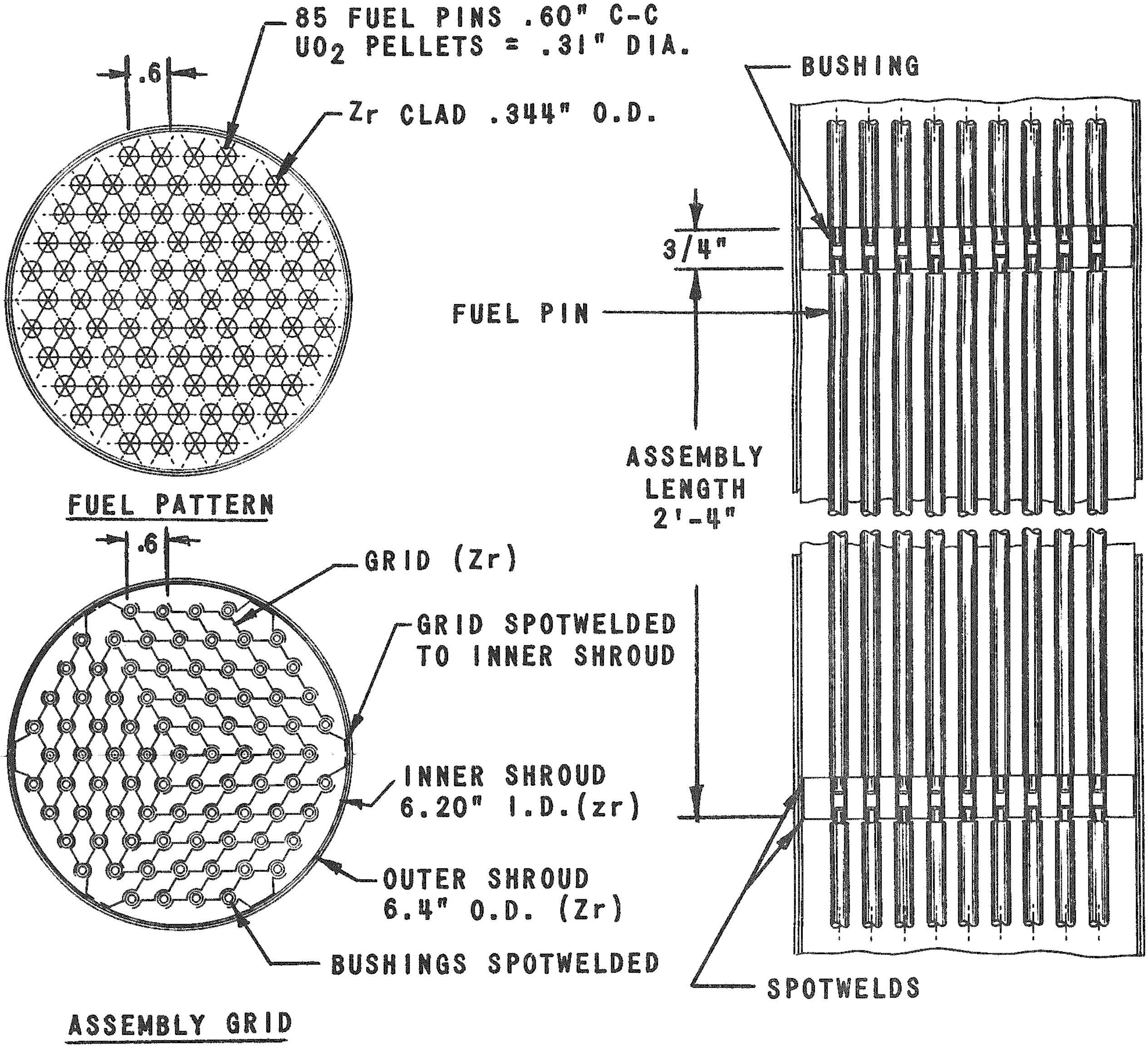


FIG. 12
BOILING FUEL ASSEMBLY - REACTOR TYPE 800-A

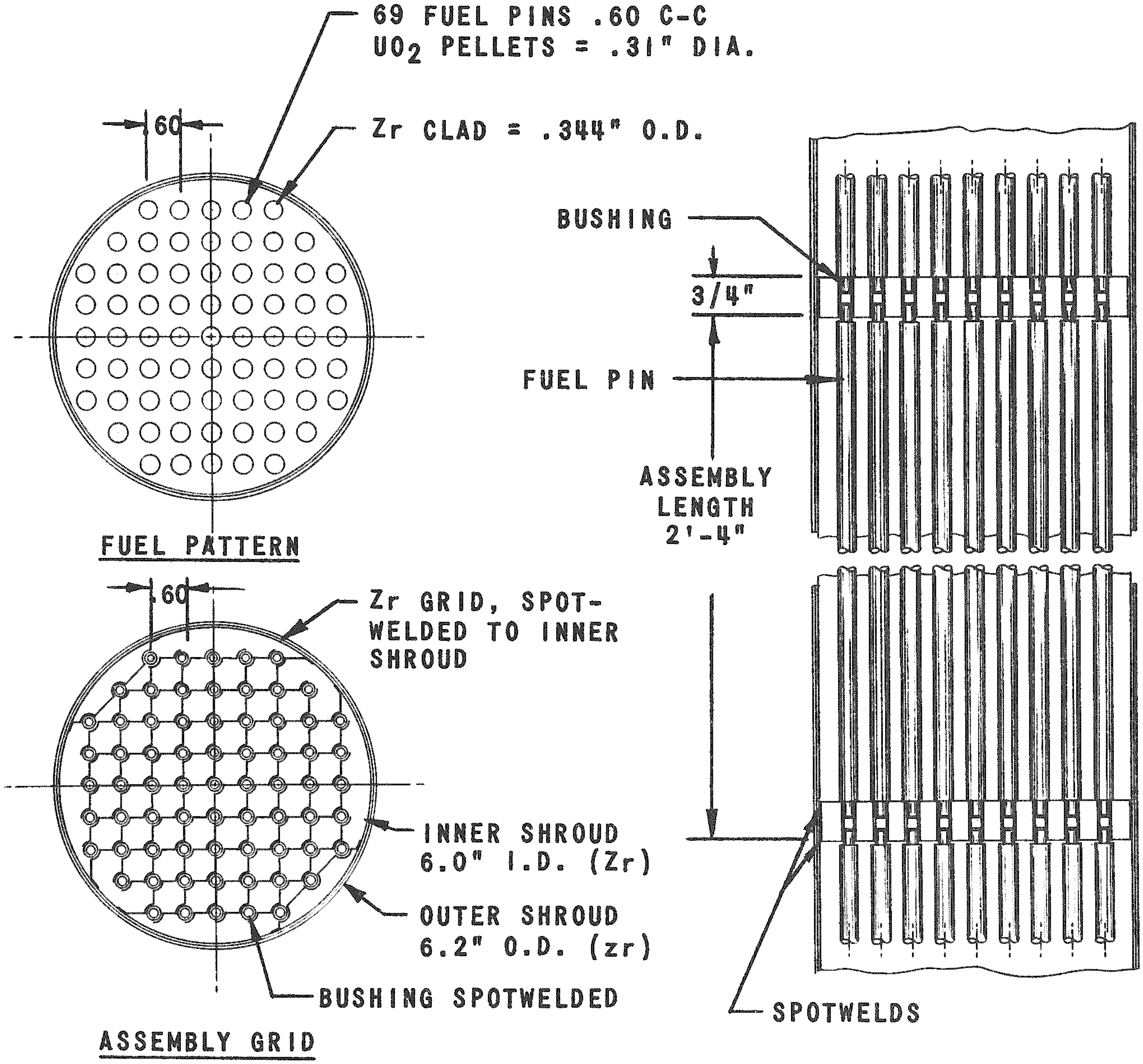


FIG. 13
BOILING FUEL ASSEMBLY - REACTOR TYPES 500NC & 800B

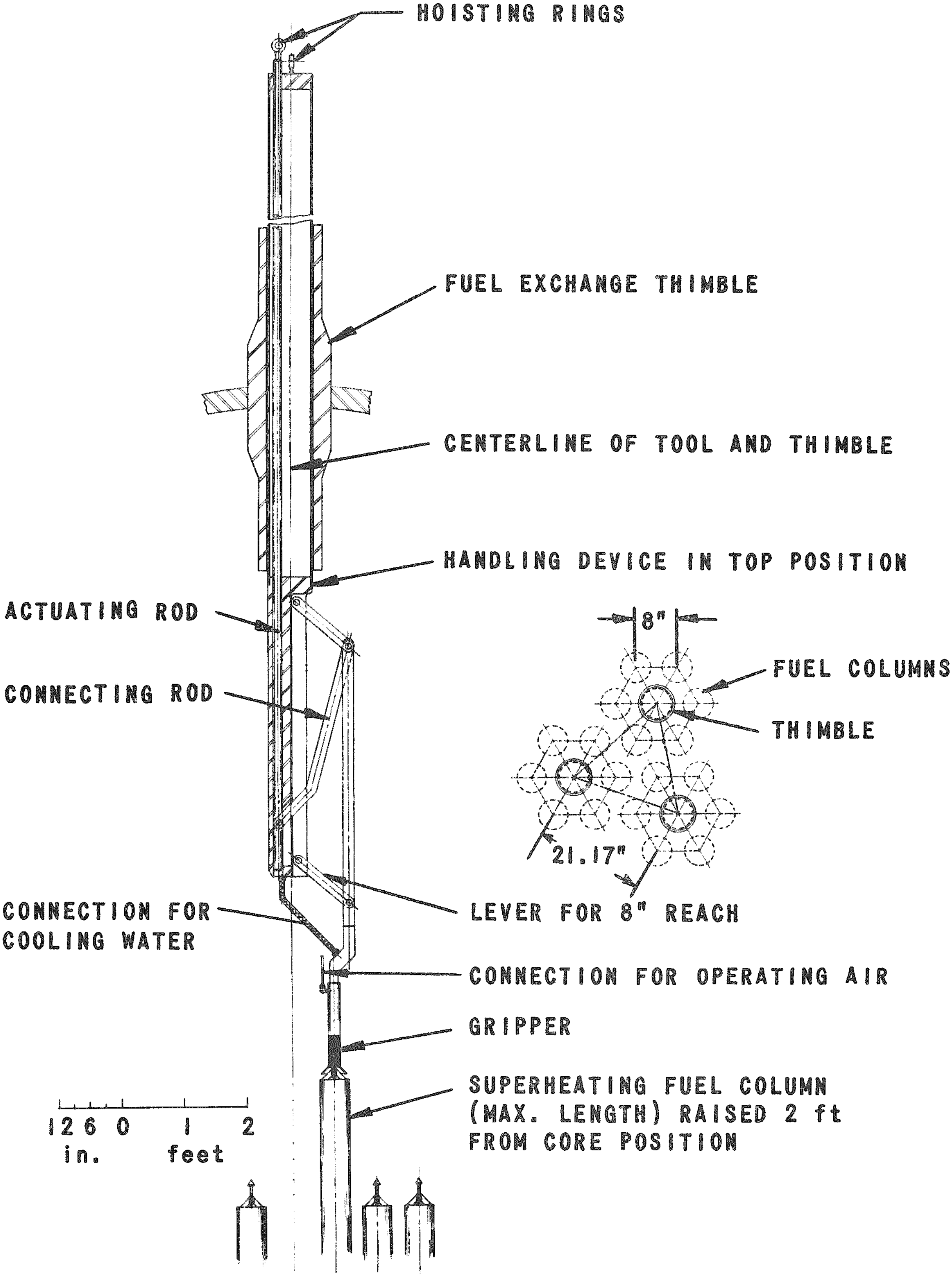


FIG. 14
FUEL HANDLING DEVICE FOR 8" REACH

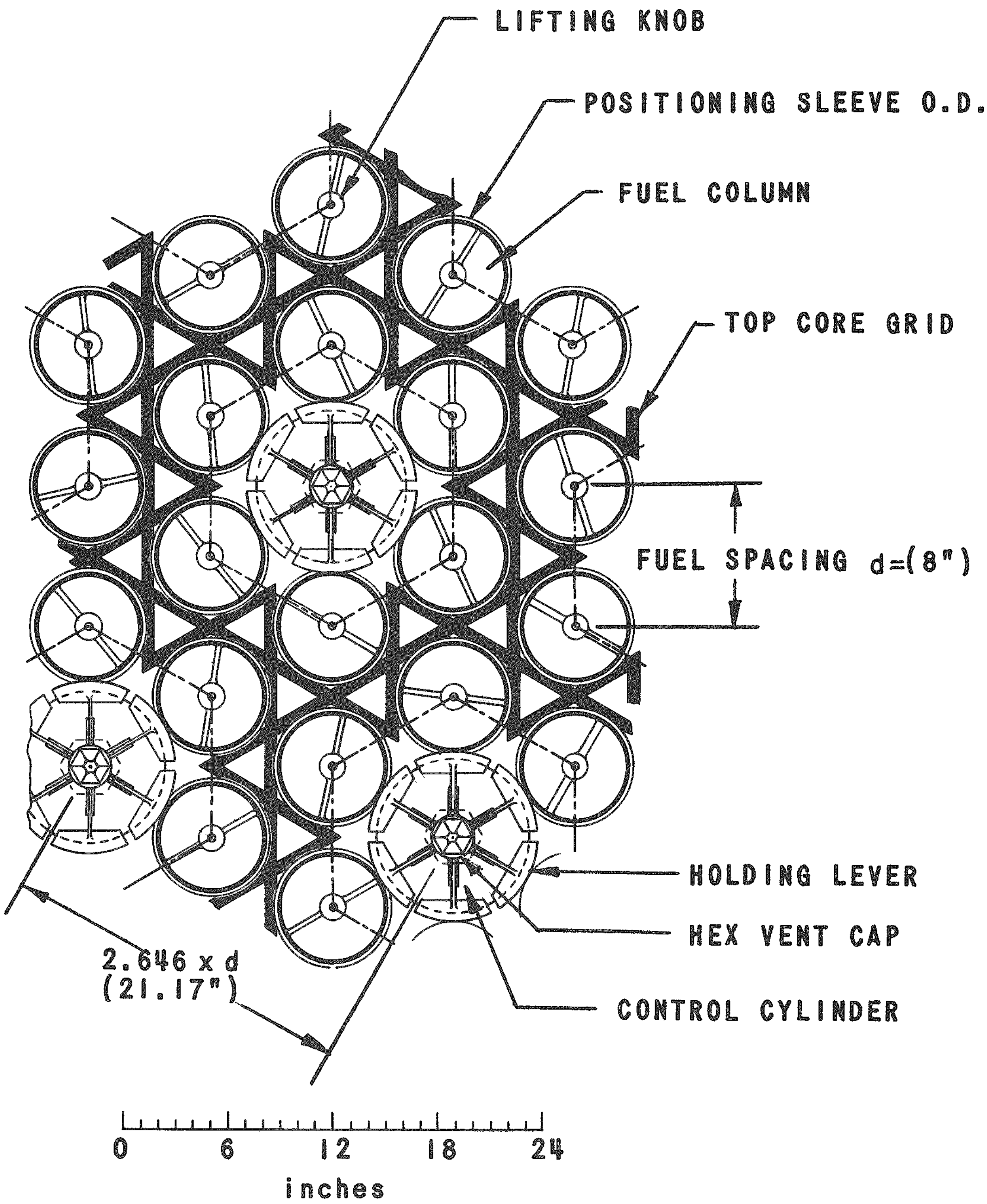


FIG. 15
TOP GRID AND FUEL HOLDING SPIDER

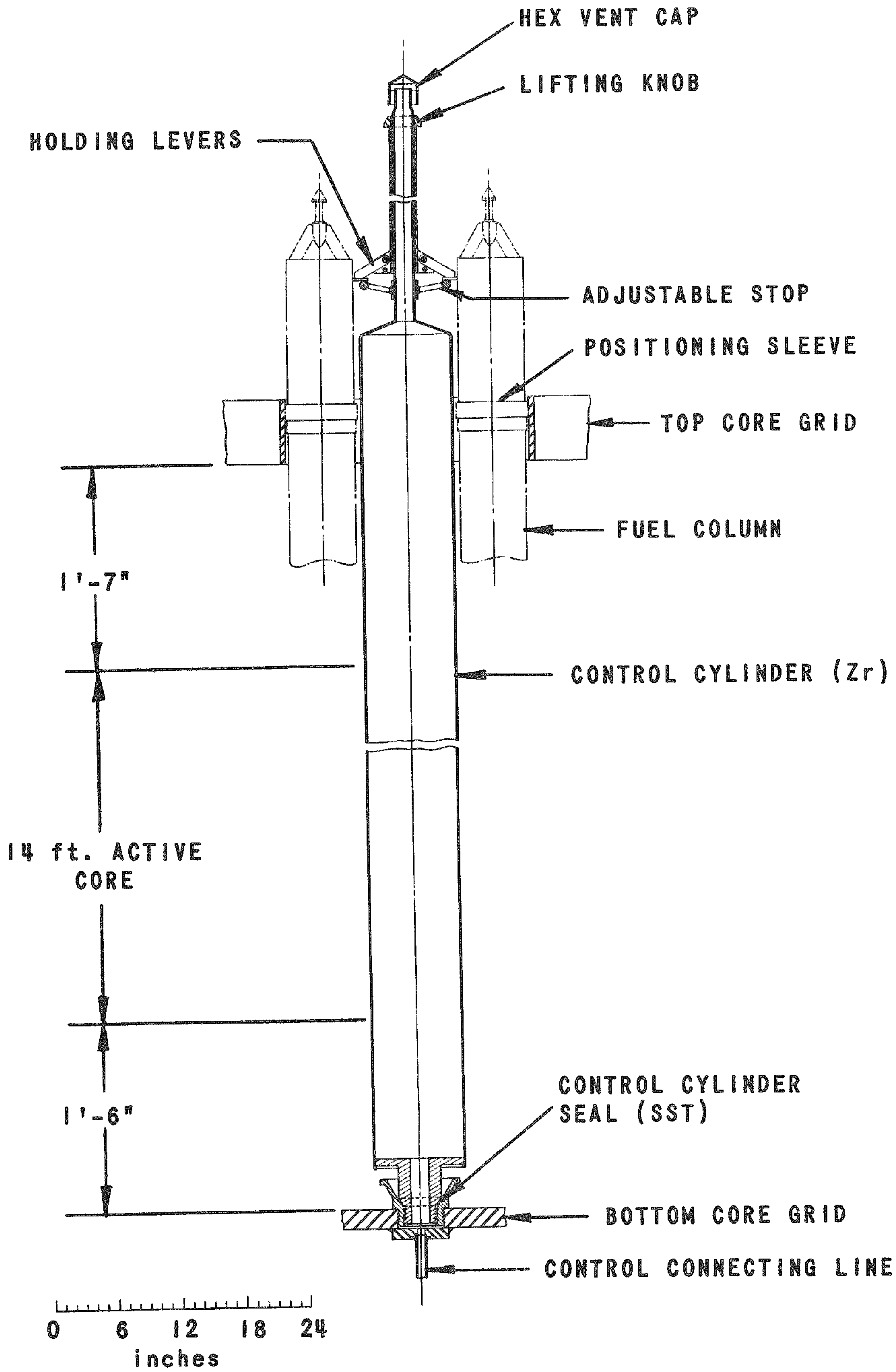


FIG. 16
CONTROL CYLINDER AND FUEL HOLDING SPIDER

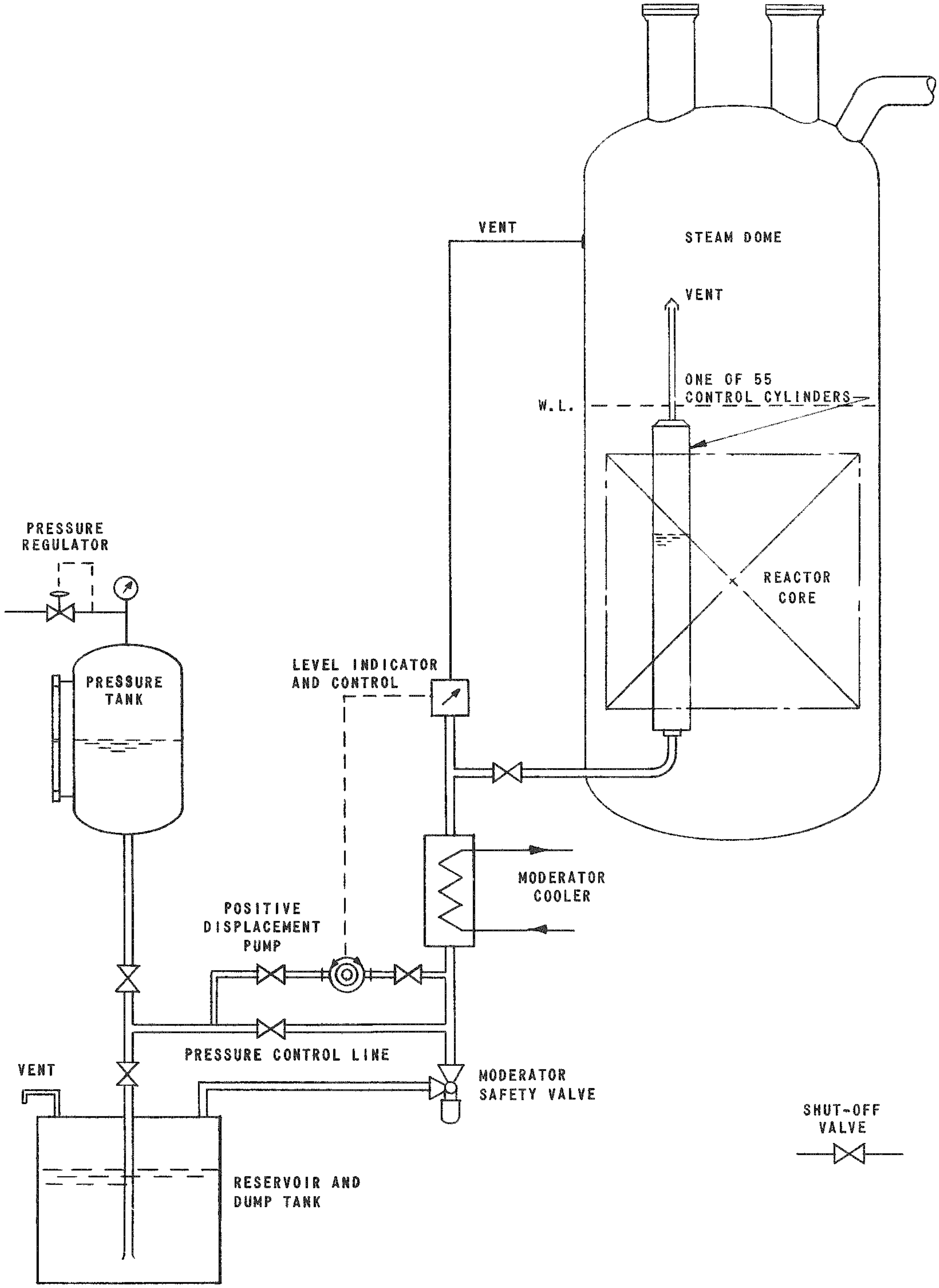


FIG. 17
MODERATOR CONTROL FLOW DIAGRAM

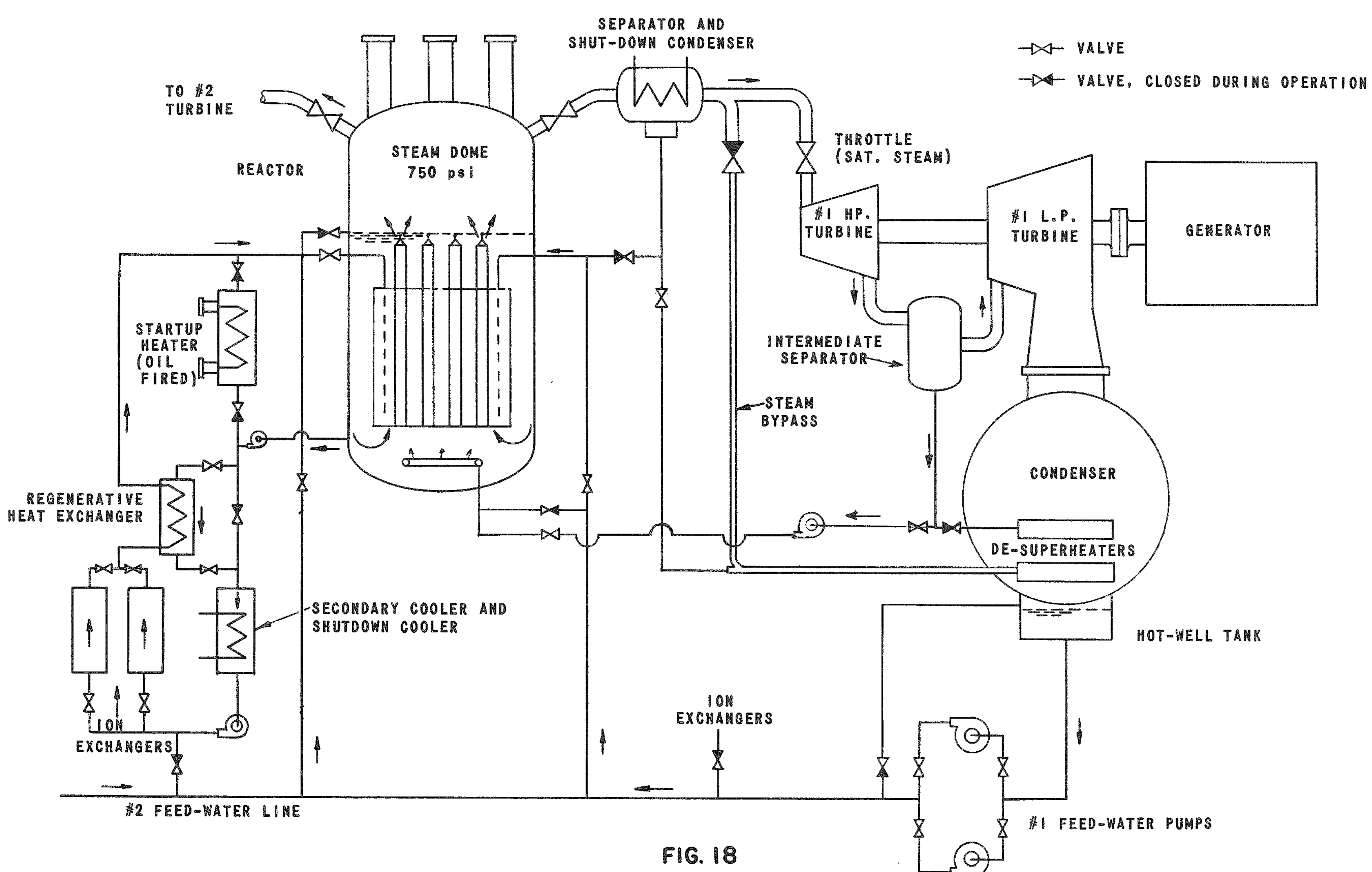


FIG. 18
SIMPLIFIED FLOW DIAGRAM
SATURATED STEAM CYCLE

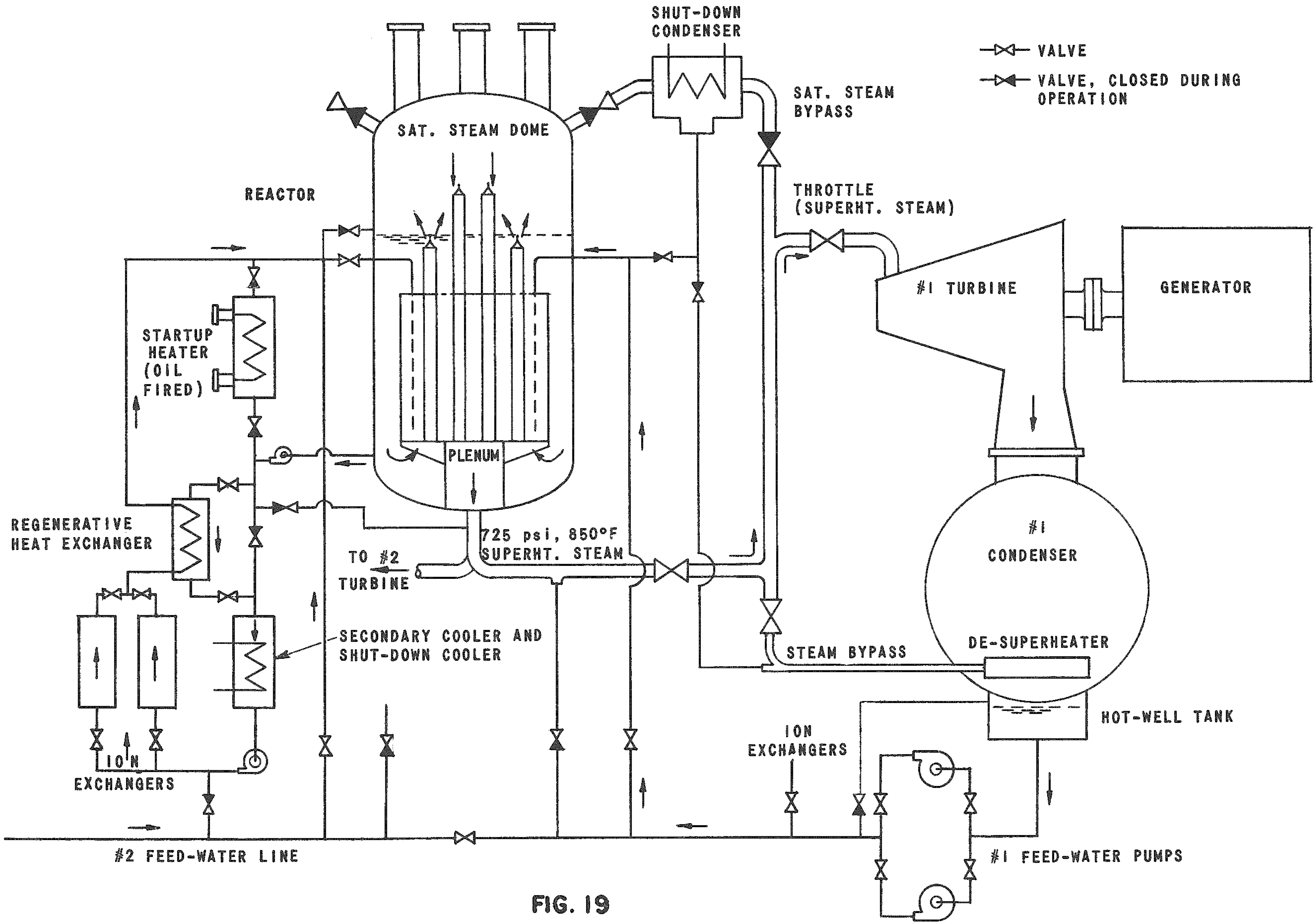


FIG. 19
SIMPLIFIED FLOW DIAGRAM
SUPERHEATED STEAM CYCLE

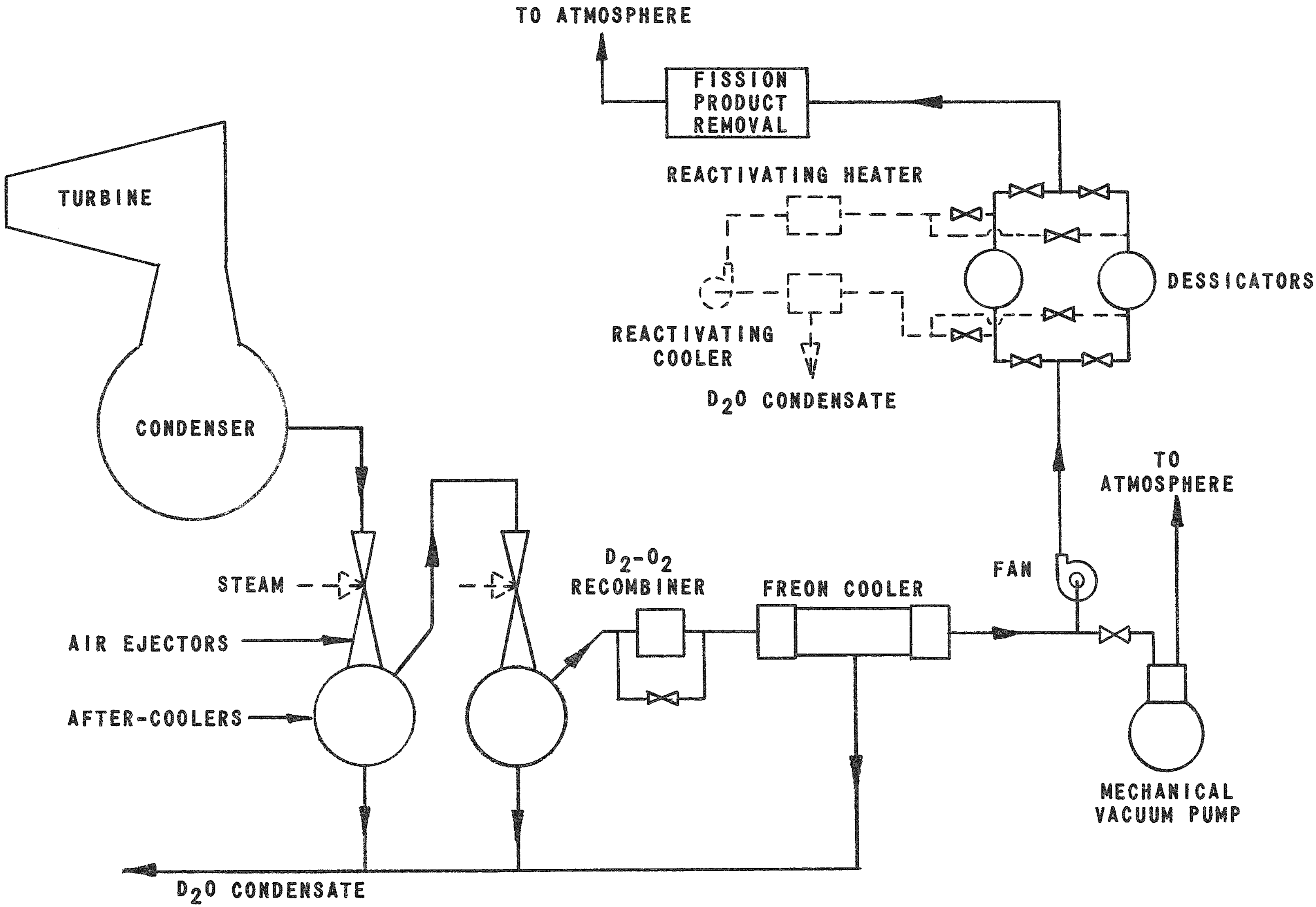


FIG. 20
VENT AIR DRYING SYSTEM AND FISSION PRODUCT REMOVAL

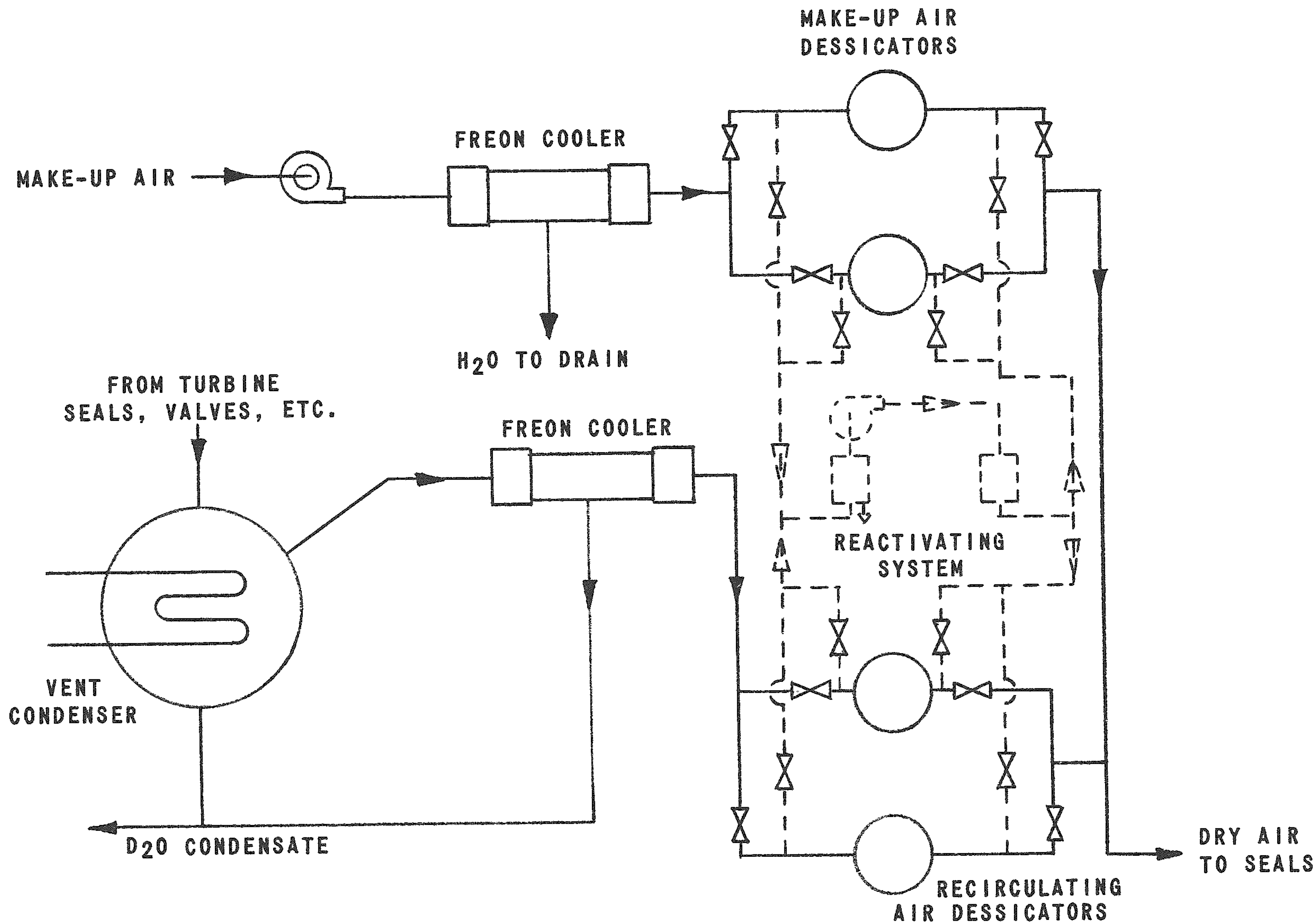


FIG. 21
GLAND SEAL AIR DRYING SYSTEM

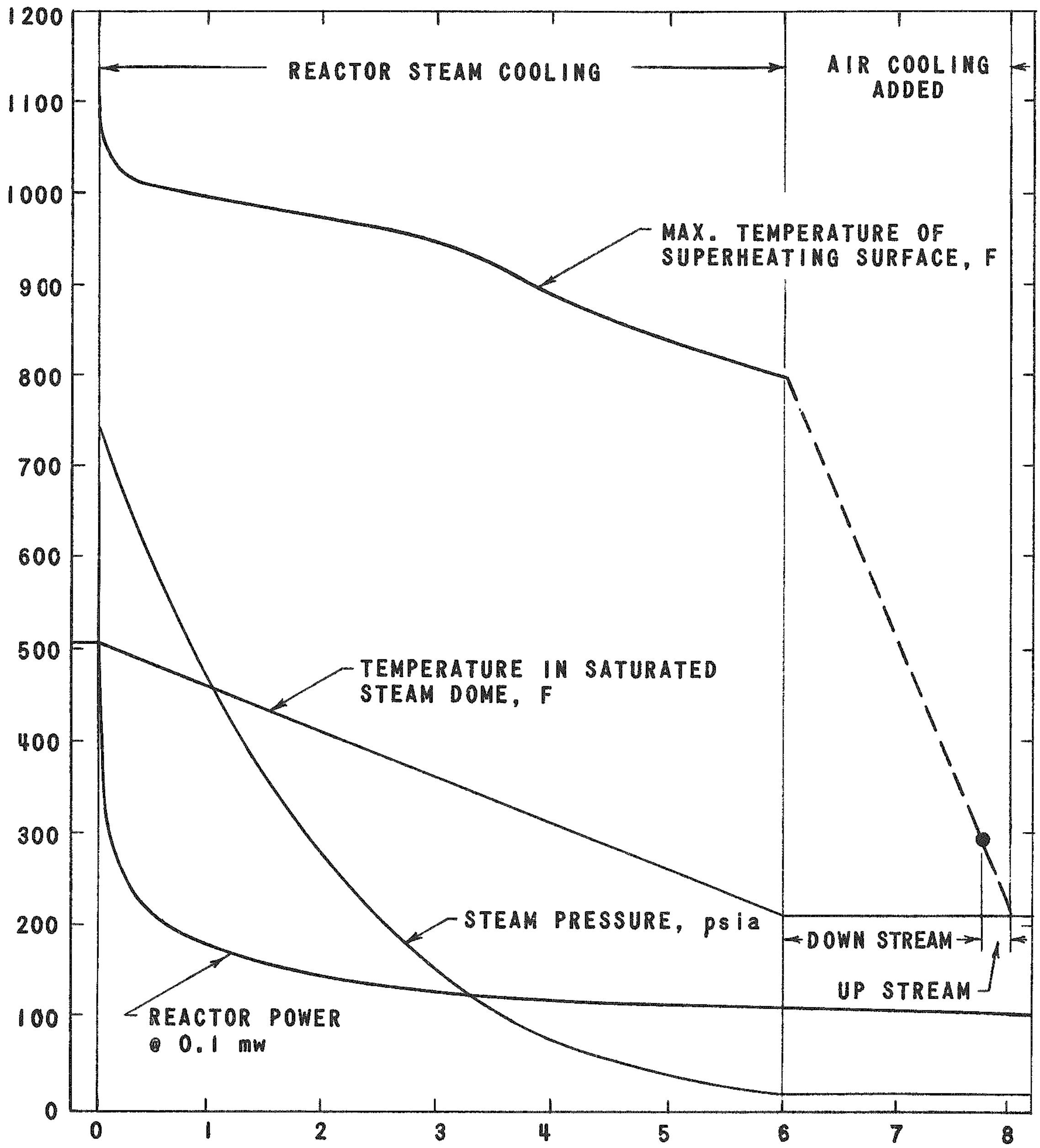


FIG. 22
TYPE 800-B REACTOR SHUTDOWN COOLING

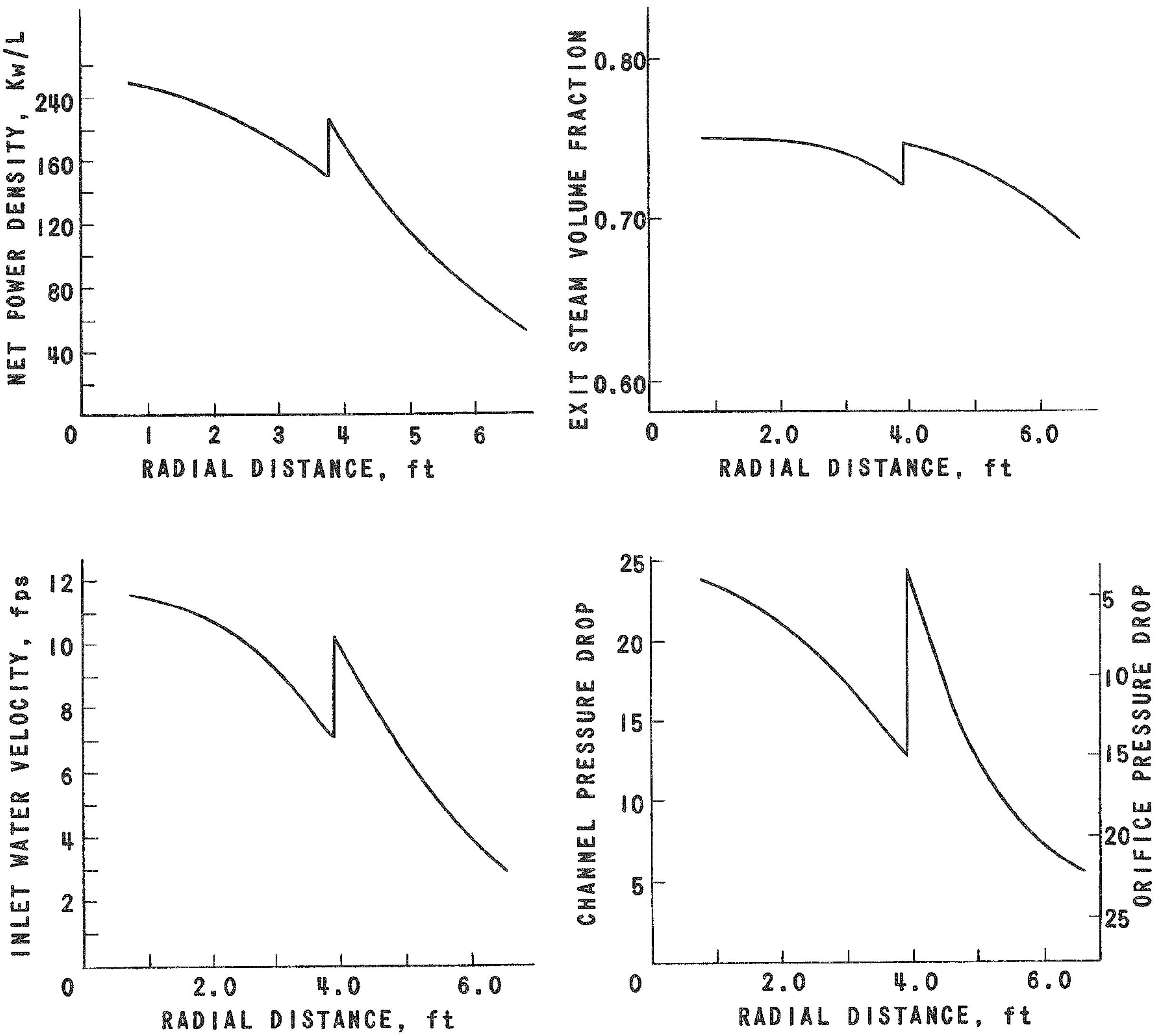


FIG. 23
PERFORMANCE CHARACTERISTICS OF REACTOR TYPE 500-FC

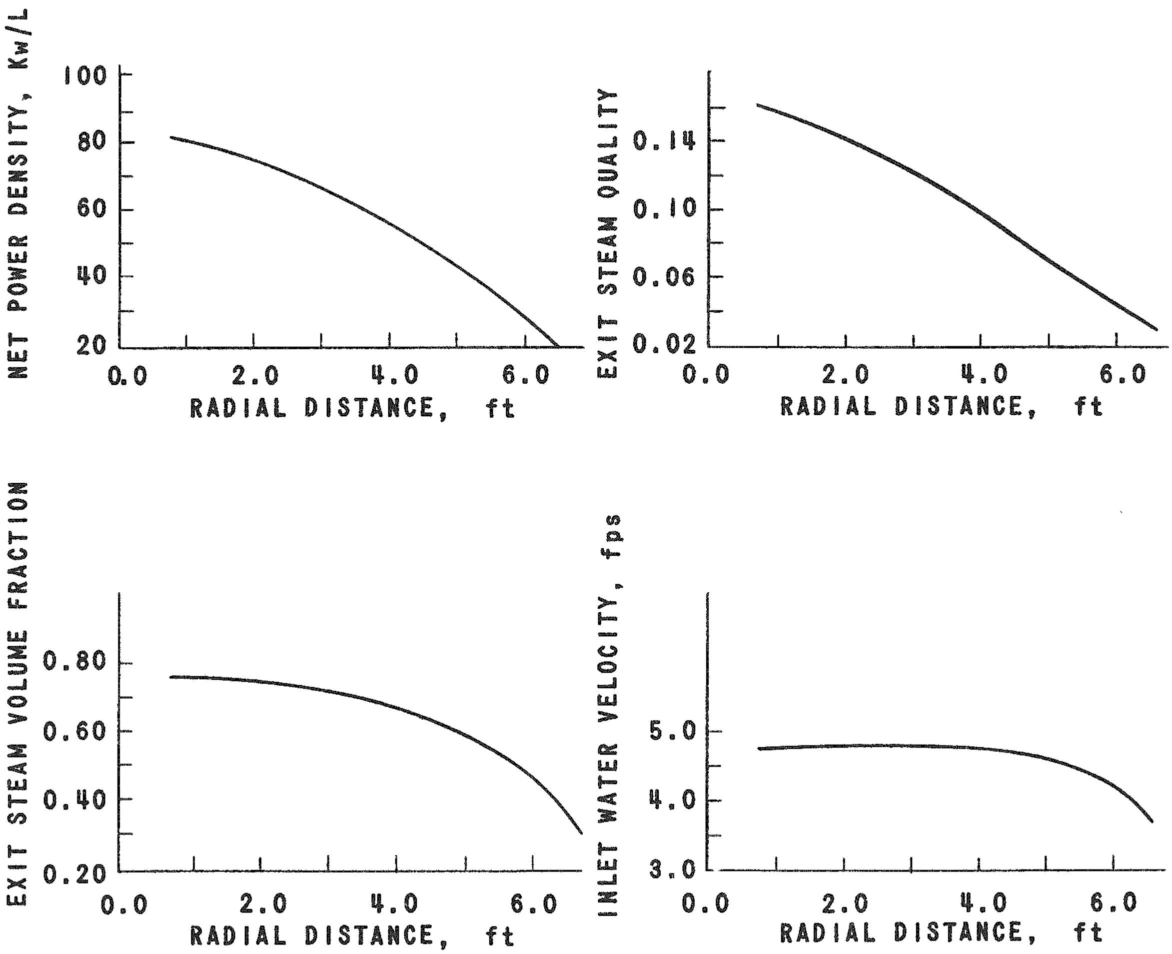


FIG. 24
PERFORMANCE CHARACTERISTICS OF REACTOR TYPE 500-NC

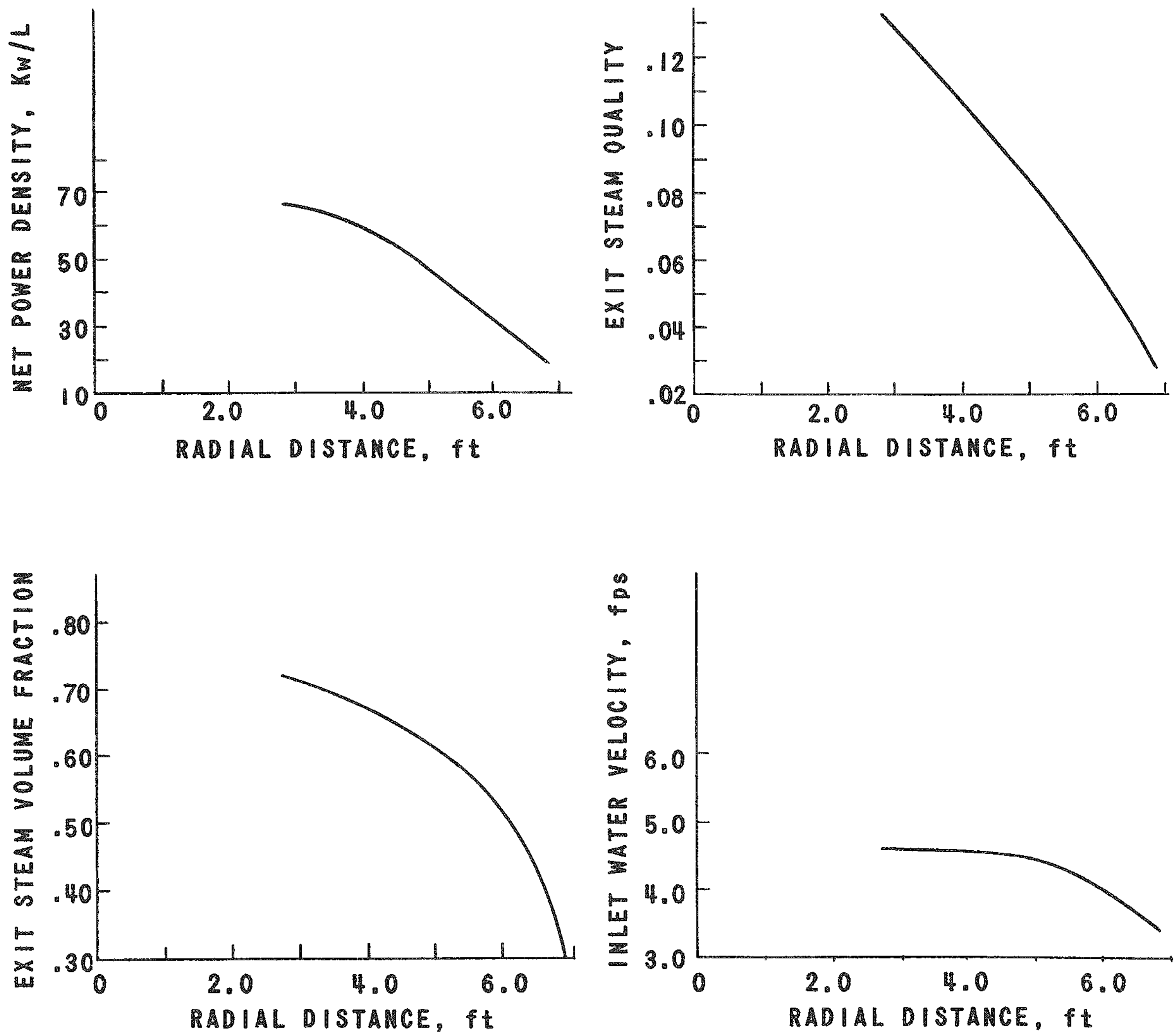
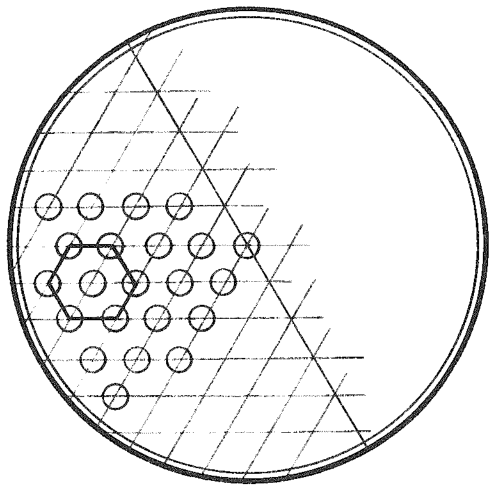


FIG. 25
PERFORMANCE CHARACTERISTICS OF THE BOILING ZONE IN REACTOR TYPE 800-A



PARTIAL DIAGRAM OF SUPERHEATER FUEL ASSEMBLY
HEXAGON DENOTES HOT SPOT CHANNEL

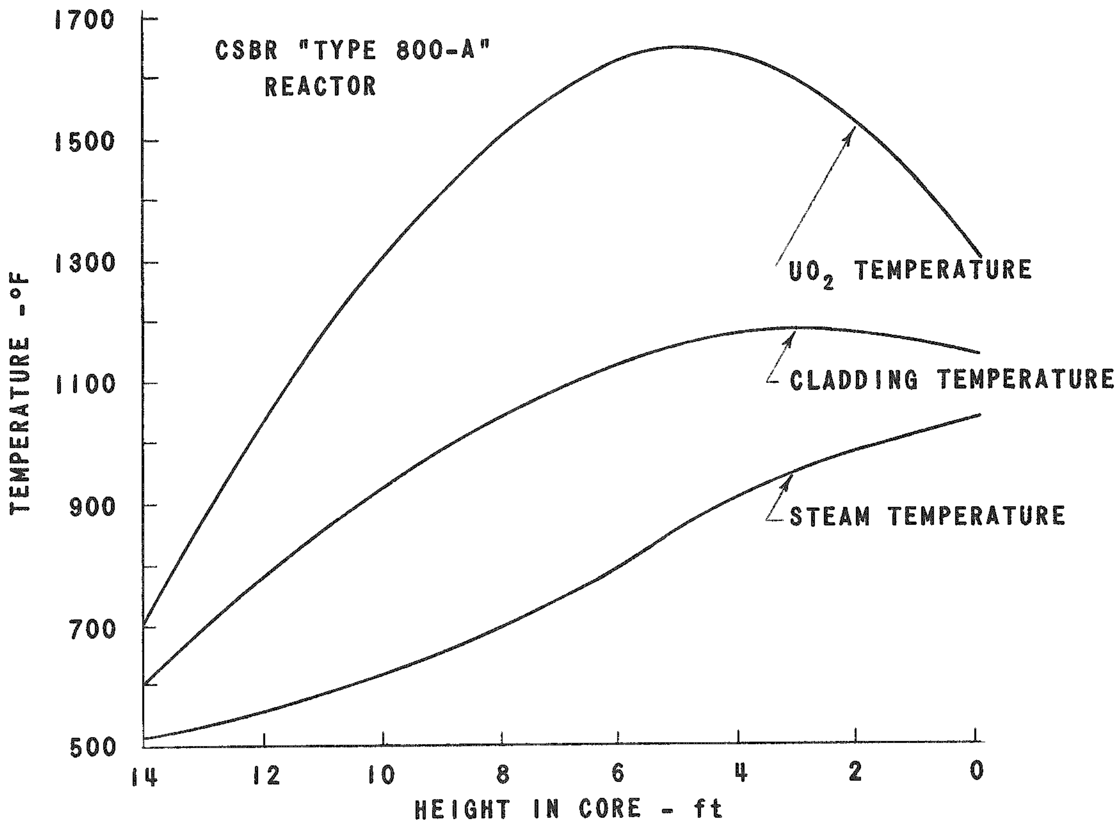


FIG.26
ESTIMATED MAXIMUM TEMPERATURES IN SUPERHEATING FUEL ASSEMBLY
REACTOR TYPE 800-A

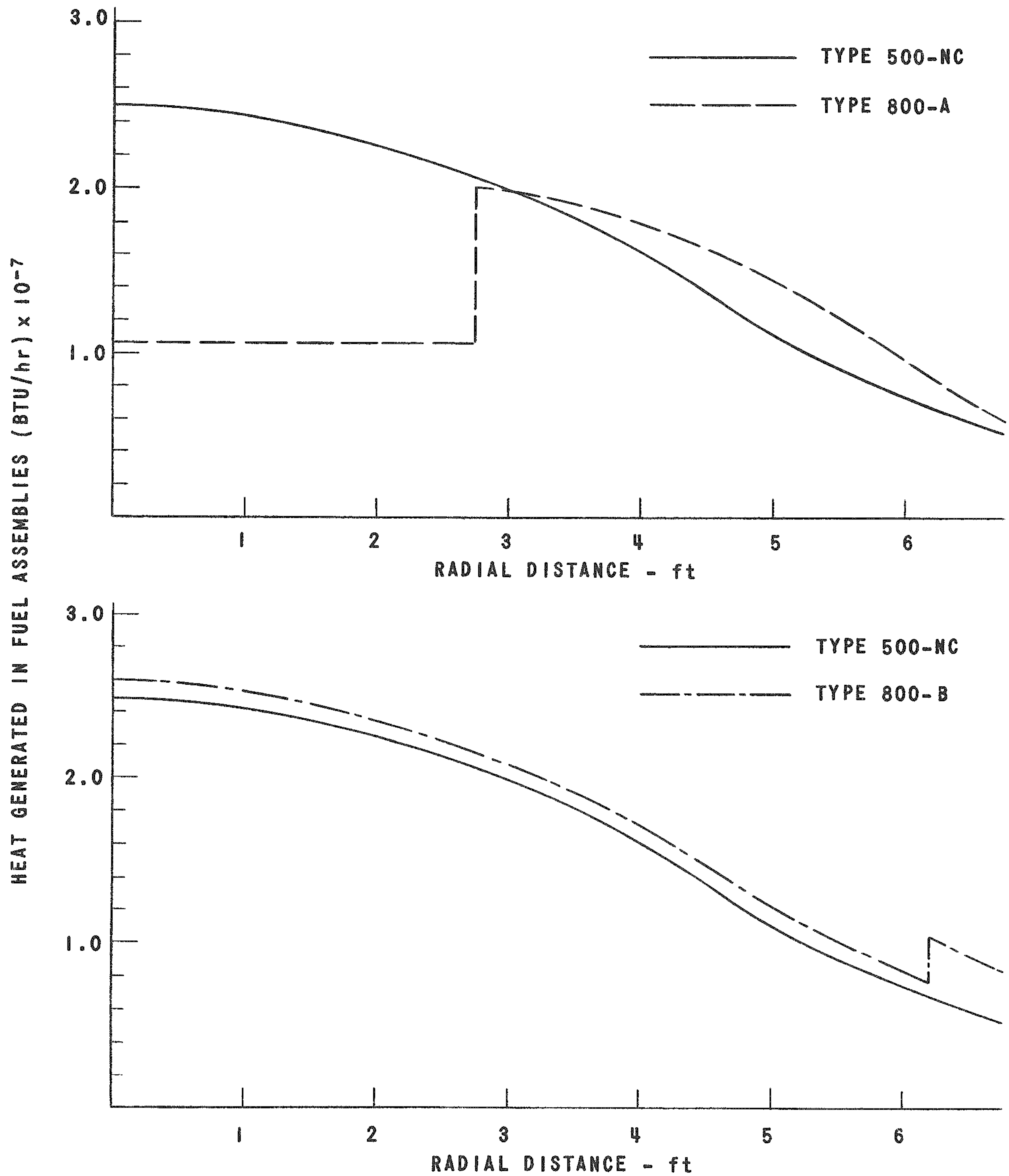


FIG. 27
COMPARISON OF POWER DISTRIBUTIONS IN REACTOR TYPES 500-NC, 800-A AND 800-B

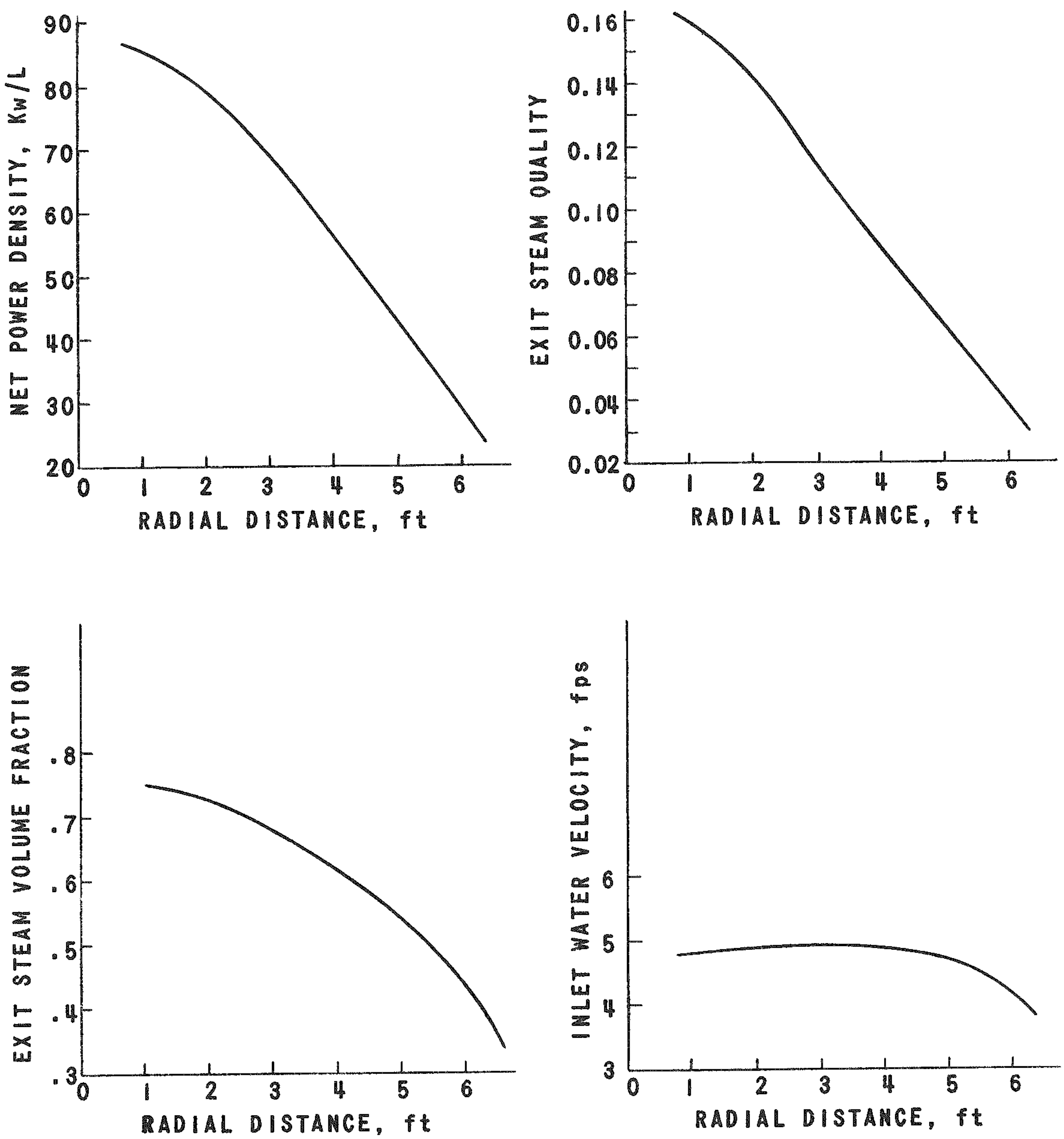


FIG. 28
PERFORMANCE CHARACTERISTICS OF THE BOILING ZONE IN REACTOR TYPE 800-B

APPENDIX: ECONOMICS OF CSBR DESIGN CONCEPTS

C. F. Bullinger

I. INTRODUCTION

An economic analysis was made of three concepts of Central Station Heavy Water Boiling Reactors: 500 FC, 500 NC, and 800 B. The analysis was designed to evaluate the relative capability of each system, in terms of reactor performance, plant cost, fuel processing and fabrication costs, and to produce electricity economically competitive with existing conventional station power plants.

The economics of Design 800 A have been omitted since initial studies revealed an unfavorable set of cost conditions.

A typical plant arrangement is shown in Figs. 1 and 2.

The results, summarized in Table I, indicate the superheated design (800 B) reflects a cost advantage which, for the most part, can be attributed to savings in the steam plant and turbine-generator. In breaking down plant costs (Table II) it becomes evident that savings of $12,000,000 or more, in capital costs must be effected to realize a saving of 1 mill/kwhr in annual capital charges.

Table I

SUMMARY OF PERFORMANCE CHARACTERISTICS AND ECONOMICS OF CENTRAL STATION D_2O BOILING REACTORS

Reactor Concept:	500 FC	500 NC	800 B
Reactor Thermal Power, mw	985	1100	1150
Gross Electrical Power, mw	280	325	370
Net Electrical Power, mw	262	291	348
Plant Factor	0.8	0.8	0.8
Plant Heat Rate, BTU/kw	11,820	11,900	10.530
Net Power Generated/year, 10^9 kwhr	1.83	2.04	2.43
Plant Cost, (Table II)			
$Million	79,250	80.081	76.572
$/kw	302	277	220
Annual Capital Charge at 15% (Table IX)			
$Million	12.918	13.336	12.795
mills/kwhr	7.0	6.6	5.2
Annual Fuel Costs and Charges (Table IX)			
$Million	10.786	12.117	12.814
mills/kwhr	5.9	6.0	5.3
Annual D_2O Cost (Table IX)			
$Million	1.88	1.76	1.76
mills/kwhr	1.02	0.87	0.75
Annual Oper. and Maint. Cost (Table IX)			
$Million	2.121	2.121	2.121
mills/kwhr	1.15	1.02	0.87
TOTAL POWER COST, mills/kwhr	15.1	14.5	12.2

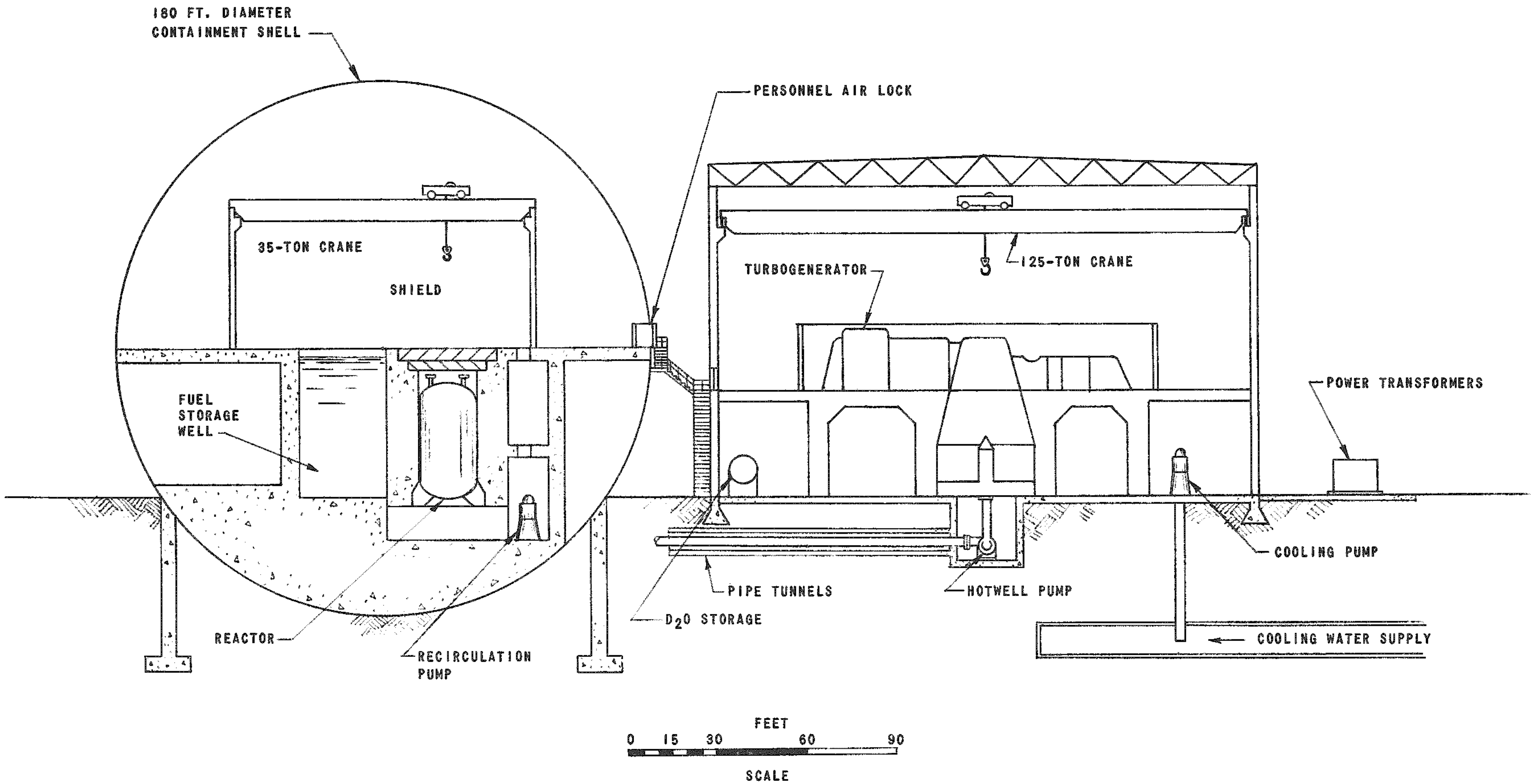


FIG. I
ELEVATION VIEW OF CENTRAL STATION D_2O BOILING WATER REACTOR

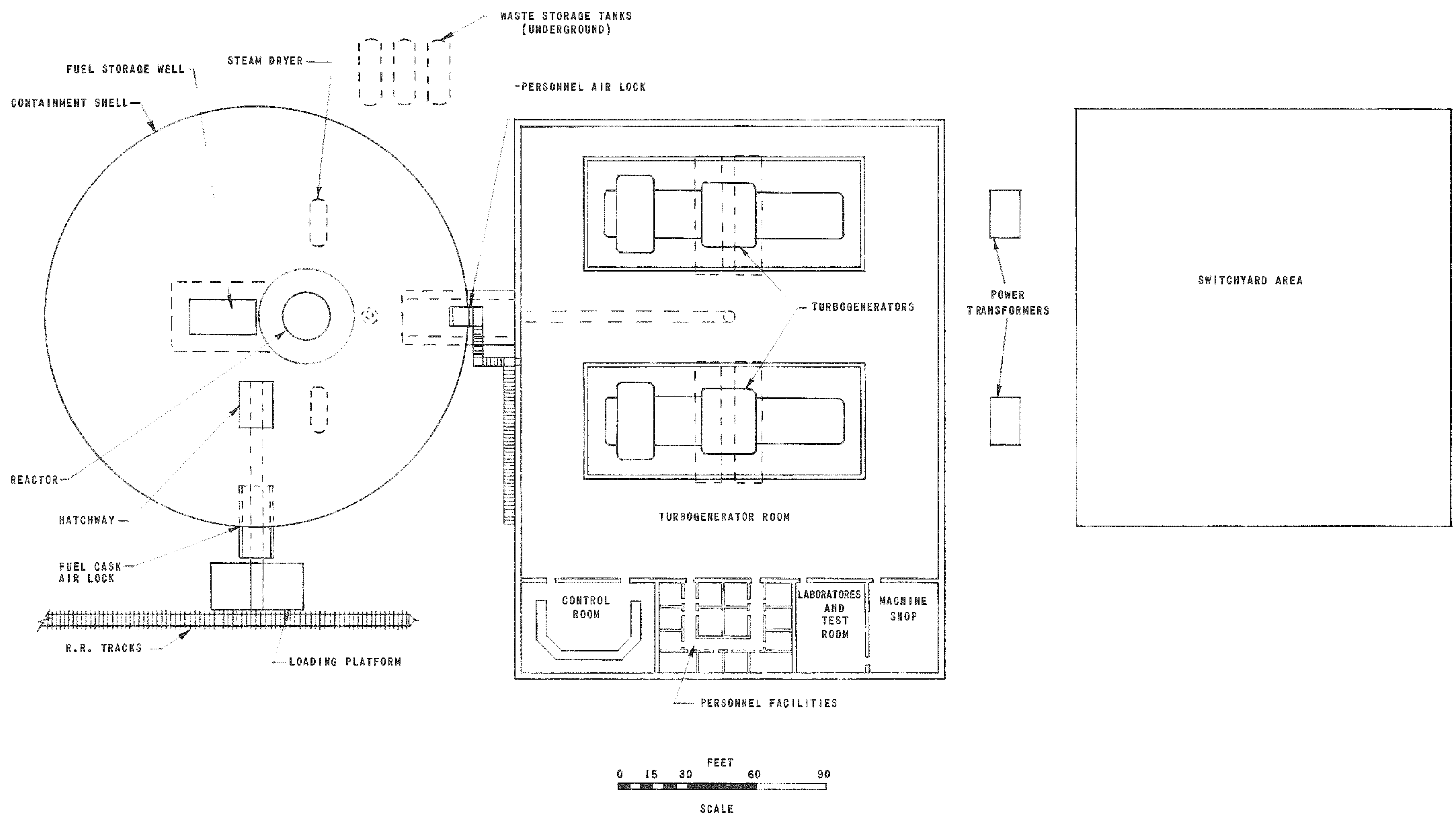


FIG. 2
PLAN VIEW OF CENTRAL STATION D_2O BOILING WATER REACTOR

Table II

ESTIMATED CSBR PLANT COSTS

Reactor Design	500 FC		500 NC		800 F	
Structures						
Land	$ 500,000		$ 500,000		$ 500,000	
Ground Improvements	2,500,000		2,500,000		2,500,000	
Turbine-Generator Building	4,200,000		4,200,000		3,700,000	
Containment Vessel	3,700,000		3,500,000		3,500,000	
Circulating Water Structures (Crib Houses, Dredging, Intake, Discharge, etc.)	3,500,000		3,500,000		3,500,000	
Miscellaneous Other Buildings	500,000		500,000		500,000	
Lighting and Miscellaneous Electrical	200,000		200,000		195,000	
Substation Foundations	125,000		125,000		125,000	
Total		$15,225,000		$15,025,000		$14,520,000
Other Reactor Plant Equipment						
Steam Separators	$ 600,000		$ 600,000		$ 600,000	
Reactor Circulating Pumps*	480,000		-		-	
Water Treating Equipment	175,000		175,000		175,000	
Miscellaneous Pumps*	60,000		60,000		60,000	
Miscellaneous Heat Exchangers	175,000		175,000		175,000	
Miscellaneous Tanks	105,000		105,000		105,000	
Piping and Insulation*	6,900,000		6,200,000		6,570,000	
Instruments and Control	90,000		85,000		85,000	
Service Water Supply	150,000		150,000		150,000	
Feed Water Pumps*	525,000		500,000		500,000	
Waste Disposal Plant	750,000		750,000		750,000	
D_2O Refining Plant	1,250,000		1,250,000		1,250,000	
Blower and Duct Work	-		-		100,000	
Filters	55,000		55,000		55,000	
Miscellaneous Other Equipment	50,000		50,000		50,000	
Total		$11,365,000		$10,155,000		$10,625,000
Turbine Plant Equipment						
No. of T-G Units	2		2		2	
T-G Output, each kw	140,000		155,000		185,000	
Steam Pressure and Temperature at Throttle	725 psia-sat.		725 psia-sat.		700 psia - 810°F	
Generators						
Type	T.C.D.F.		T.C.D.F.		T.C.T.F.	
rpm	1,800		1,800		3,600	
kva (each)	160,000		192,000		224,000	
T-G Units, accessories, etc.*	$14,000,000		$14,900,000		$11,800,000	
Condensing Equipment, Water Pumps, etc.*	4,500,000		4,875,000		5,100,000	
Piping and Insulation*	700,000		750,000		730,000	
Instrumentation, Controls, Panels, etc.	120,000		120,000		120,000	
Chlorination	75,000		75,000		75,000	
Miscellaneous Pumps*	85,000		85,000		85,000	
Air-Drying Equipment	400,000		400,000		400,000	
Total		$19,880,000		$21,205,000		$18,310,000
Electric Plant						
Main Power Equipment	$ 575,000		$ 635,000		$ 695,000	
Auxiliary Power Equipment	1,360,000		1,400,000		1,540,000	
Main Power Transformers	740,000		900,000		1,050,000	
Substation	800,000		800,000		800,000	
Temporary Power and Light	100,000		100,000		100,000	
Total		$ 3,575,000		$ 3,835,000		$ 4,185,000
Miscellaneous Power Plant Equipment						
Cranes and Hoists	$ 300,000		$ 300,000		$ 275,000	
Air Compressors	40,000		40,000		40,000	
Air Conditioning, Heating and Ventilation, etc.	300,000		300,000		300,000	
Fire Protection - CO_2	60,000		60,000		60,000	
Fire Protection - Other	25,000		25,000		25,000	
Laboratory Equipment	30,000		30,000		30,000	
Miscellaneous Bilge Pumps	10,000		10,000		10,000	
Machine Shop Equipment	75,000		75,000		75,000	
Furniture, Lockers, Office Equipment, etc.	25,000		25,000		25,000	
Communication Equipment	25,000		25,000		25,000	
Total		$ 890,000		$ 890,000		$ 865,000
Spare Parts		300,000		300,000		300,000
TOTAL		$51,235,000		$51,410,000		$48,805,000
Contingency (15%)		7,685,000		7,712,000		7,320,000
Total Estimated Construction Cost		$58,920,000		$59,122,000		$56,125,000
Top Charges						
Professional Services, Field Supervision, Administrative Expenses, Interest during Construction, Taxes, etc. (15%)		8,838,000		8,868,000		8,420,000
TOTAL PLANT COST		$67,758,000		$67,990,000		$64,545,000
Plant Cost, $/kw (net)		$ 268		$ 234		$ 185
Reactor and Associated Equipment						
Reactor Pressure Vessel	$10,512,000		$11,111,000		$11,047,000	
Fuel Handling Equipment	460,000		460,000		460,000	
Nuclear Instrumentation	120,000		120,000		120,000	
D_2O Leak Detection System	400,000		400,000		400,000	
Total		$11,492,000		$12,091,000		$12,027,000
Total Estimated Steam Plant Cost		67,758,000		67,990,000		64,545,000
TOTAL PLANT COST		$79,250,000		$80,081,000		$76,572,000
Annual Capital Charge (15%)		$11,880,000		$12,012,000		$11,485,000
Plant Cost, $/kw (net)		$ 302		$ 277		$ 220

*Includes approximately 15% allowance for D_2O handling requirements.

II. COSTS ANALYSES

The analyses are based upon plant costs compiled by Sargent and Lundy Engineers, Chicago, Illinois, and upon nuclear costs prepared jointly by the author and L. E. Link at ANL.

A. Site

Plant layout and cost estimates are based upon identical site conditions peculiar to the Middle Western States. Each plant would be located on a 1,000-acre site amply supplied with coolant from a river having a minimum seasonal variation in level. Hurricanes, floods, and earthquakes were not considered as dominant features. It was further assumed that founding conditions would accommodate soil bearings of approximately 5,000 psf.

B. Structures

The reactor and auxiliary equipment are located in a steel containment sphere (180 ft dia.) designed for internal pressure of approximately 15 psig.

The turbine-generator units, requisite auxiliary equipment, and personnel facilities are located in an adjacent building. The building is assumed to be constructed with insulated metal panel siding.

C. Reactor Core and Pressure Vessel

Each core is contained in a pressure vessel of the same basic design and size (17 ft OD).

The use of "cold" D_2O moderator necessitated employment of double-wall fuel element construction for insulation purposes. The outer shroud or wall was considered an integral part of the permanent core structure. While this approach reflected an increase in capital cost, the resultant saving in fuel cycle charges served to decrease the ultimate power cost. The costs of the permanent core structure are summarized in Table III.

The pressure vessel diameter dictates field erection and fitting out. As a consequence the fabrication and erection costs amount to \$9.00/lb. The costs of the pressure vessel and related structures are summarized in Table IV.

D. D_2O System

The D_2O inventory, including 30% excess for contingencies and reserve, is approximately 290 tons for Design 500 FC, 271 tons for Design 500 NC, and 271 tons for Design 800 B. The pressure vessel holdup in each system is 140 tons.

Table III

PERMANENT CORE STRUCTURES

Reactor Design:	500 FC	500 NC	800 B Boiling Zone	800 B Super Heat Zone
Outer Shrouds				
No. of Shrouds	228	330	270	60
Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Diameter, in.	5.05	6.20	6.20	5.35
Length, ft	17	17	17	17
Weight, lb	30	37	37	32
Control Cylinders				
No. of Cylinders	31	55	37	18
Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Diameter, in.	9.75	8.50	8.50	8.50
Length, ft	18	18	18	18
Weight, lb	196	171	171	171
Total Weight Zircaloy, lb				
Shrouds	6,850	12,210	10,000	1,920
Control Cylinders	6,100	9,400	6,300	3,100
Total Cost Zircaloy (at $70/lb)	$910,000	$1,540,000	$1,150,000	$350,000
Core Grid Plate Assembly			($1,500,000)	
Top Grid	$ 80,000	$ 60,000	$ 60,000	
Bottom Grid	130,000	120,000	120,000	
Outer Core Shell	38,000	38,000	38,000	
Forced Circulation Manifold	80,000			
TOTAL COST OF CORE STRUCTURE:	$1,238,000	$1,758,000	$1,718,000	

Table IV

PRESSURE VESSEL

Reactor Design:	500 FC		500 NC		800 B	
Vessel Shell and Connections	$4,580,000		$4,500,000		$ 4,500,000	
Thermal Shield	1,870,000		1,870,000		1,870,000	
Vessel Support	40,000		40,000		40,000	
Core Structure	1,238,000		1,758,000		1,718,000	
Insulation	60,000		60,000		60,000	
VESSELL COST:		7,788,000		8,228,000		8,188,000
Engineering and Development (10%)		779,000		823,000		819,000
Contingency and Top Charges (25%)		1,945,000		2,060,000		11,047,000
COST, $/kw		$40		$38		$32

All D_2O piping is fabricated from Type 304 stainless steel. Automatic shut-off valves are installed on interconnecting piping between the containment vessel and the turbine-generator building.

Conventional shaft-sealed pumps are employed.

An allowance of 15% was made for extra cost of leak detection and collection equipment on all D_2O system components.

E. Fuel Handling

The loading and unloading of fuel is accomplished with a fuel transfer coffin. Submerged fuel transfer was not considered because of the large D_2O inventory posed by this technique.

F. Fuel Cycle

The fuel cycle used to compute use and rental charges is shown in Fig. 3. It was assumed the cycle reached equilibrium about one year after initial startup and that a constant charge rate prevailed thereafter. The fuel cycle for each core is given in Table V.

Table V

FUEL CYCLE

			800 B	
Reactor Design:	500 FC	500 NC	Boiling Zone	Superheat Zone
Core Loading, metric ton (mt)	34.53	40.59	33.21	9.78
Fuel	UO_2	UO_2	UO_2	UO_2
Enrichment, %				
Initial (U^{235})	(Natural U)	1.15	1.15	1.9
Final (U^{235})	0.33	0.63	0.63	1.26
Final (Pu^{239})	0..25	0.36	0.36	0.36
Burnup, mwd/mt	5,000	6,000	6,000	6,500
Plutonium Production, kg/mt Uranium				
Pu^{239}	2.5	3.6	3.6	3.6
Pu^{240}	0.6	0.7	0.7	0.6
Pu^{241}	0.3	0.2	0.2	0.2
Total	3.4	4.5	4.5	4.4

Fabrication and processing costs (Table VI) for the first core are included in the capital charge assessments. One core is considered analogous to a conventional power plant coal inventory which is constantly being replenished. Fuel processing and fabrication weight losses are restricted to 1%, since larger amounts cannot be tolerated.

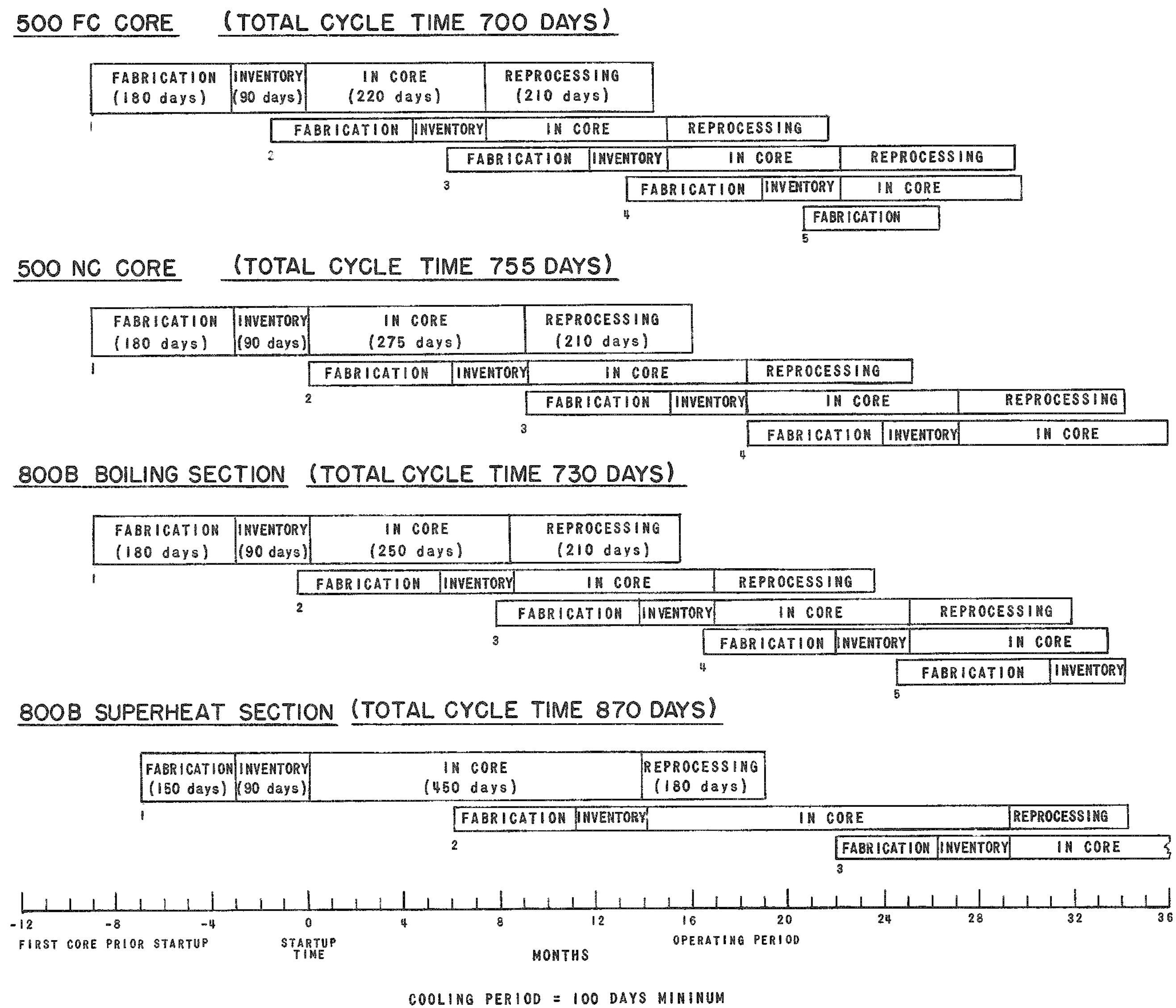


FIG. 3
FUEL CYCLE TIME FOR USE AND RENTAL CHARGES

Table VI

FUEL ELEMENT PROCESSING AND FABRICATION

Reactor Design:	500 FC		500 NC		800 B Boiling Zone		800 B Superheat Zone	
Core Loading, kg	34,530		40,590		33,210		9,780	
Fuel Processing								
Weight Loss Allowance (1%), kg	345		406		332		98	
Total Initial Weight Received, kg	34,875		40,996		33,542		9,878	
Conversion UF_6 to UO_2 at $8.80/kg		*		$360,000.		$ 295,000.		$ 87,000.
Conversion UO_2 to Pellets at $13.20/kg		$461,000.		$541,000.		$ 442,000.		$150,000.**
Fuel Assemblies								
Total Assemblies	228		330		270		60	
No. Pins/Assembly	85		69		69		85	
No. Sections/Pin	6		6		6		6	
Total Sections (28 in. long)	116,280		136,620		111,780		30,600	
Fuel/Section, kg	0.298		0.298		0.298		0.323	
Zircaloy Tubing Cost/ft	$3.66		$3.66		$3.66		$0.30 †	
Zircaloy Cost/Tube (28 in. long)	$8.40		$8.40		$8.40		$0.70	
Zircaloy Cost/Core		$1,006,000.		$1,260,000.		$1,003,000.		$21,000.
Zircaloy Tube End Fittings Cost/Set	$0.84		$0.84		$0.84		$0.79	
Zircaloy Tube End Fittings Cost/Core		97,000.		$ 114,000.		93,000.		24,000.
Zircaloy Spacer Cost/Core		76,000.		89,000.		73,000.		4,000.
Assembly Cost/Section:								
Tube, Pellets, and End Fittings	$28.		$28.		$28.		$28.	
Assembly Cost/Core		$3,256,000		$3,825,000.		$3,130,000.		$857,000.
Fuel Assembly Shrouds (Zircaloy)								
Total Weight, lb	3,300		5,600		4,600		1,100	
Fabrication Cost/lb	$70.		$70.		$70		$4	
Total Cost/Core		$231,000.		$392,000.		$322,000.		$4,000.
End Fittings and Extension/Core at $240/pair		55,000.		79,000.		65,000.		14,000.
Labor Cost/Core (at $145/Assembly)		$33,000.		$48,000.		$39,000.		$9,000.
Shipping Cost (at 2/kg)		$69,000.		$81,000.		$66,000.		$20,000.
Total Direct Cost/Core		$5,284,000.		$6,789,000.		$5,528,000.		$1,190,000.
Contingency and Top Charge (1 st Core Only), 30%		$1,585,000.		$2,036,000.			$2,015,000.	
TOTAL CAPITALIZATION COST (1st Core Only):		$6,869,000.		$8,825,000.			$8,733,000.	
Annual Capital Charge at 15%:		$1,030,000.		$1,324,000.			$1,310,000.	

*No Conversion Charge for Natural Uranium Dioxide

**$15.20/kg

† Type 347 Stainless Steel

The core direct cost for fuel processing, fabrication, labor, and material is summarized in Table VII. An allowance of 5% for spare fuel elements was made for the first core only.

All costs are based upon prices quoted by firms capable of handling fabrication contracts of the indicated magnitude. It becomes evident that for the indicated burnup values fabrication would be continuous and that on-the-site fabrication and processing might be feasible, resulting in decreased fuel costs.

The unit fuel cost is summarized in Table VIII. A plutonium credit of $12/gm was used since it is probable that D_2O plants will not be operated in time to take advantage of the $30 quoted value.

The reprocessing charges are based on quoted government rates.

Table VII

SUMMARY OF CORE DIRECT COST - ONE COMPLETE CYCLE

Reactor Design:	500 FC		500 NC		800 B Boiling Zone		800 B Superheat Zone	
Core Loading, kg	34,530		40,590		33,210		9,780	
Fabrication								
Fuel Processing								
Conversion UF_6 to UO_2	*		$ 360,000		$ 295,000		$ 87,000	
Conversion UO_2 to Pellets	$ 461,000		541,000		442,000		$150,000	
Fuel Assemblies								
Fuel Tubes	$1,006,000		$ 1,260,000		$1,003,000		$ 21,000	
End Fittings	97,000		114,000		93,000		24,000	
Spacers	76,000		89,000		73,000		4,000	
Loading and Testing	$3,256,000		$ 3,825,000		$3,130,000		$857,000	
Fuel Assembly Shrouds								
End Fittings	$ 55,000		$ 79,000		$ 65,000		$ 14,000	
Total Shroud Cost	$ 231,000		$ 392,000		$ 322,000		$ 4,000	
Final Assembly Labor	$ 33,000		$ 48,000		$ 39,000		$ 9,000	
Shipping Cost (at $2/kg)	$ 69,000		$ 81,000		$ 66,000		$ 20,000	
Total Direct Fabrication Cost:		$5,284,000		$ 6,789,000		$5,530,000		$ 1,190,000
Fabrication Cost/kg Contained Fuel	$ 153		$ 167		$ 167		$ 122	
Fuel Reprocessing								
Shipping (at $5/kg)	$ 173,000		$ 203,000		$ 166,000		$ 49,000	
Reprocessing (at $20/kg)	691,000		812,000		644,000		196,000	
Conversion UNH to UF_6	**		227,000		185,000		54,000	
Conversion Pu NH to Pu Buttons (at $ 1.50/gm)	176,000		273,000		223,000		65,000	
Total Direct Reprocessing Cost:		$1,040,000		$ 1,515,000		$1,238,000		$ 364,000
TOTAL DIRECT FABRICATION AND PROCESSING COST/CORE:		$6,324,000		$ 8,304,000		$6,768,000		$ 1,554,000
Total Cost, mills/kwhr		5.7		5.4		4.8 (4.6 avg)		3.5

*No conversion charge for natural uranium dioxide.

**Further processing not required because 0.34% depleted uranium has no resale value.

Table VIII

SUMMARY OF ANNUAL FUEL COSTS
(in mills/kwhr)

Reactor Design:	500 FC		500 NC		800 B Boiling Zone		800 B Superheat Zone	
Loading/core, mt	34.53		40.59		33.21		9.78	
Exposure, mwd/mt	5,000		6,000		6,000		6,500	
Full Power at 0.8 P.F.								
Core days	220		275		250		450	
Core years	0.60		0.75		0.68		1.23	
Core Process and Fabrication Direct Cost								
Cores/year	1.67		1.33		1.46		0.81	
Cost/core, $	6,324,000		8,304,000		6,766,000		1,554,000	
Total cost, $	10,500,000		11,000,000		9,900,000		1,260,000	
mills/kwhr		5.7		5.4		4.8		3.5
Fuel Charges and Credits								
Rental charge (3 cores at 4%), $	169,000		472,000		386,000		243,000	
mills/kwhr		0.09		0.23		0.19		0.64
Initial fuel value/core, $	1,410,000		3,930,000		3,220,000		2,026,000	
Pu credit ($12/gm)	1,404,000		2,184,000		1,718,000		516,000	
Reprocessed UF_6 credit	(None)		1,286,000		1,044,000		1,012,000	
Net fuel consumption charge/core, $	6,000		460,000		388,000		498,000	
Annual fuel consumption charge, $	10,000		625,000		566,000		402,000	
mills/kwhr		0.01		0.31		0.27		1.06
Total Core Fabrication Cost, mills/kwhr at 0.8 P.F.		5.8		5.9		5.26	5.25 avg	5.20
Fuel Operating Capital Fund (60-day), $	1,780,000		1,997,000		2,116,000			
Annual Charge (6%)	107,000		120,000		127,000			
Operating capital fund cost, mills/kwhr		0.05		0.06			0.05	
TOTAL FUEL COSTS, mills/kwhr at 0.8 P.F.		5.9		6.0			5.3	

In order to simplify cost computations, a 60-day operating capital fund was set up to disburse fuel charges.

G. Operating and Capital Charges

The annual operating and capital charges are summarized in Table IX. The capital charges are assessed at 15% and include: (1) plant costs; (2) engineering and development costs; and (3) fabrication and material costs for one core, excluding fuel.

Table IX

ANNUAL CAPITAL AND OPERATING CHARGES

Reactor Design:	500 FC Dollars	500 FC Mills/kwhr	500 NC Dollars	500 NC Mills/kwhr	800 B Dollars	800 B Mills/kwhr
Capital Costs and Charges (15%)						
Plant (Table II)	\$11,888,000	6.4	\$12,012,000	5.9	\$11,485,000	4.7
Fuel Fabrication and Processing (Table VI)	1,030,000	0.6	1,324,000	0.7	1,310,000	0.5
TOTAL:	\$12,918,000	7.0	\$13,336,000	6.6	\$12,795,000	5.2
D_2O Usage and Rental (Table X)						
Rental (4%)	\$ 650,000	0.35	\$ 606,000	0.32	\$ 606,000	0.27
Losses	1,061,000	0.58	988,000	0.47	988,000	0.41
Distillation Plant Operation	150,000	0.08	150,000	0.07	150,000	0.06
	\$ 1,861,000		\$ 1,744,000		\$ 1,744,000	
Charge for 60-day Capital Fund (6%)	19,000	0.01	17,000	0.01	17,000	0.01
TOTAL:	\$ 1,880,000	1.02	\$ 1,761,000	0.87	\$ 1,761,000	0.75
Operation and Maintenance (Table X)						
Salaries, Misc. Oper. Expense, Maint.	\$ 2,100,000	1.14	\$ 2,100,000	1.01	\$ 2,100,000	0.86
60-day Operating Capital Fund (6%)	21,000	0.01	21,000	0.01	21,000	0.01
TOTAL:	\$ 2,121,000	1.15	\$ 2,121,000	1.02	\$ 2,121,000	0.87
Fuel Costs and Charges (Table VIII)						
Process and Fabrication	\$10,500,000	5.72	\$11,000,000	5.4	\$11,160,000	4.55
Rental (4%)	169,000	0.09	472,000	0.23	629,000	0.25
Consumption	10,000	0.01	625,000	0.31	968,000	0.40
60-day Operating Capital Fund (6%)	107,000	0.06	120,000	0.06	127,000	0.05
TOTAL:	\$10,786,000	5.88	\$12,217,000	6.00	\$12,884,000	5.3
Total Power Cost, mills/kwhr at 0.8 P. F.		15.1		14.5		12.2

The annual operating costs are given in Table X. Separate 60-day operating capital funds were established for: (1) D_2O rental, loss, and recovery plant operation; (2) plant operation and maintenance; and (3) fuel cycle.

The unit costs (mills/kwhr) were calculated to two decimal places in some cases only to establish their presence in the over-all total.

Table X

SUMMARY OF ANNUAL OPERATING COSTS

Reactor Design		500 FC		500 NC		800 B	
D_2O Inventory and Operation							
Total plant holdup, tons		233		208		208	
Reserve and contingency, (30%)		67		63		63	
Total Inventory, tons		290		271		271	
Inventory value (at $28/lb),	$	16,240,000		15,180,000		15,180,000	
Rental charge (4%),	$	650,000		606,000		606,000	
	mills/kwhr		0.35		0.32		0.27
Loss at 8.5% of holdup, tons		18.95		17.65		17.65	
	$	1,061,000		988,000		988,000	
	mills/kwhr		0.58		0.47		0.41
Distillation plant operation,	$	150,000		150,000		150,000	
	mills/kwhr		0.08		0.07		0.06
Operating capital fund (60-day),	$	310,000		290,000		290,000	
Capital fund charge (6%),	$	19,000		17,000		17,000	
	mills/kwhr		0.01		0.01		0.01
TOTAL,	mills/kwhr		1.02		0.87		0.75
Operation and Maintenance							
Salaries (135 men at $10,800),	$	1,500,000		1,500,000		1,500,000	
Misc. operating expense,	$	300,000		300,000		300,000	
Maintenance,	$	300,000		300,000		300,000	
TOTAL,	$	2,100,000		2,100,000		2,100,000	
	mills/kwhr		1.14		1.01		0.86
Operating capital fund (60-day),	$	350,000		350,000		350,000	
Capital fund charge (6%),	$	21,000		21,000		21,000	
	mills/kwhr		0.01		0.01		0.01
TOTAL,	mills/kwhr		1.15		1.02		0.87
Operating Capital Fund (60-day)							
Operation and maintenance,	$	350,000		350,000		350,000	
D_2O inventory and operation,	$	310,000		310,000		310,000	
Fuel processing and fabrication	$	1,750,000		1,816,000		1,850,000	
Fuel rental and usage,	$	30,000		181,000		266,000	
TOTAL,	$	2,440,000		2,657,000		2,776,000	
Annual Charge (6%),	$	146,000		159,000		167,000	
	mills/kwhr		0.08		0.08		0.07

III. CONCLUSIONS

Plant capital charges command a unit cost with little opportunity for reduction. For instance, in plants costing 75 to 80 million dollars, it is necessary to save about 12 million dollars to effect a reduction of 1 mill/kwhr in electrical cost.

Because of the extreme integrity associated with heavy water systems, the D_2O plants costs are appreciably higher than their H_2O counterparts. However, in plants of the size discussed, the additional costs are recovered in the net mill/kwhr column. The saving realized by utilizing superheated steam indicates the direction which must be taken by reactor designers. Public utilities and turbine manufacturers are not interested in reducing the efficiency of their plants and components.

It is unlikely that operation and maintenance costs and charges can be reduced significantly.

The allowable fuel burnup of 6000 mwd/metric ton has an important bearing upon the over-all fuel cycle cost. Fuel processing periods have been reduced to a minimum. Although it is still necessary to pay rental charge on an average of three cores at any one time, the fuel rental and use charges represent 10 to 15% of the total fuel cycle cost. By lengthening the fuel burnup time, yearly fabrication costs could be reduced proportionately.

The area for cost reduction lies in decreased fabrication charges and increased core lifetime. Zircaloy is expensive; hence the dollar value of the core structure and fuel assembly parts is appreciable. The substitution of stainless steel for Zircaloy in the 500 NC concept would permit an increase in uranium enrichment from 1.15% to about 3.5% for the same dollar expenditure. However, the problem of controlling the excess reactivity from "cold" to "hot" operating condition would become acute. The net benefits with respect to increased fuel lifetime and plutonium production have not been determined.

The advent of decreased fabrication and material costs and increased core lifetime could reflect a reduction of about 50% (~3 mills) in fuel cycle costs. This is an area of reasonable doubt. In the light of evidence gradually developing, and in the face of rising construction costs, it is apparent that 10-mill power will not evolve from the present generation of nuclear reactors.