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PRELIMINARY DESIGN AND HAZARDS REPORT BOILING REACTOR EXPERIMENT V (BORAX V)

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I. INTRODUCTION

In 1957 the idea for another boiling water reactor in the BORAX series was conceived by personnel of both the Idaho Division and Reactor Engineering Division of Argonne National Laboratory. In 1958 the Laboratory proposed that an extremely flexible boiling reactor experimental facility, called BORAX V, be constructed and operated as a modification of and addition to the existing BORAX plant at the National Reactor Testing Station, Idaho.

The primary objectives of the proposed BORAX V program are to test nuclear superheating concepts, and to advance the art of boiling water reactor design by performing experiments which will improve the understanding of factors limiting the stability of boiling reactors at high power densities.

In evaluating the hazards associated with the operation of BORAX V, one should keep in mind that the remote location at the National Reactor Testing Station and the half-mile distance between control building and reactor were chosen so that the safety aspects of nuclear superheater operation and the excursion characteristics of boiling reactors could be safely investigated.

This report is preliminary. At the time of writing, the status of the BORAX V project is that the design of the reactor buildings and plant, done in collaboration with the architect-engineer has been completed and construction has just started; however, the mechanical design of the fuel and cores is still tentative. In particular, the core physics and that part of the heat engineering and hazard evaluation dependent upon it are incomplete. It is expected that the usual hazards summary report, submitted prior to reactor operation, will contain a firm design and a thorough hazards evaluation.

II. SUMMARY

Some of the plant utilized in the previous BORAX experiments is incorporated in the BORAX V system, but the reactor and its building, as well as the control building, located $\frac{1}{2}$ mile from the reactor, are new. A cutaway view of the BORAX V facility is shown in Fig. 1.

The reactor vessel is a cylinder with ellipsoidal heads, made of carbon steel clad internally with stainless steel. The inside diameter is $5\frac{1}{2}$ ft and the internal height is 16 ft. The core will be centered about 4 ft from the bottom. The top portion of the vessel will be used as a steam dome. Control rods will be driven from below the vessel.

BORAX V has been designed as an extremely flexible system. At present there are plans to provide three separate core configurations: a boiling core with a centrally located superheater; a boiling core with a peripherally located superheater; and a boiling core without superheater. In all cases it will be possible to operate with either natural or forced circulation of water through the core. The water serves as the moderator in both the boiling and superheater regions of the core and as coolant in the boiling region. The superheater is cooled by steam.

Each of the three cores is 24 in. high and has an effective diameter of 39 in. when containing the maximum of 60 four-inch-square fuel assemblies arranged in an 8 x 8-array with the corner assemblies missing. The central superheat core will contain 12 fuel assemblies in the central position, while the peripheral superheat core will contain 16 fuel assemblies, 4 in the middle of each outer row. Each square cell of 4 assemblies is surrounded by adjacent control-rod channels. Experiments will be run with the full complement of 60 fuel assemblies as well as with reduced numbers. The BORAX V reactor with a central superheater is shown in Fig. 2.

The reference design boiling fuel assemblies are composed of individual removable fuel rods (the number of rods per assembly may vary from 0 to 100) made from UO_2 of low enrichment with 0.015 in., Type 304 stainless steel cladding. The outside diameter of the rods is 0.260 in. Three different enrichments are planned: a nominal enrichment, twice nominal, and four times nominal. The degree of enrichment has not been finally determined, but is on the order of 5%.

The reference design superheater fuel assembly is made up of 5 plate-type elements with intervening water gaps. Each element is contained in a stainless steel tube which maintains a static steam-filled insulating gap between the element and the moderator water outside the channel. Each element contains 5 fuel plates separated by four 0.045-in. coolant channels. The fuel plates contain a 0.010-in.-thick, highly enriched UO_2 stainless steel cermet meat within a 0.010-in. stainless steel cladding. The two outside plates in each element contain only half as much UO_2 as the three inside plates, since they are cooled only on one surface.

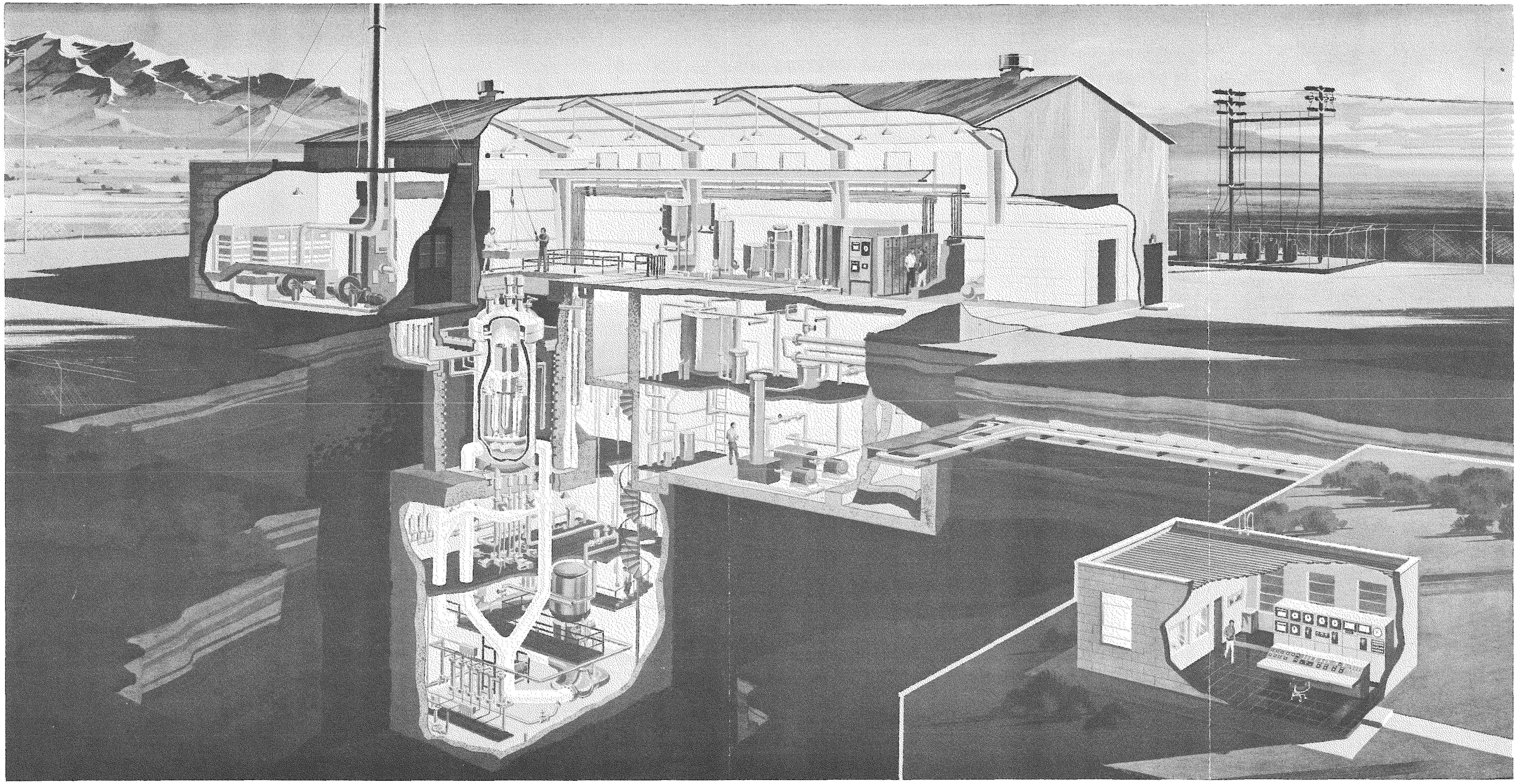


Fig. 1
BORAX V Reactor Facility

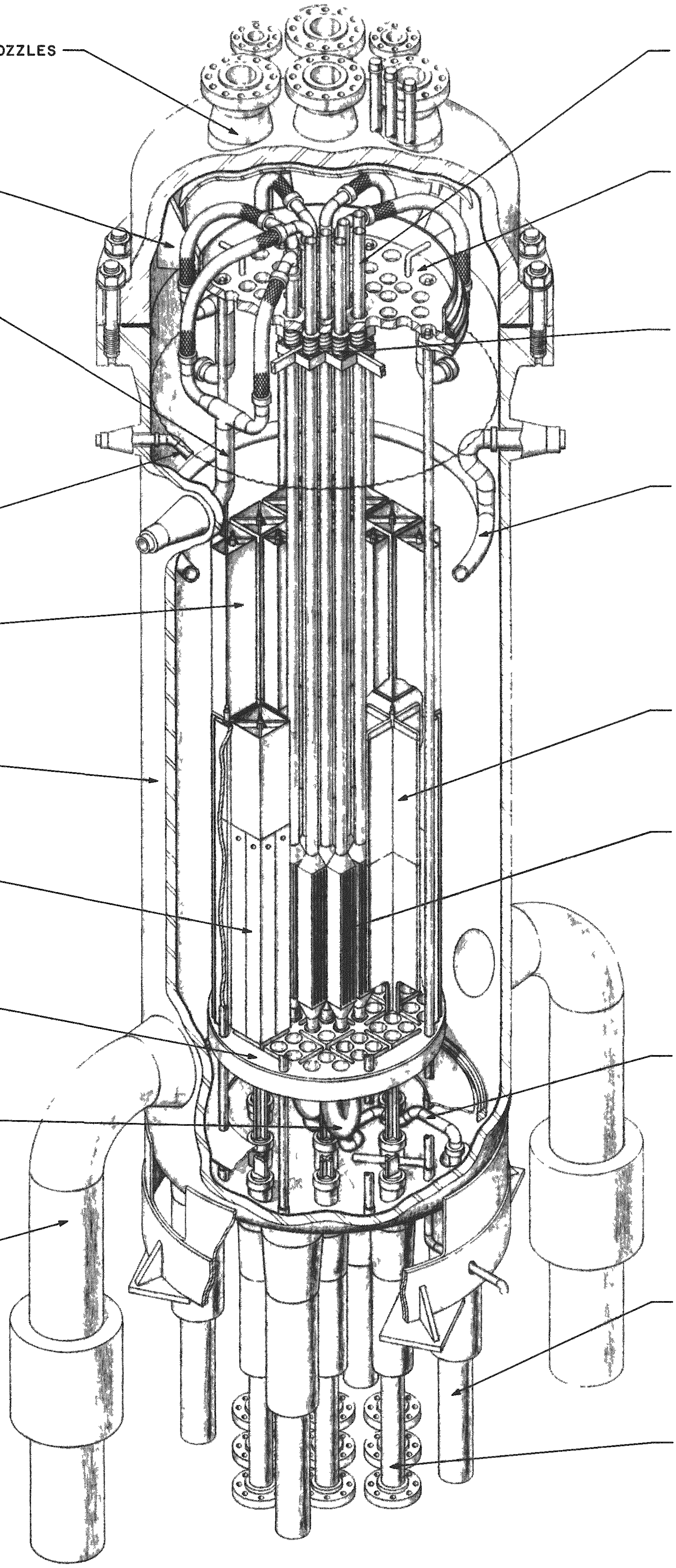


Fig. 2
Reactor with Central Superheater
BORAX V

Water flows up through the boiling core and the riser above the core by either forced or natural circulation. The steam formed in the core continues up into the steam space while the water flows down through the annular downcomer area between the core and the reactor vessel. Feedwater introduced at the top of the downcomer subcools the recirculating water and helps to increase the natural circulation rate. For the boiling core case, the steam leaves the pressure vessel through a 6-in. steam line. For the case of the superheating cores the steam from the steam dome enters the inlets to the superheat fuel assemblies, makes one pass down through half the superheat assemblies, mixes in the lower plenums, and makes a second pass up through the remainder of the superheat assemblies. It then leaves the pressure vessel through flexible tubing and the superheated-steam outlet nozzles. Because BORAX V is using the turbine-generator-condenser system from the preceding BORAX plants, the 600-psig steam must be reduced in pressure and temperature before entering the turbine.

Nine control rods, driven from below, are made of Alcoa X-8001 aluminum-clad Boral. The drive mechanisms are those formerly used on EBWR, modified for use in BORAX V so that the maximum rate of addition of reactivity is limited to 0.05%/sec. Any one of 15 scram signals causes the rods to be inserted by the pressure within the reactor vessel, augmented by springs and gravity, to produce a reduction in reactivity of at least 80% of the total rod worth in 0.2 sec. An additional emergency method of reactivity control is provided by a boron-addition system which introduces a boric acid solution into the reactor vessel as a reserve emergency shutdown mechanism.

BORAX V possesses the inherent safety common to all boiling water reactors. The additional voids formed in the moderator in the event of an excursion cause a decrease in power, though the actual void coefficient of reactivity depends on the core configuration employed.

The greatest departure in design from the previous BORAX reactors is found in the superheat region. In effect this introduces a gas-cooled zone into the normal all-boiling core and presents problems common to such reactors, e.g., decay heat removal and effect of flooding the gas passages. The decay heat-removal problem is solved by designing the superheat fuel element so that it can be radiation-cooled a short time after the reactor shuts down. In fact, the reactor can operate at 3.5 Mw with no steam flow, yet the fuel elements do not melt. An alternate system of shutdown cooling by flooding the superheater is also provided. Numerous safety interlocks are provided to prevent accidental flooding of the superheater during operation, but, if this does happen, the maximum rate of reactivity addition is only 0.25%/sec.

The maximum accident is postulated as a cold water accident which would occur if the forced-convection system begins to operate while the reactor is just critical at design temperature and the water in the forced-convection piping system is at room temperature. In such a case, cold water would enter the core and cause a reactivity increase of 6% in $\frac{1}{3}$ sec, or a rate of increase of 21%/sec. This number can be compared to the 20%/sec reactivity insertion which caused the destruction of BORAX I. Under these conditions it is possible that the reactor would be destroyed. Elaborate safety systems have been designed to insure that such an accident does not occur.

In the event a radioactive cloud is released after extended 20-Mw operation, the 300-r exclusion radius is less than one-half mile, the distance to the control building, except under extremely pessimistic conditions of inversion with low wind speed.

Table I summarizes the pertinent design characteristics of BORAX V.

Table I

DESIGN CHARACTERISTICS OF BORAX V

A. Reactor Description

1. Cores

Types: Central Superheater, Peripheral Superheater, Boiling.

Geometry: (Flexible) Right Pseudocylinders; Square, Rectangular, and Octagonal Prisms, etc.

Material: Core Structure and Control Rod Shrouds	Alcoa X-8001 Al-Ni Alloy
Material: Moderator and Reflector	H_2O
Coolant	H_2O and Steam
Active Core Height, in.	24
Maximum Equivalent Cylindrical Diameter, in.	39
Maximum Number of Fuel Assemblies	60
Approximate Fuel Loadings (60 Assemblies)	
Core With Central Superheater	
U^{235}, kg	41
U^{238}, kg	645
Boiling Core	
U^{235}, kg	35
U^{238}, kg	817
Minimum H_2O Reflector, in.	
Axial	6
Radial	11

Table I (cont)

2. Boiling Fuel Assembly

Dimensions: Square X-8001 Al Tube, in.	$3\frac{7}{8}$ by $3\frac{7}{8}$ OD by $\frac{1}{16}$ Wall
Composition, 100-Rod Assembly, (Based on 4 x 4-in. cell), Vol. %	
UO_2	26.0
Stainless Steel	7.1
Al	5.8
Balance available for H_2O	61.1

3. Boiling Fuel Rods

Fuel Material	Partially enriched UO_2
UO_2 Rod Diameter, in.	0.230
Cladding Material	Type 304 SS
Cladding Thickness, nominal, in.	0.015
Fuel Rod, OD, nominal, in.	0.260

4. Superheat Fuel Assembly

Dimensions, nominal, in.	3.875 x 3.665
Number of Elements per Assembly	5
Number of Plates per Element	5
Fully Loaded Plates	3
Half-loaded Plates	2
Meat Dimension, in.	24x3.415x0.010
Meat Composition	Cermet, highly enriched UO_2 in Type 304 SS Matrix
Cladding Material	Type 304 SS
Cladding Thickness, in.	0.010
Distance between Plates, in.	0.045
Composition, Central Superheater Assembly: (Based on a 4 x 4-in. Cell), Vol. %	
Type 304 SS Structure	21
Meat (UO_2 + SS)	5.5
Static and Flowing Steam	28
Balance Available for H_2O	45.5

Table I (cont)

5. Heat Transfer and Flow Characteristics

Reactor Reference Design Power	
(Core with Central Superheater), kw	35,700
Boiling Region, kw	29,600
Superheater, kw	6,100
Steam Flow, lb/hr	88,800
Steam Conditions Leaving Reactor,	
(Superheated), °F	850
psig	540
Average Power Density - Boiling Region	
(Based on Region Vol.) kw/l	169
Average Power Density - Superheater	
(Based on Region Vol.), kw/l	144
Power Ratios Assumed, Peak to Average:	
Axial - Entire Core	1.3
Radial - Boiling Region	1.3
Local - Boiling Region	1.15
Radial - Superheater	1.2
Engineering Hot Channel Factors, Superheater:	
Enthalpy Rise Factor	1.3
Film Temperature Drop Factor	1.3
Heat Flux, BTU/hr-ft^2:	
Boiling Region, max	300,000
av	154,000
Superheater, max	108,000
av	75,700
Temp., Reference Superheat Fuel Element,	
Center, (max) °F	1115
Temp., Reference Superheat Fuel Element,	
Surface (max) °F	1100
Temp., Reference Boiling Fuel Rod	
Centerline, (max) °F	2000
Temp., Reference Boiling Fuel Rod	
Surface, (max) °F	515
Pressure Drop through Superheater Core, psi:	
1st Pass	17.9
2nd Pass	22.6

6. Control Rods

Cruciform Blades, in.	14 x 14 x $\frac{3}{8}$ thick
"T" Blades, in.	14 x 7 x $\frac{3}{8}$ thick
Poison Material	Boral
Poison Length, in.	24

Table I (cont)

Stroke, in.:	
Boiling Core	29
Superheat Core	24
Material, Cladding and Follower	X-8001 Al
Control Rod Drive Location	Bottom of Reactor Vessel
Number of Control Rods:	
Cruciform	5
"T"-shaped	4
Total Worth, %Δk/k	~18

7. Control Rod Drives

Type: EBWR with Linear Seal, Lead Screw, and Magnetic Latch

Scram Time, Effective Stroke, sec	0.2
Max. Rate of Reactivity Insertion, %/sec	0.05

8. Reactor Vessel

Diameter, Inside, in.	66
Height, Inside, ft	16
Total Wall Thickness, in.	$2\frac{1}{16}$
Cladding Thickness, in.	$\frac{3}{16}$
Vessel Material	ASTM-A-212-B, Fine Grain Steel
Cladding Material	Type 304 SS
Operating Pressure, psig	600
Design Pressure, psig	700
Design Temperature, °F	650
Neutron Shield Material (Top Head Only)	1% boron-SS
Neutron Shield Thickness, in.	1

B. Performance

1. Reactor

Nominal Full Power, Thermal Mw	20
Maximum Plant Capacity, Thermal Mw	40
Average Thermal Neutron Flux, 40 Mw, n/cm^3-sec	5×10^{13}
Operating Pressure Range, psig	0-600

Table I (cont)

2. Steam System

Operating Pressure Range, psig	100-600
Maximum Steam Temperature, Saturated, °F	489
Maximum Steam Temperature, Superheated, °F	850
Turbogenerator System Capacity at 350 psi and 4 in. Hg, Absolute	
Input Power, Thermal, Mw	20
Steam Flow, Saturated, lb/hr	60,000
Atmospheric Vent Capacity	
Power, Thermal, Mw	20
Steam Flow, lb/hr	60,000
Turbogenerator-condenser Plus Atmospheric Vent Capacity	
Power, Thermal, Mw	40
Steam Flow, lb/hr	120,000
Electrical Generating Capacity, Mw	3.5

3. Feedwater System

Number of Feed Pumps	2
Pump Capacity, each, gpm	150
Pump Head, psig	700
Maximum Feedwater Flow, gpm	300
Feedwater Temperature, nominal, °F	100

4. Forced-convection System

Operating Pressure, psig	600
Water Temperature, °F	489
Pump Capacity, at 14 ft of Water NPSH, gpm	10,000
Pump Head, ft of Water	193
Material	Type 304 SS

5. Reactor Ion-exchange System

Maximum Temperature at Column Inlet, °F	120
Maximum System Pressure, psig	60
Maximum Flow Rate, gpm	20

6. Makeup Water Demineralizer System

Capacity, gpm	70

III. DESCRIPTION OF BORAX V FACILITY

A. Reactor Facility

1. General Features of Reactor

The BORAX V reactor system is designed with the objective of obtaining an extremely flexible experimental facility at a minimum cost.

The reactor vessel is also designed for experimental flexibility. The nine control rod drives are mounted on nozzles in the bottom head in order to leave the top of the vessel free for easy access and changing of experiments and fuel. The four forced-convection inlet nozzles in the bottom head and two outlet nozzles in the vessel shell are so located that any core can be tested with either natural or forced convection by removing or installing the portable forced-convection baffle inside the vessel. A spare water-preheat line, the main saturated-steam outlet, four experimental steam outlets also used for removal of superheated steam, two instrumentation outlets, water-level taps, poison-injection nozzle and feedwater inlet are also located in the vessel shell. The vessel top head has five nozzles for fuel-handling operations on boiling fuel assemblies, three instrumentation nozzles, and three oscillator-rod-drive nozzles.

Removable feedwater sparger rings are located either at the top of the boiling core shroud or at the top of chimneys 5 ft above the core. Feedwater may also be injected into the forced-convection system.

The integral nuclear superheating approach consisting of a two-zone boiling and superheating core has been selected for testing in BORAX V, rather than the two-reactor approach, because it is felt that in small reactors the integral superheater shows the most economic promise. However, provisions have been made in the design of the reactor shield for a second set of instrumentation holes and shield-cooling coils adjacent to the top of the reactor vessel for possible future tests of the nuclear superheating concept which embodies two cores within a single reactor vessel. In this concept a conventional boiling core controlled by drives through the bottom of the reactor vessel would furnish saturated steam to a superheating core in the upper part of the vessel, which is in turn controlled by drives mounted on the top head.

Three separate core structures will be provided: one for a pure boiling core, one for a boiling core with an integral central superheater zone, and one for a boiling core with an integral peripheral superheat zone. These core structures will be readily replaceable after removal of fuel and control rods. No bolting is required to hold a core structure in the reactor vessel. The structure rests on pads on the bottom head of the reactor vessel and is held down by gussets inside the removable top head of the vessel. The core structures will be withdrawn from the vessel as a unit by means of the building crane and stored in the dry-storage pit. The crane has a remote control station which can be shielded during these operations.

Boiling fuel assemblies are interchangeable between all three core structures. In the boiling core the number of fuel assemblies may be varied to change core diameter. For high-velocity, forced-convection tests a minimum number of assemblies may be used. The number of fuel rods per boiling fuel assembly may be varied and boiling fuel rods of three different enrichments are available. Longer cores may be tested by inserting longer fuel assemblies and control rods.

Provisions have been made in the steam system design for handling the exit steam from two in-pile loops which may be used for testing alternate or advanced superheat fuel assemblies with independent cooling systems. The in-pile loops may be used with either superheating or boiling cores.

Each of the boiling and superheat cores has 5 cruciform and 4 "T"-shaped control rods. These rods are made of Boral and have an aluminum follower. Partial independent control of superheater zone power is achieved with the control rods located in the superheat zone.

Four major reasons governed the choice of the 600-psig operating pressure:

1. Operation of 600 psig is considered adequate for all the experiments anticipated.

2. With equipment designed for higher pressures, experimental flexibility is inhibited by the increased difficulty of rapid fuel and experimental changes.

3. At pressures higher than 600 psig, the costs of pressure vessels, piping, and other hardware increase rapidly.

4. Six hundred-psig saturated-water conditions are considered the upper limit of use for X-8001 aluminum, which is proposed for some core components and possible future boiling-fuel cladding material.

2. Boiling Core Design - Mechanical

a. Boiling Fuel Assembly

The boiling fuel assembly, as shown in Fig. 3 is designed to permit a variable loading of from 0 to 100 removable rods. The boiling

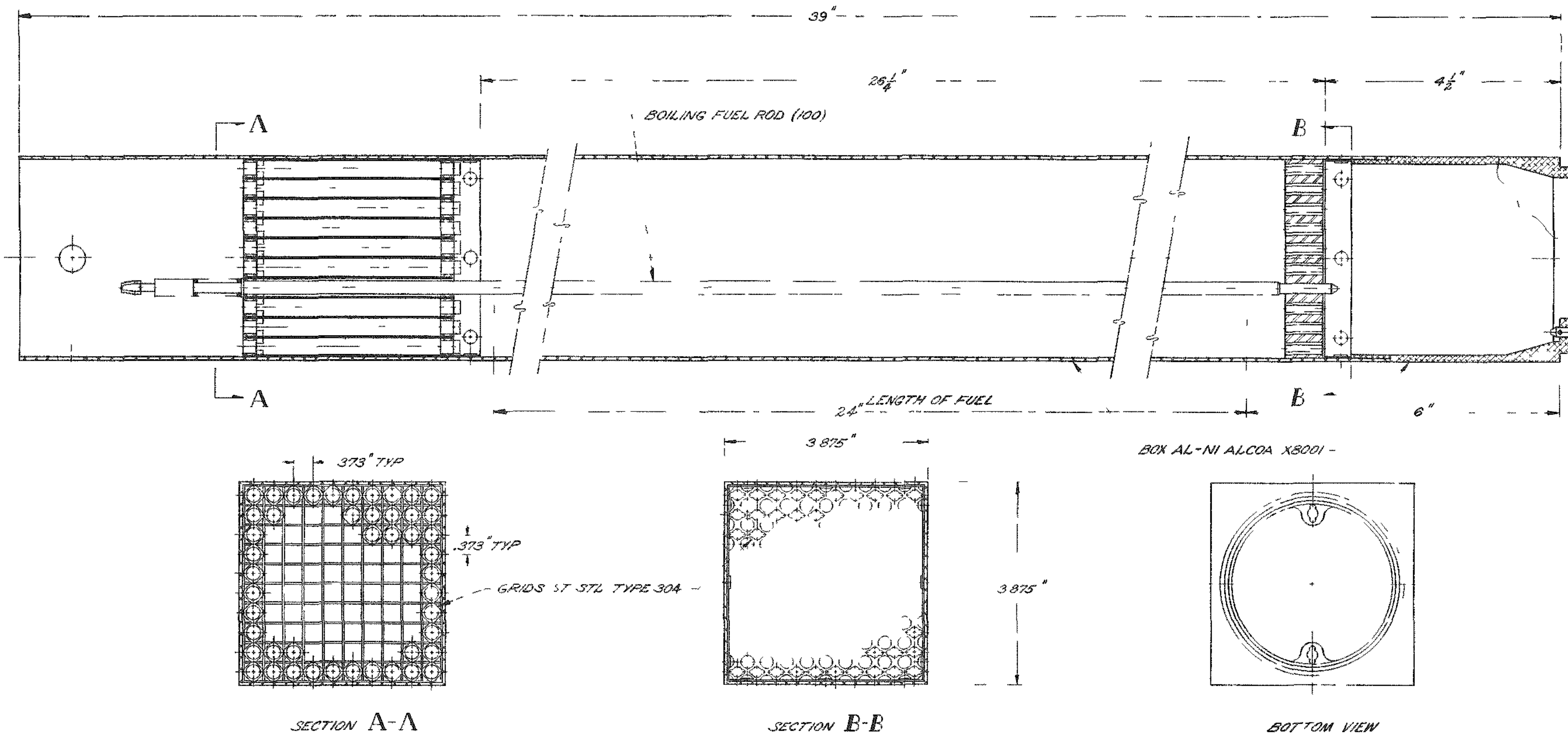


Fig. 3

Boiling Fuel Assembly - BORAX V

fuel assembly box is made up of a bottom cylindrical-to-square-transition fitting and a 3.875-in.-square tube, made of Al X-8001. Inside the aluminum tube are riveted an upper grid, 4 in. deep, and a lower grid, $\frac{3}{4}$ in. deep, of Type 304 stainless steel. A fuel rod inserted through a hole in the upper grid is guided through the corresponding hole in the lower grid. The fuel rod is then pushed down to compress the rod spring and turned 90° to latch it in place and hold the rod in tension between the two grids.

A replaceable, experimental orifice plate is provided in the lower end-fitting of the boiling fuel assembly. Holes are located in the upper end of the square tube above the fuel rods to receive the boiling-fuel-assembly grappler. Burnable poison may be installed in the form of strips spot-welded to the side of the assembly, or by means of replaceable boron rods.

b. Boiling Fuel Rod

The boiling fuel rod, as shown in Fig. 4, contains partially enriched UO_2, 24 in. long, of about 91% theoretical density, in the form of cylindrical rods or pellets of 0.230-in. diameter. The UO_2 is clad by a tube of Type 304 stainless steel with a 0.015-in. wall thickness. The stainless steel tube is either drawn or swaged down on the UO_2 to give a minimum clearance, and then the stainless steel end-fittings are welded on in a helium atmosphere. The lower end-fitting contains a latching pin. A fission gas-expansion space, which contains a tubular spacer to prevent the collapse of the cladding wall under reactor pressure, is provided between the top of the UO_2 and the upper end-fitting. On the upper end-fitting is the compression spring and washer, and a special head for receiving the fuel rod manipulator.

c. Core Structure - Boiling

Figure 5 shows a plan view of the boiling core structure. The core structure provides for a maximum of 60 fuel assemblies, grouped in cells of 4 or 3 assemblies by the control-rod-guide shroud. The shroud is made of $\frac{5}{32}$-in.-thick, X-8001 aluminum, 63 in. high, and is joined by X-8001 aluminum vertical stanchions to form $\frac{1}{2}$ in.-wide channels for five 14 x 14-in. cruciform control rods and four 14 x 7-in. "T"-shaped control rods. Tubes are attached to the outside of the shroud to hold the antimony-beryllium neutron source, a rotating oscillating rod, and startup counters. The shroud is bolted to, and the fuel assemblies rest on, a $4\frac{1}{4}$-in.-thick core support plate of Type 304 stainless steel.

Figure 6 shows an elevation of the boiling core structure. The core-support plate has 8 stainless steel, tubular legs which are guided by tapered dowels and supported by pads on the bottom head of the reactor vessel. To these legs is fastened a diffuser ring which is designed to distribute the water from the 4 inlet nozzles under forced-convection operation.

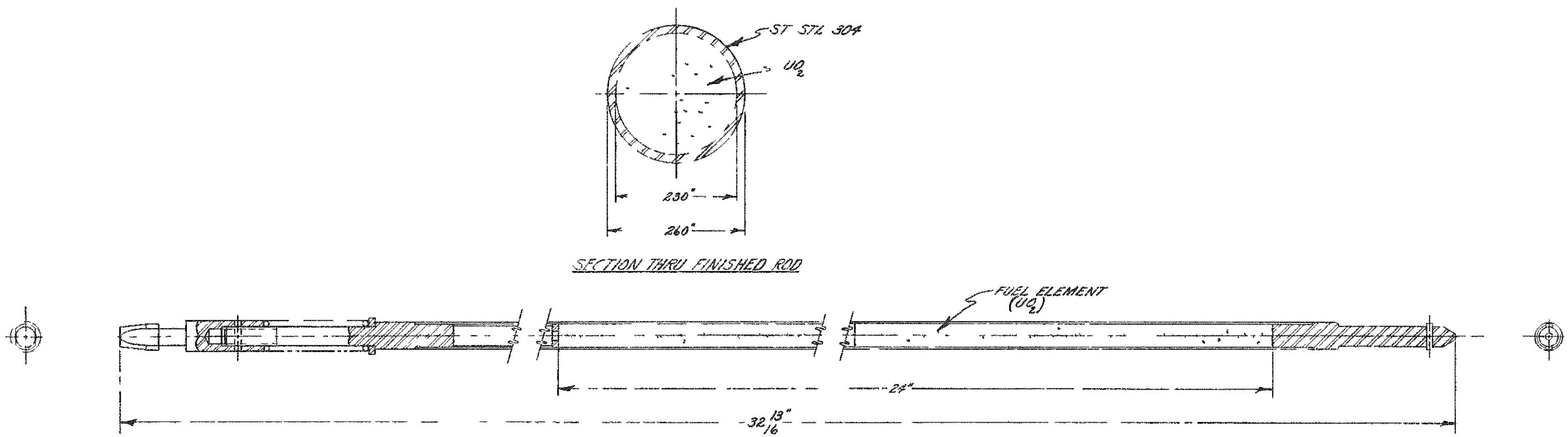


Fig. 4

Boiling Fuel Rod - BORAX V

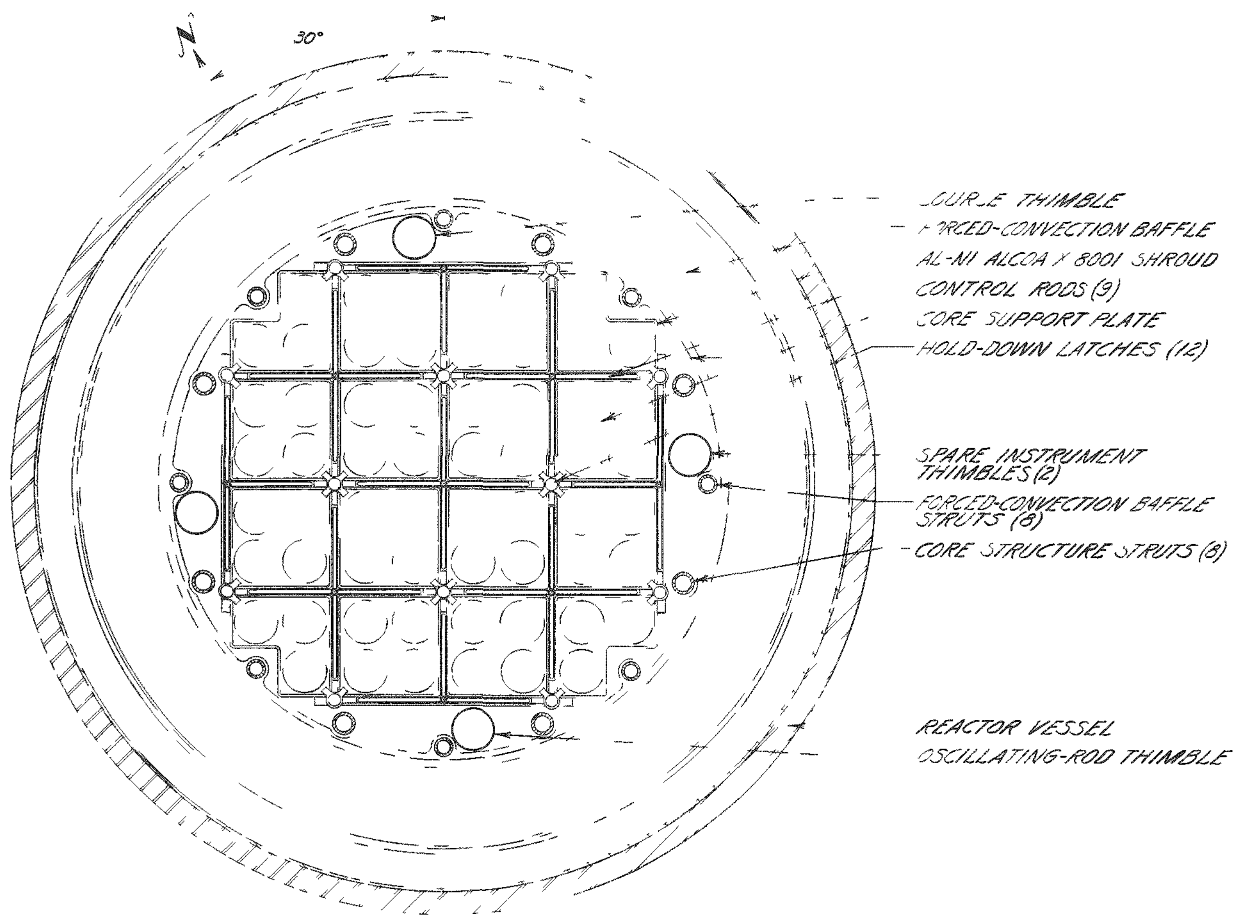


Fig. 5

Section A-A

Boiling Core Structure - BORAX V

The core support plate and aluminum shroud structure is held down against a maximum upward hydraulic load of about 28,000 lb by 8 stainless steel, tubular struts terminating at the top in a stainless steel ring flange. The ring flange in turn is held down through a 54 $\frac{3}{4}$-in.-diameter Belleville spring by 6 gussets welded to the upper head of the reactor vessel. The Belleville spring deflects to allow for variation in lengths, and differential expansion between the stainless steel core structure and the carbon steel reactor vessel. The top of the core structure is also centered by the hold-down gussets.

Each cell of 4 boiling fuel assemblies is held down and centered by a $7\frac{7}{8}$ x $7\frac{7}{8}$ -in.-square hold-down box, which in turn is held down by latches bolted to the top of the shroud. These latches may be operated from the top by a long-handled socket wrench. Additional riser height for natural circulation is provided by $7\frac{7}{8}$ -in.-square stainless steel chimneys

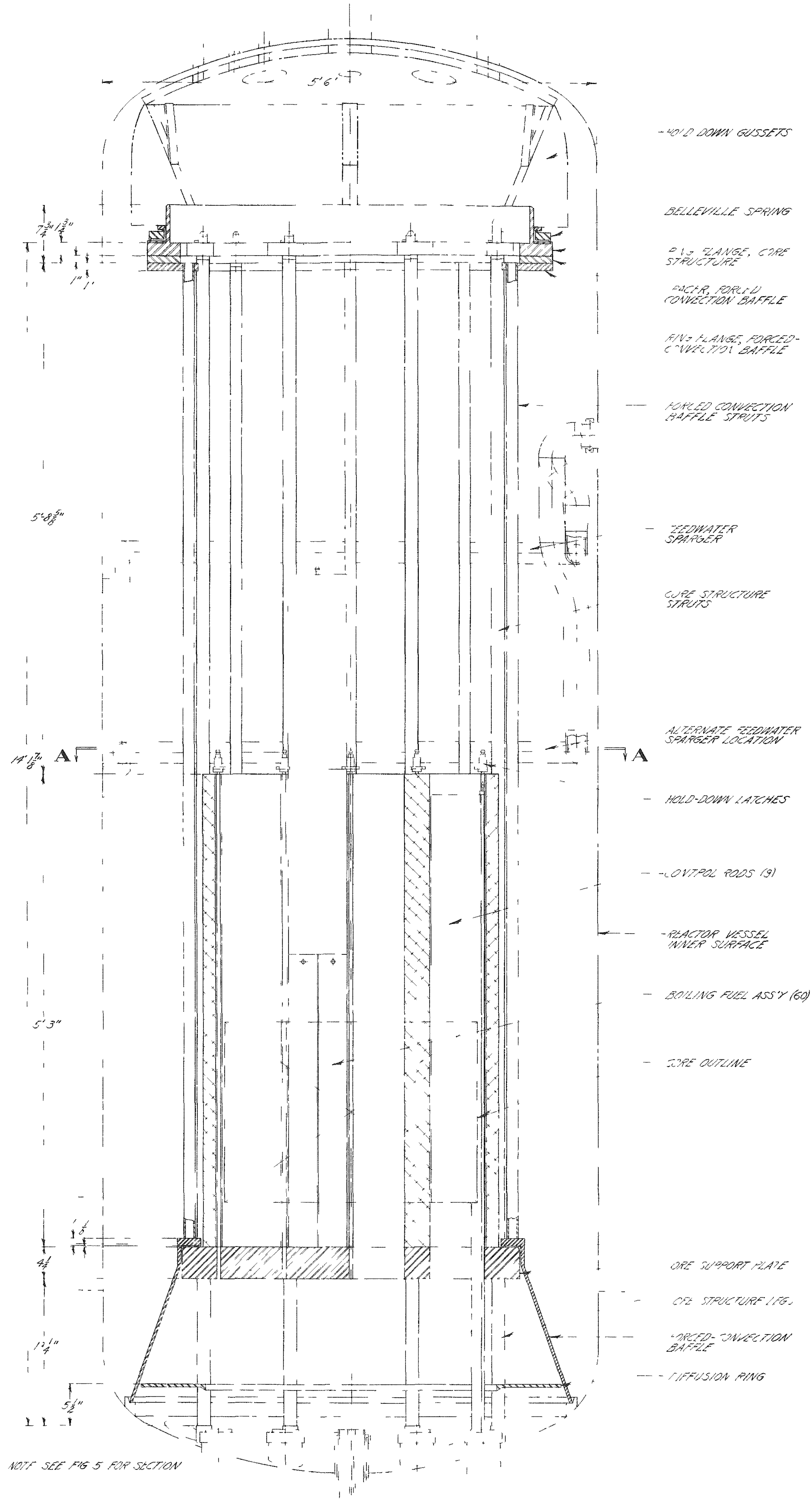


Fig. 6

Elevation Boiling Core Structure - BORAX V which pilot into the hold-down boxes and are held in place by their own weight. The cells containing three assemblies have hold-down boxes and chimneys shaped to fit inside the cell.

For forced-convection tests a conical, stainless steel baffle is installed after removing the reactor vessel upper head, the core structure ring flange with Belleville spring, and the feedwater sparger. The forced-convection baffle seats on a machined ring in the lower head of the reactor vessel and seals against the top chamfered edge of the core support plate by means of a tubular metal gasket. The baffle is clamped in place by the core structure ring flange through 8 stainless steel, tubular struts and a ring flange of its own. When installing or removing the forced-convection baffle, the lower core structure remains in place. Also, for forced-convection tests, orifice plates must be installed in the shroud at the top of each control-rod channel to regulate the amount of water bypassed for control-rod cooling. The hydraulic force across the forced-convection baffle will exert an additional 20,000-lb maximum load on the Belleville spring, making a total possible load of about 48,000 lb.

A special, low-headroom lifting fixture which engages lifting pads on the top ring flange of the core structure permits handling of the core structure as a unit by means of the building crane.

d. Experimental Components

In addition to boiling fuel assemblies and fuel rods, the following core components are used for certain experiments:

(1) Dummy fuel assemblies made of X-8001 aluminum are used to plug off unused fuel-assembly holes in the core support plate when testing small cores with forced convection.

(2) Void rods, made of empty, water-tight X-8001 aluminum tubing of the same outside dimensions as a normal fuel rod, and water rods which are void rods perforated to allow water filling, are used to determine void coefficients. Water rods may also be used to replace fuel rods in partially loaded boiling fuel assemblies to prevent flow bypass.

(3) Boron rods composed of boron-stainless steel have the same dimensions as fuel rods and may be used to adjust power distribution, as burnable poison, or to adjust reactivity.

3. Superheat Core Design - Mechanical

a. Superheat Fuel Assembly

The reference design for the superheat fuel assembly is shown in Fig. 7. The Model 2 assembly consists of five flat-plate fuel elements, with light water moderator between and outside them, joined into double-wall, static-steam-insulated inlet and outlet plenums and tubing. Near the top of each assembly is a hold-down spring and flange and near the bottom is a seal flange.

The flat-plate fuel element, Model 2, is a rigid, Nicro-brazed assembly of 5 fuel plates, 2 grooved side plates, and spacers. The fuel element is enclosed by a rectangular, Type 304 stainless steel tube which forms a static-steam-insulating annulus to reduce loss of heat from the superheated steam to the moderator. It is seal-welded to the insulating tube at the top only to allow downward expansion. A Model-2 fuel plate is 0.030 in. thick and has a sandwich construction with a 0.010-in.-thick meat of highly enriched UO_2 dispersed in a stainless steel matrix and a 0.010-in.-thick cladding of Type 304 stainless steel. Spacing between plates is 0.045 in. The two outside fuel plates are loaded only half as much as the inner three, because they are cooled on one side only.

The design of the reference superheat fuel assembly features good radiation-cooling under shutdown conditions and small flux depression across the fuel element. Fuel-plate spacing is maintained by $\frac{1}{8}$-in.-diameter spacers which are so placed as to minimize deflections caused by thermal stresses, hydraulic loads, and the two-pass, pressure-drop load transmitted through the insulating tube. Burnable poison strips may be spot-welded to the outside of the insulating tubes.

An alternate design of a water-moderated, flat-plate superheat fuel assembly, Model 1, is shown in Fig. 8. It consists of four square elements, each containing 18 fuel plates, with static-steam-insulating tubes surrounded by water moderator. This design has approximately the same homogenized composition as Model 2. It might have a lower fabrication cost, but suffers from a greater flux depression across a fuel element and has poor radiation-cooling characteristics.

Other alternate superheating fuel-assembly designs which have been studied are:

(1) a 150-rod unmoderated assembly using UO_2 stainless steel-clad rods;

(2) a 150-rod solid-moderated assembly using uranium-zirconium hydride clad in stainless steel; and

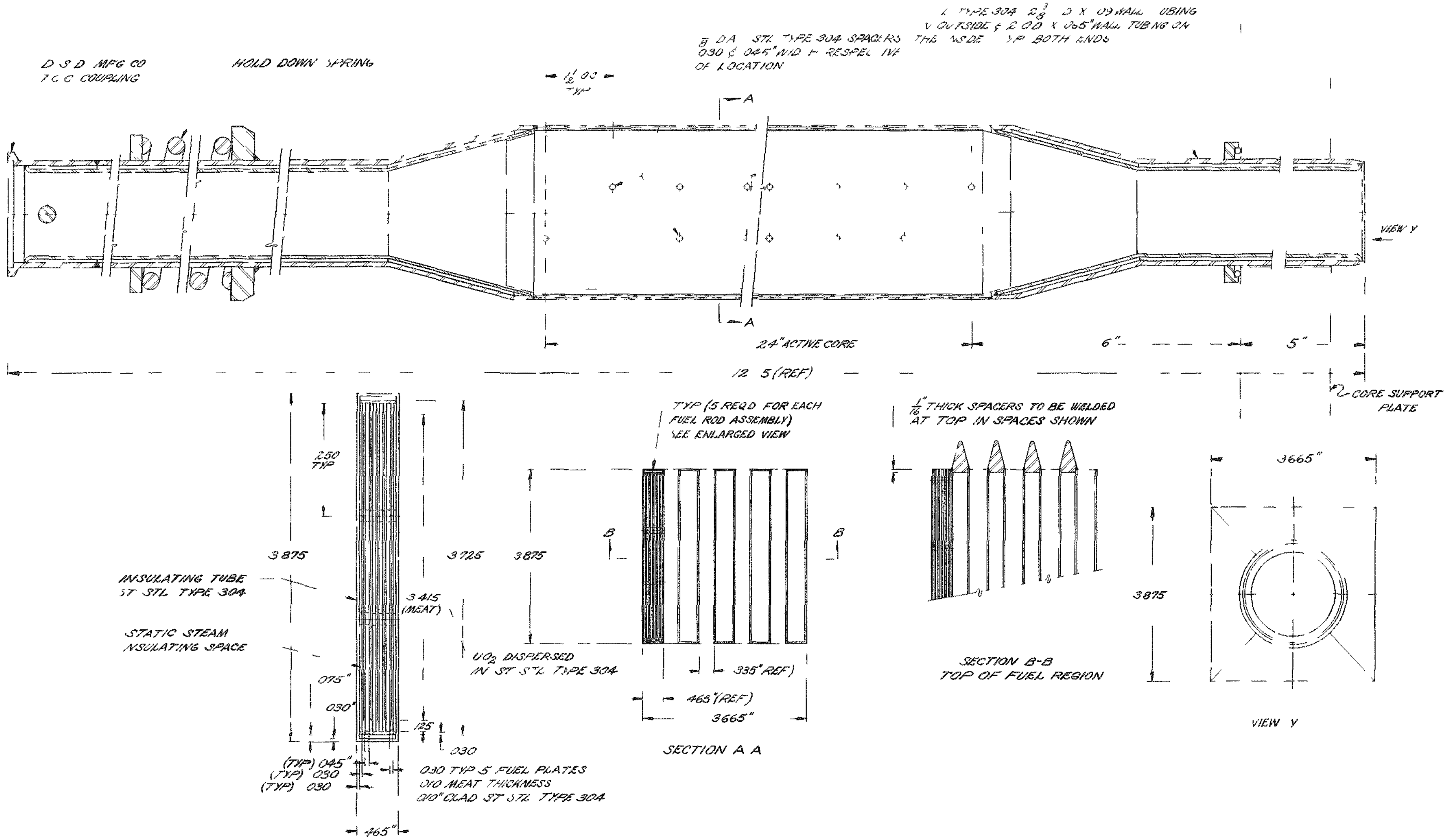


Fig. 7

Two-Pass Superheat Fuel Assembly-Model 2 - BORAX V

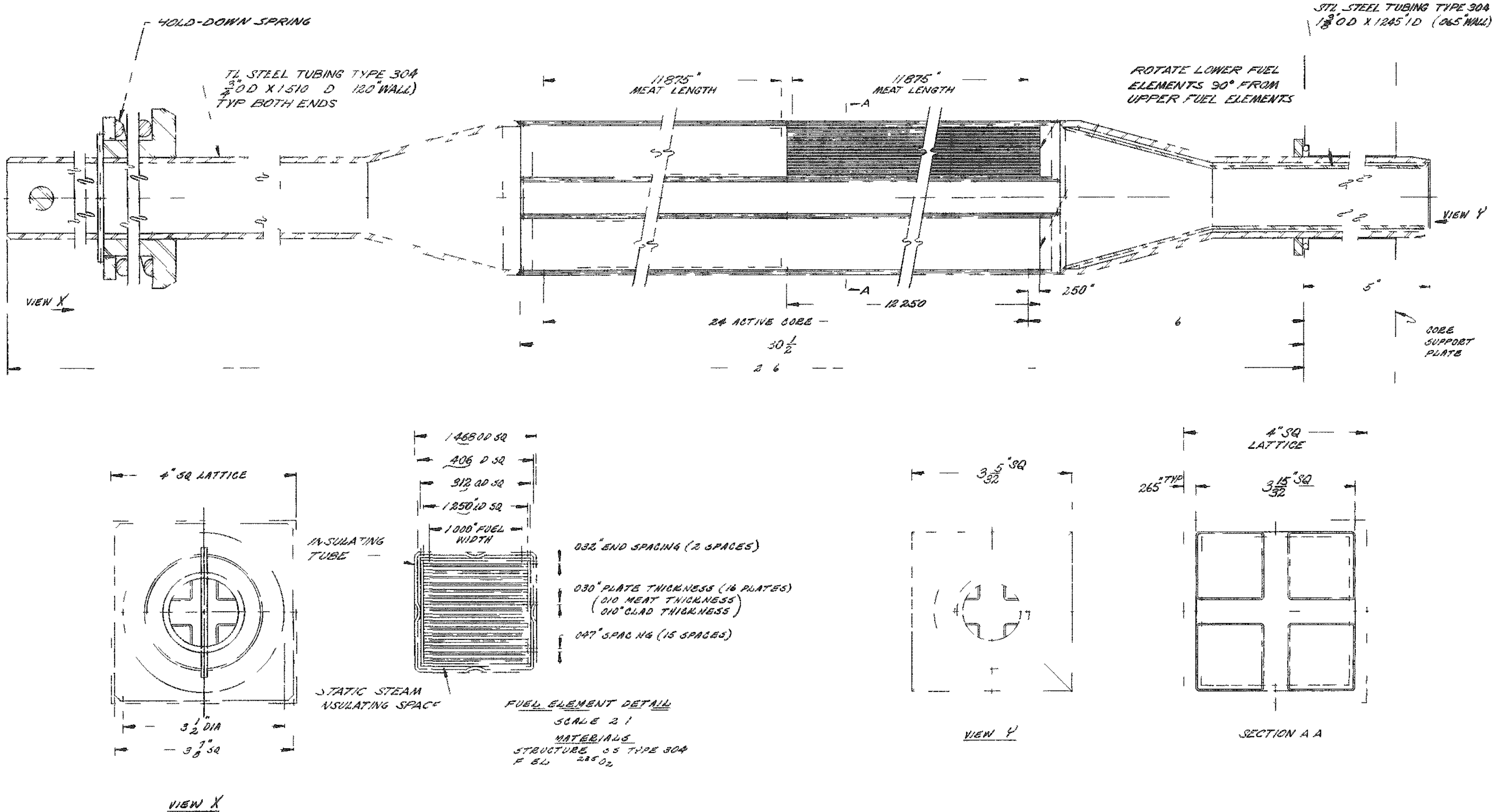


Fig. 8

Single-Pass Superheat Fuel Assembly-Model 1 - BORAX V

(3) various water-moderated designs using concentric fuel tubes made of UO_2-stainless steel cermet clad in stainless steel.

These designs were not used either because they had poor power-distribution characteristics or would require appreciable development time and effort not available in this program.

b. Core Structure - Central Superheat Zone

The core structure for the central superheat zone is shown in Figs. 2, 9, and 10. Provision is made for locating 12 superheat fuel assemblies in the center of the core surrounding the central control rod. The remainder of the fuel-assembly locations are filled with boiling fuel assemblies.

A superheat core structure has the same basic design as the structure for the boiling core. It has a similar, though shorter, X-8001 aluminum shroud with boiling fuel assembly hold-down latches, a stainless steel core-support plate, a similar leg and strut arrangement, and is similarly held down by means of the reactor vessel top head and Belleville spring. However, the core-support plates for superheat cores have additional holes drilled in them around the superheat fuel assembly locations for circulation of moderator water through these assemblies. These holes are sized for natural circulation. When forced convection is used, orifice bushings retained by the bottom sealing-flange on the superheat fuel assembly are installed to prevent bypassing an excessive amount of water. The same removable forced-convection baffle may be used on any of the boiling or superheat core structures.

There are two parallel steam-flow patterns, in each of which the saturated steam from the reactor vessel steam dome enters the top of the superheat fuel assembly. On the first pass steam flows down through three fuel assemblies, through the core-support plate into a double-walled static-steam-insulated plenum chamber and a U-pipe arrangement which passes beneath the center control rod follower. On the second pass steam flows up through three fuel assemblies and risers, which are coupled with temperature-compensating couplings, into a manifold system made of flexible, stainless steel tubing and double-wall pipe, and thence through the steam outlets located below the reactor vessel main flange.

Welded to the bottom of the two U-pipes beneath the core-support plate is a superheater drain and flood manifold system which leaves the reactor vessel via piping through one of the forced-convection system inlet nozzles. The 2-in. downcomer pipe for this system has a coupling which must be disconnected during a core structure-removal operation because of the limited crane lift height above the reactor building floor slab.

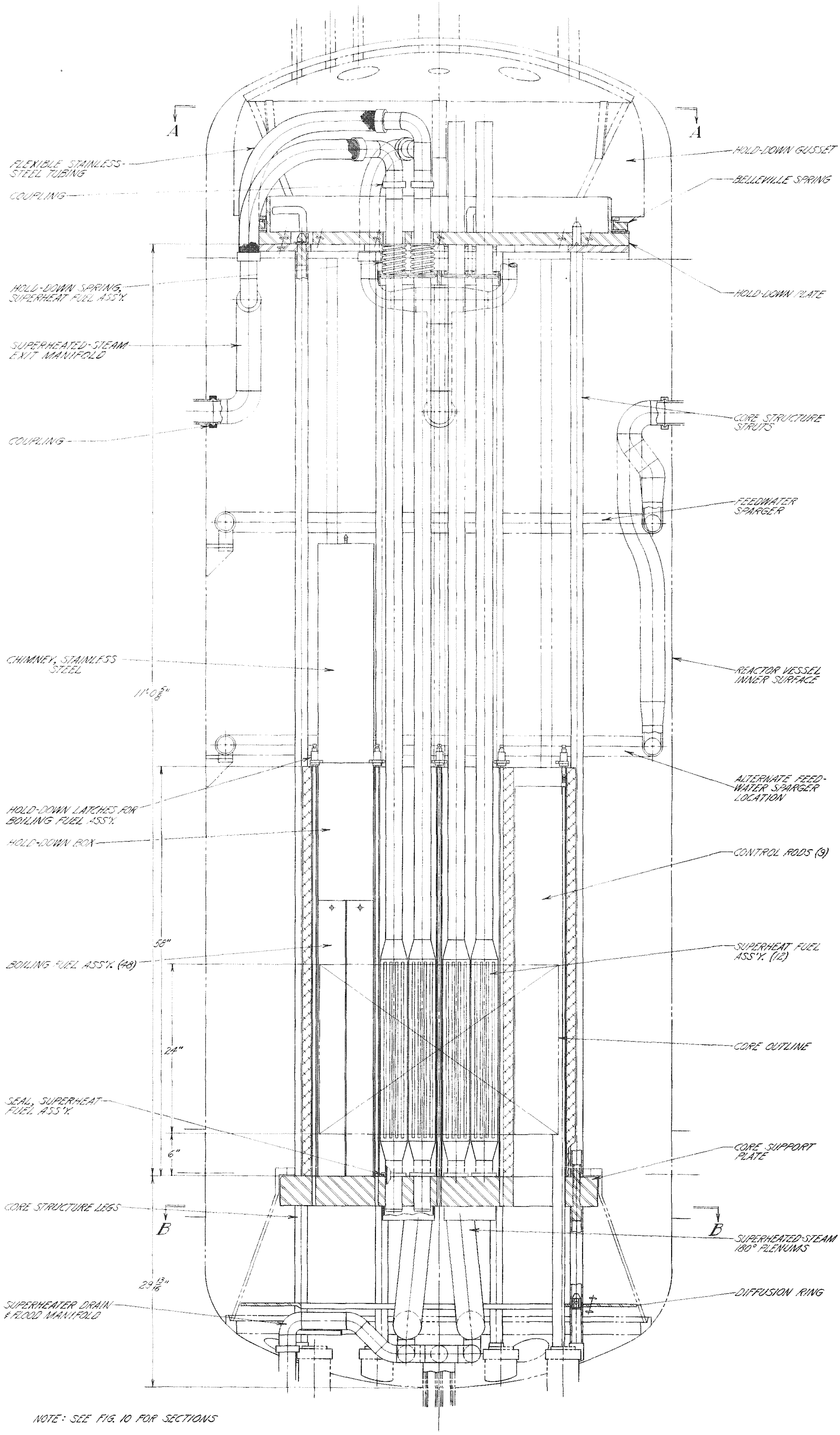
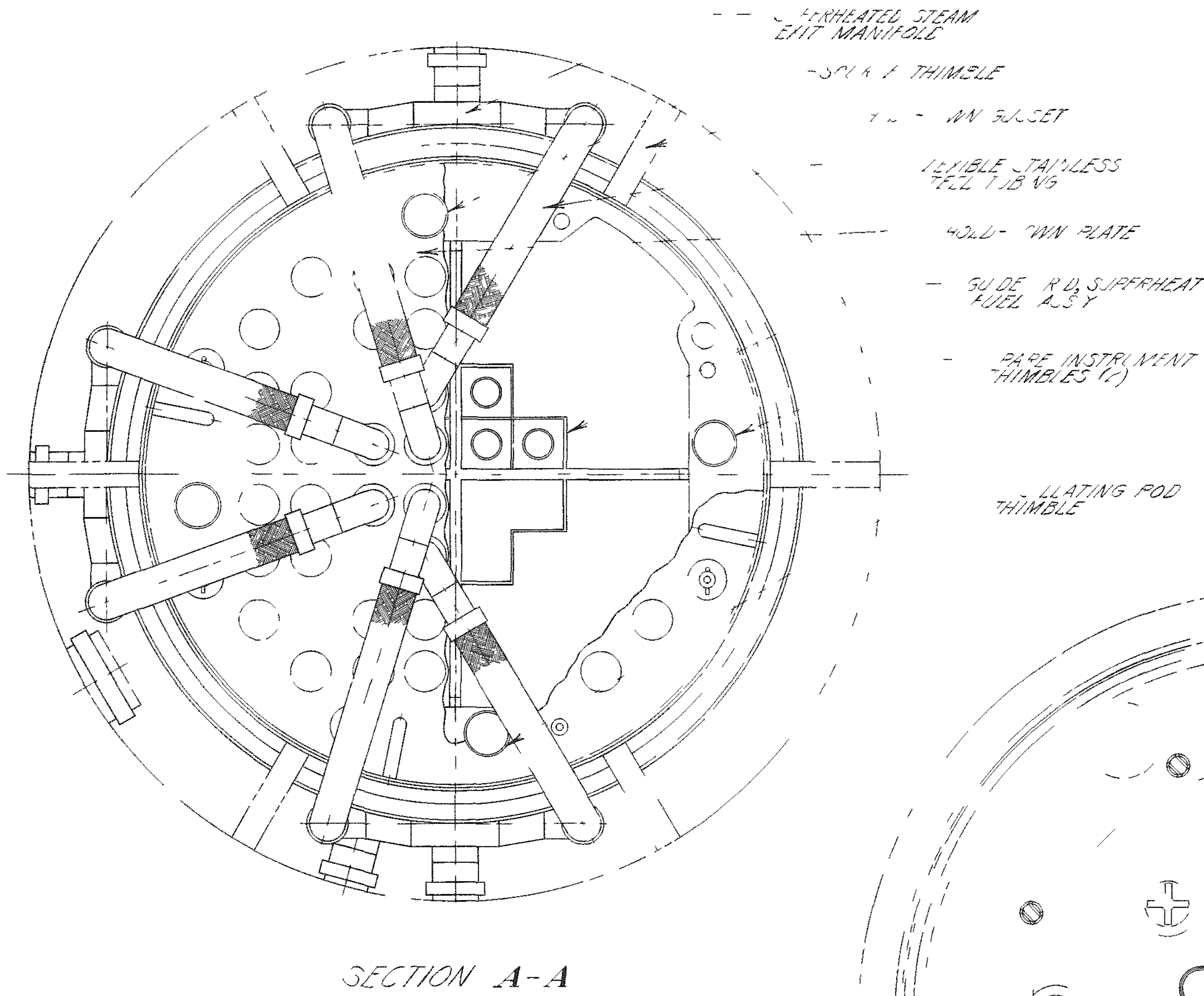


Fig. 9

Elevation Central Superheater Core Structure - BORAX V



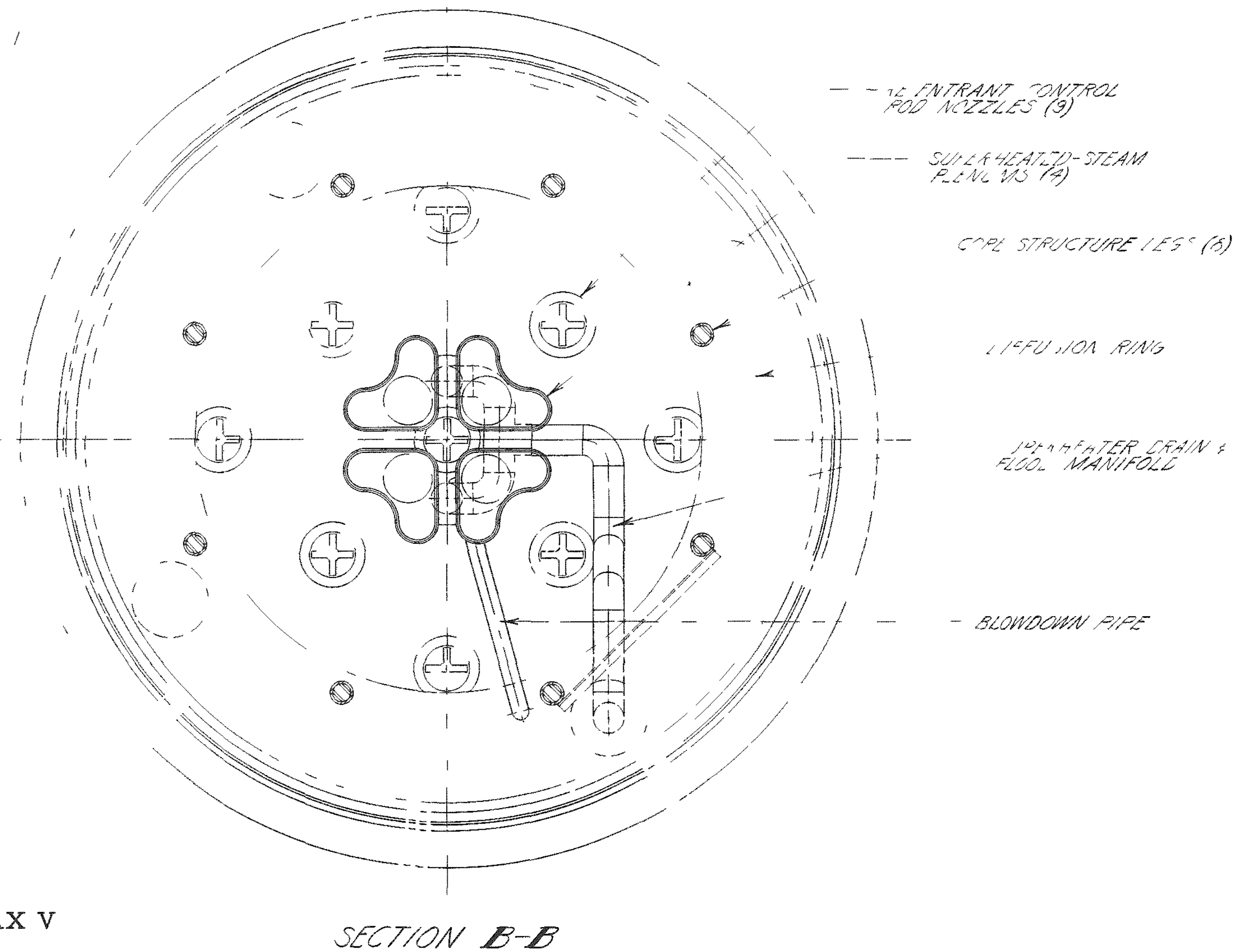


Fig. 10

Sections - Central Superheater Core Structure - BORAX V

A guide grid bolted to the top ring flange on the superheat core structure locates the upper end of the superheating fuel assemblies until the hold-down plate, which rests on the superheat fuel-assembly hold-down springs, is placed over them. This plate is held down through the Belleville spring by the reactor vessel top head. This plate also compresses the superheater hold-down springs which apply the proper gasket load to the seal between the fuel assembly and the core-support plate. The hold-down spring and flexible hose in the exit-steam manifold permit differential expansion between the fuel assembly and the core structure.

c. Core Structure - Pheripheral Superheat Zone

The core structure for the peripheral superheat zone, as shown in Figs. 11 and 12, is similar in design to that of the central superheat zone. The sixteen superheat fuel assemblies are located in the outer core edge, two on each side of the radial blade of a "T"-shaped control rod. The remainder of the fuel assembly locations are filled with boiling fuel assemblies. There are eight parallel steam-flow paths, each made up of a pair of adjacent superheat fuel assemblies. Each pair has a double-walled connecting plenum below the core-support plate. Each of these eight plenums has welded to it a superheat drain and fill line leading to a circular manifold which is connected to a single outlet pipe from the reactor vessel. Flexible metal tubing and double-wall-pipe manifolds connect the eight second-pass fuel assembly risers to the four experimental steam nozzles in the reactor vessel shell.

The superheat fuel assemblies in the peripheral superheater are held in place in the same manner as in the central superheater.

d. Seals - Superheat Fuel Assembly

In order to limit leakage of reactor water into the steam passages of the superheat fuel assemblies, an efficient mechanical seal is needed. This seal is located between the sealing flange on the superheat fuel assembly and the core-support plate. A seal test program is being carried out to test standard, commercially available, metallic seals. The choice of seals was restricted to this type because the low compressive forces required allow use of reasonably sized superheat fuel assembly hold-down springs.

The seals were tested in a vessel at 600 psig and 489°F by means of a mocked-up superheat fuel assembly with a hold-down and alignment arrangement which simulates the reference design.

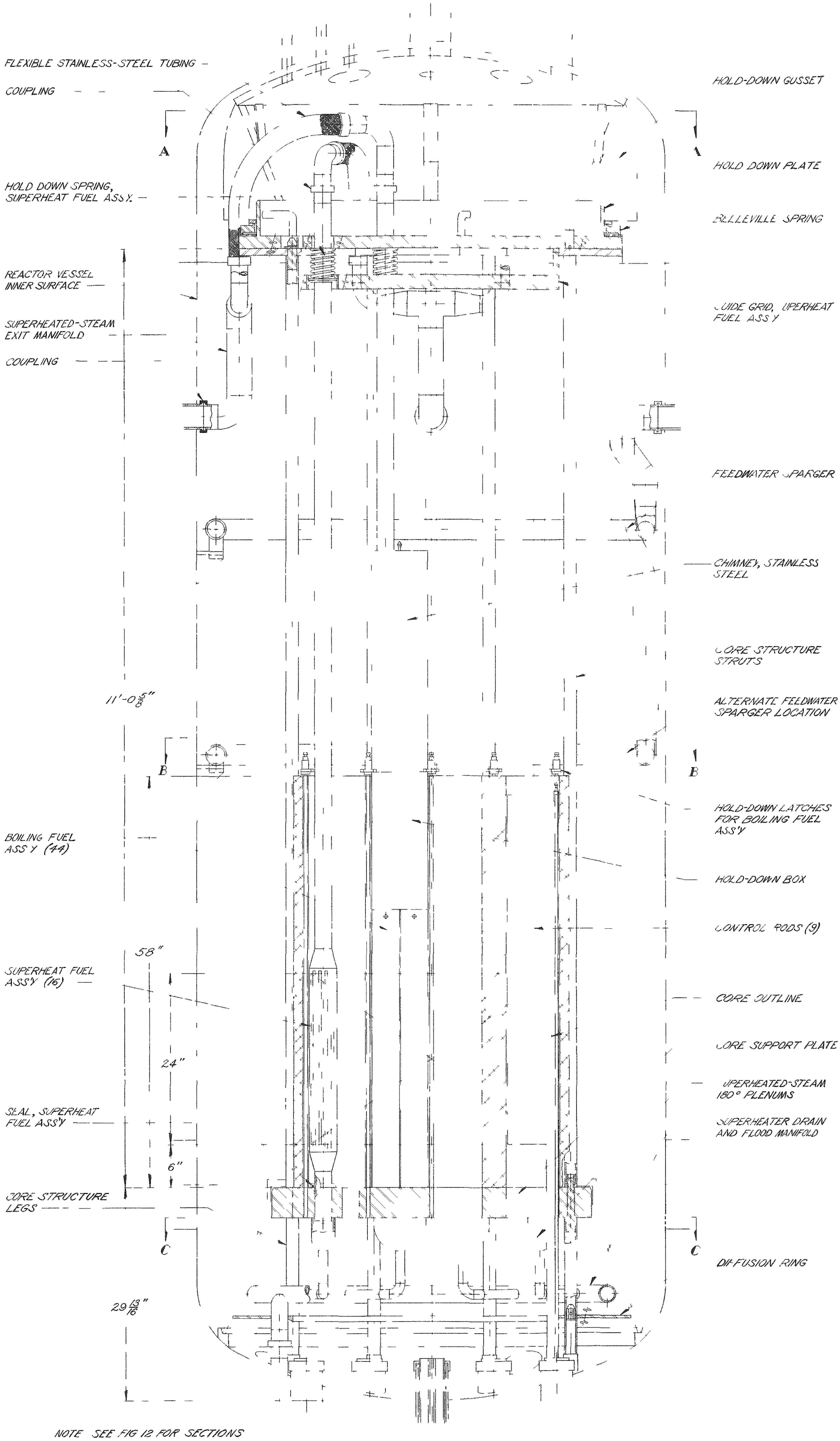


Fig. 11

Elevation Peripheral Superheater Core Structure - BORAX V

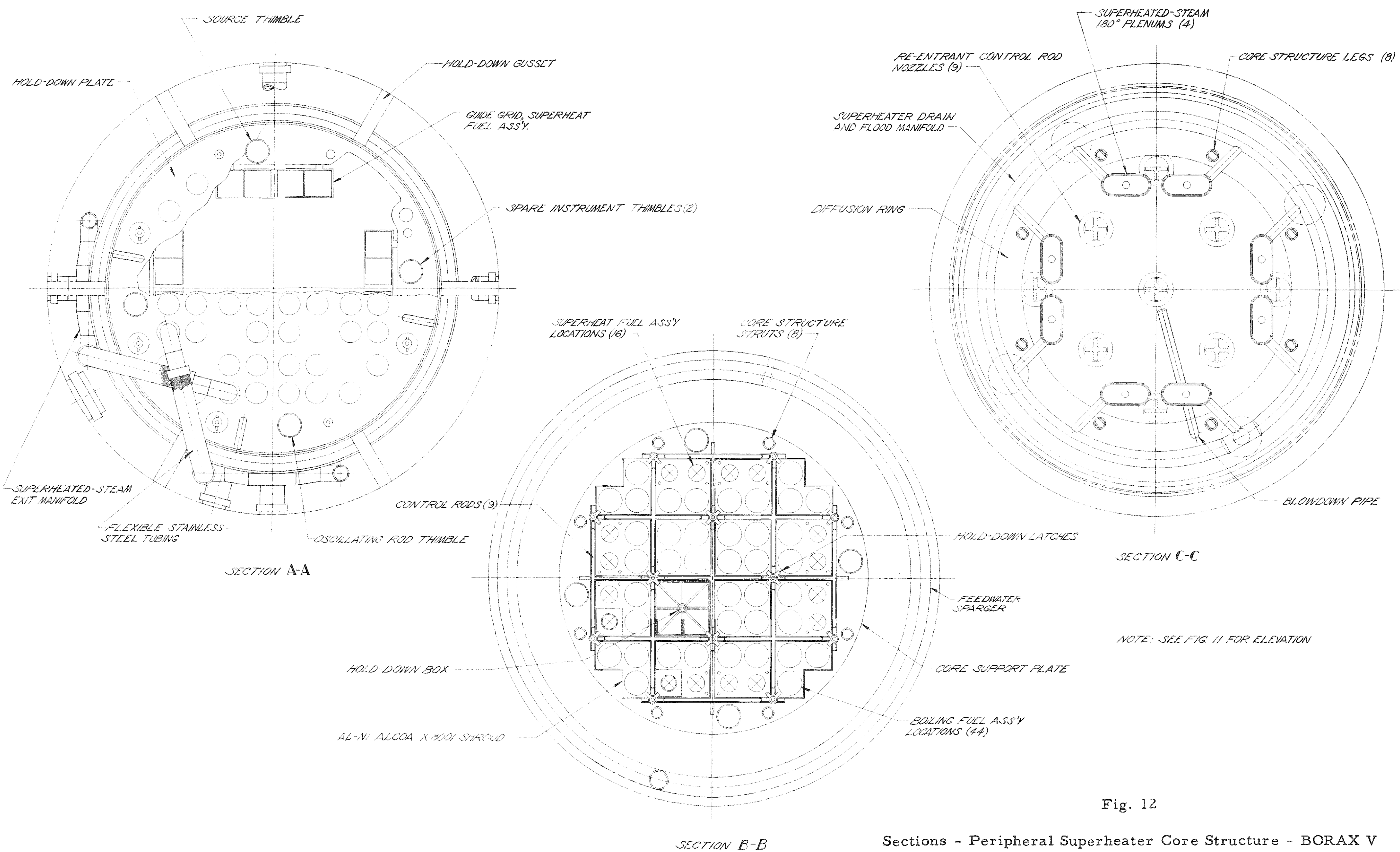


Fig. 12

Sections - Peripheral Superheater Core Structure - BORAX V

Three types of commercial seals have been tested. These are:

1. Metallic Static Seal, manufactured by Cadillac Gage Co.;
2. Skinner Static Spring Seal, manufactured by the Hydrodyne Corporation; and
3. Hollow Metal O-Ring manufactured by Advanced Products Co.

All of the seals tested were of 2.250-in. OD and all required clean seats with a ground finish for best performance. Results of short-term tests are summarized in Table II.

Table II

SEAL-TEST RESULTS

Seal Type	Material	Recommended Compressive Force, lb	Δp across Seal, psi	Max. Leakage Rate, gph
Cadillac	Inconel-X Silver-plated	140	50	0.0095
Skinner	Inconel-X Silver-plated	500	25	0.0024
Advanced	Type 321 SS Silver-plated	920	40	0.0072

As a means of evaluation, the leakage rates above may be compared with the following: If BORAX V were operating at 20 Mw with 16 superheat fuel assemblies in the core and a superheat steam exit temperature at 850°F, a leakage rate of 2.58 gal/hr/fuel assembly would be required to drop the steam temperature by 10°F.

Long-term tests are continuing on the two seal types which required the lowest compressive force.

4. Core Design - Heat and Hydraulics

From the standpoint of heat transfer and fluid flow, the design of the core for a boiling water reactor with an integral superheater is complicated by the thermal coupling which exists between the boiling and the superheating regions. For a given operating pressure and exit steam temperature, the fraction of the reactor power generated in the superheater remains virtually constant regardless of the total reactor power level. For BORAX V, operating at 600 psig with an exit steam temperature of 850°F, 83% of the power is used to turn the feedwater into saturated steam, and 17% is used to superheat the saturated steam from 489°F to 850°F. The problem is to design the two sections so that they operate properly as a unit while staying within the individual limitations of heat flux, surface temperature, and pressure drop.

The core size, as well as the design of the fuel rods and boiling fuel elements, are based almost entirely on mechanical considerations. The principal effort on the boiling region consists of determining the performance capability of the design. The situation for the superheater region is quite different, however, and the design of this fuel element is still in progress. The boiling and superheat fuel assemblies described herein are the present reference design, but are subject to change as work on the final design proceeds.

a. Experimental Work

ANL has investigated flow characteristics and critical heat flux in the boiling region, and is currently conducting tests to determine convective heat transfer coefficients for steam in both turbulent and laminar flow.

(1) Flow Characteristics

Experimental studies have been performed: a) to investigate the effect on flow oscillations of the orificing action of the boiling fuel assembly grid plates at the entrance and exit of the boiling fuel element: b) to establish the flow characteristics for a natural-circulation system that was hydrodynamically similar to the core and riser combination in the proposed reactor; and c) to obtain a feeling, in view of the lack of data, for the burnout heat flux at this pressure and under these conditions.

At the time of these experiments the proposed boiling assembly was to have a maximum of 100 removable fuel rods, 0.290 in. OD with an active fuel length of 24 in. The fuel rods were to be located in a 10 x 10 square lattice with 0.375 in. between centers. The simulated flow passages consisted of four stainless steel tubes, 0.332 in. ID by 24 in. long, with parallel flow discharging into a common riser that was 36 in. in length. The tubes had approximately the same equivalent diameter and flow area as the proposed 100-rod boiling element. The fuel assembly grid plates were simulated by orifices placed at the entrance and exit of the electrically heated tubes. It was felt that the 100-rod boiling element was the most restrictive with respect to flow; hence, simulation of this element should provide data on the lower limits of flow and instability. The primary difference between the experimental system and the reactor was that in the fuel assembly there is boiling on the outside of rods with interconnected flow passages, whereas in the experimental test section the boiling took place inside of tubes and there was no interconnection.

The results of these tests indicated that high power densities could be achieved with stable flow conditions in parallel, natural-circulation channels, hydrodynamically similar to the BORAX V flow channels, even with these channels restricted at both entrance and exit. Flow oscillations first became apparent at a power density of approximately 460 kw/l of coolant at 600-psig pressure. The corresponding heat flux was 306,000 Btu/hr-ft^2. A similar test was run at a lower pressure of 150 psig and no flow oscillations were detected at power densities up to 300 kw/l of coolant. The 150-psig test was discontinued at this point to prevent possible damage to the test section.

To establish the effect of orifices on the flow characteristics of this system, the orifices were drilled out and reamed to the diameter of the tubes. Three runs were made at high power, and the velocity in the tubes was found to increase approximately 10%.

Figure 13 is a plot of the experimental data, showing the tube inlet velocity as a function of power at 600 psig. It should be noted that in the experimental system the inlet subcooling for a given channel power density was considerably higher (by a factor of about 5) than it will be in the reactor. This will have the effect of shifting the velocity-power density curve to the left.

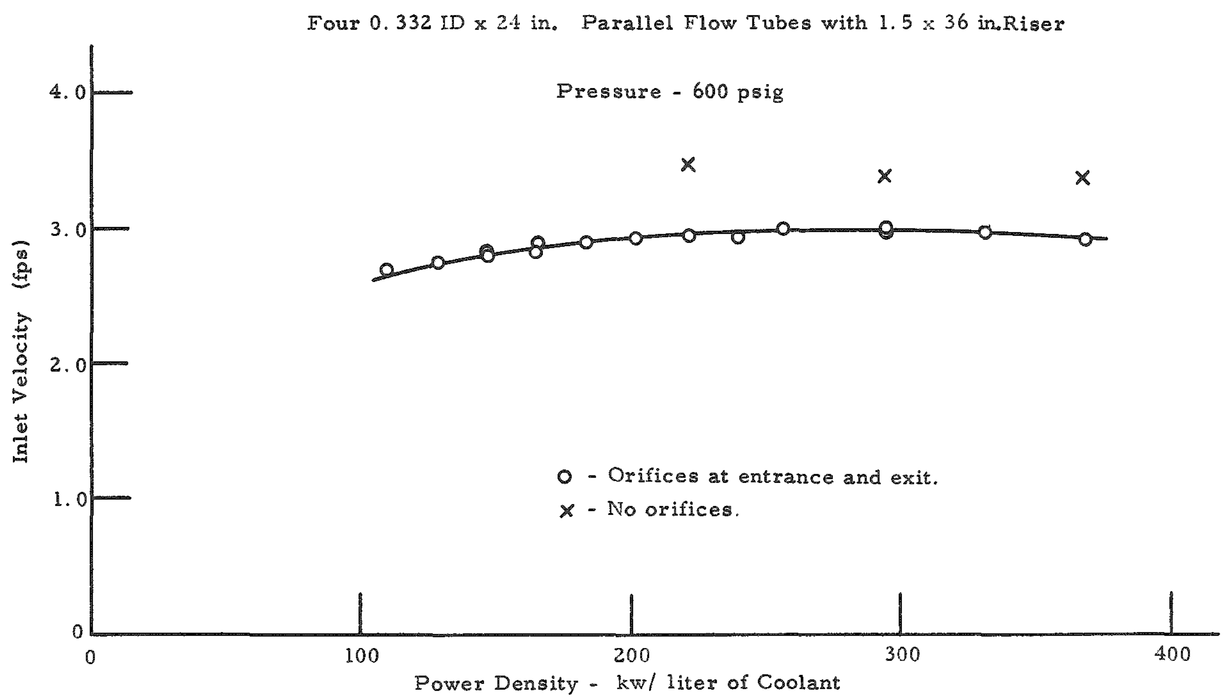


Fig. 13

Natural-Circulation Performance - BORAX V

(2) Burnout Tests

The only previously available data relating to burnout heat flux at 600 psi and applicable to BORAX V are some Argonne forced-circulation data for a round tube, 0.306 in. ID x $23\frac{1}{4}$ in. long. These dimensions are essentially the same as those of the electricically heated test section and the 100-fuel rod boiling assembly. Unfortunately, the data did not extend into the heat flux-quality range of interest and there are no data at this time for estimating the effect of changing equivalent diameters.

To obtain a feeling for the burnout limitations, the electrically heated test section used for the flow tests was instrumented with a burnout detector, and power was applied until burnout conditions were indicated. The orifices had been reinstalled at the entrance and exit of all the heated tubes for this test. Table III gives the pertinent data for the three burnout points obtained.

Table III

EXPERIMENTAL BURNOUT DATA

Burnout Heat Flux, Btu/hr-ft^{2}x10^{-6}	Power Density, kw/l	Coolant Velocity, fps	Inlet Coolant Subcooling, °F	Mass Flow Rate, lb/hr-ft^{2}x10^{-6}	Exit Quality	Exit Enthalpy, Btu/lb
0.349	521	2.47	59	0.445	0.22	634
0.363	543	2.52	62	0.453	0.22	633
0.356	533	2.67	61	0.480	0.19	615

There were indications of flow oscillations prior to all three burnout points. As mentioned previously, the oscillations started at a power density of about 460 kw/l of coolant. Extrapolation of the Argonne forced-circulation burnout data to the region of interest predicts a burnout heat flux of approximately 500,000 Btu/hr-ft^2.

The only conclusion that may be drawn from this burnout test is that burnout may be more of a problem than initially anticipated. Application of the absolute value of these experimental burnout points to the reactor is virtually impossible because these points are the outcome of only one combination of the pertinent variables. Furthermore, the reactor will differ from the experimental system in three respects:

(a) The experimental system had a uniform heat flux, where the reactor will have some axial power variation that will place the most probable burnout point in a position where there is neither a maximum heat flux nor a maximum bulk enthalpy.

(b) The inlet subcooling for the channels in the test was much higher than it will be in the reactor. Thus, for the same power input, the reactor channel will have a higher exit quality and bulk enthalpy.

(c) The flow channels in the test were not interconnected as they will be in the reactor.

How interconnected channels will influence burnout at 600 psi is not known. Westinghouse[1] has reported burnout data for parallel flow through rod bundles (square array, 9 rods, 0.300 in. OD x 10 in. long, 0.360 center-to-center at 2000 psi) and concluded that design equations for boiling in single channels may safely be applied to rod bundles.

(3) Superheated Steam Heat Transfer Experiments

Heat transfer tests are under way to determine convective heat transfer coefficients for steam in both laminar and turbulent flow. The correlation of steam heat transfer data is complicated by: a) the disagreement of various references on the value of certain steam properties, particularly of the viscosity and thermal conductivity; and b) the fact that steam, unlike most common gases, exhibits rapidly changing properties as it is heated above saturation temperature. The specific heat at 600 psia, for example, decreases rapidly as the steam is heated from 486°F, reaches a minimum at about 900°F, and then begins increasing slowly. Data on turbulent-flow steam heat transfer, particularly at high heat fluxes, are scarce, and various authors use different steam properties as a basis for their correlations. Laminar-flow steam heat transfer data are nonexistent, although important in analyzing reactor temperatures during shutdown and startup cooling. Heat transfer data from a prototype channel will reduce the uncertainty in the heat transfer correlation and allow a reduction in the hot-channel factor.

Heat transfer tests will be performed in a rectangular channel with dimensions similar to the proposed superheater channel. Local heat transfer coefficients will be measured so that the entrance effects (including the effect of rapidly varying steam properties) and the effect of the large aspect ratio may be analyzed. The range of variables to be covered are:

Pressure	Up to 600 psi
Heat Flux	Up to 500,000 Btu/hr-ft^2
Velocity	Up to 300 fps
Steam Temperature	Up to 1000°F

Initial data will be obtained from a round tube to obtain operating experience, check out instrumentation, and check some of the available correlations. The rectangular test section has been manufactured and is essentially ready for operation.

b. Analysis

A parametric study of the core heat transfer which corresponded to a similar study of the core physics was set up. Both studies provided for varying the number of fuel rods in a boiling element, the void fraction in the boiling region, and the dimensions of the superheater. Work on the parametric study is continuing.

(1) Boiling Region

The parametric study for the boiling region was organized to allow determination of performance capability before core power distributions became available. This study utilized RECHOP, an IBM 704 code, to calculate the hydrodynamic performance characteristics of a boiling water reactor core.(2) Several problems were run, with very good agreement between the experimental and analytical results, to check this code against the experimental work reported earlier.

Although most of the boiling region fuel element design, to date, has been based on purely mechanical considerations, the height of the riser above the fuel assembly and the number of rods per fuel assembly were not so fixed. A series of problems was run to determine the optimum riser height at which the increase in driving head, due to additional height, was offset by the increased frictional pressure drop. In all cases considered, as the riser height increased, the head gain exceeded the loss. The design height of 5 ft above the top of the core was established to leave sufficient space above the riser for steam separation.

Since nuclear considerations have a strong bearing on the number of fuel rods in a boiling assembly, this item was made a variable in the study. For assemblies composed of 50, 75, and 100 rods, calculations were made with varying values of inlet subcooling (4°F steps, from 2 to 30°F) and exit void fraction (5% steps, from 10 to 70%) for a total of 312 cases. The significant results for each given number of fuel rods and subcooling were plotted, and a typical plot (for a 100-rod assembly with 14°F subcooling) is shown in Fig. 14. Radial flux plots corresponding to a core with the proper number of rods in a boiling assembly can be used to integrate the total flow and steam flow across the core. For one subcooling (to be found by iteration) there will be an energy balance between the incoming feedwater, the reactor power, and the outgoing steam. Thus, the performance of the boiling region can be predicted. This method can be used whether a central or peripheral superheater is used, since the integration procedure will take this factor into account.

The experimental work on boiling burnout reported earlier indicated that the BORAX V power level would be limited by the boiling region rather than by the superheater. A maximum allowable

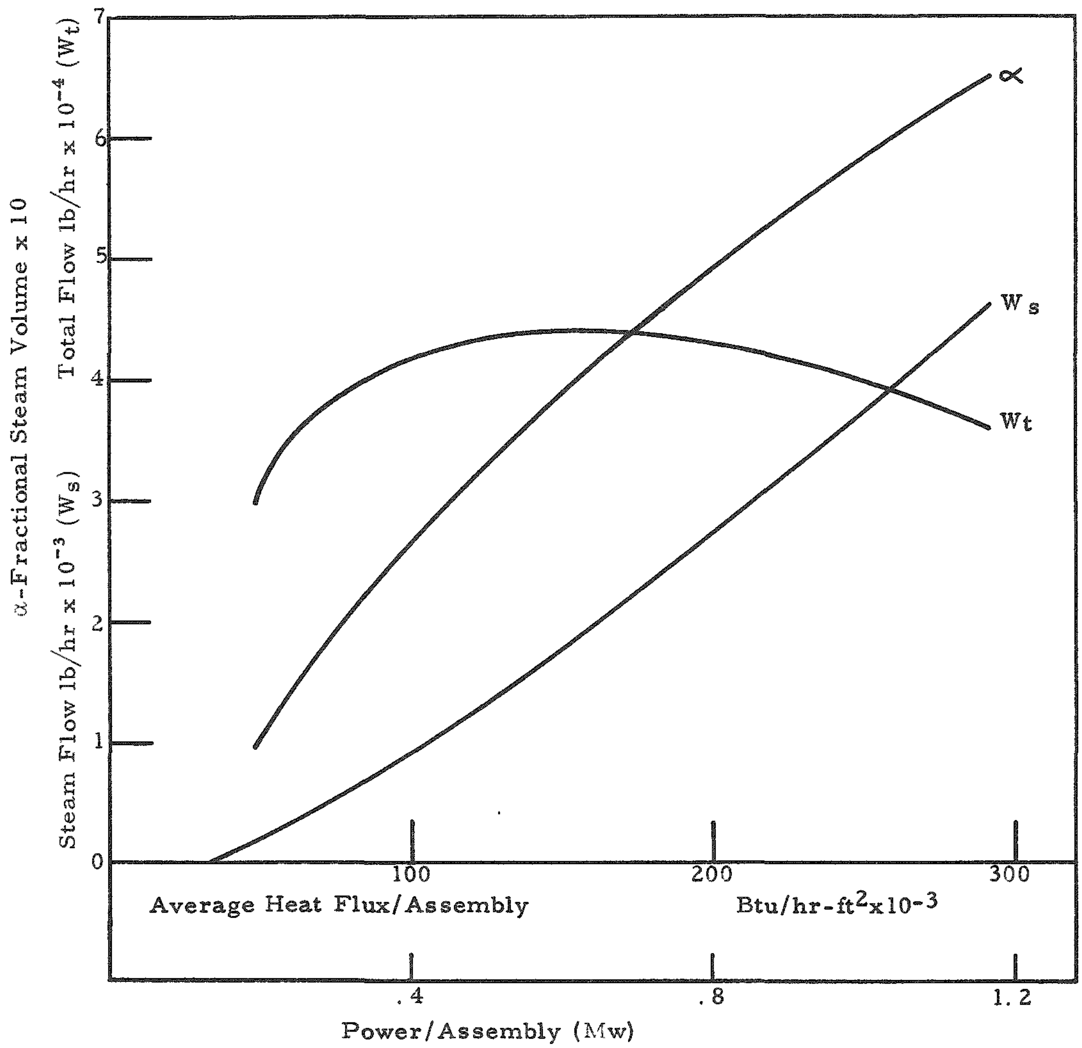


Fig. 14

Typical Hydrodynamic Parametric Study Results - BORAX V

heat flux of 300,000 Btu/hr-ft^2 was established. This value is believed to be safely below the measured burnout heat flux of 349,000 to 363,000 Btu/hr-ft^2, since the peak heat flux in the reactor due to axial neutron-flux distribution would be considerably upstream from the point where the fluid enthalpy was a maximum (i.e., at the core exit). As more experimental evidence becomes available, it is expected that the permissible heat flux will be increased.

The allowable power level for the boiling region was determined so there would be a logical basis for designing the superheater. This power level was found by using the 300,000-Btu/hr-ft^2 maximum heat flux and assuming a radial peak-to-average flux ratio of 1.3, an axial peak-to-average of 1.3, and a local flux-peaking ratio of 1.15, with a resultant average heat flux of 154,000 Btu/hr-ft^2. Based on forty-eight 100-rod boiling assemblies in the central superheater, the permissible power from the boiling region is 29.6 Mw (average power density of 169 kw/l of core volume). The total reactor power is 35.7 Mw, since the boiling region produces 83% of the total. The assumed 1.3 radial peak-to-average flux ratio may be too low but it was used to estimate the required superheater performance. If the assumed number is low, the result is a conservatively designed superheater and a corresponding reduction of the permissible power level in the boiling region.

(2) Superheater Region

The design of the superheater fuel element is based on the following requirements: a) exit bulk steam temperature of 850°F; and b) maximum limit of fuel element surface temperature is 1200°F.

In addition, it is considered desireable to have the superheater element produce as much power as the boiling element it replaced, and for the pressure drop through the superheater system to be a minimum, compatible with the requirements above. The element should also have the following characteristics: a low nuclear disadvantage factor; the ability to radiate enough heat so it will not melt in the event of a loss-of-coolant accident; good thermal stress properties; a minimum of structural material, good radiation-damage resistance, and relatively easy and inexpensive fabrication. The present reference design meets the two principal requirements above and also satisfies, to a resonable extent, the desirable features.

After investigating the feasibility of superheat fuel assemblies with no moderator and with solid moderators, it was decided to use an assembly with a water moderator between fuel elements. An assembly (Model 1) which received a considerable amount of attention is shown in Fig. 8. During this time, the conceptual design was based on a one-pass superheater, and the plates in this element were rotated 90° halfway through the core in order to promote mixing and thereby decrease the hot-channel enthalpy-rise factor. It was subsequently learned that the desired mixing would not occur.[3] The decision was then made to go to a two-pass core. Since the desired mixing effect was absent and the large number of fuel plates prevented any radiation cooling and imposed an excessive nuclear disadvantage factor, the Model-1 design was discarded.

The present reference design is shown in Fig. 7. With this type of element, limited radiation cooling is possible and the nuclear disadvantage factor is reduced to the order of 1.05, thereby decreasing the nuclear hot-channel factors. It is possible to use the same concept (i.e., an element made up of subassemblies of a series of fuel plates and coolant channels, separated by moderator) with the fuel contained in concentric tubes rather than in flat plates. A 4x4-in. assembly could thus, for example, be made up of 9 subassemblies about 1.2 in. OD, or of a series of concentric regions of fuel tubes and coolant channels separated by moderator regions. The heat transfer analysis for the flat-plate element is directly applicable to a concentric tubular geometry.

The reference design produces 35.7 Mw and is limited by the peak heat flux of 300,000 Btu/hr-ft^2 in the boiling region. Under these conditions, the reference superheater fuel element meets the design requirements, and the maximum surface temperature is 1100°F (rather than the 1200°F design point). The coolant temperature and the fuel element surface temperature as a function of the distance through each pass is shown in Fig. 15. This plot is based on the central superheater configuration, since the 12 fuel elements used therein present the most severe conditions of heat flux and pressure drop.

The superheated steam convective heat transfer film coefficient used to obtain these results is that of McAdams, et al.(4) and is given by:

$$h = 0.0214 \left(1 + 2.3 \frac{D}{L}\right) \frac{k}{D} \mathrm{Re}^{0.8} \mathrm{Pr}^{1/3}$$

where h is in Btu/hr-ft^2, D is equivalent diameter in feet, L is length in feet, k is thermal conductivity in Btu/hr-ft-°F, Re is the Reynolds number (dimensionless), and Pr is the Prandtl number (dimensionless) and all properties are evaluated at the film temperature.

The film coefficient found above was multiplied by 0.83 to account for the maximum deviation reported in the original work, and by 0.90 to account for use of flat plates. This last factor was used to assure conservatism of design, although there is considerable difference of opinion as to whether such a factor is necessary.(5) Additional assumptions in regard to flux distributions, engineering hot-channel factors, etc., are presented in Table I, which also gives information on the performance characteristics.

One feature of the reference design which is of particular interest is the capability of radiating heat to the moderator across the static-steam-insulating gap. As shown in Fig. 16, if the fuel element is emitting heat at 1% of the design rate (750 Btu/hr-ft^2), then the outside fuel plate must radiate 3000 Btu/hr-ft^2, and its temperature is found to be 980°F (the intersection of line A with the 490°F line representing the temperature of the stainless steel insulation plate). The next fuel plate must then radiate 2250 Btu/hr-ft^2 to the first fuel plate, and its temperature is found to be 1165°F (line B). The central fuel plate, which radiates half of its output in each normal direction, must then radiate 750 Btu/hr-ft^2 to the plate at 1165°F, and thus must be 1205°F (line C), which is approximately the design limit. This analysis neglects the fact that the axial power distribution has

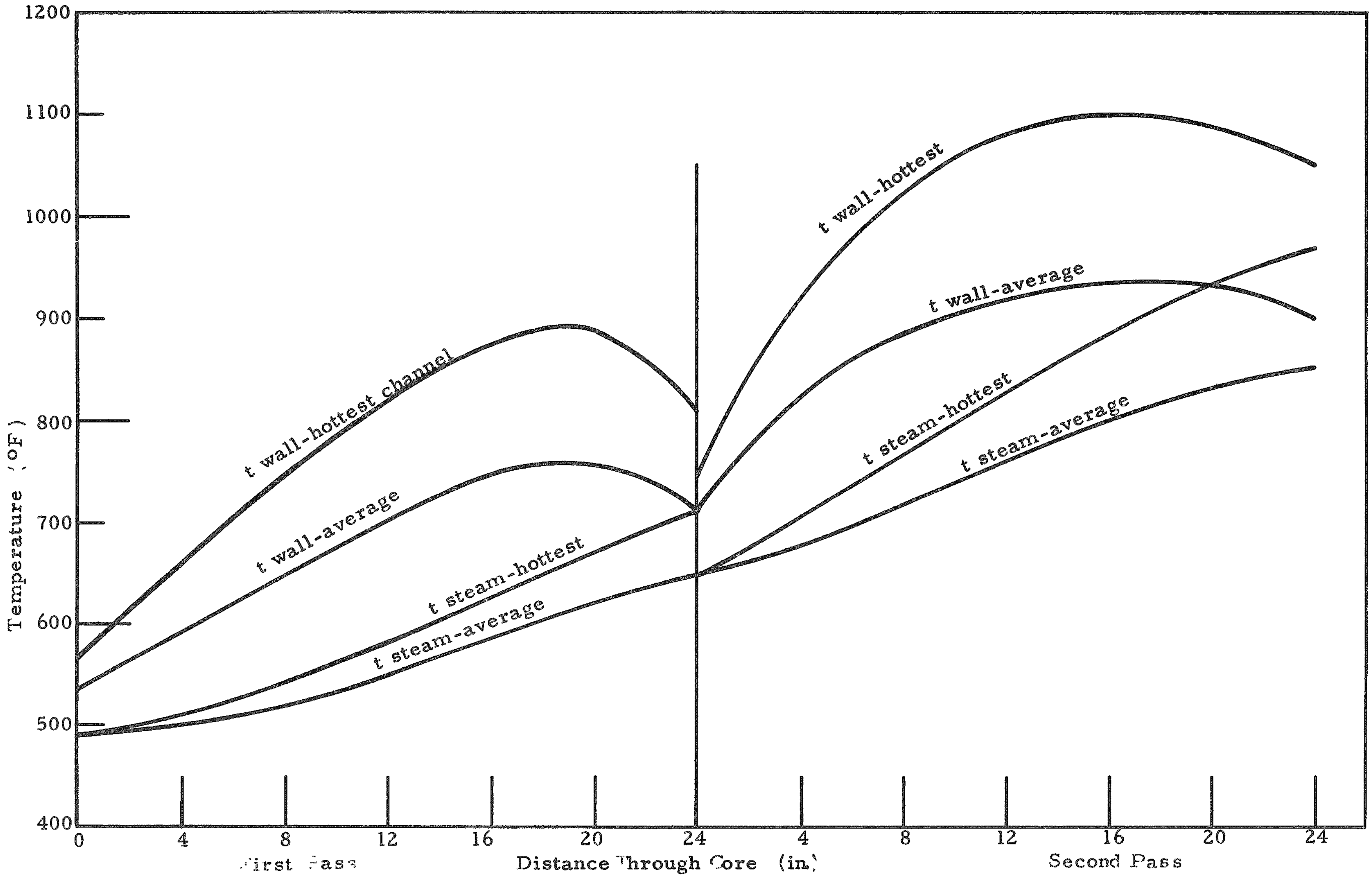


Fig. 15

Superheated Steam Temperature and Fuel Element Surface Temperature vs. Distance through Core - BORAX V

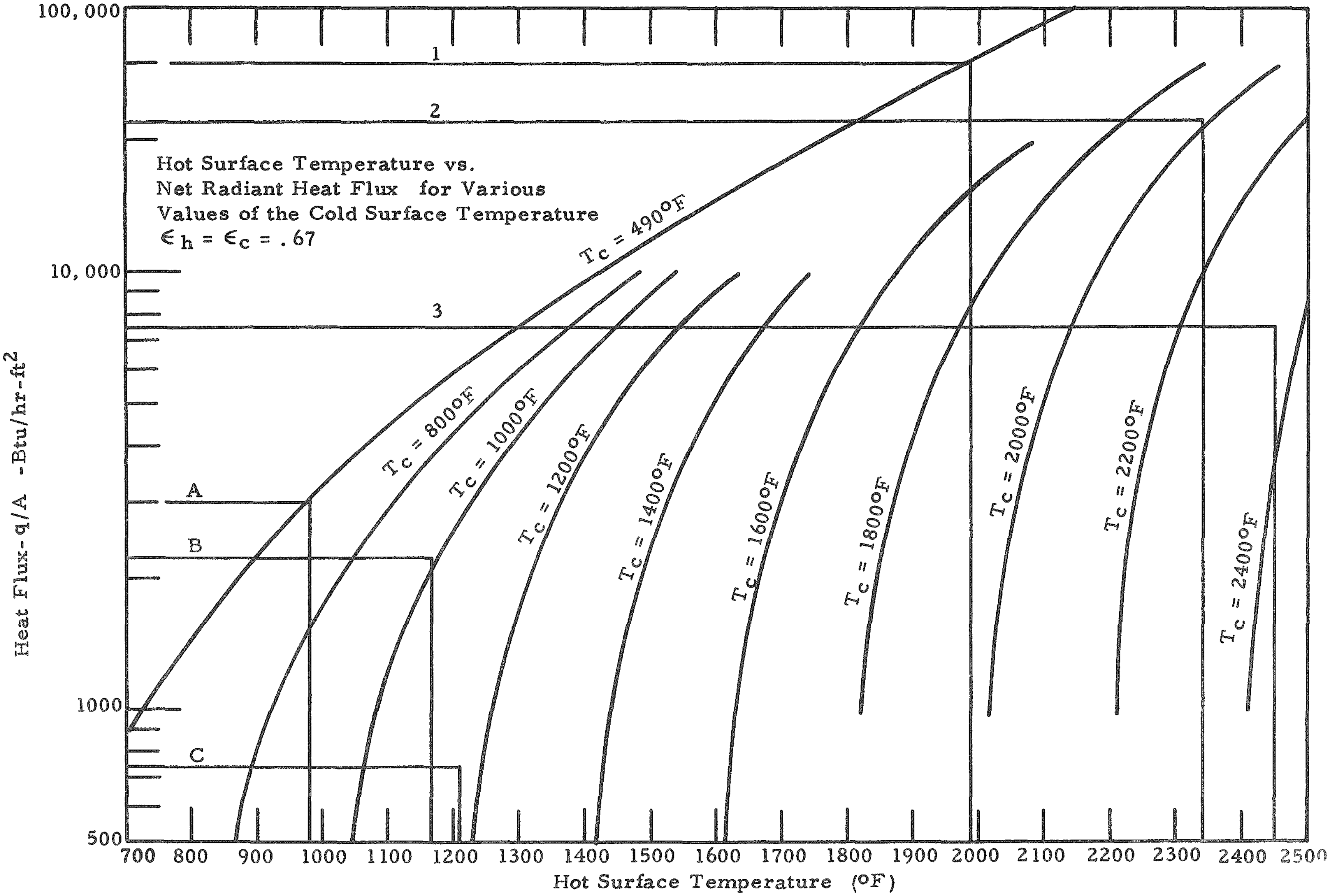


Fig. 16

Radiation-Cooling Curves - BORAX V

a peak-to-average ratio of 1.3, but this is largely compensated for in the case of decay heat removal by the fact that less than 80% of the decay heat is generated in the fuel plates. Thus, it is possible to remove the superheater decay heat by radiation alone when the power level has dropped to 1% of maximum design. The time for this drop is a function of the prior operating history, both in regard to power level and time at power, but is only 300 sec after 12-hr operation at 20 Mw. Similarily, Fig. 16 shows that for a flux of 7500 Btu/hr-ft^2 from each fuel plate, equivalent to 3.5-Mw reactor power, the center plate temperature reaches only 2450°F (below the melting point of Type 304 stainless steel) if the cooling is by radiation alone. Therefore, the possibility of melting any of the superheater fuel plates is small.

Under normal operating conditions, there will also be some heat radiated to the moderator, but the radiant heat loss, together with the heat loss due to conduction and convection across the insulating static-steam gap, will be only 1 to 2% of the superheater output.

5. Core Design - Reactor Physics

a. Introduction

The purpose of this section is to describe briefly methods and results of BORAX V core physics calculations. The reactor criticality calculations were performed by codes RE-6 and RE-8.(6) The cross sections for these codes are obtained with the aid of codes MUFT(7) and SOFOCATE.(8) Most of the criticality calculations were based on two-group cross sections, but the agreement with several calculations based on four-group cross sections was quite satisfactory; therefore, no distinction is made between two-group and four-group results. All results of this section are preliminary and subject to revision.

One of the first objects of the core physics calculations is to determine configurations and loadings for the two regions of the core such that (1) the system is critical at power, and (2) the proper fraction of the power, 17% in the present case, is produced in the superheater. Distributions of power density in both regions and of voids in the moderator are also of immediate interest. Promising configurations and loadings were examined for change in reactivity on flooding of superheater, for void and temperature coefficients of reactivity, and for adequacy of control rods.

b. Core with Central Superheater

Since the boiling region of this core contains lumped fuel and an appreciable amount of U^{238}, a resonance self-shielding factor must be obtained for use with MUFT and a thermal disadvantage factor for use with SOFOCATE. An estimate for the resonance self-shielding was obtained from the results of Dresner[9] and of Hellstrand [10] Some difficulties arose in the calculation of thermal disadvantage factors for a P_3 code for cylindrical geometry[11] and the method of Amouyal[12] yielded values of the disadvantage factor which increased with decreasing moderator thickness in the region of interest. Thie and his co-workers[13] have reported the same sort of physically questionable results from SNG [14] The disadvantage factors were finally calculated by a modification of a method suggested by Copic[15] and Thie.[13] The method used involved (1) calculation of diffusion-theory flux ratios for the cylinder, (2) transformation of the cylinder to an "equivalent" slab; (3) calculations of P_3[16] and diffusion-theory flux ratios for this slab and (4) estimation of P_3 flux ratios for the original cylinder

The fuel rod for which the disadvantage factors mentioned above were calculated was an early concept of the design It was a UO_2 rod, 0.584 cm in diameter with a density of 7 01 g/cm^3. The rod was clad with aluminum, 0.762 cm in diameter. All the earlier design calculations were based on this aluminum-clad rod In order to obtain preliminary results for this design, the disadvantage factors calculated for this rod were applied to the rod shown in Fig 4. This stainless steel-clad rod, with oxide density 90% of theoretical is envisaged in all other calculations in this section.

In order to obtain an estimate of the axial power-density distribution in the boiling region with nonuniformly distributed voids, the axial power density distribution, curve A of Fig. 17, was assumed. This power-density curve, combined with representative results from the heat transfer parametric study, led to the curve of axial distribution of void fraction shown in Fig. 18 For the purposes of calculating the corresponding axial power density, the boiling core with no superheater was divided into five "layer cake" regions with uniform, but different, void fractions in each region. The resulting smoothed axial power-density distribution is curve B of Fig. 17 The ratio of maximum-to-average power and the position of the maximum power density are almost unchanged, so it is reasonable to assume that the axial distribution of void fraction in Fig. 18 is fairly good. The introduction of a superheating region probably makes no drastic change in these axial distributions.

The arrangement of the 12 central superheating assemblies is shown in Fig. 19. The superheating assembly actually considered here is Model 1, as shown in Fig 8 except that, as noted on Fig. 20 and in Tables IV and V, the volume fraction of stainless steel differs from the volume fraction derived from the drawing Since the superheater contains very little U^{238}, the resonance self-shielding factor is unity Disadvantage factors were calculated by replacing the fuel square with a cylinder and applying a P_3 code[11] and/or the method of Amouyal [12]

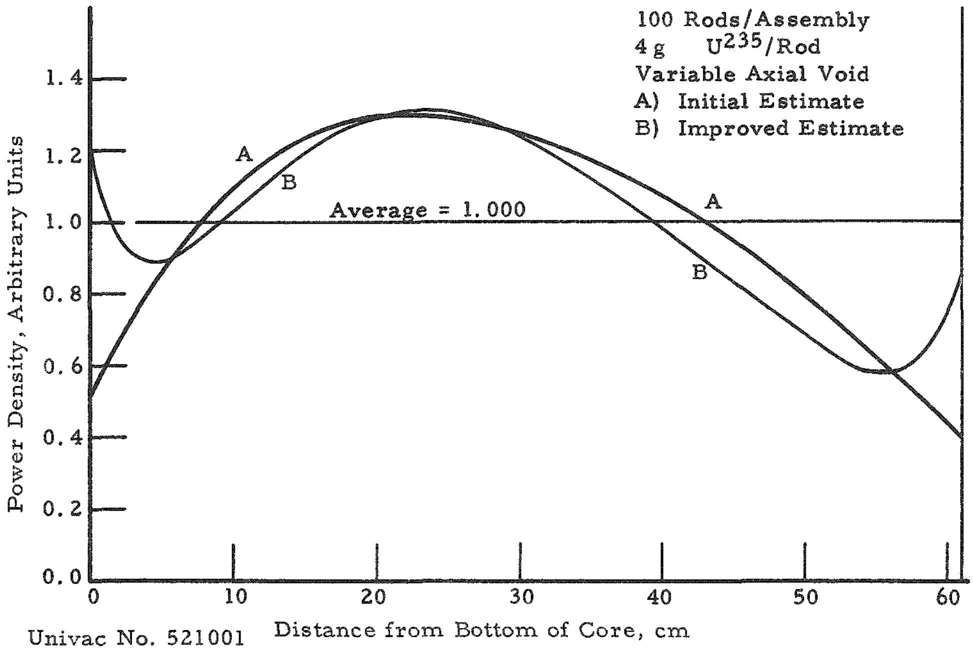


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Fig. 17

Axial Power Density Boiling Core - BORAX V

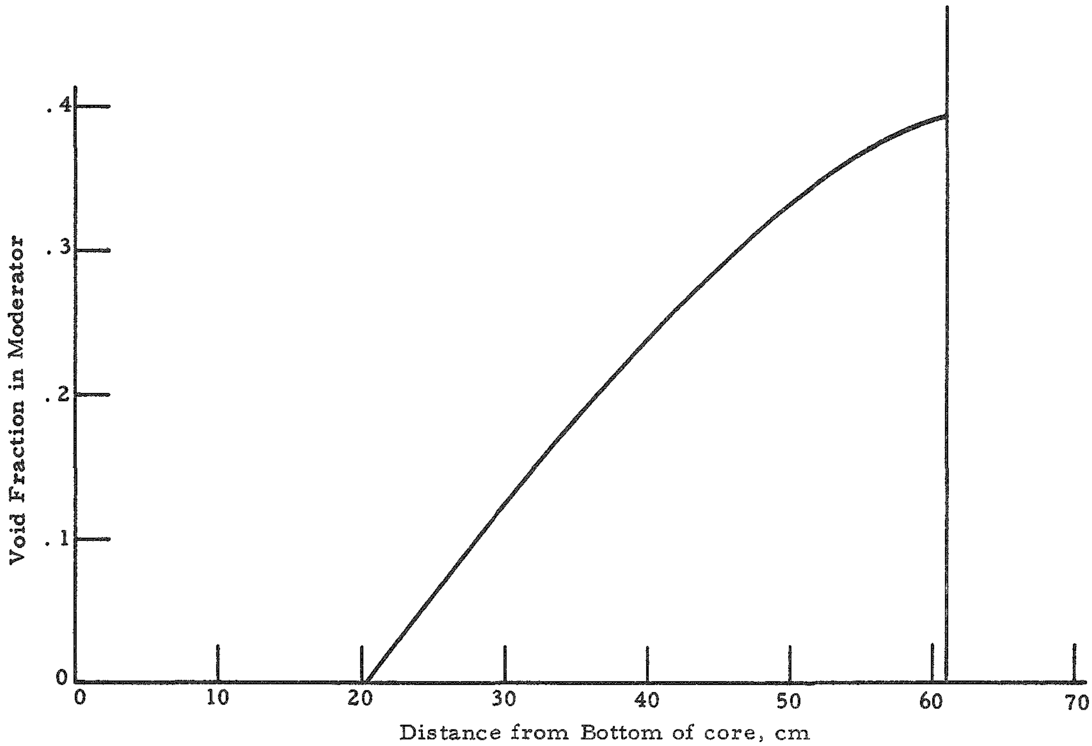


Fig. 18

Axial Void Fraction Boiling Core with 100 Rods/Assembly - BORAX V

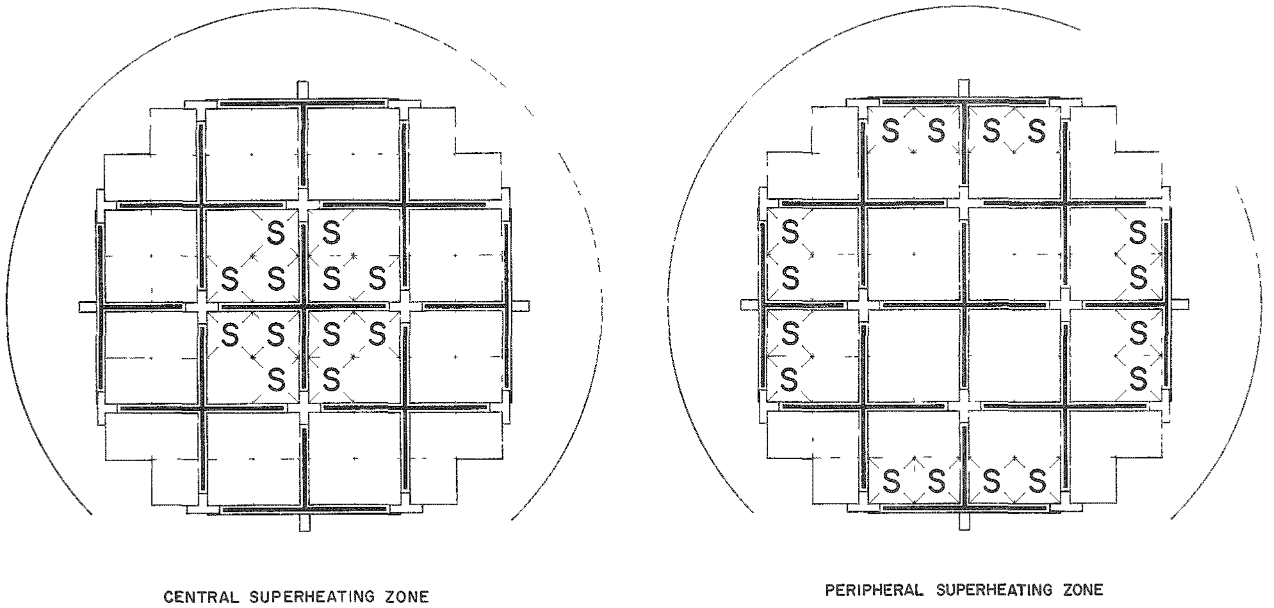


Fig. 19

Loading Diagram - BORAX V

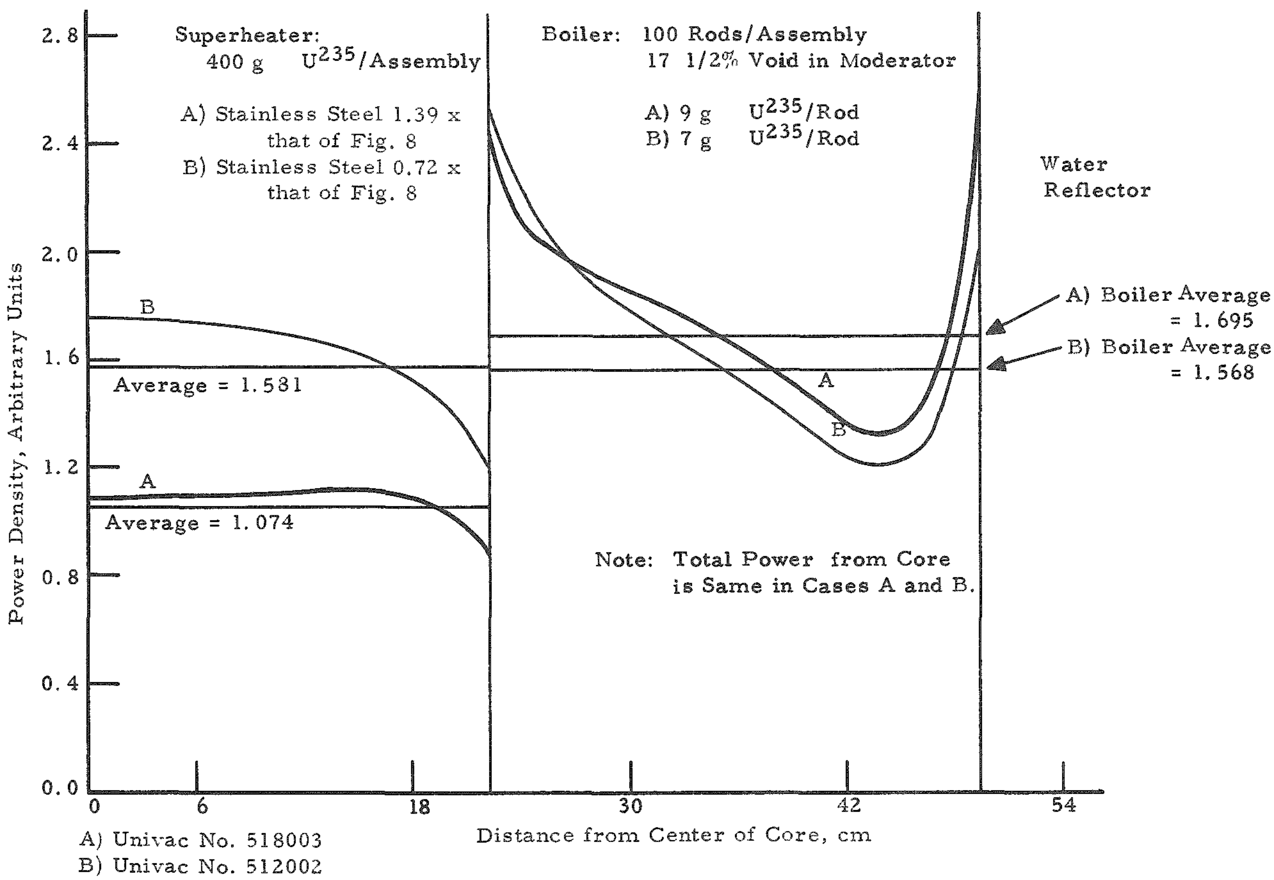


Fig. 20

Radial Power Density Central Superheater - BORAX V

Table IV

CHARACTERISTICS OF REACTOR WITH CENTRAL SUPERHEATER

No. Fuel Rods/Boiling Assembly	g U^{235}/ Fuel Rod	g U^{235}/ Super-heater Assembly	Fraction of Power from Super-heater, %	Max/Av Radial Power Density		Void Coeff. (17 1/2% to 0%), %Δk/k/% Void	Normal Flooding Worth of Superheater (System at 253°C with 0% Void), %Δk/k	Total Change of Reactivity From Cold Flooded System to System at 253°C and 17 1/2% Void, %	Temp Coeff. (for Flooded System between 20°C and 253°C) %Δk/k/°C
				Super-heater	Boiling Zone				
75	7.7	400	13.8	1.02	1.21	-0.21	+0.3	+11	-0.031
100	10.6	300	10.6	1.04	1.63	-	-	-	-
100	9.5	400	13.6	1.04	1.56	-0.25	+0.5	+12	-0.029
100*	8.2	510	17	1.04	1.46	-	-	-	-

Notes: Stainless steel volume 1.39 x that of Fig. 8. System critical at power with 17 1/2% void in boiling zone moderator.

*Numbers in last row of this table obtained by extrapolation from data of second and third rows.

Table V

ADDITIONAL CHARACTERISTICS OF REACTOR WITH CENTRAL SUPERHEATER

No. Fuel Rods/Boiling Assembly	g U^{235}/ Fuel Rod	g U^{235}/ Super-heater Assembly	Fraction of Power from Super-heater, %	Max/Av Radial Power Density		Void Coeff., %Δk/k/% Void	
				Super-heater	Boiling Zone	(17 1/2% to 10% Void)	(17 1/2% to 0% Void)
75	5.6	400	22.1	1.08	1.44	-0.20	-
100	7.5	300	15.2	1.07	1.60	-0.26	-0.19
100	6.2	400	21.3	1.12	1.68	-0.23	-
100*	7.1	330	17	1.08	1.62	-0.25	-

Notes: Stainless steel volume 0.72 x that of Fig. 8. System critical at power with 17 1/2% void in boiling zone moderator.

*Numbers in last row of this table obtained by interpolation of data of second and third rows.

Typical radial curves of power density are shown in Fig. 20. Some preliminary results of the calculations are presented in Tables IV and V, from which the following rough results are obtained: the central superheater core, with the Model 1 assembly of Fig. 8, is critical and producing 17% of the power in the superheater when the 48 boiling assemblies contain 4800 fuel rods with a total of about 36.5 kg of U^{235} and the 12 superheating assemblies contain a total of about 4.9 kg of U^{235}. Under these conditions, the ratio of maximum-to-average power density is about 1.06 in the superheater and about 1.55 in the boiling zone. The void coefficient of reactivity between $17\frac{1}{2}$% and 0% void is roughly -0.25%/% void, and the temperature coefficient of reactivity between 20 and 253°C is roughly -0.03%/°C. The increase in reactivity on normal flooding of the superheater is about 0.5%. The total change in reactivity from the flooded system at room temperature and pressure to the system at power is about 12%. Account has not been taken of reactivity losses due to fission product poisoning or burnup.

c. Core with Peripheral Superheater

The superheating assembly actually considered here is again that of Fig. 8 except that the volume fraction of stainless steel is 28% less than the volume fraction derived from the drawing.

The arrangement of the 16 peripheral superheating assemblies is shown in Fig. 19 and a typical radial plot of power density is shown in Fig. 21. Some preliminary results of the calculations are presented in Table VI.

Table VI

CHARACTERISTICS OF REACTOR WITH PERIPHERAL SUPERHEATER

No. Fuel Rods/Boiling Assembly	g U^{235}/Fuel Rod	Fraction of Power from Superheater, %	Max/Av Radial Power Density		Void Coeff. ($17\frac{1}{2}$% to 10% Void), % $\Delta k/k$/% Void
			Superheater	Boiling Zone	
75	5.1	16	1.22	1.43	-0.26
100	6.1	15	1.43	1.28	-0.32

Notes: System critical at power with 400 g U^{235}/superheat assembly and $17\frac{1}{2}$% void in boiling zone moderator.

d. Boiling Core

Calculated results are available for a boiling core at power, made of 60 fuel assemblies with 100 fuel rods per assembly. The moderator contains $17\frac{1}{2}$% void, uniformly distributed. These results

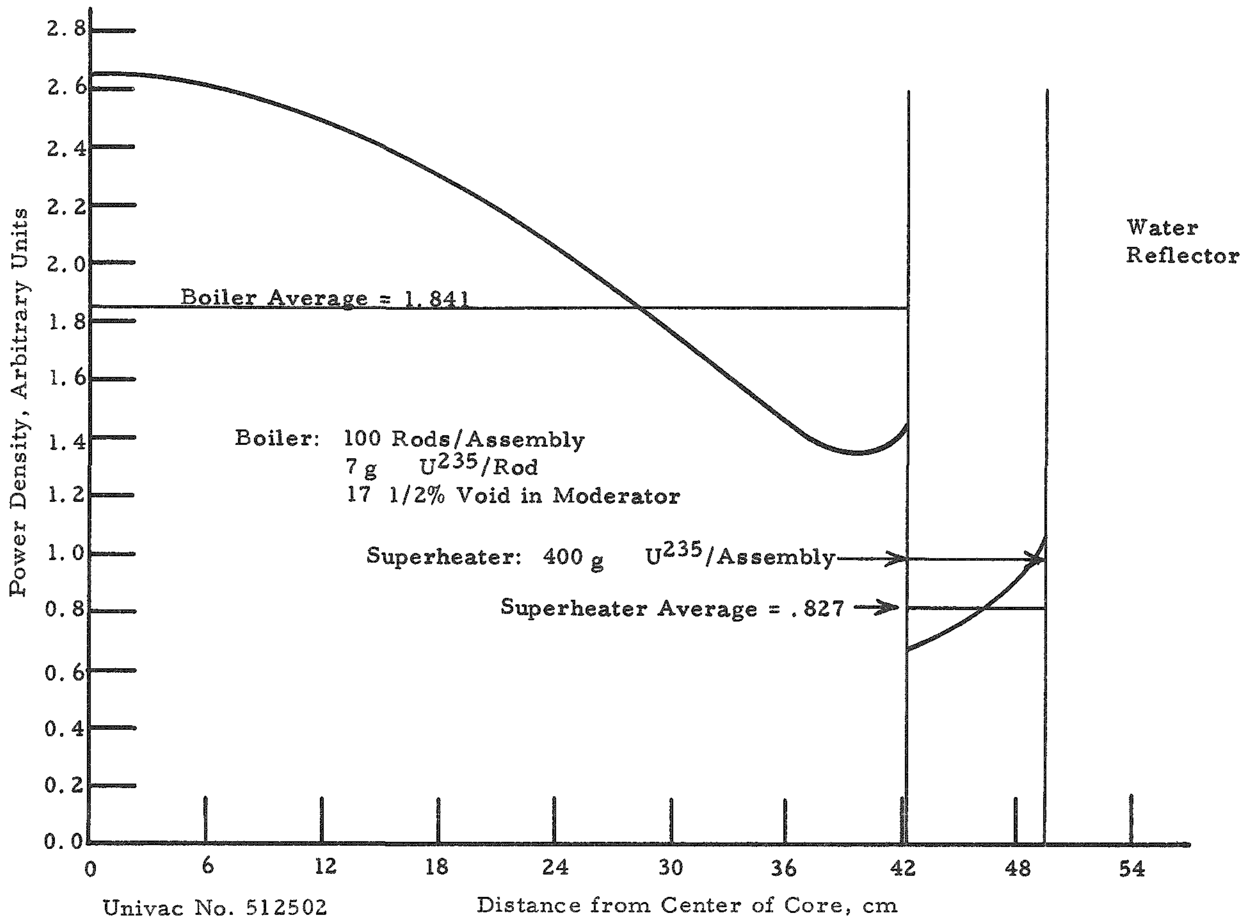


Fig. 21

Peripheral Superheater Radial Power Density - BORAX V

indicate that the system is critical with about 5.7 g of U^{235}/fuel rod and that the radial maximum-to-average power-density ratio is about 1.51. A typical plot of radial power density is shown in Fig. 22.

e. Control Rod Worth

Preliminary calculations on a boiling core for total control rod worth have been made by a semi-empirical method based on experimental correlation.(17) The value obtained is 18% $\Delta k/k$, which is probably low. The most pessimistic reasoning would indicate a minimum value of 16% $\Delta k/k$. Since 12% excess reactivity is probably the maximum that must be allowed for void and temperature effects, there appears to be some margin for fission product poisoning and shutdown.

f. Fuel Assembly Worths

Two-group calculations were performed by hand to determine the worth of an assembly in the cold core without superheater. The

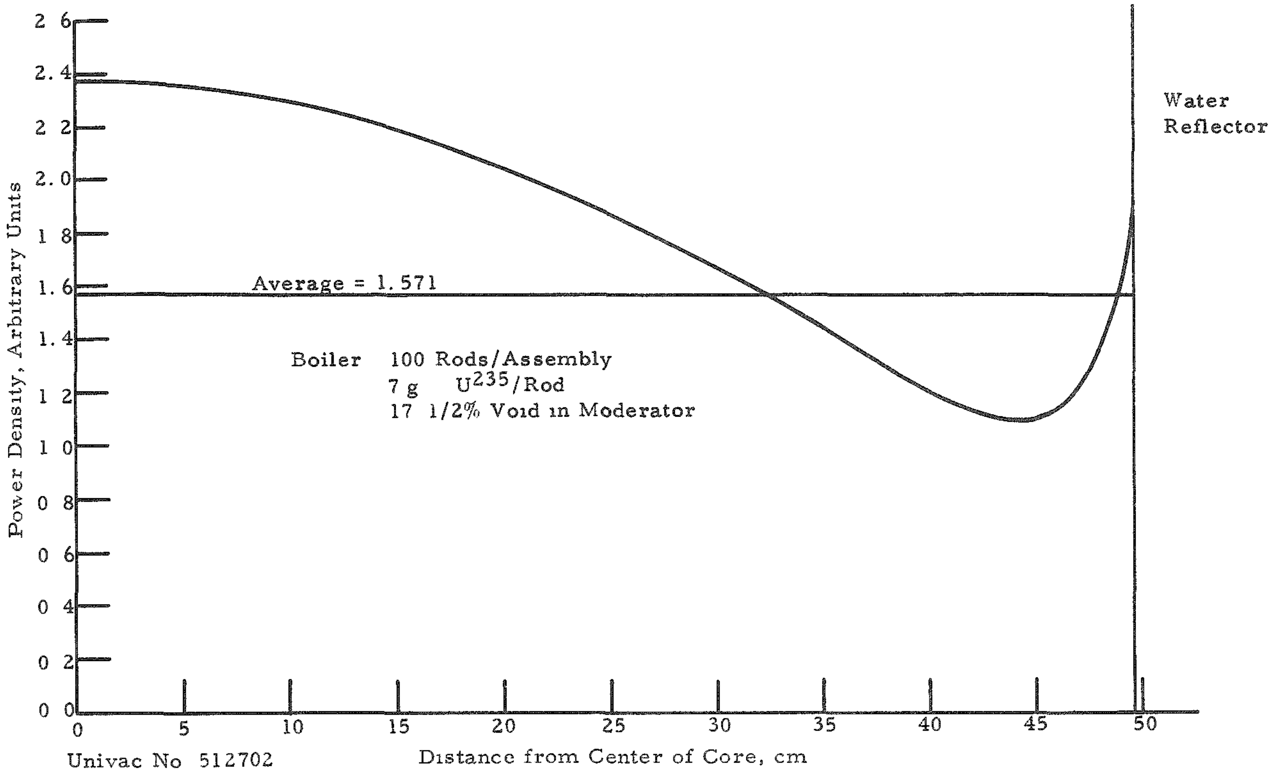


Fig. 22

Boiling Core Radial Power Density - BORAX V

assembly considered contained 100 fuel rods with 7 g U^{235} per rod. The worth of the assembly is about 1% $\Delta k/k$ at the edge of the core and about 5% $\Delta k/k$ at the center.

g. Plans for the Future

Some rather difficult problems arise in the study of the experimental boiling core because this core can be a quite small, heterogeneous system of intermediate enrichment with an atom ratio of U^{235}/H as large as 0.05. More careful investigation of these problems has been postponed because of the greater urgency of questions related to the superheating core.

In the superheating core, the calculated loadings for the central and peripheral superheaters will probably be unequal. It may be desirable to avoid making a set of superheating assemblies with one loading for the central superheater and a second set with another loading for the peripheral superheater. In order to do this, one might use boron poison strips with the higher enrichment.

6. Reactor Vessel

a. General

The general form of the reactor pressure vessel, as shown in Figs. 2 and 23, is a right cylinder closed at each end by an ellipsoidal head. The top head of the vessel is joined to the vessel proper by two bolted and gasketed flanges. The inside surfaces of the vessel are clad with stainless steel. The structural portion is fine-grained carbon steel and the assembly of the several parts is by fusion welding. Thermal insulation is stainless steel wool within a stainless steel skin. Support is by means of a bottom-mounted skirt, and the vessel is installed in a ventilated reactor pit. In order to provide the versatility desired in an experimental facility, the vessel is equipped with numerous nozzles of various types, as shown in Fig. 23 and Table VIII (page 61).

b. Design Criteria

Table VII shows the principal design and expected operating conditions.

Table VII

REACTOR VESSEL DESIGN CRITERIA FOR BORAX V

Vessel Height, Inside, Exclusive of Cladding, ft	16
Vessel Diameter, Inside, Exclusive of Cladding, in.	$66\frac{3}{8}$
Cladding Thickness, Nominal, in.	3/16
Design Pressure, psig	700
Design Temperature, °F*	650
Operating Pressure, psig	600
Operating Temperature, °F*	490
Hydrostatic Test Pressure, psig	1275
Neutron and Gamma-induced Thermal Stress in Vessel Shell, Max, psi	6000
Load Transferred to Top and Bottom Head by Core Hold-down System, lb	75,000
Load Transferred to Bottom Head Due to Weight of Fuel Core Support, etc., lb	20,000
Impact Load, Downward, on Any Single 3 or 6-in. Nozzle in Top Head, lb	5000
Impact Load, Downward, on 3 or More Nozzles in Top Head, lb	15,000
Impact Load, Downward, on Each Control Rod Nozzle, lb	3000
Force, Due to Piping, on Each Forced-convection Suction Nozzle, Max, lb	3600
Force, Due to Piping, on Each Forced-convection Inlet Nozzle, Max, lb	2800
Force, Due to Piping, on 6-in. Steam Nozzle, Max, lb	1250
Force, on Each of the Remaining Nozzles, lb	500
Moment, Due to Piping, on Each Forced-convection Suction Nozzle, Max, in.-lb	129,000
Moment, Due to Piping, on Each Forced-convection Inlet Nozzle, Max, in.-lb	27,400
Moment, Due to Piping, on 6-in. Steam Nozzle, Max, in.-lb	48,000
Moments on Remaining Nozzles	Negligible
Integrated Fast Neutron Flux at Vessel Wall, Min, nvt	5×10^{18}
Calendar Lifetime, Due to Fast Neutron Exposure at Vessel Wall, Min, yr	5

*The steam outlets and two of the instrumentation nozzles are thermally sleeved and are designed to conduct superheated steam at 900°F. The feedwater, spare heater inlet, and poison-injection nozzle are thermally sleeved and are designed to conduct water at 60°F.

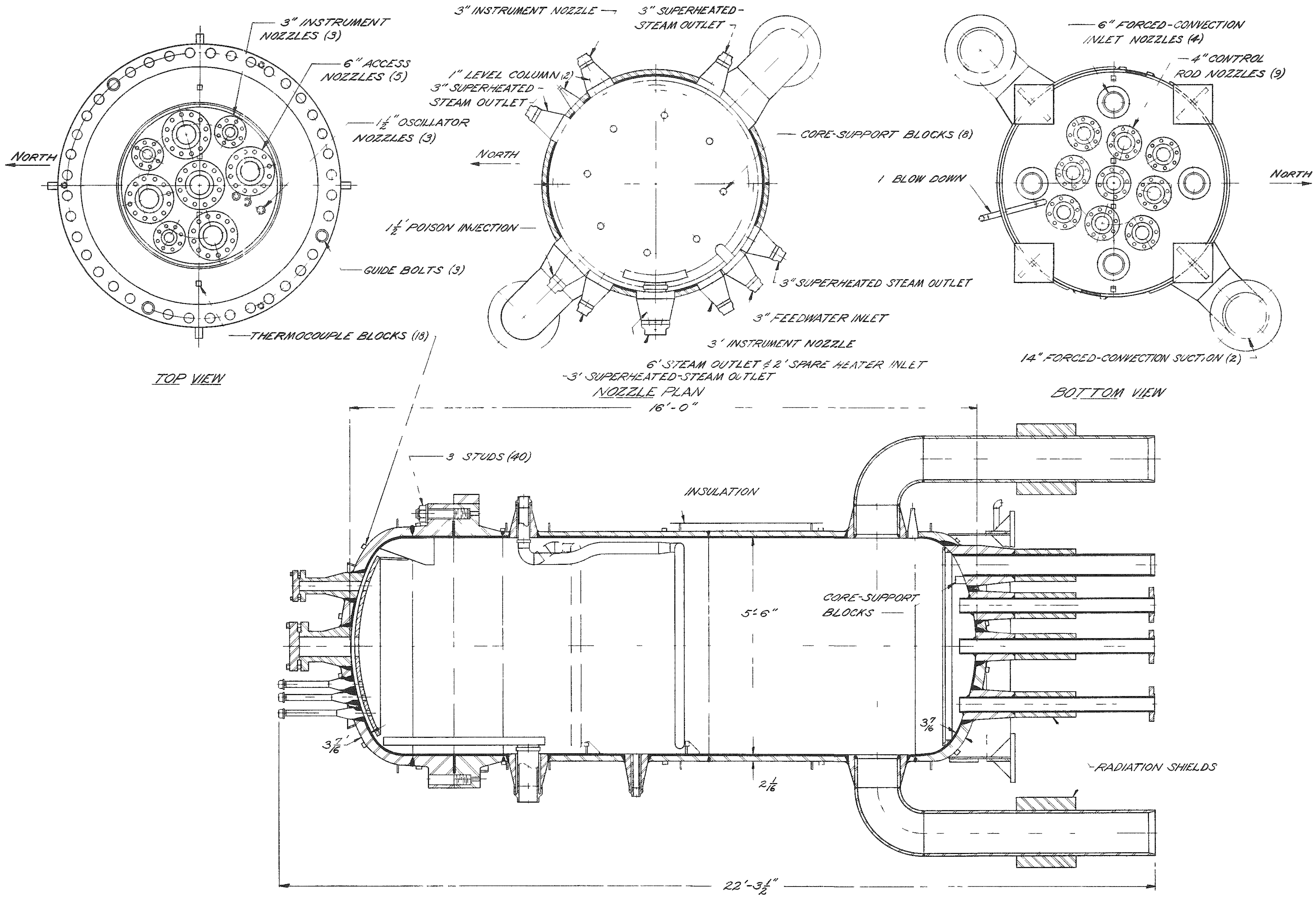


Fig. 23

Reactor Pressure Vessel - BORAX V

With one notable exception, vessel design and fabrication are based on the ASME Boiler and Pressure Vessel Code, Section I, 1956 Edition, including addenda and interpretations. In the interests of conservatism, the Code-allowed working stress of the BORAX V vessel metal has been reduced by 3000 psi, an amount equal to one-half the calculated neutron and gamma-induced thermal stress. Compliance with State of Idaho requirements is automatic, since the State has adopted the ASME Code, without exception.

c. Vessel Life

There appear to be three areas of general agreement, pertinent to the design of this vessel, on the subject of change of characteristics or damage in structural metals due to fast neutron irradiation. These are:

(1) Fast neutron radiation does cause some undesirable changes in structural metal, the change of primary importance being elevation of the ductile-brittle transition temperature.

(2) Appreciable irradiation-induced changes in A-212 steel do not occur below 5×10^{18} nvt.

(3) Metal characteristics in heat-affected zones adjacent to welds may exhibit the highest rates of change.

There is no general agreement regarding rates of change with increasing irradiation, temperature effect on rate of change, the effects of neutrons at energies below 1 Mev, and the effects of rate of dose.

With the foregoing in mind, the following considerations and treatments were applied in this phase of the vessel design:

(1) BORAX V is an experimental facility and so has a relatively short useful life, estimated at about 5 years.

(2) Design factors are conservative, but gross overdesign is avoided for reasons of economics. Total fast neutron exposure is therefore set at 5×10^{18} nvt. Based on a calculated 1.5×10^{11} fast neutrons/cm^2-sec at the vessel inner wall at 20-Mw thermal power, and with the anticipated operating schedules, this exposure will be acquired in about five calendar years.

(3) In line with accepted practice, the vessel structural metal is made to fine-grain specifications in order to depress the transition temperature.

(4) A system for monitoring vessel condition is provided by a series of coupons made from the vessel plate material and installed at the vessel inner wall in the maximum-flux zone. At appropriate times, coupon sets will be removed for testing. The coupons include base metal, weld metal, and heat-affected metal.

(5) Attempts will be made to measure integrated fast neutron exposures of the vessel wall and the coupons by suitably placed flux monitors.

d. Shell, Heads and Flanges

The structural portions of the vessel shell and heads, except as noted in Section "g" below, are carbon steel, A-212, Grade B Firebox Quality, made to A-300 fine-grain requirements with 6 to 8-Mc Quaid-Ehn grain size. Shell wall thickness is $1\frac{7}{8}$ in. and the heads $3\frac{1}{4}$ in., less cladding. The heads are 2:1 ellipsoidal and are seamless.

The inboard surfaces of the shell and heads are clad by metallurgical bond, according to A-264 requirements, with Type 304 stainless steel. No credit for cladding strength has been taken in the vessel wall calculations.

The structural portions of the vessel flanges are carbon steel, A-105, Grade II, made to A-300 Class I requirements with 6 to 8-Mc Quaid-Ehn grain size. The upper flange is $8\frac{1}{4}$ in. thick and the lower is $7\frac{11}{16}$ in. The greater thickness of the upper flange is due to the moments created by the core hold downs.

There will be 40 flange bolt assemblies, each consisting of 3-in., A-193, Grade B-7A steel, profiled and bored studs, with gauging pins and A-194, Grade 2H nuts. The required bolt prestress is calculated to be about 26,000 psi.

The flange bores are clad with Type 304 stainless steel, electrode-deposited by a method which limits the carbon content to 0.08% maximum in the outer 0.100 in. of the cladding. The gasket area of the flanges is inlaid with Type 309-moly stainless steel.

The lower flange is faced to receive two concentrically located, spiral-wound stainless steel and Teflon gaskets. The inner gasket alone is the normal pressure seal, with the outer gasket functioning as a standby seal. Between the two is a leak-off groove. Leakage is piped away from the vessel to permit monitoring of inner gasket integrity.

e. Neutron Activation in Upper Head

Unless preventive measures are taken, the buildup of Fe^{59} in the upper vessel head, from the neutron-gamma reaction on Fe^{58}, leads to a personnel-exposure problem when the head is removed or replaced because of neutron streaming from the virtually dry superheat assemblies.

Cross-section considerations make it clear that the major source of activation is the thermal neutron group. This is effectively eliminated by the use of a neutron shield consisting of a 1-in. -thick dished plate of Type 304 stainless steel with 1% boron. The nozzle openings in the head are similarly shielded. The largest remaining effect will then be due to the fast neutron component, which the neutron shield cannot effectively reduce. These fast neutrons, degrading through inelastic collisions, will lead to a certain amount of Fe^{59} in the head.

For purposes of calculation, all fast neutrons were assumed to originate from a point source at the geometrical center of the reactor. Consideration of geometry and attenuation factors gave a value of 3.7×10^{10} n/cm^2-sec for the effective fast neutron flux at the neutron shield. A ten-group energy analysis of the fast flux in iron, coupled with cross-section values, makes it possible to establish the specific neutron-absorption rate of Fe^{58} as a function of distance into the iron. This rate, 1.5×10^3 absorptions/cm^3-sec/incident fast neutron, proves to be essentially constant with distance and, corrected for the ratio of activation to absorption and for isotopic abundance, yields a resultant 1.7×10^6 activations/cm^3-sec/fast neutron. The use of standard methods of calculating shielding and the incident fast flux indicated a saturation radiation level of 200 mr/hr at the surface of the vessel head.

f. Radiation Heating in Shell

To obtain the maximum condition, all calculations were based on 40-Mw operation and a core volume of 170 liters. The following values were used for the concentration of core constituents:

H_2O (Density 0.8 g/cm^3)	1.12×10^{22} atoms/cm^3
Al	2.10×10^{22} atoms/cm^3
U^{235}	1.64×10^{20} atoms/cm^3
U^{238}	2.85×10^{21} atoms/cm^3

It was assumed that the source of the neutron and of prompt and delayed gamma radiations from the core could be treated as an infinite plane. Nine energy groups of gammas were considered, namely: 0.5, 1.0, 1.5, 2.0, 2.5, 3.0, 4.0, 6.0 and 8.0 Mev. Prompt gammas from neutron capture in the water moderator-coolant were also considered. The following results were obtained for a $2\frac{1}{16}$-in. shell wall (including the $\frac{3}{16}$ in. stainless steel cladding) first, with a shield, consisting of a 1-in. stainless steel plate with 1% boron, and, second, with no shield:

RADIATION-INDUCED THERMAL STRESS

Mev/cm^3-sec	$Watts/cm^3$	Δt, °F	Thermal stress, psi
With Thermal Shield Inner Edge 3.74×10^{12} Outer Edge 0.63×10^{12}	 0.60 0.10	8.7	2520
Without Thermal Shield Inner Edge 9.55×10^{12} Outer Edge 1.26×10^{12}	 1.53 0.20	19.2	5560 (say 6000)

These calculated radiation stresses, when added to the balance of the anticipated stresses, indicate that thermal shielding opposite the core is not justified. This is particularly true when one considers that the maximum radiation stress will be achieved infrequently. Furthermore, the allowable design working stress in the structural material has been reduced by 3000 psi.

g. Nozzles and Reinforcements

The vessel nozzles and nozzle extensions are A-312, Type 304 stainless steel with Schedule 80 wall thickness, except for the level column and blowdown nozzles, which are Schedule 160.

For the thermally sleeved nozzles, a "Y"-type, one-piece junction device has been chosen for connecting the nozzle outboard ends to the sleeves in order to place the weldments away from the areas of high thermal-stress concentrations.

The nozzle reinforcements are A-312 and A-240 stainless steel. While this will certainly cause ring stresses at the point of juncture to the carbon steel vessel walls, calculations indicate these stresses will not be excessive and certain distinct advantages will be achieved. The foremost among these has to do with nozzle placement and alignment, which are stringently specified. This construction permits vessel stress-relieving to be completed after the welding-in of the reinforcements, but prior to installation of the nozzles. After relieving, the reinforcements are precision-bored to receive and position the nozzles. The specified placement tolerances are thus readily obtained. The remaining welds are stainless-to-stainless, eliminating the need for later stress-relieving with its inherent warpage potential.

The nozzle complement is set forth in Table VIII.

Table VIII

NOZZLE LIST FOR THE REACTOR VESSEL

Function	I. P. S. Size		
	Number	in.	Comment
Feedwater Inlet	1	3	Thermally Sleeved
Spare Heater Inlet	1	2	Thermally Sleeved
Poison Injection	1	$1\frac{1}{2}$	Thermally Sleeved
Level Column	2	1	- - - - -
Control Rod	9	4	Externally and Internally Shielded
Steam Outlet	1	6	Thermally Sleeved
Steam Outlet	4	3	Thermally Sleeved
Blowdown	1	1	- - - - -
Instrumentation	3	3	Internally Shielded
Instrumentation	2	3	Thermally Sleeved
Access	5	6	Internally Shielded
Oscillator	3	$1\frac{1}{2}$	Internally Shielded
Forced-Conv. Suction	2	14	Externally Shielded
Forced-Conv. Inlet	4	6	Externally Shielded
Flange Leak-off	1	3/4	- - - - -
Total Nozzles	40		

h. Nozzle Shielding

In keeping with accepted practice, the several nozzles and nozzle extensions which penetrate the shield pit walls are equipped with steel radiation sleeves, or shields, or are offset to provide a stepped configuration. In addition, the 9 re-entrant control-rod nozzles are fitted with internal shielding.

i. Internal Attachments

The several reactor vessel internal supports, blocks, rings, etc., are the Type 304 stainless steel and are attached to the vessel structural metal by welding. Finish-machining of these items will be done after stress-relieving.

j. Thermocouples

Eighteen thermocouple blocks, each containing 2 thermocouples, are attached to the outside surface of the vessel for temperature monitoring.

k. Vessel Supports and Guides

The vessel is supported from the bottom head by an A-7 steel skirt and four sole plates. The sole plates are attached, by welding to 14-in.-wide flange beams cast into the bottom reinforced concrete support and shield slab.

The upper 8 in. of the skirt contains 20 vertical slits to allow for vessel thermal expansion.

Four equally spaced guide plates are provided at the vessel flanges. Matching guides are anchored to the reactor pit walls. These plates and guides permit thermal expansion to take place but resist lateral vessel movement.

l. Thermal Insulation

Vessel insulation consists of a 3-in. blanket of Type 430, fine stainless steel wool batts, applied at a density of 5 lb/ft^3. These batts are supported by a series of A-107 steel rings shop-welded to the vessel and are enclosed by a skin of 20-gauge Type 304 stainless steel, wired and bolted in place. Certain sections are removable for access to flange bolts, etc. Handhole covers are provided opposite each thermocouple block for access during preliminary testing. It is estimated that heat loss through this insulation will not exceed 25 kw.

The vessel nozzles and nozzle extensions inside the reactor pit are insulated to appropriate thicknesses and enclosed in 24-gauge galvanized iron jackets.

Insulation deterioration is expected to be at a minimum since the reactor pit is force-ventilated and the products of dissociation are thereby removed.

m. Stress Relieving

Stress-relieving, according to the ASME Code, is done at a maximum temperature of 1000°F in order to avoid sensitization of the stainless steel components.

n. Inspection and Testing

An extensive program of inspections and tests has been formulated to assure a degree of quality control of materials and workmanship commensurate with the intended use of this vessel.

7. Reactor Control and Instrumentation

a. Control Rods

BORAX V has a total of 9 control rods. Five of the rods, located with one at the center and 4 equispaced on a 12.5-in. radius, are cruciform shaped with 14-in. blades. The remaining 4 rods, located at the midpoint of the core edges, are "T"-shaped, each with a 14-in. and a 7-in. blade.

The design of a cruciform rod is shown in Fig. 24, and the "T"-shaped rods are similar in construction. The poison section of a control rod is made of $\frac{1}{4}$-in. thick Boral which is covered by a $\frac{1}{16}$ in.-thick X-8001 aluminum cladding spot-welded to the Boral and seal-welded at the edges to give a total blade thickness of $\frac{3}{8}$-in. The Boral section of each rod is vented to permit outgassing. Each rod has a $\frac{3}{8}$-in.-thick X-8001 aluminum follower to reduce flux peaking in the control rod channels. Riveted to the bottom of each rod is a 3 x 3 x $\frac{3}{8}$-in. cruciform stainless steel extension shaft which extends down through a reactor vessel control-drive nozzle and connects to a control-rod-drive-mechanism extension rod. Control rods may be removed from the reactor core structure after disconnection from the control-drive mechanism by means of the grappler head at the top of the rod.

Two sets of control rods are required. In the superheat cores the control-rod stroke and follower length is limited to 24 in. by the flood and drain manifolds at the bottom of the core structure. The control-rod stroke and follower length in the boiling core are 29 in., to give an unpoisoned upper reflector 5 in. thick for certain experiments.

b. Control-rod Drives

The control-rod drive mechanisms for BORAX V are those originally used on EBWR, which are available because of modifications to EBWR. They are being changed to meet the stroke and speed requirements of BORAX V and to incorporate improvements and simplifications demonstrated to be desirable from EBWR operation.

Figures 25 and 26 show the drives after modification and ready for installation. The original EBWR stroke was 48 in. For BORAX V this will be shortened to a 30-in. maximum. Adjustable limit switches then will reduce this to either 29 in. for the boiling core, or to 24 in. for superheating core operation. The original drive motors with speed-change gears are available with the drives. These are used on the 4 "T"-shaped outer rods and the center rod. Because of the greater worth of the center rod its drive motor is geared to give a slower speed. New two-speed motors are used with the other 4 cruciform intermediate rod drives, giving a slower

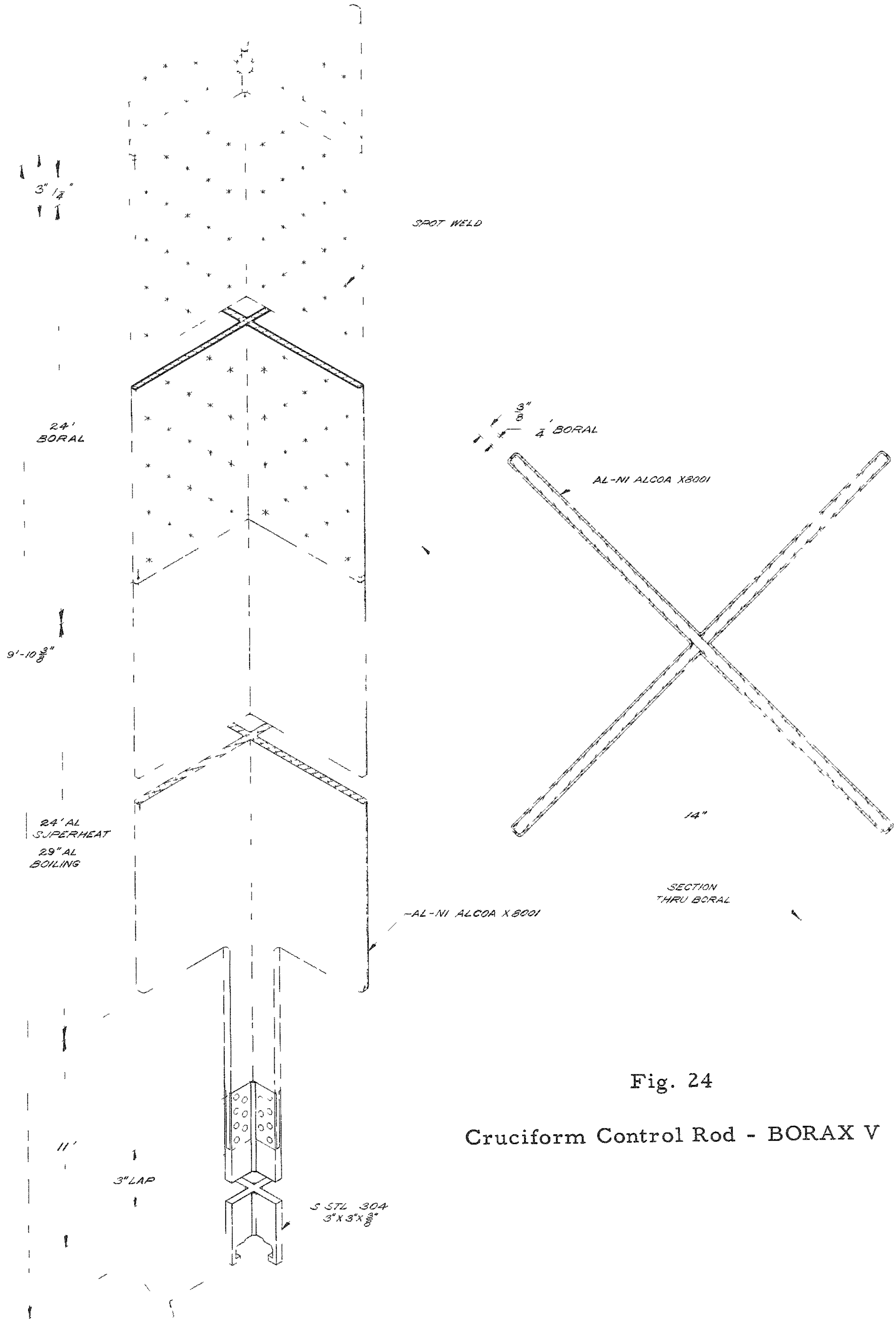


Fig. 24

Cruciform Control Rod - BORAX V

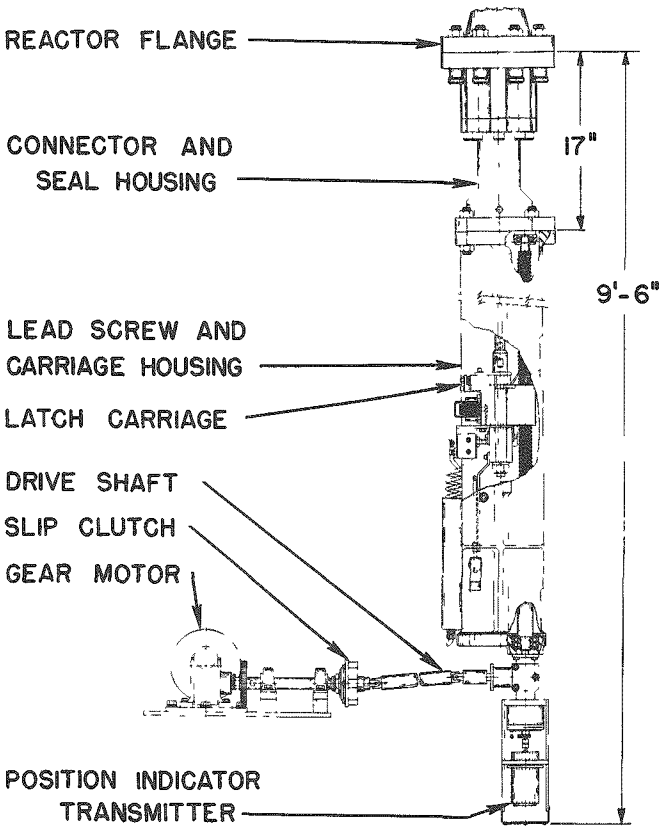


Fig. 25

Control-rod Drive Mechanism

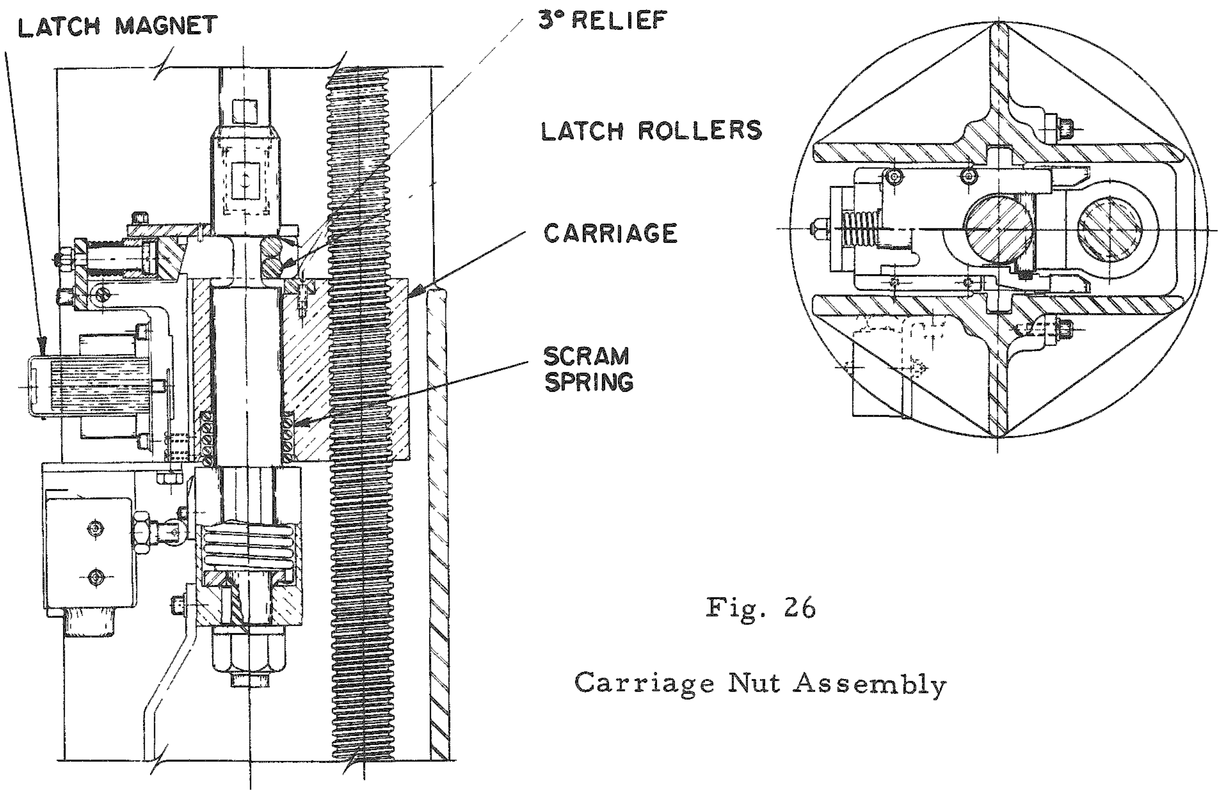


Fig. 26

Carriage Nut Assembly

speed when all 4 rods are operated as a group than when operated individually. In any case the maximum rate of reactivity addition by means of control-rod withdrawal is limited to 0.05%/sec.

To adapt the position indicator to the BORAX V system, the gears driving the selsyn generator are changed to give a ratio of 2 revolutions of the selsyn to one of the lead screw, or 10 revolutions of the selsyn per in. of travel.

Two changes are being made as a result of EBWR operating experience. The external dashpot was found to be unnecessary and has been removed, and the flushing arrangement has been changed to eliminate trouble from radioactive crud settling in the seal housing. A new flushing-water inlet between the guide bushing and seal is used, with the outlet remaining below the seal in the lantern ring section. A blowdown pipe is connected above the guide bushing and is used for intermittent flushing during periods of shutdown.

Beyond these changes the original EBWR description applies. The mechanisms are located outside the reactor shield and below the reactor. A screw shaft and screw nut are used to translate rotating drive motion into the linear motion of the control rod. The control-rod shaft is connected to the screw nut by a triggering device in the form of a roller latch and 15-volt dc solenoid. Interruption of the electrical power de-energizes the solenoid, thereby releasing the latch and scramming the rod.

The control rod and drive are connected by a chrome-plated extension shaft which passes through the pressure-breakdown seal. Figure 27 shows the seal and the connecting joint.

When a rod is raised, a 140-lb spring is compressed between the rod guide and latch carriage. At reactor pressure below 300 psi this spring is most effective and maximum rod velocity is reached in 4 in. However, the main propulsion force is always reactor pressure. For a scram at 600 psi the time for the rods to travel 24 in. (the point at which the dashpot begins to act) is estimated to be 0.2 sec.

c. Reactor Control, Electrical

The reactor is controlled from a building located one-half mile from the reactor. The control console as shown on Fig. 28 is designed for two operators to control both the reactor and the process systems. Process system control is described in Section III, c.

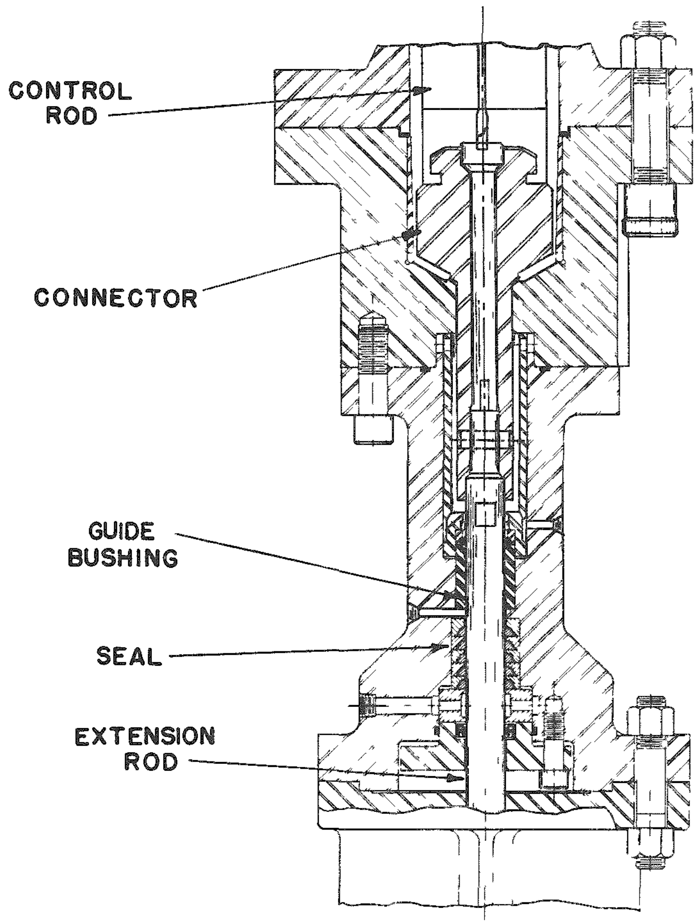


Fig. 27

Connector and Seal Assembly

(1) Control System

The reactor is controlled by moving control rods in and out of the core. The control rods are driven by 3-phase, 60-cycle, 480-volt, reversing motors. The intermediate rods have two-speed reversing motors, while the center and outer rods have single-speed motors. A permissive interlock requires that all control rods be at the full "in" position before the control circuits can be energized. Likewise, all scram signals must be reset. Operating the reactor-on key switch (key switch No. 1) then energizes the latch solenoids for each rod and allows the operation of the control circuits. The motors are operated through conventional motor-starters with overload protection. The control circuits are operated at 120 volts ac.

The control rods are raised and lowered by the operation of 3 lever switches at the control console. One lever switch controls the center rod only, a second operates the intermediate rods, and a third the outside rods. Interlocks provide that only one of the three lever switches at a time is effective in the "raise" position. A selector switch for the intermediate rods and one for the outside rods allows each rod to be operated individually or 4 rods to be operated as a group.

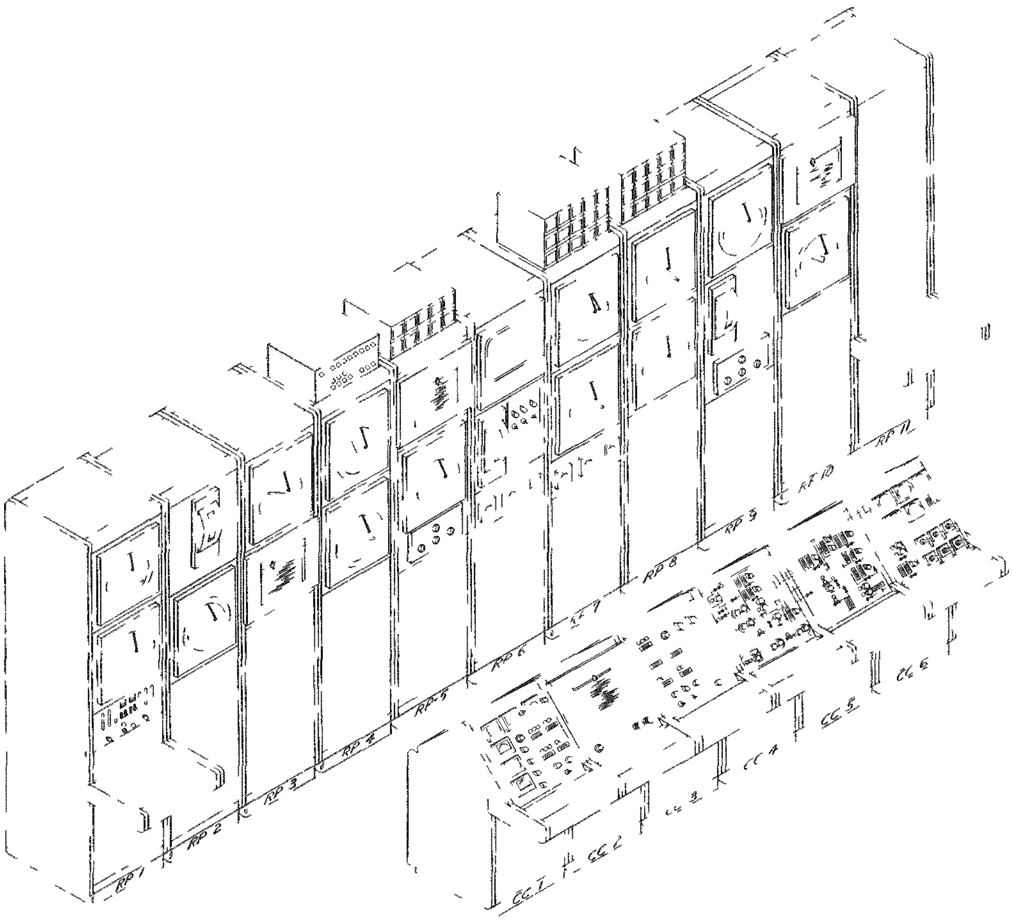


Fig. 28

Recorder Panel and Control Console - BORAX V

To keep the rod-motion time as short as possible, yet within safe limits of reactivity-addition rate, the intermediate rods, when operated as a group, move at one-fourth the speed they have when operated individually. Because the outside rods have a lower reactivity worth, operating them at the same speed, both for individual and group operation, permits reasonable rod-motion times and safe reactivity-addition rates. Change gears will allow a variation of control-rod speed depending on the control-rod worth for various loadings.

All control rods are driven into the reactor automatically under any of 15 scram conditions. In addition, three rod-drop buttons are provided for testing of control-rod operation. One button operates the center rod only. The other 2 buttons operate with the intermediate and outside rod-selector switches to drop the rod or the particular group selected. Operation of a scram or drop button interrupts the latch-solenoid circuit, causing the rod to move to the full "in" position under the forces of gravity, reactor pressure, and the accelerating spring. The latch solenoids are 15-volt, dc-operated, and are supplied by a rectifier taking power from the 120-volt ac system.

(2) Position Indication

The control-rod position is indicated by a selsyn transmitter and receiver with a mechanical counter read-out for each rod. The system will provide for reading the rod position to the nearest 0.01 in. The transmitting selsyn is geared to the drive lead screw. Therefore, the control-rod position and the drive position are the same only when the rod is latched to the drive lead screw.

Three position-indicating lights are operated from relays which, in turn, are operated from limit switches. These lights indicate when the rod is in the full "in" position, the full "out" position, and when the rod is latched to the drive.

(3) Scram System

Table IX shows a list of the conditions and their operating points which cause a scram. An annunciator is provided to give visual indication of a scram. Manual reset is required.

There are 4 key switches (numbers 2 to 5) in addition to the reactor-on key switch which can be used for bypassing scram interlocks during certain periods of operation.

Key switch No. 2 is provided for bypassing the short-period scram, during normal operation after startup.

Table IX

SCRAM CONDITIONS

1. High Flux No. 1	125% of operating power or power not to exceed burnout heat flux
2. High Flux No. 2	125% of operating power or power not to exceed burnout heat flux
3. Short Period	Less than 5 sec
4. Low Reactor Water Level	Variable set, (5 to 15 ft) above bottom of reactor vessel
5. High Reactor Water Level	Variable set, (5 to 15 ft) above bottom of reactor vessel
6. Low Main-Steam Flow	2000 lb/hr
7. Low Forced-convection Flow	80% of operating flow
8. High Superheat Fuel Temperature*	1300°F
9. High Condenser Pressure	2 psig
10. Superheater Flood Valve, Open	- - -
11. Low Air Pressure	100 psig
12. High Reactor Pressure	640 psig

The following conditions cause a scram, but do not operate an alarm annunciator:

13. Manual Scram (Control Desk)
14. Manual Scram (Reactor Building)
15. Loss of Electrical Power

*This item is developmental and may not be in the reactor.

Key switch No. 3 is provided for bypassing the low steam-flow scram during startup and is put into operation after sufficient steam begins flowing through the superheater. This key switch also places in operation an interlock circuit to open the superheater vent valve by a preset amount on any scram, thereby providing emergency cooling of the superheat fuel elements.

Key switch No. 4 is provided for bypassing the superheater flood valve, open scram. When the reactor is being heated with the superheater flooded, this scram must be inoperative.

Key switch No. 5 is provided for bypassing the low forced-convection low scram. When critical experiments or natural-circulation tests are being performed, it is necessary to operate without forced-convection flow. To test the effects of varying the forced-convection flow rate, the scram operating point can be lowered.

To prevent inadvertent flooding of the superheater, the superheater flood valve can only be opened by a switch which has a spring-loaded guard.

(4) Alarm System

An alarm and annunciator system is provided in the control room to monitor the reactor and process system. All alarms initiate visual and audible signals. Acknowledging the alarm silences the audible signal and retains the visual indication. Clearing of the alarm condition removes the visual signal. All alarm circuits are supplied by the emergency power system. A list of alarms is given in Table X.

Table X

ALARM CONDITIONS

1.	High Reactor Water Level*	Variable set (2 in. below scram setting)
2.	Low Reactor Water Level*	Variable set (2 in. above scram setting)
3.	High Reactor Pressure*	625 psig
4.	Low Reactor Pressure*	15% below operating
5.	Low Condenser Vacuum	15 in. of Hg
6.	High Shield Temperature*	200°F
7.	Low Main-steam Flow*	4000 lb/hr
8.	High Steam-to-Turbine Pressure*	360 psig
9.	High Steam-to-Turbine Temperature*	595°F
10.	Low Main-air Pressure	125 psig
11.	High Steam Temperature*	875°F
12.	High Reactor Ion-exchange Water Temperature	120°F
13.	Low Boron Tank Level	46 lb
14.	High Forced-convection-pump Bearing Temp.*	150°F
15.	High Circulating-pump Bearing Temp*	150°F
16.	Low Demineralized-water-storage Level**	36 in.
17.	Low Feedwater Pressure	650 psig
18.	High Feedwater-storage-tank Level**	60 in.
19.	Low Feedwater-storage-tank Level**	36 in.
20.	Low Turbine-oil Pressure	7 psig
21.	Low Turbine Gland-water Level	12 in.
22.	Condenser Circulating Pump, Off**	- -
23.	Condensate Pump No. 1, Off**	- -
24.	Condensate Pump No. 2, Off**	- -
25.	Cooling Tower, Fan, Off**	- -
26.	High Air Activity, Turbine Building*	***
27.	High Air Activity, Reactor Building*	***
28.	High Reading, Fission Break Monitor*	***
29.	High Area Activity, Turbine Bldg., Main Floor*	***
30.	High Superheat-fuel Temperature*	1250°F
31.	Low Forced-convection Flow*	85% of operating level
32.	Low Circulating-pump Flow*	100 gpm
33.	Turbine Throttle, Tripped	- -
34.	Chamber High-voltage, Off for Safety, Period and Linear Circuits**	- -
35.-50.	Spares	- -

*These circuits do not alarm when electric power to the sensing instrument fails.

**These circuits require electric power to the sensing instrument and alarm on power failure. The other circuits do not require power to the sensing instrument and alarm on closing of a contact.

***Set points to be determined by operating experience.

d. Nuclear Instrumentation and Radiation Monitoring

(1) Nuclear Operating Instrumentation

A diagram of the reactor operating nuclear instrumentation for BORAX V is shown in Fig. 29. Instrument holes in the shielding wall between the access shaft and the reactor pit are shown in Fig. 32 (page 85). Cooling coils are attached to the outside of the instrument-hole liners. Figure 30 shows the operational range of the various nuclear instruments.

The nuclear operating-instrument circuits are listed as follows:

(a) Safety Circuits (High-flux Scram) (Two)

The safety circuits are ANL Model CD104 Safety Trip Circuits. They have a range from 10^{-8} to 10^{-4} amp in decade steps and a trip circuit which is adjustable over a decade range. Outputs are provided at the control console for remote indication, range selection and trip-point adjustment.

(b) Period Meter (One)

The period meter is an ANL Model ICD-6. It has a log current range of 10^{-11} to 10^{-3} amp, a period range of ∞ to $\pm$ 5 sec, and a positive-period trip at 5 sec. Outputs are provided at the recorder panel for remote log-current indication and recording, and remote period indication.

(c) Startup Circuits (Two)

The startup circuits are BF_3 proportional-counter circuits consisting of two pulse amplifiers, two scalers and a dual-channel log count-rate-meter. The log-count-rate-meter has a range of 10^1 to 10^5 counts/sec. Startup detectors will be located to give a suitable count rate at source strength.

(d) Linear Operating Instrument (One)

The linear operating instrument is an ANL Model IA 10 Linear dc amplifier. It has a range of 10^{-12} to 10^{-3} amp in decade steps. Outputs are provided at the control console for remote indication, recording, and range selection.

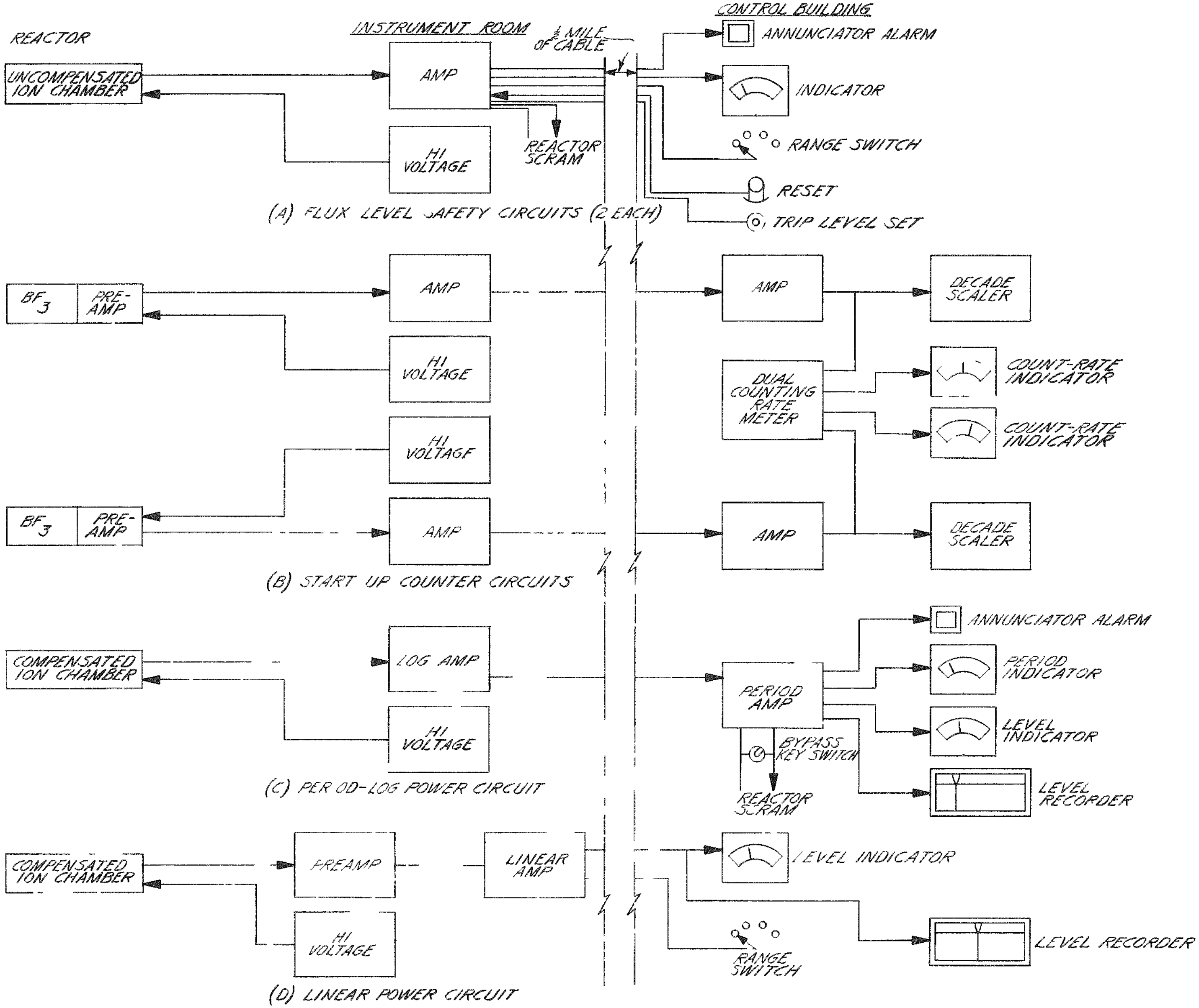


Fig. 29

Nuclear Instrumentation - BORAX V

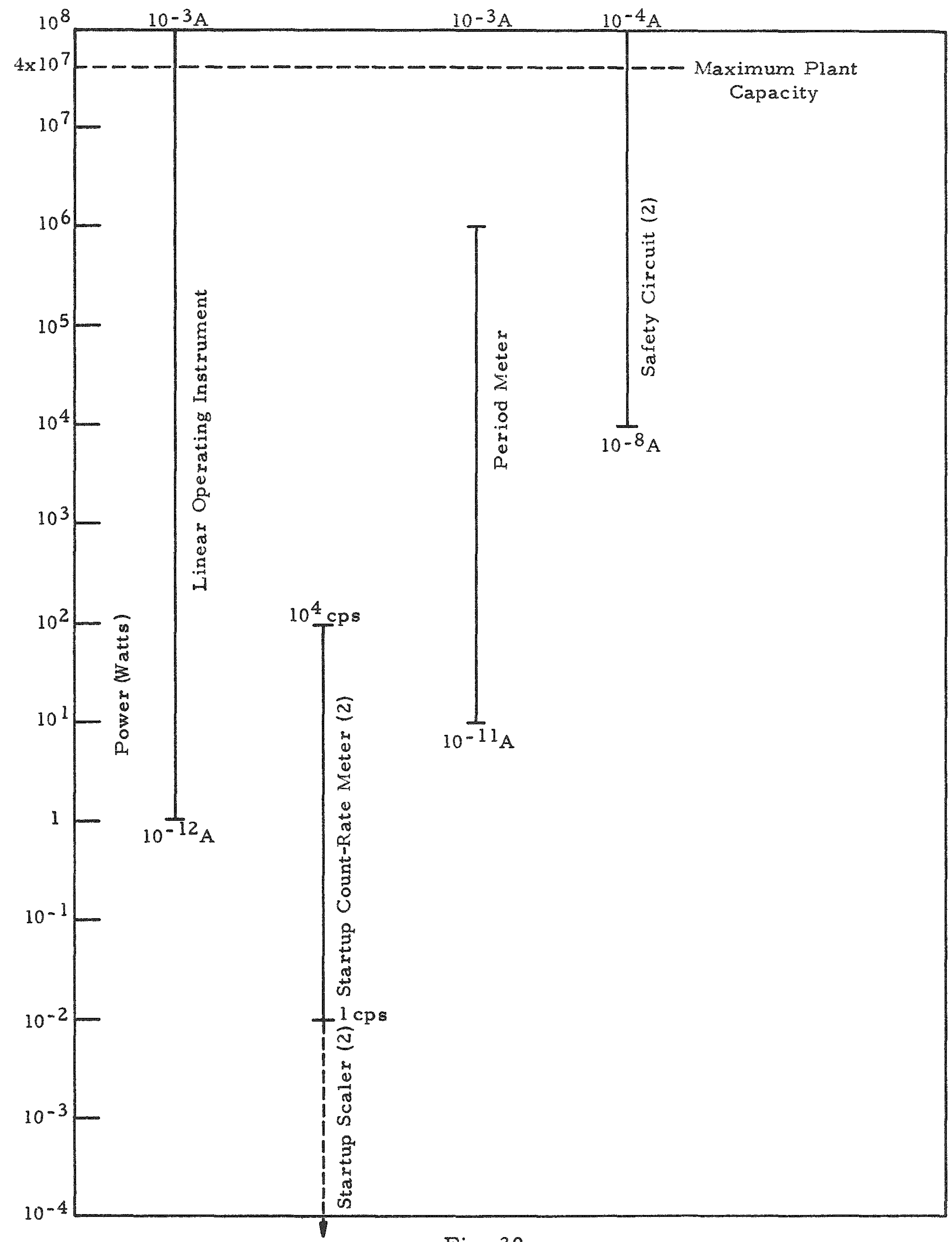


Fig. 30

Nuclear Instrumentation Operating Ranges - BORAX V

(2) Radiation Monitoring System

A diagram of the radiation monitoring system for BORAX V is shown in Fig. 31. The system gives warning of any abnormal release of radioactivity, including that from a fuel element rupture. It is discussed in detail as follows:

(a) Area Monitor (One)

The area monitor is a six-channel, gamma-sensitive monitor, each channel having a range of 0.01 mr/hr to 10 r/hr. In the reactor building, the detectors are located on the main floor and in the equipment pit, access shaft, and subreactor room. In the turbine building, detectors are located in the basement and on the main floor. Outputs are provided from each channel for indication and alarm both locally and in the control building.

(b) Continuous Air Monitors (Two)

A continuous air monitor of the moving-filter type is located on the main floors of the reactor and turbine buildings. Each provides an output locally and to the control building for high air-particulate activity alarm.

(c) Fission Break Monitors (Two)

Two fission break monitors are provided; one monitors the condenser air-ejector system during turbine operation, the other monitors the reactor water through a system of small ion-exchange columns and is used primarily for low-power operation when the turbine is not in use. The two detecting systems are identical, consisting of a scintillation detector, pulse amplifier, single-channel, pulse-height analyzer and log count-rate meter. Outputs from each system provide indication, recording and high-level alarm in the control building.

(d) Airborne Particulate Monitor (One)

During turbine operation an airborne particulate monitor of the fixed-filter type samples and records the particulate activity in the condenser air-ejector system.

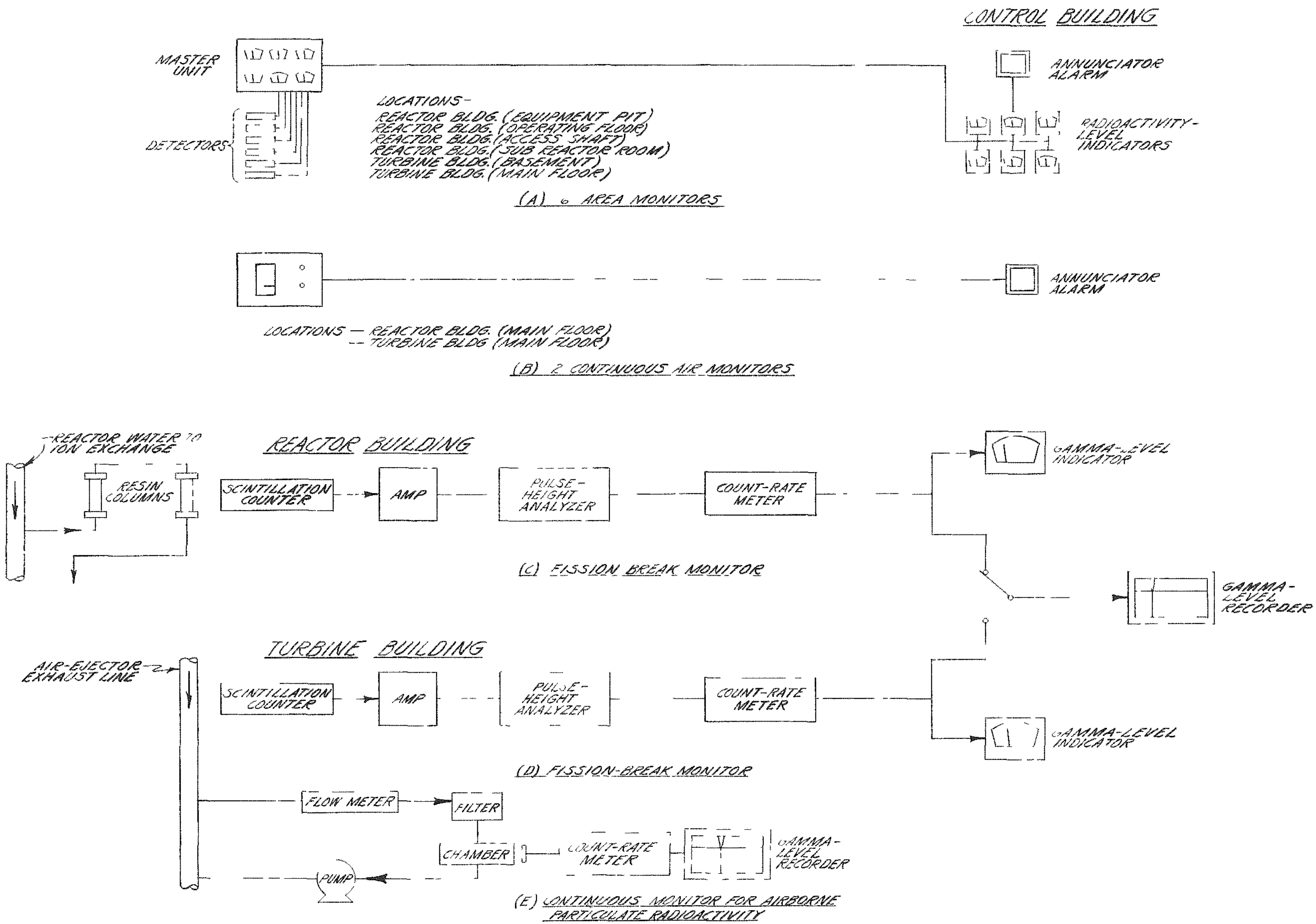


Fig. 31

Radiation Monitors - BORAX V

8. In-Core Instrumentation, Development

a. In-Core Measurement Desired

An intensive effort is being made to develop good in-core instrumentation for BORAX V. The instrumentation discussed here is developmental and experimental, and is not required for operation. To provide operating and proof-of-design types of information, it is intended to measure the following variables:

(1) Flow Rates

(a) Boiling Assembly Entrance
(b) Boiling Assembly Exit
(c) Superheat Assembly
(d) Downcomer

(2) Temperatures

(a) Fuel Temperature of Boiling Rods
(b) Water-channel Temperatures in Boiling Assemblies
(c) Superheater Fuel Plate Temperature
(d) Superheated Steam Exit Temperatures
(e) Saturated Steam Temperatures
(f) Degree of Subcooling

(3) Steam Void Ratio at Boiling Assembly Exit

(4) Neutron Flux

(a) Axial and Radial Distributions
(b) Absolute Magnitudes

(5) Reactor Vessel Pressure

b. Use of In-Core Data

(1) Operating Information and Protection

An attempt will be made to monitor selected fuel-plate temperatures on a periodic basis. A system, which includes an alarm light for each point monitored, is provided. It allows selection of any point for continuous recording and scram protection to provide the reactor operator with an additional aid during startup and shutdown.

(2) Design Information

Through adequate measurements of the previously listed variables it should be possible to determine the following steady-state and transient information:

(a) Power generation per fuel assembly and at various locations.

(b) Flow profile within the core as a function of power, pressure, location, etc.

(c) Boiling fuel rod central temperature and temperature response to power transients.

(d) Steam void ratio at boiling fuel assembly exit as a function of power, saturation pressure, location, etc.

(e) Effective boiling length of boiling fuel assembly for various functions of power, pressure, location, etc.

(f) Recirculation ratio as a function of power, pressure, riser height, core size, etc.

(g) Subcooling of core inlet water for various conditions of reactor operation.

(h) Steam temperature.

(i) Neutron-flux distribution within the core for given steady-state power levels.

(j) Absolute neutron flux at specific locations as a function of feedwater temperature, flow, core size, power, etc.

Overall instrument system performance is aimed at a minimum (3 decibels) sine-wave response up to 2 cycles per second for flow, temperature, and void devices. Ionization chamber systems are designed to follow the maximum anticipated frequency of 10 cycles per second. The required response of the pressure-measuring system will be further evaluated.

c. Description of Proposed Instrumentation

The instrumentation is intended to be installed in two major divisions: that which is attached to two removable boiling fuel assemblies and two removable superheat fuel assemblies, and that which is attached to core structures. The instrumentation in the removable fuel assemblies is intended to allow determination of data at different core locations by means of relocating the instrumented assemblies. The fixed instrumentation within the vessel is intended for measurement of data at fixed locations.

All instrumentation is installed so electrical and instrument taps necessary for operation terminate at a junction box outside the reactor vessel. The junction box is attached to the fuel assembly by means of a hollow steel tube which contains individual pressure tubes and metal-sheathed electrical conductors. This connecting tube offers some support for handling of the fuel assembly, and the individual tubes and electrical leads are brazed or welded into the connecting tube to form a pressure seal. The connecting tube is then welded to an instrument nozzle blind flange to seal the reactor vessel.

Connections from the junction box to external recording or indicating instruments are through "Cannon" or "Bendix"-type watertight connectors. This arrangement allows handling of the instrumented fuel assemblies in a manner similar to the handling of standard fuel assemblies, i.e., it allows removal or replacement of the vessel head over a junction box, or moving the assembly in or out of the vessel with the head in place.

The following describes the instrumentation in terms of the above divisions and types of instruments to be used:

(1) Boiling Fuel Assembly Instrumentation

(a) Flow Rates

Special devices are necessary to measure the flow rates through these assemblies. A modified type of a commercially available turbine-type flowmeter is being evaluated. Because natural circulation in the core makes it necessary to have a low head loss, the turbine-type flowmeter may be unsatisfactory. Therefore, a development program on a velocity-sensing-type flowmeter is being undertaken.

(b) Void Ratio at Exit

Through measurement of volumetric flows at inlet and exit, the steam content of water leaving the exit can be determined. This technique has been proved out-of-pile at ANL using gamma-attenuation methods. An alternate technique, requiring calibration for flow effect, is that of measuring the hydrostatic head of a column of steam-water mixture leaving the assembly.

A choice between the two above methods will be made after additional testing.

(c) Fuel Temperature

To obtain temperatures at the radial centers of selected boiling fuel rods, it is planned to install thermocouples down the axes of the rods to the calculated axial hot spots.

At the present stage of design, the magnitude of the fuel rod center temperature is uncertain. If the predicted temperature is below 2500°F, platinum, platinum-rhodium thermocouples with tantalum sheaths will be used. If the temperature is above 2500°F, it will be necessary to resort to materials and techniques only partially proved. The present plan is to use a 0.040-in. OD, tantalum-sheathed thermocouple made of a tungsten-rhenium pair, insulated with beryllium oxide or other suitable refractory. The selected fuel rods are partially assembled with hollow fuel pellets to allow insertion of the thermocouples. The assembly of fuel tube, thermocouple, and fuel pellets is seal-welded in an inert atmosphere and leak-tested to provide a replaceable fuel rod.

(d) Water-channel Temperatures

By using a vacant fuel rod space for installation of a thermocouple "rake," the axial water-temperature distribution is measured at intervals of approximately 2 in. The thermocouples intended for use are 0.040-in. OD, stainless steel-sheathed, chromel-alumel of matched characteristics, and aged to provide high stability.

(2) Superheat Assembly Instrumentation

The techniques described previously for pressure-sealing, lead terminations, etc., apply also to the superheat fuel assembly instruments. Two instrumented superheat assemblies and one spare are provided to make the following measurements:

(a) Flow Rates

Superheat assembly steam flow is measured by either a modified turbine-type meter or with a venturi tube and external pressure transducer.

(b) Fuel Plate Temperatures

Stainless steel-sheathed chromel-alumel thermocouples are assembled into the edges of selected fuel plates at calculated axial hot spots to measure these temperatures. Welding or brazing these thermocouples into the plates will insure rapid response. This attachment technique should prove easier and more reliable than surface attachments. The 0.040-in. OD thermocouples will lead through the static-steam-insulating spaces to their points of installation.

(3) Fixed Instruments

(a) Superheated Steam Exit Temperatures

Stainless steel-sheathed, chromel-alumel thermocouples,, $\frac{1}{16}$-in. OD, are installed in the entrances to the first and second-pass manifolds from individual superheater assemblies. Thermocouple extension leads are brought out through vessel nozzles below the closure flange. Pressure seals are made for each lead in the same manner as for fuel assembly instrument leads.

(b) Saturated Steam Temperature

To obtain a measurement of reactor vessel steam temperature, 4 thermocouples are installed as fixed instruments in the vessel steam dome. All thermocouples are $\frac{1}{16}$-in. OD, stainless steel-sheathed construction. Two will be chromel-alumel, and 2 iron-constantan. The latter two are available for recording on the process-temperature recorder (TR-2) and the former is used for subcooling data. Leads are brought through a vessel nozzle below the flange.

(c) Subcooling

To determine subcooling of the water in the reactor vessel, 2 thermopiles are installed to sense the temperature differential between the upper vessel region and core inlet region.

For integrity and accuracy these thermopiles are fabricated using multiple stainless steel-sheathed, chromel-alumel thermocouples, $\frac{1}{16}$-in. OD, assembled into a stainless steel tube-type housing. Leads are brought out through a nozzle below the vessel closure flange and terminated in a conventional thermocouple connection head.

The subcooling temperatures will be used along with saturated-steam temperature, feedwater flow, and feedwater temperature to obtain downcomer flow rate by a heat-balance technique.

(d) Neutron Flux

Miniature commercial ionization chambers can be made to sample the neutron flux by means of guide tubes to various core locations. Thus, plots of relative flux at various core locations can be obtained. These same chambers can be used for transient testing.

(e) Reactor Vessel Pressure

A strain gage-type transducer is proposed for measurement of pressure transients within the vessel. The difficulty lies in finding a transducer which is insensitive to gamma and neutron irradiation. Commercially available transducers meet the temperature requirements, but may not meet the radiation-sensitivity requirements.

9. Fuel and Radioactive Material Handling and Storage

a. Boiling Fuel Assemblies

Fuel handling of boiling fuel assemblies is normally done through the five 6-in.-diameter nozzles in the top head of the reactor vessel. This is accomplished by using a modified, existing coffin with hoist, underwater lights, underwater viewer, an offset manipulator, an offset socket wrench and other long-handled tools.

The proposed reloading procedure for a boiling fuel assembly is as follows:

(1) Depressurize the reactor vessel and flood to top of vessel. Remove the shielding slabs and blind flanges on the 6-in. head nozzles as required.

(2) Place the coffin on one nozzle; insert an underwater light, a viewer, a manipulator, etc., through the other nozzles.

(3) Using the proper tool, lift up a chimney and set it aside on top of the other chimneys; operate the hold-down latches with a socket wrench; lift out the hold-down box and set it aside on top of the chimneys.

(4) Using the offset manipulator, grasp the grappler hanging from the coffin hoist and engage the top of the boiling fuel assembly. Lift up the fuel assembly with the manipulator until it is clear of the chimneys, move it laterally, and center it beneath the coffin. During this operation, the coffin hoist cable must also be reeled in simultaneously. When the coffin hoist has taken the load of the fuel assembly, disengage the manipulator, lift the assembly into the coffin, and shut the bottom coffin door.

(5) Using the building crane, transport the coffin into the water storage pit, open the coffin door, and lower the fuel assembly into a fixture in the pit. Disengage the grappler and remove the coffin.

(6) Individual fuel rods of a boiling fuel assembly in the water storage pit may be handled under water by means of underwater lights and a fuel rod manipulator.

(7) Boiling fuel assembly loading procedure is the reverse of the above. For removal of instrumented boiling fuel assemblies, it may be necessary to remove the top head of the reactor vessel.

b. Superheat Fuel Assemblies

The proposed reloading procedure for a superheat fuel assembly is as follows:

(1) Depressurize and flood the reactor vessel to a level just below the main vessel flange. Remove the shielding slabs, insulation over the head-bolts, and the reactor vessel head. Place a floodable, temporary vessel extension or a portable shield plate in place if necessary.

(2) Disconnect the couplings at the top of the superheat fuel assembly risers and bend the flexible manifolds out of the way against the vessel wall. Lift off the hold-down plate along with the Belleville spring.

(3) Using the building crane, center the coffin over a superheat fuel assembly. Using the coffin hoist and grappler, lift a fuel assembly into the coffin. With the building crane transport the coffin and superheat fuel assembly to storage.

(4) Superheat fuel assembly loading procedure is the reverse of the above.

c. Other Reactor Components

The reactor vessel head must also be removed to lift the following large reactor or experimental components from the vessel: chimneys, hold-down boxes, forced-convection baffle, feedwater sparger, core structures, control rods, oscillating rod, and certain in-core instrumentation. The Sb-Be source and some instrumentation can be handled through nozzles in the reactor vessel head.

d. Storage Facilities

Storage facilities for fuel and other radioactive materials are provided as follows:

(1) The existing water-storage pit is used for temporary storage of fuel and other radioactive components. It is also provided with underwater lights, fixtures and

tools for underwater reloading of boiling fuel, boron, water and void rods, and underwater handling and servicing of other radioactive components such as instrumented fuel assemblies, sources, oscillating rods, etc.

(2) A fuel-storage area containing tubes both inside and outside the existing BORAX reactor vessel is located in the old reactor pit. The space between these tubes is filled with gravel and capped with concrete. An existing high-density concrete rolling shield is used to cover this area.

(3) A new dry-storage pit with concrete slab covers is located in the southwest corner of the reactor building. This pit is used to store core structures, forced-convection baffle, feedwater sparger, control rods, and other large, radioactive items. These radioactive items are transferred from the reactor pit to the dry-storage pit by means of the building crane which can be controlled from a remote control station located against the west wall of the reactor building. The remote control station can be shielded by temporary concrete slabs if necessary.

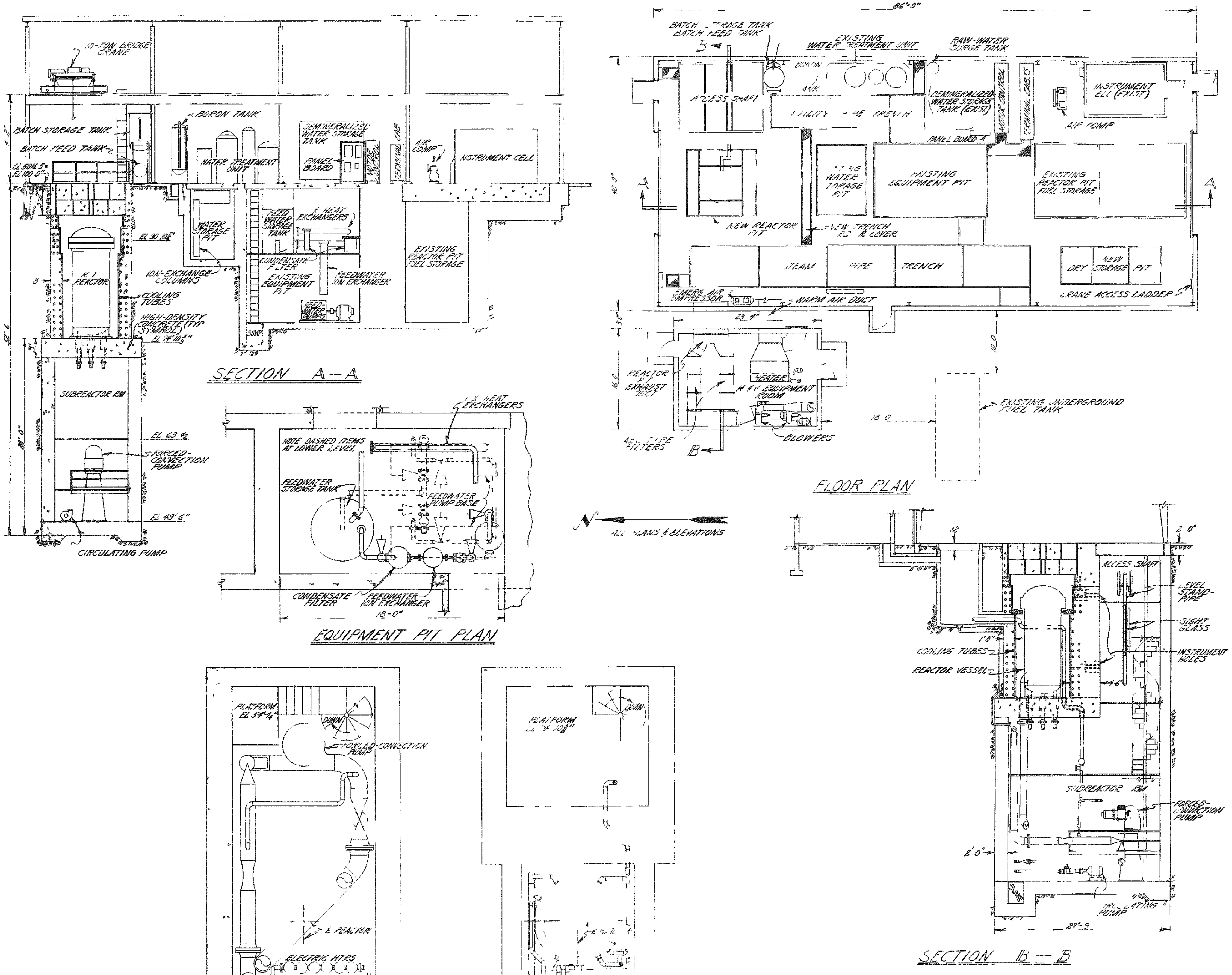
10. Biological Shielding

a. Description

The biological shielding for the BORAX V reactor is shown in Fig. 32. The main floor of the reactor building is made accessible during operation by shielding of the core with a total thickness of 5 ft 2 in. of high-density (215 lb/ft^3) concrete slabs covering the reactor pit, 2 ft of ordinary concrete covering the access shaft, and 1-ft-thick concrete covers over the piping trenches, equipment pit, and dry-storage pit. The access shaft and subreactor room are only partially shielded and are not accessible during operation. The reactor pit is shielded horizontally by $4\frac{1}{2}$ ft of concrete on the access shaft side and by earth on the other three sides. Between the floor of the reactor pit and the ceiling of the subreactor room is a 3-ft-thick slab of high-density concrete. Penetrations into the reactor pit are offset or stepped as required. The piping trench between the reactor building and the turbine building is covered with 6 in. of concrete and 3 ft of earth.

b. Shielding Design

This section is divided into two parts: shielding requirements at power and shielding requirements for access to reactor components after shutdown. The calculations for the first part are based on a



PLAN BETWEEN EL 63-4½" & EL 49'6"   PLAN BETWEEN EL 50'10½" & EL 74'-10½"

NEW REACTOR PIT PLAN

FIG. 32 BORAX V

REACTOR BLDG. FLOOR PLAN & ELEVATIONS

K-130

40-Mw reactor operating full time with personnel being exposed to a maximum of 7.5 mr/hr, the tolerance for a 40-hr week. With respect to shutdown conditions, the reactor is assumed to have operated long enough to be in near equilibrium with its fission products. Higher short-time exposures might be expected in performing work on the reactor after shutdown. In both cases extreme values are used; hence, the shielding is conservative. Three types of radiation are considered; fast neutron (>1 Mev), thermal neutron, and gamma ray. The gammas are further broken up into prompt fission gamma, fission product gammas, and capture gammas formed by the absorption of thermal neutrons in the shielding medium.

Additional shielding provisions will be necessary if experiments are conducted with a separate superheat core in the upper end of the reactor vessel. In areas where high-density concrete is specified, the calculations were made on the basis of magnetite-type concrete (ρ = 215 lb/ft^3).

(1) Shielding at Power

(a) Top of Reactor Pit

The most severe shielding problem occurs when the reactor core is loaded with a central superheater. This permits open radiation channels through the water above the core. The top shield was sized for this condition.

Thermal Neutron Flux. A 1-in.-thick, 1% boron-steel plate inside the head reduces to a negligible amount the thermal neutron flux entering the shield. The effect of the neutron flux which has been thermalized in the shield can be neglected because it is small with respect to the flux of fast neutrons and gamma rays.

Fast Neutron Flux. The fast neutron flux is high because of the lack of moderator in the superheater channels. A fast neutron flux of 3.7×10^{10} n/cm^2-sec from a geometrically reduced plane source was considered incident upon the shielding. Sixty-two in. of high-density concrete reduce the fast neutron flux dose to less than 4 mrem/hr.

Gamma Flux. The prompt fission gammas were divided into 8 energy groups from 0.5 to 8 Mev. The prompt gamma dose was then calculated using each group as a plane source at the reactor shield and summing the individual effects. The secondary gammas resulting from neutron captures were determined in the shield by assuming an exponential source of gammas which followed the fast neutron flux. The final shield thickness of 62 in. of high-density concrete was determined principally by this last form of radiation.

(b) Access Shaft and Trenches

In determining the thickness of the cover over the access shaft and the placement of ducts and trenches, the practice was to assure that a shielding thickness at least equivalent to the number of attenuation lengths through the top shield be maintained in any straight-line direction from the reactor core.

(c) Activation of the Steam

Figure 32 shows the location within the building of the trenches containing the 6-in.-diameter main-steam line, and the smaller superheated steamlines. Barring fission product release, the major activity to be considered is 6.2-Mev gamma rays from the 7.6-sec N^{16}. A specific activity in the steam line of 5.5 x 10^6 Mev/g-sec of water was calculated for this isotope provided all the N^{16} is entrained in the steam. The radiation level was then calculated considering the steam pipe as a line source. One foot of concrete reduces this radiation level to below tolerance.

The condensate returned from the turbine building does not represent a hazard due to the holdup in the feedwater system and the short half-life of N^{16}.

(2) Shielding for Shutdown

Three areas were considered with respect to accessibility after shutdown: a) the access shaft, b) the subreactor room, and c) the top of the reactor vessel. The numerous penetrations into the access pit and the subreactor room complicate the shielding calculation; therefore, the determination of shielding for direct radiation leakage is somewhat of a best guess. Since access to the reactor after shutdown is usually limited to short exposure times, higher-than-normal levels of radiation can be tolerated. For shutdown calculations, the radiation from the fission products was divided into 7 energy groups of gammas from 0.4 to 3.0 Mev. A delay of 2 to 24 hours after shutdown is recommended before access is permitted to the reactor vessel components.

(a) Access Shaft

The shielding wall between the reactor and access shaft is $4\frac{1}{2}$ ft of ordinary concrete and is augmented by the foot of water surrounding the core. All pipe penetrations into the reactor pit are offset by at least $1\frac{1}{2}$ diameters of the duct. Special plugs are designed for the instrument holes to minimize leakage through the shield.

(b) Subreactor Room

The shielding above the subreactor room is 3 ft of high-density concrete. All pipe penetrations are offset by $1\frac{1}{2}$ diameters, and none of these pipes are directly in line with the core. To eliminate streaming along the control-rod drive and forced-convection nozzles, a stepped bushing of steel is inserted, as shown in Figs. 23 and 32. The greatest leakage is through the large, offset steam-line penetration, but the steam line can be filled with water after shutdown.

(c) Access to Top of the Reactor

Access above the reactor with the shield slabs removed requires flooding of the superheater channels with water and raising the level of the water to a depth of 8 ft above the core. As shown in Fig. 32, the radiation which is scattered up the area between the pit walls and the reactor vessel is shielded by a ring of 6-in.-thick lead shielding blocks, mounted just below the flange of the reactor vessel.

c. Heat Generation in the Concrete Biological Shield

An estimation of the heat generated in the concrete biological shield was based on a consideration of the following sources of radiation heating:

(1) Core gammas (both prompt and delayed)

(2) Secondary gammas from neutron absorption in water

(3) Secondary gammas from neutron capture in the reactor vessel shell and intervening thicknesses of concrete wherever applicable.

(4) Fast neutron moderation in the concrete.

A summary of results for radiation heating as a function of distance into the concrete at core center line is given in Fig. 33. A similar summary as a function of elevation and along the floor of the reactor pit is given in Fig. 34. If the reactor pit shield is uncooled, the maximum concrete temperature at 40-Mw reactor power has been calculated to reach about 520°F. Since concrete deteriorates and spalls at this temperature, a shield-cooling system is provided as shown in Fig. 32 and described in Section III, B, 10.

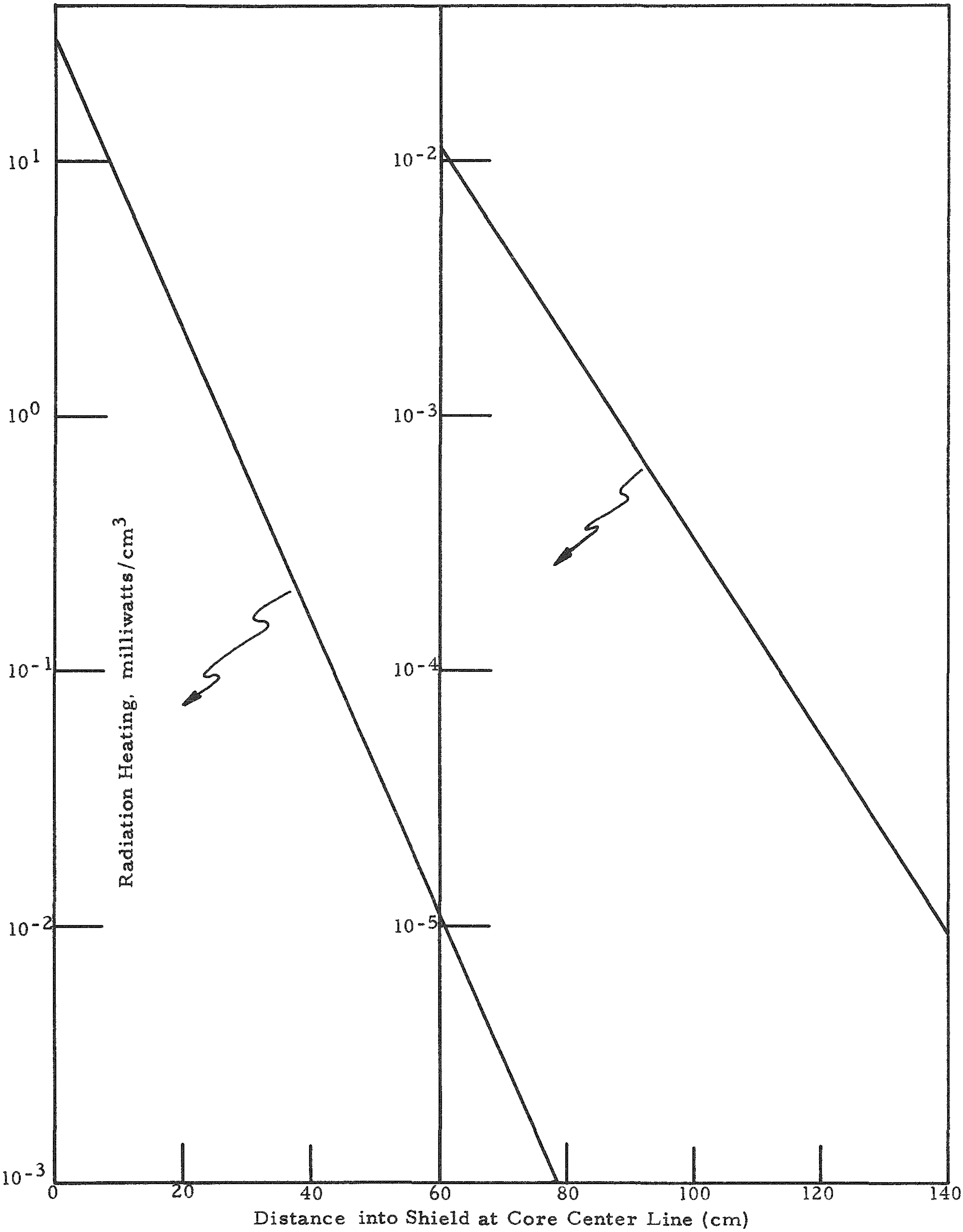


Fig. 33

Radiation Heating of Shield Versus Distance into Concrete - BORAX V

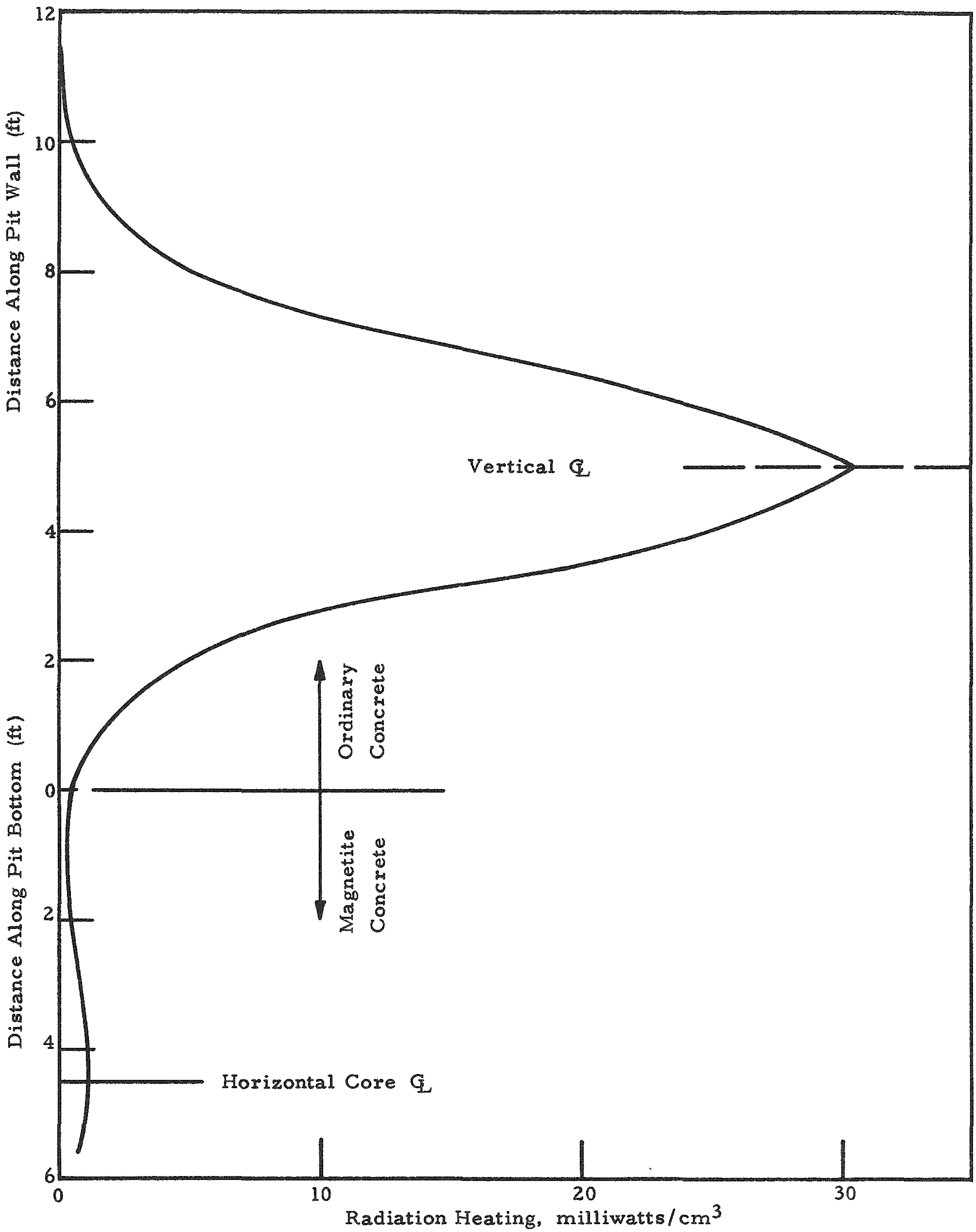


Fig. 34

Radiation Heating of Shield along Surface of Reactor Pit Wall - BORAX V

B. Process Piping Systems

1. Superheater Vent, Flood, and Drain Systems

a. Superheater Vent System

The function of the superheater vent system is to insure that a sufficient flow of coolant steam is maintained through the superheater during reactor startup and shutdown operation, to prevent overheating. The superheater vent system consists of flowmeter FI-4 and a manual or automatic air-operated superheater-vent valve (HIC-5aV) connected to the main-steam piping, as shown in Fig. 35. This system in the "vent" condition permits steam to flow from the steam dome of the reactor vessel through the superheater into the main-steam piping and then to atmosphere.

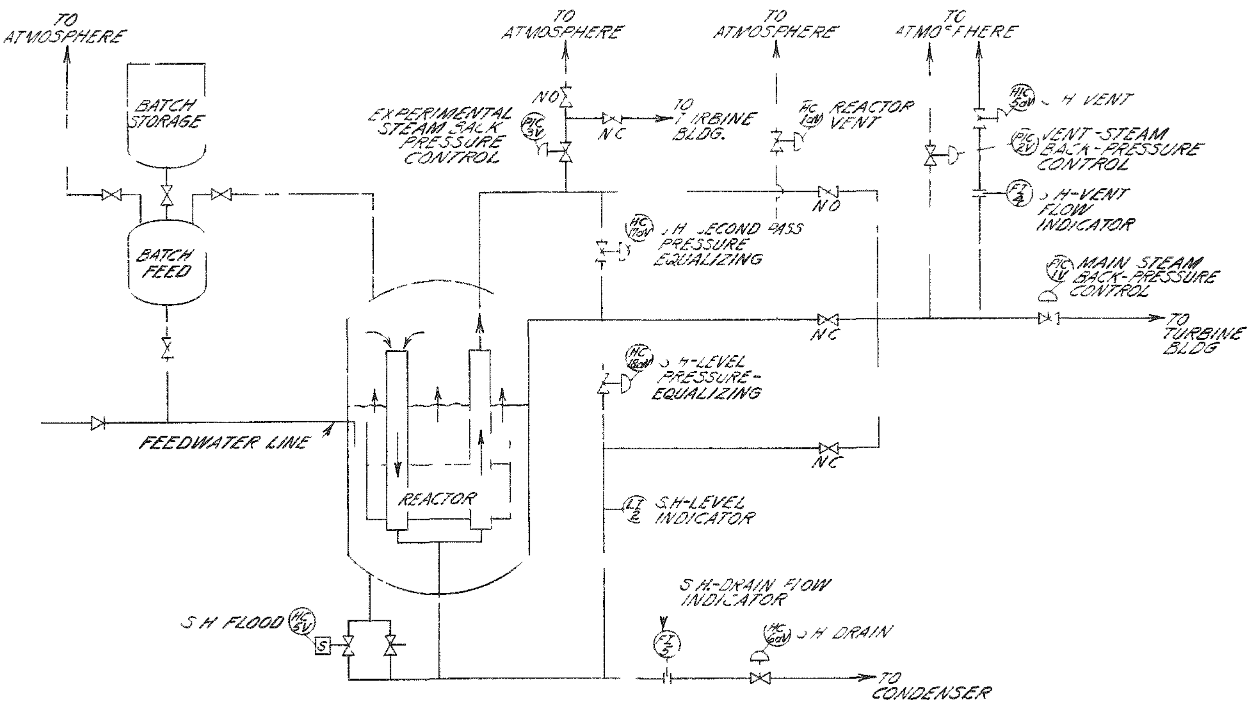


Fig. 35
Superheater Startup and Shutdown System - BORAX V

Valve operation is normally elective and adjustable for flows of from 0 to 8000 lb/hr. In event of a scram, the valve is opened automatically to a preset flow position calculated to provide adequate immediate cooling. The material of construction is Type 304 stainless steel.

This is an essential system and the valves and controls are therefore connected to the emergency power system.

b. Superheater Flood System

The superheater flood system is made up of superheater flood valve (HC-5V) (solenoid trip and air reset), a manual gate valve which parallels valve HC-5V, two air-operated pressure-equalizing valves (HC-17aV and HC-18aV), superheater water-level indicator (LI-2), and a piped connection to one of the reactor vessel forced convection inlet lines, all as shown in Fig. 35.

This system permits flooding of the superheater for additional cooling, if desired, during the shutdown cycle. Calculations indicate that flooding will be accomplished in about one minute.

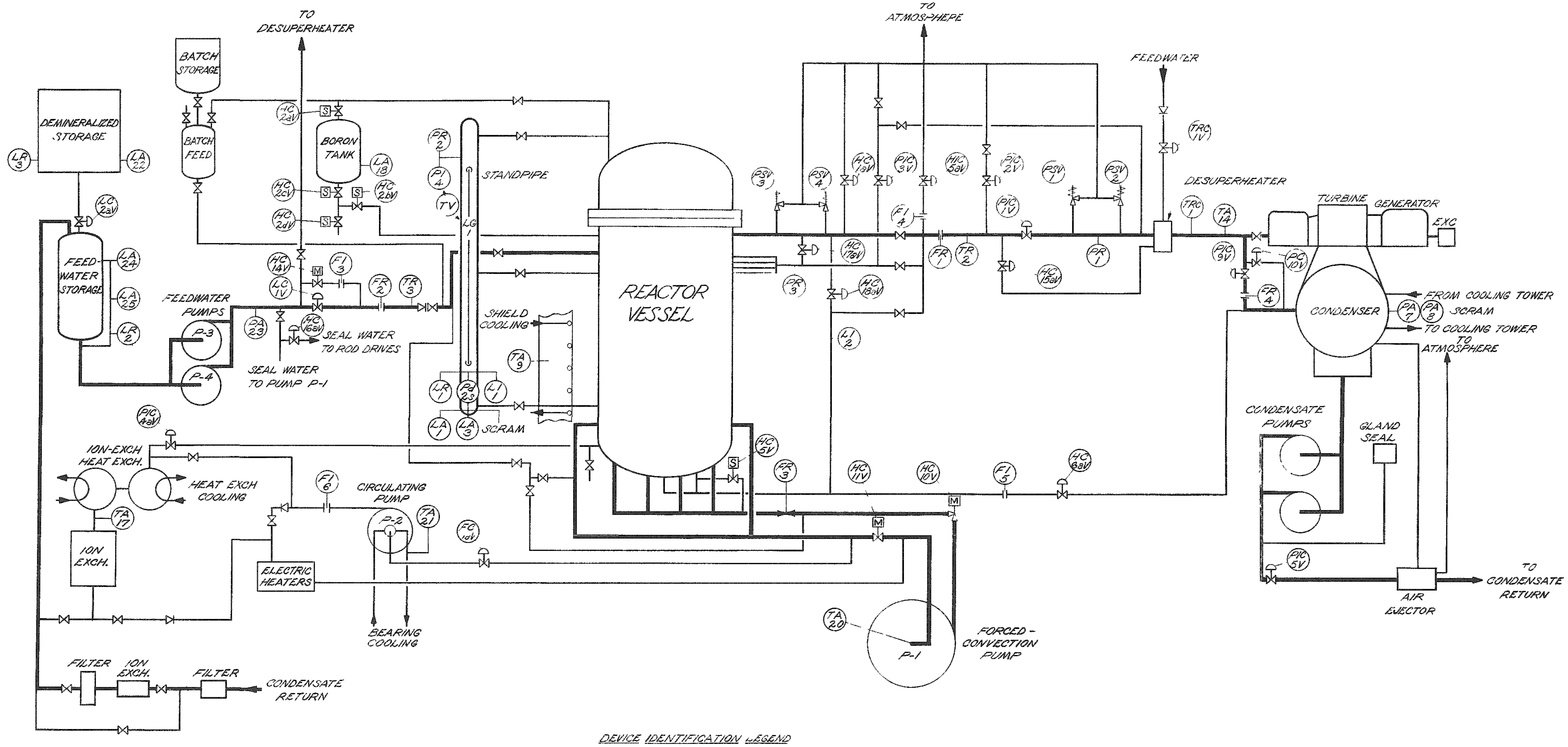
In the "flood" condition, the superheater vent valve is closed and the flood valve and two pressure-equalizing valves are open. The flood valve permits water to flow by gravity from the boiling section of the reactor vessel, through the forced-convection line and into the superheater. The superheater second-pass pressure-equalizing valve (HC-17aV) assures that pressure in the superheater second pass is the same as that in the boiling section during flooding. Superheater level-indicator pressure-equalizing valve (HC-18aV) equalizes the pressure across level indicator LI-2, permitting a true superheater water-level indication when the flood valve is reclosed. The manual gate valve serves the same purpose as the superheater flood valve in case of failure of the flood valve and is manipulated from the operating floor by means of an extension rod. The material of construction is Type 304 stainless steel.

This, too, is an essential system and therefore the valves and controls are connected to the emergency power systems.

c. Superheater Drain System

The superheater drain system consists of an air-operated superheater drain valve (HC-6aV), a superheater drain-flow indicator (FI-5), the previously mentioned pressure-equalizing valves and level indicator LI-2, and a piped connection to the turbine condenser, all as indicated on Figs. 35 and 36.

When the reactor has been brought to pressure, the flooded superheater must be drained prior to the start of steam flow. In the "drain" condition, the flood valve is closed and the drain valve and two pressure-equalizing valves are open. The drain valve permits water to flow out of the superheater at about 120 gpm to the turbine condenser where it is received and cooled. The superheater pressure-equalizing valve provides a supply of boiling section steam at the exit of the superheater second pass thus assures proper blowing down of the second pass.



DEVICE IDENTIFICATION LEGEND
FOR
MAIN PROCESS FLOW AND INSTRUMENTATION

FI-3	LOW RATE-FEEDWATER-FLOW INDICATOR
FI-4	SUPERHEATER-VENT FLOW INDICATOR
FI-5	SUPERHEATER-DRAIN FLOW INDICATOR
FI-6	REACTOR CIRCULATING PUMP-FLOW INDICATOR
FR-1	MAIN-STEAM-FLOW RECORDER
FR-2	MAIN-FEEDWATER-FLOW RECORDER
FR-3	FORCED-CONVECTION-PUMP FLOW RECORDER
FR-4	TURBINE-BYPASS STEAM-FLOW RECORDER
PI-4	REACTOR-VESSEL-PRESSURE INDICATOR
PR-1	STEAM-TO-TURBINE PRESSURE RECORDER
PR-2	REACTOR-VESSEL-PRESSURE RECORDER
TR-2	REACTOR WATER OR STEAM-TEMPERATURE RECORDER
TR-3	FEEDWATER-TEMPERATURE RECORDER
TRC-1	STEAM-TO-TURBINE TEMPERATURE RECORDER
LR-1	REACTOR-WATER-LEVEL RECORDER
LR-2	FEEDWATER-STORAGE-LEVEL RECORDER
LR-3	DEMINERALIZED-WATER-STORAGE LEVEL RECORDER
LI-1	REACTOR-WATER-LEVEL INDICATOR
LI-2	SUPERHEATER-WATER-LEVEL INDICATOR
TV	TELEVISION MONITOR
LG-1	REACTOR-WATER-LEVEL GLASS
Pd-2s	HIGH AND LOW REACTOR-WATER-LEVEL SCRAMS
LA-1	HIGH REACTOR-WATER-LEVEL ALARM
LA-3	LOW REACTOR-WATER-LEVEL ALARM
LA-18	LOW BORON-TANK-LEVEL ALARM
LA-22	LOW DEMINERALIZED-WATER-STORAGE LEVEL ALARM
LA-24	HIGH FEEDWATER-STORAGE-LEVEL ALARM
LA-25	LOW FEEDWATER-STORAGE-LEVEL ALARM
PA-7	HIGH CONDENSER-PRESSURE SCRAM
PA-8	LOW CONDENSER-VACUUM ALARM
PA-23	LOW FEEDWATER-PRESSURE ALARM
TA-9	HIGH SHIELD-WALL-TEMPERATURE ALARM
TA-14	HIGH STEAM-TO-TURBINE-TEMPERATURE ALARM
PR-3	SUPERHEATER-PRESSURE-DROP RECORDER
TA-17	HIGH ION-EXCHANGE-TEMPERATURE ALARM
TA-20	HIGH FORCED-CONVECTION-PUMP BEARING-TEMPERATURE ALARM
TA-21	HIGH CIRCULATING-PUMP-BEARING COOLING-TEMPERATURE ALARM
HC-1aV	REACTOR-VENT VALVE
HC-2aV	BORON TANK PRESSURE-EQUALIZING VALVE
HC-2bV	BORON-TO-REACTOR VALVE
HC-2cV	BORON-SEAL VALVE
HC-2dV	BORON-LEAKAGE VALVE
HC-5V	SUPERHEATER-FLOOD VALVE
HC-6aV	SUPERHEATER-DRAIN VALVE
HC-10V	FORCED-CONVECTION-PUMP-DISCHARGE VALVE
HC-11V	FORCED-CONVECTION-PUMP-SUCTION VALVE
HC-14V	LOW RATE FEEDWATER-CONTROL VALVE
HC-15aV	DESUPERHEATER ATOMIZING-STEAM VALVE
HC-16aV	SEAL-WATER SUPPLY VALVE
HC-17aV	SUPERHEATER SECOND-PASS PRESSURE-EQUALIZING VALVE
HC-18aV	SUPERHEATER LEVEL-INDICATOR PRESSURE-EQUALIZING VALVE
TRC-1V	DESUPERHEATER FEEDWATER-CONTROL VALVE
PS-1-2-3-4	PRESSURE-RELIEF VALVES
FC-1aV	CIRCULATING-PUMP-SUCTION VALVE
LC-2aV	DEMINERALIZED-WATER-STORAGE TO FEEDWATER-STORAGE CONTROL VALVE
LC-1V	MAIN FEEDWATER-CONTROL VALVE
PIC-1V	MAIN STEAM BACK-PRESSURE-CONTROL VALVE
PIC-2V	VENT STEAM BACK-PRESSURE-CONTROL VALVE
PIC-3V	EXPERIMENTAL STEAM BACK-PRESSURE-CONTROL VALVE
PIC-4aV	ION-EXCHANGE PRESSURE-CONTROL VALVE
PIC 5V	GLAND-SEAL-WATER PRESSURE-CONTROL VALVE
PIC-9V	TURBINE-BYPASS PRESSURE-CONTROL VALVE
PC-10V	STEAM-TO-CONDENSER DUMP VALVE
HIC-5aV	SUPERHEATER-VENT VALVE

Fig. 36
Main Process and Instrumentation Systems - BORAX V

Drainage flow is indicated by flowmeter FI-5 and complete drainage is first evidenced by a change in indicated flow rate and may be confirmed by closing the superheater drain valve and then observing the superheater level indicator (LI-2).

The material of construction is Type 304 stainless steel, except for the piping between the superheater drain valve and the condenser which is carbon steel.

2. Batch Feed System

The superheater requires a continued flow of steam after reactor shutdown for as long as the decay heat remains above 0.4 Mw. As a result, an assured means of making up evaporation losses is required in order to maintain an adequate reactor water level.

The batch emergency feedwater system is provided to fulfill this need in the event of power or feedpump failure. The system consists of a 350-gal plastic-lined, steel batch-storage tank and a 150-gal Type 304 stainless steel, batch-feed pressure tank, piped and valved as shown in Fig. 35. Piping and valves are Type 304 stainless steel. Both tanks are normally filled with demineralized water.

To initiate water addition, reactor steam is manually admitted to the top of the 150-gal pressure tank and the line from the tank drain to the reactor feedwater line is valved open. When the tank pressure is the same as reactor pressure, water flows by gravity into the reactor. After the tank has drained, it is manually isolated from the reactor, vented to atmosphere, refilled by gravity from the 350-gal storage tank, and the pressurizing and drain cycle is repeated. In this manner, makeup equivalent to about $2\frac{1}{2}$ ft of reactor water can be added in approximately one hour. If necessary, raw water can be added to the reactor vessel in this manner, and under extreme conditions, the contents of the boron-addition tank can also be fed into the reactor vessel.

3. Boron-addition System

The boron-addition system is a reserve means of poisoning the reactor and making it subcritical. It is a stainless steel system, consisting of a 270-gal pressure tank with two 12-kw immersion heaters and temperature controls, and an assembly of piping and valves, all arranged as shown in Fig. 36. The tank is normally charged with 20 kg of boric acid, H_3BO_3, dissolved in 200 gal of demineralized water.

In the storage condition, boron-tank pressure-equalizing trip valve (HC-2aV), boron-to-reactor valve (HC-2bV), and boron-seal valve

(HC-2cV) are closed and boron-leakage valve (HC-2dV) is open. In this condition, the tank is isolated from the reactor vessel and, with the boron-leakage valve open, chance leakage of boric acid into the reactor vessel is prevented. Each of these valves is a solenoid-trip type.

Poisoning is initiated at the control building by a single control which causes valve HC-2dV to trip closed and valves HC-2aV, HC-2bV, and HC-2cV to trip open, admitting reactor steam into the top of the tank and establishing a drain route from tank to reactor. After the tank has reached reactor pressure, the boric acid solution flows by gravity into the reactor vessel.

A 20-kg charge of boric acid is calculated to be adequate for most, but not all, of the possible core loadings. When a greater poisoning capacity is needed, the immersion heaters permit storing of a heated charge containing more than 100 kg of boric acid.

Even though this is a slow-acting system, the valves and controls of this system are connected to the emergency power supply. In addition, each of the trip valves is equipped with an auxiliary manual trip. Since inadvertent poisoning creates a special problem, the electric control circuitry has been arranged to provide the best degree of protection against accidental dumping.

4. Forced-convection System

During the course of experiments to investigate higher core power densities and coolant flow velocities, a forced-convection system will be utilized on BORAX V. This stainless steel system, as shown on Fig. 36, includes a Worthington 10,000-gpm centrifugal pump, vertically mounted below the reactor in the subreactor room. The pump has a NPSH of 14 ft, a total discharge head of 193 ft, and is driven by a 450-hp squirrel-cage induction motor.

Forced-convection flow rates may be remotely regulated by means of motor-operated, forced-convection-pump discharge valve (HC-10V). A venturi-type flowmeter is provided, with remote indication in the reactor control room. A large portion of the piping and controls from the unused EBWR forced-circulation system will be utilized in the BORAX V system.

A shown on Fig. 23, the forced-convection outlet from the reactor vessel is through two 14-in. nozzles near the bottom of the downcomer area. These 14-in. nozzles extend through the reactor pit floor slab into the subreactor room and are flanged to 14-in. piping which enters a common 16-in. pump-suction line. The pump discharge passes through a 14-in.

line and is then equally distributed into the bottom of the reactor vessel through four 6-in., forced-convection, reactor inlet nozzles. A diffusion ring at this point breaks down the nozzle effects and equalizes flow distribution before the water enters the core.

It is necessary to have the forced-convection baffle installed in the reactor vessel during forced-convection experiments to provide the necessary division between vessel inlet and outlet flow paths.

5. Steam Systems

The BORAX V steam system is unusually flexible, as is shown on Fig. 36. It is capable of removing saturated steam from the pressure vessel steam dome and superheated steam from nozzle outlets in both the steam dome area and the bottom of the vessel. The steam produced can be utilized in the turbogenerator unit, bypassed to the condenser, and/or exhausted to the atmosphere, as necessary in the performance of various experiments.

A new steam main, completely entrenched from the reactor to the turbine building, is provided in place of the old overhead line which has been removed. The reactor is operable at all pressures up to 600 psig. Normal turbine inlet steam conditions are 350 psig with a maximum allowable superheat of 150°F, or a temperature of 585°F. At reactor pressures between 350 and 600 psig a pressure reduction upstream from the turbine must be provided, and at steam temperatures above 585°F desuperheating is necessary. Below 250 psig the turbine is inoperable and steam flow must be bypassed directly to the condenser or to the atmosphere. Steam produced in excess of approximately 20 thermal Mw must be vented to atmosphere, since this is the maximum capacity of the existing turbine-generator-condenser system.

Materials of construction are equal to A-376, Type 304 stainless steel, A-335, grade P-11 chrome-moly steel and A-106, grade B carbon steel, depending on location and service conditions.

The major components of the BORAX V steam system are as follows:

(a) Internal and External Superheat Piping

It is proposed that both central and peripheral superheater fuel elements be tested in the BORAX V reactor. For the two-pass reference design superheater, the steam produced is removed, as shown in Fig. 9 and 11, through stainless steel flexible tubing coupled to the tops of the individual fuel elements. The flexible tubes are fastened to lengths of solid, double-wall pipe attached to the four 3-in. superheated steam vessel outlet nozzles by temperature-compensating couplings. The four 3-in. superheated-steam outlet lines are joined in a common manifold connected to the main 6-in. steam line.

Should single-pass superheat be produced, it will be removed through the reactor vessel forced-convection inlet nozzles via plenum chambers and piping attached to the core support plate. A 6-in. superheated-steam line will then carry the steam from the subreactor room up through the reactor pit into the main 6-in. steam line in the pipe trench.

(b) Main Steam Line

The main 6-in. steam line extends from the reactor pressure vessel through the pipe trench to the turbine building and turbogenerator unit. A large expansion loop is located in the trench between buildings to allow for pipe expansion. Located along the main steam line are the following: flow, pressure, and temperature-measuring instruments, a 3-ft removable test section, a calorimeter, sampling and sample injection lines, control valves, shut-off valves, and many branch lines to make a completely flexible steam system.

(c) Back-pressure Stations

The main-steam, back-pressure control valve (PIC-1V) located in the main-steam line is designed to operate as a remote, variable-set valve. It is required to flow saturated or superheated steam at capacities up to 60,000 lb/hr at a set-point between 300 and 600 psig. This valve is sensitive to upstream pressure and opens only when reactor pressure is increased above the preset pressure. In this manner it serves as a reducing station and a reactor containment device.

At powers above 20 Mw, or when it is desirable to flow steam directly to the atmosphere, a vent line with vent-steam, back-pressure control valve (PIC-2V), upstream from the main-steam, back-pressure control valve, is utilized. This valve and line is designed to flow steam at 0 to 60,000 lb/hr; it has a remotely adjustable, variable set-point between 100 and 600 psig, and it operates in the same manner as the main-steam, back-pressure control valve.

(d) Safety Valves and Vents

Inasmuch as the main-steam system is normally operated at two different pressures, two sets of dual-safety valves are required. The two valves for the reactor vessel and high-pressure steam line have settings of 650 and 670 psig, respectively, with a combined capacity of 120,000 lb/hr of saturated steam. The two safety valves for the low-pressure portion of the steam system have settings of 388 and 400 psig and also have a combined capacity of 120,000 lb/hr. All four discharge to the atmosphere through a 6-in. stack on the west side of the reactor building.

A 2-in. superheated-steam vent line with superheater-vent valve (HIC-5aV) is installed as previously described to insure that steam flow through the superheater continues on shutdown. An additional reactor-vent system allows purging and blowdown of reactor pressure via the reactor-vent valve (HC-1aV).

For independent, single-fuel assembly experiments, a separate flow path, flow measurement, and the experimental-steam, back-pressure control valve (PIC-3V) are provided for steam flowing through two of the superheated steam outlet lines.

(e) Desuperheater Station

Turbine-temperature limitations require that a desuperheater (DE-1), as shown on Fig. 36, be installed in the main steam line to insure a maximum steam-to-turbine temperature of 585° F. A feedwater line is piped to the desuperheater station and steam is provided via the desuperheater atomizing-steam valve (HC-15aV) to improve feedwater atomization at low flows.

(f) Steam Bypass and Separation

The existing turbine-bypass, pressure-control valve (PIC-9V) has been retained and can be set at the turbine building to control turbine throttle pressure. This valve normally bypasses steam directly to the condenser to maintain a set pressure so incremental changes in turbine load affects bypass flow rate.

A new steam separator is located immediately upstream from the main turbine stop valve for quality improvement during saturated-steam operation.

(g) Turbine-Generator-Condenser Unit

The existing turbine-generator-condenser unit at the BORAX site is utilized, with a few minor modifications, as a heat sink for the BORAX V reactor.

The turbine-generator is a 1926 Westinghouse unit rated at 3750 kva. The main, water-cooled, surface condenser is located directly beneath and supported from the turbine. The existing 6500-gpm Allis-Chalmers main-circulating pump is used to circulate cooling water from the condenser to the existing cooling tower. Return flow from the tower is by gravity. A parallel loop in the cooling-water system circulates water through a water rheostat which is used to apply electrical loads, as desired, to the main generator.

The existing set of two-stage air ejectors, operated from the main steam supply, are used to create the normal operating pressure of about 4 in. of Hg absolute on the main condenser.

Although the turbine gland-seal water continues to be supplied from the existing gland-seal-water storage tank, a modification to the system is made. Gland-seal-water control valve (PIC-5V) is installed in the main condensate-return line and maintains the necessary head to elevate condensate to the tank.

A new air-ejector, atmospheric-exhaust line is installed to expel noncondensable condenser gases to the atmosphere. A new filter and a blower are located in the atmospheric exhaust line so that the filter remains under negative pressure during operation. This prevents leakage of radioactive gases into the turbine building and insures that they are expelled to the atmosphere.

6. Condensate and Feedwater System

The completely closed condensate and feedwater system can be seen in Fig 36. The two existing condensate pumps and one condensate booster pump remove condensate from the condenser hotwell and pump it back to the reactor building via a new condensate-return line located in the pipe trench. After passing in series through an existing condensate filter, a full-flow, mixed-bed condensate demineralizer, and a new filter, the condensate enters the existing stainless steel feedwater-storage tank which is relocated to the upper level of the equipment pit. The demineralizer may be bypassed if desired. Two new 150-gpm centrifugal feedwater pumps are located on the lower level of the equipment pit, taking suction from the storage tank and discharging into a common main-feedwater line.

A feedwater-regulating system utilizing automatic and/or manual control permits maintenance of reactor water at desired levels over a wide range of powers. The main-feedwater-control valve (LC-1V) is required to pass the design flow of 275 gpm. In parallel to the normal main feedwater control valve is a motor-operated, low-rate, feedwater-control valve (HC-14V) to provide manual control at low flow rates (from 0 to 30 gpm).

The feedwater line downstream from the regulating valves is piped so that there are three alternate flow paths for the water to enter the reactor vessel. By manipulation of the proper valves feedwater may enter the vessel through the internal feedwater sparger ring, the forced-convection inlet piping, or the forced-convection outlet piping.

Feedwater is also supplied to the steam desuperheater station DE-1. The desuperheater, located in the pipe trench on the west side of the reactor building, is equipped with an automatic control valve (TRC-1V) which is capable of passing a maximum of 15 gpm of desuperheating feedwater. This results in a reduction of steam temperature from 850°F to 585°F, for a maximum of 60,000 lb/hr of 600-psig superheated steam.

The reactor seal-water system also originates at the feedwater regulating-valve header. This 1-in. line is piped into the subreactor room, supplying seal water to the forced-convection pump glands and the control-rod-drive seals. Flow to the control-rod seals first passes through a filter, through a seal-water supply valve (HC-16aV), and then through individual rod-seal, inlet and outlet valves and rotameters, thus allowing accurate control and monitoring of seal water for the control rod drives.

Reactor leakage from the control-rod seals is piped to the subreactor room sump, while the return from the forced-convection pump seals, approximately 6 gpm, is returned to the feedwater storage tank.

The range of reactor power is normally from 0 to 20 Mw, with a maximum short duration capacity of 40 Mw. This requirement, together with the needs of the ion-exchange, desuperheater, and seal systems, establishes the need for two 150-gpm feed pumps.

7. Reactor Preheat System

The reactor preheat system, shown on Fig. 36, is composed of a bank of immersion-type electric heaters with a rated capacity of 144 kw, a 150-gpm circulating pump, both located in the subreactor room, and suitable piping valves, and equipment to make a complete system.

This system is used during startup operations or whenever it is desirable to heat the reactor water without nuclear heat. It is especially useful during initial plant startup since it allows plant checkout and hot-critical experiments to take place without undue irradiation problems.

A secondary function of the system is to heat uniformly and simultaneously the reactor vessel and forced-convection piping, when necessary, to prevent excessive stresses due to differential expansion.

With the forced-convection piping in place, the circulating pump suction line is connected to the forced-convection-pump suction piping immediately upstream from the closed pump suction valve. The circulating-pump discharge passes through the heater bank and flows back into the forced-convection-pump suction line immediately downstream from the pump-suction valve. Circulating water then passes through the forced-convection pump and returns to the reactor vessel via the reactor forced-convection inlet piping. This arrangement insures flow and temperature equalization throughout the system.

When the forced-convection piping is not in place it is necessary to make use of the spare, 2-in. preheat-water line and one of the reactor vessel forced-convection inlet nozzles as connecting points for the circulating-pump suction and discharge lines.

A third function of the circulating pump is to move reactor water through the ion-exchange system as described in Section III, B, 8.

8. Reactor Ion-exchange System

The reactor ion-exchange system is designed to clean reactor water continuously, with the reactor operating or shut down, at a range of flow rates from 0 to 20 gpm. This system, as shown on Fig. 36, is required to maintain reactor water quality, pH, and chloride concentration within very close limits.

Another function of this system is removal of boric acid from reactor water when necessary.

With the reactor shut down or at very low pressures, it is necessary to use the circulating pump to force reactor water through the ion-exchange system. Whenever reactor pressure is greater than about 50 psig, it is necessary to secure the circulating pump and close the circulating-pump suction valve (FC-1aV) in order to protect the low-pressure heat exchangers and ion-exchange columns. During these periods, ion-exchange flow is maintained by reactor pressure alone.

When the circulating pump is used, the ion-exchange supply water flows from a branch line at the pump discharge to the existing heat exchangers which are relocated to the upper level of the equipment pit. A globe valve at the heat exchanger inlet is used for throttling purposes to adjust the flow rate as desired.

When reactor pressure is available to create ion-exchange supply flow, water leaves the reactor through the 1-in. reactor-blowdown line and flows to the heat exchangers in the equipment pit. Flow rates through the system at from 0 to 20 gpm are adjustable through use of the ion-exchange pressure-control valve (PIC-4aV) located at the inlet to the heat exchangers. A pressure breakdown orifice downstream from the control valve reduces supply pressure as necessary for system protection. Adjustment of the control valve is accomplished at a control panel located on the east side of the reactor building operating floor.

Ion-exchange supply water flows in series through the first two-pass heat exchanger and the second four-pass heat exchanger and is cooled to a maximum of 120°F, which is the recommended maximum temperature for the ion-exchange resins. Water then flows directly to the ion-exchange columns in the water-storage pit. The system can be manually adjusted to give flow through cation, anion, or mixed-bed columns as desired. This discharge from the columns is then sent back to the reactor vessel if the circulating pump is in use, or to the feedwater-storage tank if reactor pressure is creating the ion-exchange flow.

9. Demineralized-water System

A $1\frac{1}{2}$-in. raw-water supply line connects with the existing demineralized-water makeup system located in the reactor building. Flow through this makeup system is rated at a maximum of 70 gpm.

The demineralized water produced is stored in an existing 5000-gal demineralized-water storage tank in the reactor building. From here makeup water is gravity-fed automatically by control valve LC-2aV to maintain a proper operating level in the feedwater-storage tank.

When operating at 40 Mw, it is necessary to vent approximately 20 Mw of steam to atmosphere because of the limitation of the existing steam-consuming equipment. This in turn creates a loss of about 120 gpm of feedwater, which must be made up from the demineralized-water system. If this system is operated at a rate of 60 gpm, there remains 60 gpm to be made up from the 5000-gal storage tank. Therefore, this is the limiting factor which establishes the duration of 40-Mw operation at about $1\frac{1}{2}$ hr.

10. Raw-water Systems

A raw-water distribution header, located on the east wall of the reactor building, is supplied by a 4-in. supply line from the EBR-I well. Distribution lines from this header include: a $1\frac{1}{2}$-in. line to the demineralized-water make-up system, a $\frac{3}{4}$-in. line to provide cooling water to the sample test station, a 2-in. line to feed the reactor-shield cooling system and supply gland-cooling water to the circulating pump, a 3-in. cooling-water supply line to flow in series through the two ion-exchange heat exchangers, and a $1\frac{1}{2}$-in. line to supply water to the water-storage pit. In addition, the raw-water supply header contains a $1\frac{1}{2}$-in. firehose connection and three $\frac{3}{4}$-in. hose connections.

It is estimated that a raw-water supply of 175 gpm is sufficient to fulfill the total demand of the above systems. There is at least 200 gpm of raw water available for use at the BORAX site.

The shield-cooling system, mentioned above and shown on Figs. 32 and 36, requires a maximum flow of approximately 30 gpm. This system includes: a series of 1-in. pipes located about the horizontal reactor centerline against the inside steel surface of the reactor pit; a parallel system of $\frac{1}{2}$-in. piping 12 in. from the pit inner wall, and $\frac{1}{4}$-in. tubing cooling-water coils on the instrument-hole liners. To provide for the possibility of investigating a separate superheat core in the steam dome area of the vessel, an auxiliary set of cooling coils is imbedded in the pit wall near the top of the reactor but is left unconnected until required.

At maximum rated power, the raw-water makeup required by the main-condenser cooling tower is approximately 112 gpm. For conservation purposes, the cooling-water return lines from the ion-exchange heat exchangers, the shield-cooling coils, and the pump glands are all combined in a 4-in. line piped to the turbine building and discharged into the condenser cooling-water piping.

11. Other Systems

a. Sampling Systems

Several sampling connections are provided for monitoring the various plant systems. These are piped to a laboratory-type sampling station equipped with condensers, valving, etc. Reactor water pH and resistivity are monitored by means of continuous recording equipment.

b. Drains

All systems are equipped with drains which discharge into one of three local sump pits. Pumps deliver the sumpage to an existing leaching pond. High-temperature drains are suitably vented. Disposal of liquid radioactive waste is controlled by monitoring.

c. Water-Storage Pit Filter System

A 10-gpm circulating pump and cellulose filter are provided to keep the water in the water-storage pit transparent.

C. Process Instrumentation and Control Systems

The equipment included for process control includes instruments, transmitters and valves in the various systems which are described herein and shown in Figs. 35 and 36.

1. General Safety Considerations

(a) Signal Transmission between Reactor and Control Buildings

All instrument signals are carried electrically between the reactor and control buildings. Proportional-type signals use a biased signal wherever possible (i.e., 4 to 20 mamp proportional to 0 to 100% of process variable) to allow distinguishing between an extreme value of a variable and a failed instrument.

(b) Failure of Instruments

Design is based on the use of "fail-safe" circuits so that scrams are actuated on loss of power. Air-operated valves are installed to close on loss of air to minimize the interconnection of systems and to seal the reactor from the atmosphere in case of accident.

(c) Valve Position Indicators and Lamps

All valves which are critical in terms of safety to plant and personnel are equipped with limit switches and/or stem position transmitters. Position is noted in the control building by indicating lamps or electrical meters calibrated to read in percentage of valve opening. Limit switches are also used in interlock circuits to prevent undesired operations, such as starting the forced-convection pump with its suction and/or discharge valves closed.

(d) Process System Interlocks

Table XI shows the electrical interlocks which are provided to give protection to personnel and equipment.

Table XI.

PROCESS SYSTEM INTERLOCKS

A. Superheater Startup and Shutdown

1. The superheater-drain valve (HC-6aV) cannot be opened unless the superheater-flood valve (HC-5V) is closed.

   This interlock prevents draining of the boiling regions of the reactor through valve HC-6aV.

2. The pressure-equalizing valves (HC-17aV and HC-18aV) open and close with opening and closing of the superheater-drain valve (HC-6aV). They also open and close with the opening and closing of the superheater-flood valve (HC-5V). The equalizing valves can be opened independently of the superheater-flood and drain valves providing the back-pressure-control valves (PIC-IV and PIC-2V), the reactor vent valve (HC-1aV), and the superheater vent valve (HIC-5aV) are closed.

   The boiling section of the reactor and the superheating section of the reactor are separated mechanically within the reactor vessel. When the reactor is producing superheated steam, they must stay isolated. When the superheater is being flooded or drained, the top of the superheat fuel assemblies, the superheater-level-measuring line, and the pressure vessel steam dome must be at the same pressure (interconnected) in order to allow the proper flow of water. After the flood valve has been closed and the superheater has been drained, the superheater water level may be checked. To provide a true reading on the superheater-water-level indicator the superheater-level-measuring line must be connected to the pressure vessel steam dome. This is accomplished by independent operation of the equalizing valves. Inadvertent operation of this independent means of opening the equalizing valves while the reactor is flowing steam would bypass the second pass of the superheater.

3. The superheater-flood valve (HC-5V) can be tripped open only when the reactor is shut down.

   Introducing water to the superheater while the reactor is running could cause a reactivity addition and thermally shock the fuel plates.

4. When the low-steam-flow scram is not bypassed by key switch no. 3 the superheater-vent valve (HC-5aV) is automatically opened a preset amount and the main-steam and vent-steam back-pressure-control

Table XI (Cont'd.)

valves automatically close on a scram. On the failure of ac power, valve HC-5aV can be cycled from full open to full close using emergency air and dc power.

These provisions assure the flow of an adequate amount of cooling steam through the superheater in emergency conditions.

B. Boron-addition System

The boron dump valves operate in sequence. Operation of the pushbutton initiates the closing of the boron-leakage valve (HC-2dV); closing of this valve initiates opening of the boron-tank pressure-equalizing valve (HC-2aV); opening of this valve initiates the opening of boron-seal valve (HC-2cV) and the boron-to-reactor valve (HC-2bV). All valves are tripped by emergency 24-volt dc and must be manually reset.

This system assures that leakage from the boron tank will be to waste and not to the reactor. When boron flows into the reactor, the leakage valve closes first to prevent dumping the boron to waste. The pressure-equalizing valve opens ahead of the seal and reactor valves to permit proper flow of water to the reactor.

C. Forced-convection System

1. The forced-convection pump-suction valve (HC-11V) must be full open and the discharge valve (HC-10V) must be opened a predetermined amount (for minimum flow) in order to run the forced-convection pump.

   Operation of this pump without flow could cause it to overheat in a very short time.

2. Water must be flowing in the circulating system or the forced-convection system before starting up the reactor. (This interlock will be operative only during forced-convection operation.)

   The forced-convection system water and piping must be kept at the same temperature to prevent excessive mechanical stresses and the sudden introduction of cold water to the reactor with a resulting addition of reactivity.

3. The forced-convection pump cannot be started while the reactor is operating.

   Rapid changes in forced-convection flow rate could cause large reactivity changes.

Table XI (Cont'd.)

D. Preheat and Ion-exchange System

1. The ion-exchange, pressure-control valve (PIC-4aV) closes and the circulating pump will not run if the temperature of the water going to the ion-exchange columns exceeds 120°F.

   The ion-exchange resin will be damaged if the temperature exceeds 120°F.

2. The electric preheaters are not energized unless the flow through the preheaters is greater than 100 gpm.

   Flow below 100 gpm would cause the electric heaters to burn out.

3. The circulating-pump suction valve (FC-1aV) is also operated by the circulating pump motor starter.

   This valve prevents having reactor pressure on the circulating-pump packings during normal operation and prevents having reactor pressure on the ion-exchange system if the manual isolating valves are not properly closed.

4. The circulating-pump-bearing temperature must be below 150°F in order to run the pump.

   Cooling water must be supplied to the bearings of this pump to maintain satisfactory temperatures on the pump packing.

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2. Superheater Startup and Shutdown System

The valves in this system are all operated from the control building through electrical connections, although in some cases local operators are provided for emergency valve operation. A combination of solenoid trip-type valves and air-operated valves is used.

a. Superheater-vent Valve

In emergency conditions the superheater-vent valve opens automatically. Normally the valve is controlled manually at the control console. The valve is air-operated and the remote electrical control signal is converted at the valve into an air signal. This valve is operated from the emergency systems.

The superheater-vent flow indicator (FI-4) associated with this system gives the reactor operator information needed to regulate flow. A temperature recorder also provides information on superheater temperature at specific locations.

b. Superheater Water-level Indicator

This level indicator (LI-2) is used by the reactor operator to determine floodwater level in the superheater during startup and shutdown operations of the reactor. It operates from a differential-pressure transmitter connected to a pipeline external to the vessel.

c. Superheater Drain-flow Indicator

This flow indicator (FI-5) is used for superheater startup operations. Its purpose is to indicate flow of water and steam from the superheater during the draining phase of the startup operation.

3. Boron-addition System

To accomplish boric acid addition, valve operation is initiated from the control building. All valves are solenoid-operated from the 24-volt dc supply, and must be energized to complete the action. Valve control circuits are interlocked for proper sequence of operation, and indicating lamps are provided for monitoring the operation of all valves involved. Boric acid is injected by gravity feed.

A local pressure gage, local level gage glass, a temperature indicator and low water-level alarm in the control building are provided for the boron tank. An electric heater and automatic thermostat control are provided to maintain the boron tank at elevated temperatures when required.

4. Forced-convection System

The forced-convection pump is started and stopped by push-buttons on the control console. The suction and discharge valves are electric-motor operated and are positioned by control switches on the control console.

Flow rate in this system is recorded. The signal is received from a differential-pressure transmitter connected to a flow tube section.

5. Steam-pressure Control System

The steam-pressure control system regulates flow of reactor steam to maintain reactor vessel pressure constant at a preselected point.

Two air-operated valves with two separate controllers maintain set pressure during steam flow transients created by load changes or variation in reactor steam generation rate. Through the main-steam, back-pressure control valve (PIC-1V), reactor steam is routed to the steam turbine via the desuperheater. Through the vent-steam, back-pressure control valve (PIC-2V) the excess above turbine capacity is vented to atmosphere.

The following instruments are included to monitor this system:

a. Reactor Vessel-pressure Recorder (PR-2)
b. Superheater Discharge-pressure Recorder (PR-2A)
c. Steam-to-Turbine Pressure Recorder (PR-1)
d. Steam-temperature Recorder (TR-2)

6. Desuperheater and Turbine Steam Systems

Reactor steam, both saturated and superheated, is passed through a desuperheater which is automatically controlled to maintain steam conditions at the turbine inlet between saturation conditions at 250 psig and superheat conditions of 585°F at 350 psig.

The temperature recorder-controller sends a signal to the air-operated desuperheater-feedwater control valve (TRC-1V) which admits feedwater to the desuperheater.

A pressure-regulating valve (PIC-9V) maintains constant steam pressure at the turbine inlet, bypassing excess to the condenser through a pressure-controlled, air-operated valve. A flow recorder monitors this flow.

7. Feedwater Control System

Feedwater admission to the vessel is controlled by two independently operated valves. The main-feedwater control valve (LC-1V) is capable of regulating feedwater rates over the range from 0 to 275 gpm, and is air-operated by an automatic or manual control signal output from the three-element control system. The second low-rate-feedwater control valve (HC-14V) is sized to pass a maximum of 30 gpm and is manually controlled through an electric motor to provide small feed rates to the vessel.

The automatic level-control system is of the three-element type similar to that used in conventional power plant boiler-level control. A reference signal, proportional to the desired water level, is compared with a signal proportional to actual water level. The error developed is then added to the error signal, produced by comparing steam flow and feedwater flow signals. The sum of these signals is then used to actuate the main-feedwater control valve which corrects the reactor water level. Each of the three signals (feedwater flow, steam flow and reactor water level) is capable of adjustment to provide the necessary gain and time-constant factor for proper reactor water-level control. The effect of steam flow and feed flow can be made negligible in the system. This mode of operation is termed a "single element" control since only the water-level signal is used to control actual water level. Requirements for the control system and components are set so that for all normally anticipated transients in steam-flow rate and power, the water level is maintained within ±1 in. of the set-point. The set-point is adjustable from 5 to 15 ft above the bottom of the reactor vessel.

Recorders monitor steam flow rate, steam temperature, feedwater flow rate, and reactor vessel-water level. Manual compensation for temperature effects on steam mass-flow rates is made by electrical-compensating circuits which are to be set by means of a calibrated-dial potentiometer. Integrated feed and steam flow is indicated on each recorder.

Feedwater pumps are turned on and off at the control console. A flow indicator for measuring the 0 to 30-gpm rate from the manually operated low rate-feedwater valve (HC-14V) is provided. A separate, wide-range, reactor water-level indicator is provided for observing draining and flooding. The gage glass provides backup indication of reactor water level through the television system.

8. Reactor Preheat System

The electric preheaters and circulating pump are turned on and off from the control console by switches located in the reactor building. The heating rate can be set at 36, 72, 118, or 144 kw. A flow indicator in the reactor building allows monitoring of the flow rate in this system.

9. Ion-exchange Pressure-control System

This system is designed to limit the pressure on the primary side of the reactor ion-exchange heat exchanger to its safe recommended value and is operated from the control console. The control system regulates by comparing the actual line pressure with a value set on the indicating controller and operating the pneumatic, ion-exchange pressure-control valve (PIC-4aV) to vary flow to the heat exchanger. The flow rate is monitored by a local rotameter. A pressure-reducing orifice in the line limits the maximum downstream pressure at 20 gpm to 50 psig for a maximum upstream pressure of 600 psig.

10. Process Temperature-indicating System

This system consists of a potentiometer-type temperature indicator, associated thermocouples, and a selector switch to monitor process temperatures as required, and has a capacity of 24 temperature points. Process-system temperatures measured are:

a. Steam-to-Turbine
b. Forced-convection Water from Reactor
c. Forced-convection Water to Reactor
d. Preheater-water from Reactor
e. Preheater-water to Reactor
f. Shield-cooling Inlet
g. Shield-cooling Outlet
h. Feed Pump Inlet
i. Ion-exchange Heat Exchanger Inlet
j. Ion-exchange Heat Exchanger Outlet
k. Forced-convection-pump Casing
l. Boron Tank Water

11. Superheat Fuel Element Temperature

Two superheat fuel assemblies are installed with thermocouples attached to them. A monitoring system measures these temperatures once each 120 sec and sounds an alarm should any temperature exceed the design maximum. The temperature from any one of these thermocouples can be recorded by selecting the particular point with a selector switch. An independent, superheat fuel element, "hot spot"-temperature scram and alarm circuit is planned.

12. Reactor Vessel and Shield Wall Temperature-recording System

This system consists of an automatically cycled multipoint temperature recorder and associated thermocouples. Thermocouples are positioned on the external wall of the reactor vessel and within the reactor pit shield wall and floor to obtain records of temperature gradients.

Reactor vessel temperatures are measured at 18 locations which are all vertically coplanar, and paired to locate 9 installations diametrically opposite to 9 similar installations. A symmetrical grouping on the vessel is thus accomplished.

Shield-wall thermocouples are positioned at points within the wall calculated to be within the zone of maximum heat generation. An additional thermocouple, imbedded in the concrete shield wall, activates a high-temperature alarm by means of an alarming-pyrometer indicator in the event of excessive temperature. The pyrometer can be connected to any shield-wall thermocouple which exhibits the highest operating temperature.

13. Miscellaneous Control Building Meters

The demineralized-water and the feedwater-storage tank level recorders are also located in the control building.

14. Control-rod Seal-water Flowmeters and Valves

The seal-water supply valve is operated from the control building. Inlet and discharge flow to each rod drive mechanism is measured by 18 "rotameter"-type flowmeters, and is adjusted by valves associated with each flowmeter. Valves and flowmeters are located in the subreactor room.

15. Local Pressure Gages

The following pressure gages are installed locally in the reactor building:

a. Feedwater from Ion-exchange System
b. Water from Electric Preheaters
c. Reactor Standpipe
d. Boron Tank
e. Water to Reactor (Forced-convection Line)
f. Water from Reactor (Forced-convection Line)
g. Batch-feed Tank
h. Forced-convection Pump, P-1 (Discharge)
i. Circulating Pump, P-2 (Discharge)
j. Circulating Pump, P-2 (Suction)
k. Feedwater Pump, P-3 and P-4 (Suction)
l. Feedwater Pump, P-3 (Discharge)
m. Feedwater Pump, P-4 (Discharge)
n. Emergency Air-compressor
o. Emergency Air Header
p. Condensate Filter
q. Condensate Ion-exchange Filter

16. Television Monitor

A remotely controlled television camera located in the access shaft allows observation of the reactor vessel water-level sight glass, reactor vessel pressure gage, and general conditions of the area and equipment. The viewing screen is located in the control building.

17. Instrument Compressed-air Systems

Normal air supply for operation of valves is furnished by two compressors and is monitored by low-pressure alarms. An emergency air compressor is electrically interlocked to start at a preset low pressure, and by means of suitable check valves in the system, furnishes air only to the superheater-vent valve, the experimental-steam back-pressure control valve, the reactor-vent valve, and the two pressure-equalizing valves. These 5 valves are required to isolate the reactor vessel.

D. Buildings and Electrical Service Systems

1. Reactor Building

The new BORAX V building is a prefabricated, Butler-type, 40 x 86 x 20 ft high (vertical wall height). Figure 32 shows details of the building, pits, trenches, and major equipment.

The existing equipment pit houses the new feedwater pumps, ion exchange heat exchangers, condensate filters and ion exchanger, feedwater-storage tank and water-storage-pit filter system. The existing water-storage pit continues to be used and a new dry-storage pit is located at the southwest corner of the building. The reactor vessel is mounted in a new reactor pit. The forced-convection pump, circulating pump, and electric preheaters are in the subreactor room. Access to the subreactor room is through an access shaft by means of a spiral stairway.

Conventional structural concrete is used throughout for all footings, foundations, walls, etc., except for the shielding around the reactor. Magnetite high-density concrete shields the reactor and water cooling coils imbedded in the pit walls surrounding the reactor vessel prevent overheating and spalling of the concrete.

All plant piping and valves are in shielded trenches below the floor line accessible for servicing. Steam lines lie in the steam pipe trench and run underground from the reactor building to the turbine building. Electrical and control wiring runs in floor trenches or trays, and conduit along the walls.

All footings are new. The loads exerted by the frame of the building, due to both the heavier building and the crane load, necessitated footings of greater bearing areas. The north wall is 25 ft north of the old north wall. Because of the existing concrete instrument room at the south end of the reactor building the building south wall is in the same location as in the old building.

Access to the reactor building is provided for both personnel and freight by a 16 x 13-ft-high rolling steel door and by 3 x 7-ft hollow-metal doors at each end, and one 3 x 7-ft door at the middle of the west wall. Eight 4 x 6-ft fiberglass-plastic windows on both the east and west walls admit daylight to the interior.

The reactor building is insulated, making occupancy practical during all seasons of the year. A new oil-fired hot-air furnace with ducts throughout the building comprises the new heating plant. To house the furnace, pit ventilation blowers and filters, a heating and ventilating building of pumice block construction is located next to the reactor building, as shown in Fig. 32.

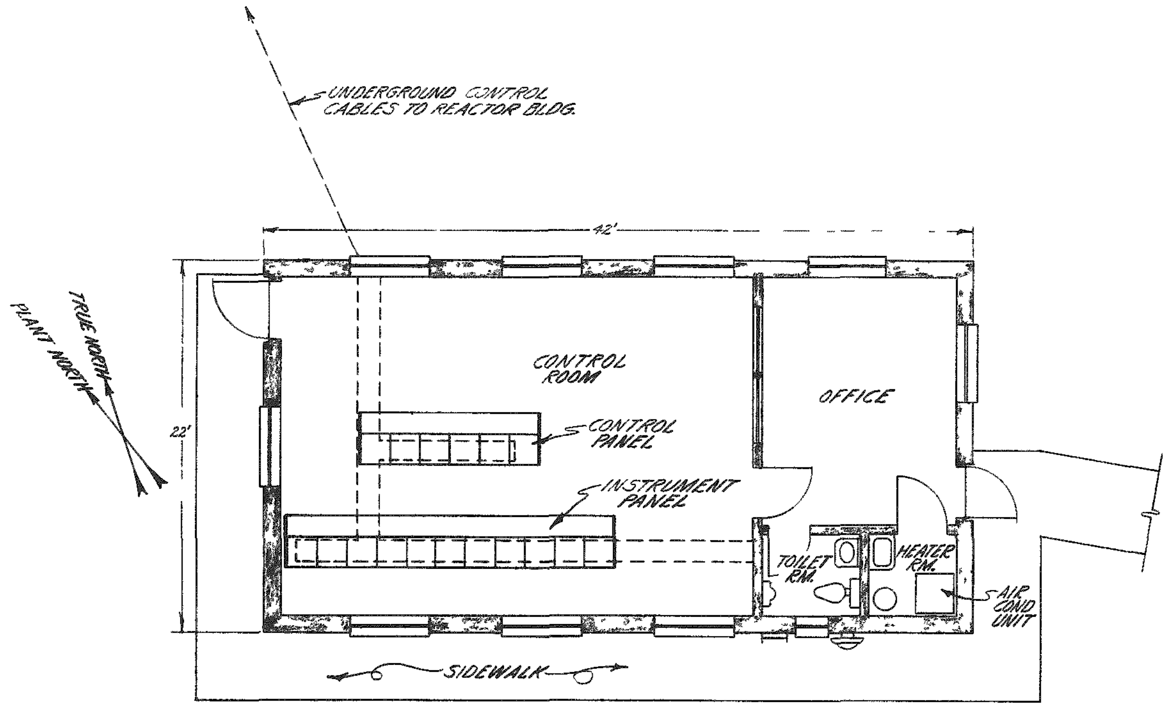

Fig. 37

Control Bldg. Floor Plan - BORAX V

Two 3000-cfm blowers are installed for pit ventilation. For summer operation both are needed during reactor operation to supply sufficient air to keep the subreactor room temperature below 120°F. When cooler ambient temperatures prevail, as during the winter months, one blower will suffice. Inlet air is drawn in through the hot air furnace and warm air expelled into the building through a distribution duct along the west wall. The ventilation blower draws the building air into the east pipe trench, down the access shaft, up through the reactor pit, out the steam pipe trench, through a filter bank, and then discharges it up the stack.

The reactor building 10-ton bridge crane is suspended from the building frame. The crane serves almost the full area of the building and all pits, with a maximum hook height of 15 ft above the floor and ability to reach $51\frac{1}{2}$ ft below floor level.

Crane speeds are as follows:

Hoist - 10 to 20 fpm in 5 steps, with auxiliary drive separately controlled to give inching speed of 2 fpm.

Trolley - 25 to 50 fpm max, in 3 steps.

Bridge - 25 to 50 fpm, in 3 steps.

Speed controls must pass through the slow-speed range first. Two pushbutton stations are provided with the crane: one pendant from the trolley with a cable tension reel; the other a remote control station mounted on the west wall.

2. Control Building

The new control building, shown in Fig. 37, is located about one-half mile southeast of the reactor building and is adjacent to the EBR-I site. It has pumice block walls, a tiled concrete slab floor, and a flat built-up roof with a steel roof deck. The building, which measures 20 x 40 ft inside, is divided into a control room, office, utility room and toilet. Covered trenches are provided in the control room floor, beneath the control panel locations, for control and instrument cable.

Constant temperature is maintained in the control building by a combination heating and air-conditioning unit.

3. Turbine Building and Cooling Tower

The existing Butler-type turbine building and the redwood forced-draft cooling tower will be reused essentially unchanged.

4. Electrical Service Systems

a. General

Electric power is distributed through a radial system. Since the plant is an experimental facility supplying an artificial load, a high degree of continuity of service is not required. Therefore the simplest, safest, least expensive power distribution system is used. Figure 38 is the single-line diagram for electrical power distribution.

Power is supplied to the EBR-I and BORAX area by a $3\frac{1}{2}$-mile-long distribution line from the Central Facilities Area. In order to limit voltage fluctuations caused by starting the 450-hp forced-convection pump, it was necessary to underbuild the existing 12,470-volt line with a parallel circuit.

b. Auxiliary Power

The forced-convection pump is supplied through a 500-kva step-down transformer and 4160-volt motor controller. The motor starts across the line and the controller provides short circuit, overcurrent, and undervoltage protection.

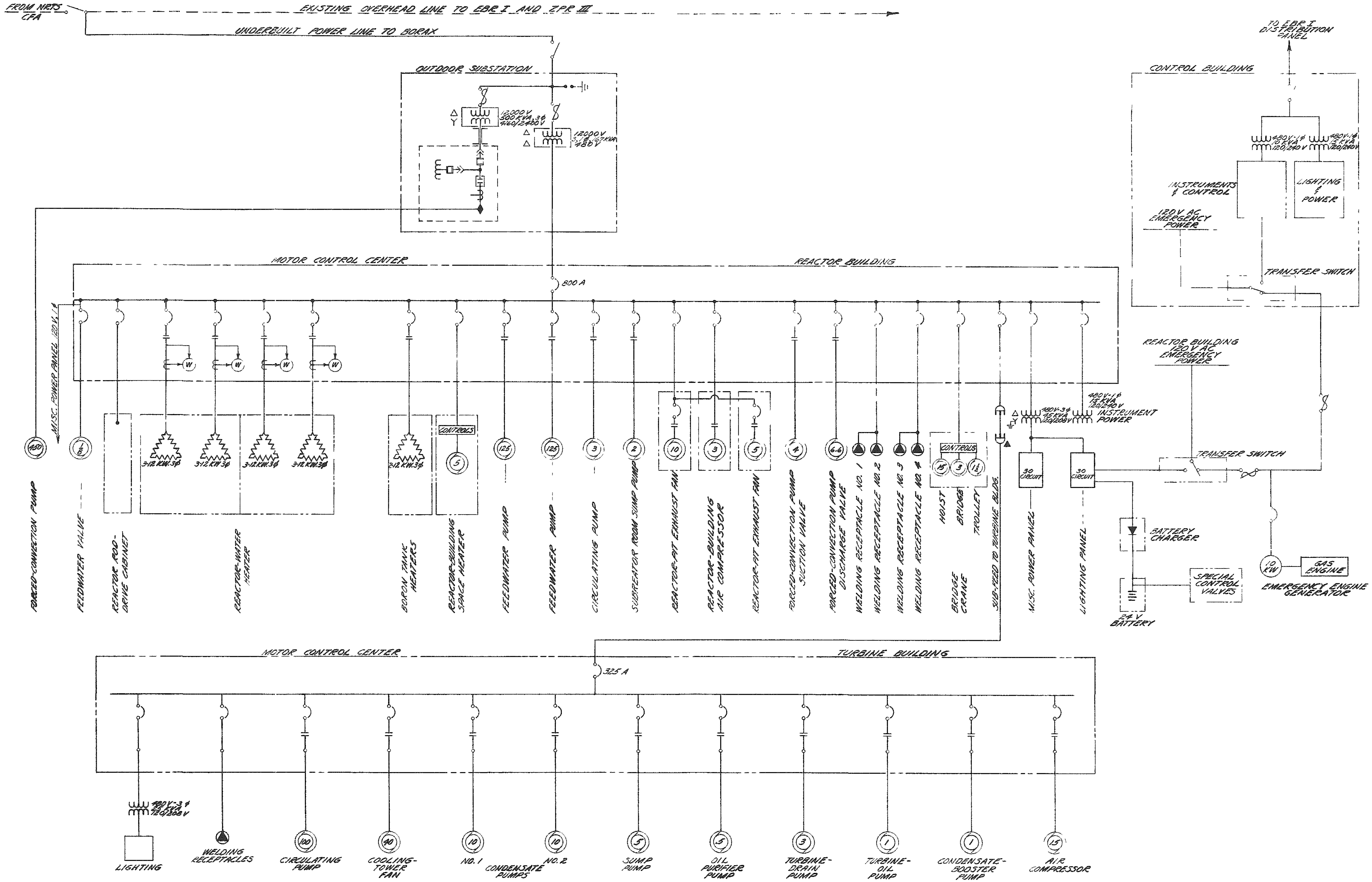

FIG. 38
ELECTRICAL SINGLE LINE DIAGRAM
BORAX V

The remainder of the plant auxiliaries are supplied by a bank of three 167-kva transformers which step down from 12,470 to 480 volts. The reactor building motor-control center supplies power to motors, resistance heaters, reactor control circuits, lighting, and instruments. Control of these devices is accomplished in the control building and/or in the reactor building, depending on the service required. Standard overload short circuit and undervoltage protection is provided. Auxiliaries for the turbine-generator are supplied through an overhead feeder from the reactor building to the turbine building. The existing motor-control center provides local control and protection for motors and other equipment in the turbine building.

c. Turbine-Generator

The turbine-generator can be controlled from the turbine building, or from the control building after the turbine is at rated speed and the manual throttle valve is fully open. The power produced by the generator at 2400 volts is dissipated in a water rheostat. The controls consist of generator breaker, field breaker, governor control and water-rheostat level control. Protection for the turbine-generator is provided by an overspeed trip and differential current relays.

d. Control Building Power

Electric power to the control building is supplied from the EBR-I distribution system at 480 volts. Failure of power to either the reactor building or the control building will shut down the reactor.

e. Lighting

Incandescent lighting is used for all buildings. There is a step-down transformer, operating at 480 volts to 120/240 volts, in each building. Standard circuit breaker distribution panels provide protection and control.

f. Instrumentation

Isolating transformers for instrument power are included at the reactor building and the control building to step down from 480 volts to 120/240 volts. This power is used for nuclear instruments, process instruments, and special test instruments.

g. Emergency Power

There are two sources of emergency power. One is the 24-volt dc battery and the other is the gasoline-engine-driven, 115-volt, 60-cycle generator.

The 24-volt battery has an 8-hr rating of 100 amp-hr and is considered the most reliable source of power.

The emergency generator has a capacity of 10 kw, and starts automatically on power failure at the reactor or the control buildings. Automatic transfer switches connect essential instruments and controls to this power source when the generator reaches operating voltage. A $1\frac{1}{2}$-hp, 6-scfm compressor operates from this generator to supply air to the essential control valves.

The items of equipment connected to the emergency power supplies are listed in Table XII.

Table XII

EQUIPMENT CONNECTED TO EMERGENCY POWER SUPPLIES

A. Emergency Air

1. Superheater-vent Valve (HIC-5aV)
2. Reactor-vent Valve (HC-1aV)
3. Experimental-steam, Back-pressure-control Valve (PIC-3V)
4. Pressure-equalizing Valves (HC-17V and HC-18V)

B. 120-v ac Power

1. Emergency Air Compressor
2. Reactor-water Level Indicator (LI-1)
3. Superheater-water Level Indicator (LI-2)
4. Main-steam Flow Recorder (FR-1)
5. Linear Flux Recorder
6. Reactor-pressure Recorder (PR-2)
7. Superheater-vent-valve Control (HIC-5)
8. Experimental-steam, Back-pressure-control-valve Control (HIC-3)
9. Reactor-vent-valve Control (HC-1aV)
10. Superheater-vent Flow Indicator (FI-4)

C. 24-v dc Power

Reactor-vent Valve (HC-1aV)
Superheater-flood Valve (HC-5V)
Pressure-equalizing Valves (HC-17V and HC-18V)
Boron-injection Valve (HC-2aV, HC-2bV, HC-2cV and HC-2dV)
Turbine-generator Breaker Trip

5. Communications and Alarms

The normal means of communication between the control building, the reactor building and the turbine building is a two-way intercom system. There are four stations in the reactor building, one station in the turbine building and one at the control console. Interconnected with the intercom system is a paging system which permits the control console operator to call the turbine and reactor buildings. One of the stations in the reactor building can also be used for paging.

For emergency conditions and for equipment calibration and checking, a sound-powered phone circuit is provided with plug-in jacks at various points in the reactor and control buildings as well as at the control console.

Each building has a fire-alarm signal box connected to the site fire-alarm system. Gongs are also provided.

One Bell System phone is located in each building and connected to the EBR-I switchboard.

There is an evacuation alarm horn and evacuation alarm switch in each building. This system is tied into the EBR-I system.

A flashing red light indicates that the reactor is in operation. Two lights are located inside, two outside the reactor building, and one in the subreactor room. Warning horns located inside and outside the reactor building and in the subreactor room sound for 10 sec on reactor startup.

IV. MANAGEMENT

Operation of this reactor facility shall be through a three-level organization: managerial, supervisory, and operational.

The managerial level will consist of an Operations Manager and an Operations Engineer. Both of these positions shall be filled with personnel appointed by the Director of the Idaho Division.

The Reactor Operations Manager shall be responsible for the program and safe operation of the reactor facility. He shall have complete authority within limits as set forth in the Hazards Summary Report, Operating Manual, and directions from the Division Director. He shall be directly responsible to the Division Director.

The Operations Engineer shall serve as an assistant to the Manager and shall act for him when specifically instructed. He may not, without authorization of the Division Director, act in place of the Operations Manager.

Supervision of actual operation for each shift shall be by a Chief Operator who shall be appointed by the Manager. He shall comply with standard operating procedure and may not deviate from this by his own volition except to take the necessary action to safeguard the facility in case of an unpredicted emergency.

Actual operation of the reactor controls or of any auxiliary system which might affect the reactor in any manner shall be by Reactor Operators. One of these may be designated Operator Foreman. He shall have responsibility for such routine matters as maintenance, inspection, and other duties as delegated by the Chief Operator.

All qualifications to the above positions shall be approved by the Manager and by the Division Director after proof of ability has been presented.

During normal power operation, two Operators shall be required in the control building and one in the turbine-reactor area. One of the control building Operators shall observe and control the nuclear operations of the plant. Normally, the second Operator shall observe and control the process portions of the plant and shall assist and relieve the nuclear Operator as required. The Operator at the turbine and reactor buildings shall observe all equipment therein for normal operation, start and stop equipment as needed, and perform routine operating duties. For open-vessel operation and certain special experiments the reactor building shall be vacated.

Reactor keys shall be in the custody of the Operations Manager when not in the custody of the responsible operations supervisor.

V. REACTOR OPERATIONS

Initially, the entire plant shall be checked out, including operational tests, hydrostatic tests, and instrument calibration. After this, the operation can be divided into three classes; Loading, Open-vessel, and At-pressure. For any class of operation, a check-out procedure shall be followed daily to assure that essential equipment is operating satisfactorily.

A. Loading

An antimony-beryllium neutron source shall always be installed in the reactor. For initial and other special loadings it is necessary to locate the neutron detectors in temporary positions inside the reactor vessel. Loading shall proceed as required by changes in multiplication. Two fission counters and 2 ionization chambers shall be located so changes in multiplication can be observed by both the control room operator and the loaders.

All loading changes (unloadings, loadings, or any movement of fuel) shall be made under the immediate supervision of the Operations Manager or an Operations Engineer. Proposed loading changes shall be checked before performance by at least 2 qualified persons for anticipated effect, and during the change for proper placement of the fuel.

A health physics representative shall be present during all loading changes to assure that proper radiation monitoring instruments are available, and that proper radiation protection precautions are taken.

B. Open-vessel Operation

After each change in loading, the reactivity effect shall be measured by low-power critical runs. Normally, in this type of operation, all personnel shall be evacuated from the reactor building before control rods are moved. Open-vessel operation shall be construed to mean any attempt to make the reactor critical without the top shielding in place. For low-power flux-distribution measurements, etc., after suitable radiation surveys, personnel may re-enter the building.

C. At-pressure Operation

There are 4 general classes of power operation for this reactor: 1. natural-convection boiling with superheater; 2. natural-convection boiling; 3. forced-convection boiling; 4. forced-convection boiling with superheater. There are 3 core structures: two for superheating operation, and one for boiling only. After inserting the desired core structure the reactor is loaded with the vessel lid removed. When the desired loading is reached and the open-vessel operation is completed, the reactor vessel lid shall be bolted in place and the shielding positioned. The following operating description shall pertain to normal operation after necessary "At-pressure" calibrations have been performed and approval for operation has been given. Since operation of a reactor with nuclear superheating is unique and untried, a detailed description of proposed operation under this condition is presented. The following steps shall be taken to bring the reactor to full power and back to shut-down conditions:

1. Natural-convection Boiling with Superheater

For both central and peripheral superheater operation the procedures shall be the same. The only difference is that in the first case the central control rod partially controls the superheater, and in the second case the outside rods partially control the superheater. The operating procedures shall be as follows:

a. Starting Procedure

Perform the startup checkout. Heat the reactor to operating temperature and pressure. Procedures will differ here depending on whether or not the superheater is flooded. These procedures are as follows:

Superheater Flooded

(1) Withdraw control rods to attain reactor criticality.

(2) Increase the reactor power to the level necessary to heat the reactor and vessel at a rate of temperature rise no greater than 200°F/hr. Electric preheaters may be used in addition to, or in place of, reactor power.

(3) When the pressure reaches about 575 psig (before the main-steam, back-pressure-control valve opens), the reactor power shall be reduced by inserting control rods in preparation for draining the superheater. Superheater control rods shall be fully inserted. Calculations show that at 0.4 Mw the superheater elements are safely cooled by radiation cooling. The superheater shall then be drained by closing the superheater-flood valve and opening the superheater-drain valve to blow the water from the superheater to the condenser. When the drain valve is opened, the pressure-equalizing valves open to vent the superheater second pass and the superheater water-level-measuring column to the reactor steam dome. Before closing the drain valve, the superheater-vent valve shall be opened to flow sufficient steam to atmosphere in order to keep

the superheat fuel-element temperature below 1200°F. The pressure-equalizing valves close when the drain valve is closed.

(4) Withdraw the boiling section control rods to raise reactor power and return to operating pressure. The power shall be limited to give a rate-of-temperature rise of 200°F/hr. Eighty percent of the power produced by the reactor must be used to flow steam through the superheater for fuel element cooling. As the pressure rises, the main-steam, back-pressure-control valve opens to permit flow of steam to the turbine building. As soon as steam flow to the turbine building becomes adequate for superheat fuel element cooling, the superheater-vent valve shall be closed.

Superheater Not Flooded

(Since the superheat fuel elements are adequately cooled by radiation when the power production is below 0.4 Mw, it may sometimes be desirable to leave the superheater unflooded between shutdown and the next startup. Under this condition of startup the reactor heating rate is limited to 0.3 Mw reactor power plus the electric heater power. The pressure may be any value from atmospheric to 600 psig.)

(1) Withdraw the control rods to attain reactor criticality, primarily using boiling-zone control rods.

(2) Increase the power to 0.3 Mw to heat the reactor to either 575 psig, or the pressure at which the main-steam, back-pressure-control valve begins to open.

(3) The superheater-vent valve shall be opened to flow sufficient steam for superheater cooling and the reactor power is raised to further increase pressure and begin flow through the main-steam, back-pressure-control valve. When a steady flow of steam through the back-pressure-control valve has been established, the superheater-vent valve may be closed.

b. Procedure for Reaching Full Power

(1) As steam flow to the turbine begins, the desuperheater shall be placed in operation.

(2) With 350-psig saturated steam at about 6000 lb/hr flowing through the turbine-bypass valve, the turbine can be started and slowly brought to full speed.

(3) To reach higher reactor power and superheated steam conditions, the control rods for the boiling section and the control rods for the superheat section shall be adjusted. The turbine load shall be connected and increased simultaneously. When the steam flow has reached 15,000 lb/hr the steam temperature may be increased to the desired conditions. The power produced in the boiling section and in the superheater can be adjusted to a limited extent by use of control rods in the respective sections.

(4) The turbine-condenser system dissipates approximately 20 thermal Mw. For higher powers the vent-steam back-pressure-control valve can pass steam to atmosphere. Operation at these higher powers is possible for only a short time because of the limited demineralized-water capacity.

(5) To shut the plant down, the reactor power shall be reduced by first inserting the superheater control rods and then the boiling-region control rods. The turbine load shall be correspondingly reduced. The steam temperature to the turbine is reduced by the desuperheater as required until the turbine is unloaded. Steam flow shall be maintained at a rate which limits the superheat fuel-element temperature to the design maximum. When the control rods are fully inserted and the superheat fuel-plate temperature has been sufficiently reduced, either the flood valve may be operated to flood the superheater, or cooling may be accomplished by radiation alone.

2. Natural-convection Boiling

To operate without a superheater the appropriate reactor core structure is used. With the core structure in place, the reactor is loaded with fuel as described previously. The procedures for starting up and shutting down will be similar except for the operating temperature which is at saturation. Flooding, draining and desuperheating systems are not operated.

3. Forced-convection Boiling

The same core support structure used for natural-convection boiling is used for this method of operation. The baffle for directing forced-convection flow must be inserted in the reactor vessel. The forced-convection piping must be attached to the reactor vessel. Reactor loading, startup, and shutdown procedures are similar to those used for natural-convection boiling.

When the forced-convection piping is connected, the reactor vessel and the piping must be kept at the same temperature to prevent excessive stresses and to prevent large reactivity changes caused by the introduction of cold water. Also, the possibility of variations in reactivity due to large changes in forced-convection flow must be anticipated.

4. Forced-convection Boiling with Superheater

It is possible to use the forced-convection system while superheating. Flow capacity is reduced slightly because one of the forced-convection inlet nozzles is used for the entrance of the superheat drain and flood pipe. All of the precautions and scrams required for both forced-convection and for superheating shall be effective.

VI. EXPERIMENTAL PROGRAM

The proposed program of experiments in BORAX V is divided chronologically into two phases. The initial series of experiments after plant shakedown will be on superheating cores and fuel assemblies using primarily natural convection. The purposes of these tests will be to prove thoroughly the integral superheat core concept and investigate the safety aspects of integral nuclear superheaters; to compare the merits of a central versus a peripheral superheat-zone location; to experimentally determine the statics and dynamics of the coupled cores; and to test superheat fuel elements of both the reference and advanced designs.

The second phase of experimentation will be done on a pure boiling core. The primary purpose of these tests will be to improve understanding of the factors which control the stability of boiling reactors at high power densities and limit the maximum stable power capability. Tests will be run using both natural and forced circulation.

A tentative experimental program is summarized in the following paragraphs:

A. Startup and Initial Operation

The reactor will be started up for the first time with a boiling natural-circulation core using boiling fuel assemblies which will later be used in the superheating cores. The reactor will be operated at power long enough to check out and calibrate the process piping, instrumentation and control systems, and to train operators. In-core instrumentation in the boiling fuel assemblies will also be proof-tested at this time. Finally, the forced-convection baffle will be installed and operation with the forced-convection system will be tested briefly. The fuel and control rods will then be unloaded and the boiling core structure removed.

B. Superheating Experiments

After installation of a natural-circulation superheat core structure (probably the central superheater), the reactor will be built up to full size and the usual calibrations, measurements, and superheater flooding worth will be made at zero power. Before proceeding to higher powers, the superheater-vent, drain, and flood systems will be tested. In-core instrumentation will be used to observe core characteristics during operation. If necessary, power distribution in the boiling zone may be adjusted to attain the desired powers. The first superheat core will be operated until feasibility has been demonstrated. A forced-convection baffle may be installed and the superheating reactor tested with forced convection.

Following completion of the experiments on the first superheat core, the core structure will be replaced and a similar series of tests will be made on the second type of superheat core, probably the peripheral superheater.

It is expected that superheat fuel elements of advanced design will be tested either in the superheat cores, if they are available at the time, or later, by means of in-pile loops during the boiling core tests.

C. Boiling Core Experiments

Proposed experiments which are designed to gain understanding of boiling reactor characteristics may be divided into three categories: static measurements, dynamic measurements, and engineering tests. The aims of these experiments follow.

1. Static Measurements

These measurements will include the usual measurements and calibrations of various reactivity effects associated with the approach to critical, low-power, and full-power operation of a reactor. Of prime interest here are reactivity changes affected by variations in operating conditions.

2. Dynamic Measurements

These measurements are aimed at investigation of the stability characteristics of the system, with emphasis on predictions of the maximum stable power achievable. Oscillator techniques are one approach to these measurements, although other techniques, such as noise analysis, may be used where appropriate.

3. Engineering Tests

These tests, which might actually be included under the above tests, will be aimed at determining the effect of the many possible design variables on the maximum stable power output of boiling reactors. Some of the variables are fuel-rod spacing or coolant-channel size, core size, flux flattening, chimney height, water level, coolant-channel inlet and outlet restriction, etc. Such studies should permit a more confident optimization of the design of future boiling reactor systems.

D. Water Chemistry Experiments

It is intended that water chemistry experiments be conducted with every type of core tested in BORAX V. Both the reactor vessel internals and the process piping system have sampling taps at strategic points. Sampling lines are run to a central sampling panel located on the main floor of the reactor building.

The routine water chemistry program planned for BORAX V will consist primarily of measurements to determine pH, conductivity, concentration of chlorides, suspended and dissolved solids, corrosion rates, radiolytic gas-evolution rate, and identification of radioisotopes in gas, steam and water.

Special tests which have been proposed include:

1. investigation of the causes of variation in steam activity;
2. studies of total solids in reactor water and their effect on scale formation on fuel elements;
3. studies to find methods of reducing radioactivity in the steam plant; and
4. studies of water decomposition rates.

VII. HAZARDS EVALUATION

A. Inherent Safety of Boiling Reactors

A reactor may be called inherently safe against reactivity additions if it is able to undergo these additions without deleterious effects to the surroundings, or, more restrictively, to the reactor itself. The inherent safety capabilities of boiling water reactors as a type have been shown to be high in the previous BORAX experiments (BORAX I-IV) and the continuing SPERT experiments. Detailed analyses of some of these experimental results have been applied to the EBWR, a boiling reactor of different design, to indicate similarly a rather marked degree of inherent safety. BORAX V should exhibit similar characteristics.

The limiting safe accident, defined as that accident which just melts the hottest point of the hottest fuel element, for any particular reactor must be a function of the design of that reactor. It is realized that generalization from particular studies may lead to difficulties. This will be especially true for the BORAX V design, which incorporates a highly enriched, steam-cooled, superheating section either at the core center or at the core periphery. Nevertheless, the boiling core effects will shut down the reactor without harm under certain conditions. No detailed study of a limiting safe accident has yet been carried out for BORAX V. However, certain qualitative statements about the inherent safety of this reactor can be made. These will of course, be supplemented by detailed studies in the final hazards summary report.

The most direct evidence of the inherent safety of BORAX V stems from the transient tests of BORAX IV. Both reactors have about the same void coefficient (0.25% $\Delta k/k/\%$ void) and fuel size with the same conductivity and specific heat. The cladding of the BORAX V fuel is 0.015 in. of stainless steel, while that of BORAX IV was 0.013 in. of lead plus 0.020 in. of aluminum, thus the resistivities are not greatly different. BORAX IV experienced an excursion with a period of 0.083 sec corresponding to a step reactivity increase of 0.74% with no damage. Indeed the total energy release to peak power was less than 10 Mw-sec. It is calculated that BORAX V with a central superheater can undergo an excursion of 140 Mw-sec from operating temperature before melting the superheat fuel plates. This result is based on the heat capacity of the superheater with no heat loss. It seems fairly certain that BORAX V can undergo excursions even more violent than the BORAX IV super-prompt-critical excursion without damage. (On the same basis, a 273-Mw-sec excursion from operating conditions is necessary to just melt the hottest boiling fuel rod.)

A comparison of BORAX V characteristics with EBWR characteristics indicates that the ρ-C_p values for the two fuels are comparable, i.e., the heat per unit fuel volume for a given temperature rise is comparable, and the heat conductance of the claddings is comparable, with EBWR having a thicker cladding of slightly lower conductivity. The void coefficient of BORAX V is at least twice that of EBWR (-0.25 $\Delta k/k/\%$ void for BORAX V as opposed to -0.12 $\Delta k/k/\%$ void in a 4-ft EBWR core and -0.067 $\Delta k/k/\%$ void in a 5-ft EBWR core). Hence, again, it is likely BORAX V will be safe under transients terminated by steam formation with periods shorter than 0.083 sec (since EBWR calculates a safe period for steam shutdown of 0.057 sec).

The Doppler effect in the U^{238} contained in BORAX V fuel should contribute significantly to the shutdown. Calculations of the Doppler effect will be included in the final report.

B. Possible Hazards

1. Hazards of Superheater Operation

a. Loss of Coolant

Since a superheat fuel assembly in BORAX V is essentially gas-cooled, and also has a low heat capacity due to its thin-plate construction, the primary hazard associated with superheater operation is loss of coolant-steam flow followed by rapid heating and possible meltdown with attendant release of radioactivity. For instance, at a reactor power of 20 Mw and no steam flow, the maximum superheat fuel element temperature rises from 1200°F to the melting point, 2600°F, in about 8 sec.

Loss of steam-coolant flow can be caused by manual closing of a steam stop-valve or closing of a back-pressure-control valve due to loss of instrument air or to control-power failure. (The back-pressure-control valves are designed to fail-close in order to contain steam, water, or radioactivity within the reactor vessel.)

The superheat fuel elements are protected against damage from loss of coolant steam by: (1) a power failure scram; (2) a low-instrument-air-pressure alarm; (3) a low-steam-flow alarm set at 4000 lb/hr, with scram set at 2000 lb/hr, (4) a high-superheated-steam-temperature alarm set at 875°F; (5) an emergency air and power system to insure operation of essential valves and equipment, and, if possible, by (6) a high-superheat-fuel-element temperature alarm set at 1250°F, with scram set at 1300°F

Steam flow through the superheater may also be stopped by inadvertent flooding of the superheat fuel assemblies. If the flooding is done by raising reactor-vessel-water level to the top of the superheat fuel assembly inlet, the reactor is protected by a high-water-level alarm and scram and reactor water-level indicators. If flooding is accomplished by opening the superheater-flood valve, the flood-valve-open scram operates.

If this interlock fails, the reactor power can rise due to the positive flooding coefficient of reactivity. In this case the power-level-trip safety circuits scram the reactor. Once the superheater is flooded and shut down, it is cooled by either boiling or natural circulation of the flood water.

Any of the aforementioned alarms give the reactor operator a warning to shut down and open the superheater-vent valve to keep the superheat fuel elements cool. On any scram the superheater-vent valve opens automatically to a preset position if air is available. Further means of insuring operation of the superheater-vent valve is provided by an emergency air compressor powered by the emergency generator, and stored air in the air compressor tanks. Manual operation of this valve through a reach rod is also possible.

b. Flooding Superheater

Programmed flooding of the superheater is performed after shutdown, with control rods fully inserted and at a time when the temperature difference between the reactor water, the flooding medium, and the superheat fuel plates is low, to give a minimum thermal shock to the fuel plates. Under this condition, even though flooding has a positive reactivity effect, the reactor remains subcritical due to the large amount of negative reactivity in the control rods. If the superheater is flooded during operation, it is with 600-psig, 489°F saturated water. Under this condition, the total flooding reactivity worth of the central superheater zone is calculated to be +0.5%. The worth of the flooded peripheral zone has not yet been calculated. The maximum rate of flooding by means of raising the reactor-vessel-water level to the superheat fuel assembly inlets at 600 psig in the reactor vessel is limited to 300 gpm by the capacity of the two feedwater pumps. This amounts to a reactivity addition rate of about 0.25%/sec for the central superheater. The flooding rate via the superheater flood valve is only 100 gpm.

Additional safety protection against conditions resulting from a flooded superheater is inherent in the reactor itself. When saturated water strikes the high-temperature superheat fuel plates it flashes into steam. If scrams and alarms fail to operate and the superheater is completely flooded, the power level rises until it levels off due to boiling in both the superheat and boiling zones.

c. Excursions and Chugging

Assuming the reactor is "chugging" (violently oscillating in power) or assuming the rapid addition of a large amount of reactivity which causes an excursion, it can be postulated that, as a result of the time lag in flow of additional cooling steam from the boiling zone behind the power increase in the superheat zone, the superheat fuel elements may overheat and melt. One Mw-sec of reactor energy release causes a 6.2°F rise in superheat fuel plate temperature. Therefore, a 140-Mw-sec excursion from full-power operating conditions, taking into account the assumed radial and axial power distributions shown in Table I, results in only a 2450°F maximum fuel plate temperature.

Protection against this accident, in addition to the negative power coefficient of reactivity, is given by the time delay required to reach the melting temperature, which allows the power-level scrams and, perhaps, the high superheat-fuel-element temperature scram to operate.

d. Superheat Shutdown and Startup

Normal superheat or shutdown procedures as presented in Section V, C, do not constitute a hazard since steam flow is maintained until the reactor post-shutdown decay heat is down to less than 1% of maximum power or about 0.4 Mw. If the steam flow is stopped below this power, radiation cooling keeps the fuel-plate temperature below 1200°F. In fact, radiation cooling will limit the temperature at 3.5 Mw at 2450°F. At this temperature, which is below the melting point of Type 304 stainless steel, severe scaling, distortion, and damage to fuel elements will probably occur, but no radioactivity should be released unless this condition persists for sometime.

The superheater startup operation takes place either immediately after shutdown when steam is still flowing, as in the case of a spurious scram, or long enough after shutdown to have allowed the reactor decay heat to reach a low level. Another postulated danger point might be a startup operation with flooded superheater. Draining a flooded superheater to the condenser requires about 60 sec. During this time, before steam starts flowing through the superheater, the fuel elements are adequately cooled by radiation alone. Moreover, operating procedure is to reduce reactor power before draining the superheater. In addition, superheater draining has a negative reactivity effect. Thus this situation actually poses no hazard.

2. Core Melting

The BORAX V core can be made to melt and possibly vaporize by the sudden addition of a large amount of excess reactivity. Loss of water from the core, though it will shut the reactor down, can result in core melting from decay heat under certain conditions of reactor operating history.

If the reactivity addition is sufficient to vaporize fuel elements, an explosion similar to the terminal experiment which destroyed BORAX I could result. The immediate area around that reactor was heavily contaminated with fission products and fuel element fragments, while a small radioactive cloud moved off downwind. There is no danger of concentration of fuel elements to form a second critical mass in this type of accident.

If the superheat fuel elements melt due to loss of coolant, the finely divided UO_2, which is dispersed in stainless steel, is carried with the molten metal. The molten mixture may flow to the lower part of the core, or to below the core, before it is sufficiently dispersed to cool and solidify. Since the superheat zone of BORAX V is part of a coupled reactor, it alone does not have enough fuel under any circumstances to go critical.

The boiling fuel rods are made of sintered UO_2 rods or pellets clad with stainless steel. The melting point of stainless steel is about 2600°F, while that of UO_2 is about 5000°F. Because of the low conductivity of UO_2, the temperature at the center of the fuel rod is much higher than the clad surface temperature. If the rate of heat removal on the clad surface is decreased by steam-blanketing or loss of water, the cladding melts before the UO_2, releasing some of the gaseous and volatile fission products. When sufficient cladding has melted, the pellets and fragments fall onto the lower grid in the boiling fuel assembly. If the heat from the pellets and fragments causes melting of the stainless steel grid and aluminum fuel assembly box, the pellets and fragments may fall through the holes or melt through the stainless steel of the thick core-support plate, coming to rest on the bottom of the reactor vessel. The nine re-entrant control-rod nozzles in the bottom of the reactor vessel serve to separate the mixture of stainless steel and UO_2 pellets and fragments so that, in the unlikely event water is still present, no critical mass is formed. By this time, the UO_2 is cooled down by dispersion in the molten steel. The mixture does not have sufficient heat content to melt the re-entrant control rod nozzles or melt through the bottom of the reactor vessel.

3. Cold-water Accident

Since the reactor has a negative temperature coefficient, introduction of cold water increases the reactivity. The temperature coefficient for the cores is estimated to be about $0.03\% \Delta k/k\text{-}^{\circ}C$. Cold water can be injected into the reactor by two possible methods:

(a) If the feed pumps are turned on with the reactor just critical at operating temperature, cold water can be injected at a maximum possible rate of 300 gpm. The minimum operating volume of water in the reactor vessel is about 1600 gal. Four minutes are required to fill the reactor vessel and the water temperature is reduced by 85°C. This results in a rate-of-reactivity addition of about 0.01%/sec for a total addition of 2.55%. The reactor power level merely rises to a point at which the increase in temperature and steam formation nullify the increase in reactivity due to the addition of cold water.

(b) The worst cold-water accident that can be postulated involves the forced-convection system. Assume that the reactor with central superheater is just critical at operating temperature, the forced-convection baffle is installed, and the forced-convection pump is off. Also assume the operating procedure to keep the forced-convection system hot has been ignored and that the low forced-convection flow and low circulating-pump-flow scrams, alarms, and interlocks do not operate, so that the temperature of the water in the forced-convection system is at room temperature. (Incidentally, under this condition, the lower nozzles on the reactor vessel are severely stressed.)

When the 10,000-gpm pump is turned on with system valves open, full flow is reached in about 6 sec. The 489°F water in the core is replaced with 70°F water in 0.33 sec, giving a reactivity addition rate of 21%/sec. The slug of cold water passes through the core for about 6 sec. As noted previously, no excursion analyses have yet been made. To prevent this cold-water accident, the following interlocks are provided:

(1) A flow interlock, to assure that either the 10,000-gpm forced-convection pump or the 150-gpm circulating pump is flowing water through the forced-convection system and reactor vessel before reactor startup. This insures equal temperature throughout the system prior to startup.

(2) A pump starter interlock, to prevent starting of the forced-convection pump after the reactor is in operation. This prevents a cold-water accident even though interlock (1), above, has failed.

(3) A low-flow alarm and scram on the forced-convection system.

(4) A low-flow alarm on the 150-gpm circulating system.

Interlocks (3) and (4) above either scram the reactor or warn the operator if flow rate decreases in the forced-convection system. Cooling down of this system is thus inhibited and protection is afforded against a cold-water accident or high reactor-vessel-nozzle stresses.

The maximum possible cold-water reactivity addition rate from the 150-gpm circulating pump is only 0.015 of the rate associated with the forced-convection pump and is not deemed hazardous.

4. Power and Air Failure

In case of loss of power from the incoming utility line, the solenoid-operated latches holding the control rods fail-safe and release. The rods are inserted into essentially their maximum effective shutdown position in less than 0.2 sec.

An independent 24-volt dc battery, emergency-power system, which is considered the most reliable source of power, energizes certain critical valves.

A second emergency-power system is supplied by a gasoline-engine-driven generator which starts automatically on utility-power failure at either the reactor or the control building. Automatic transfer switches connect certain essential instruments and controls to this power source when the emergency generator reaches operating voltage.

Loss of utility power eventually results in a loss of instrument-air pressure for control-valve operation. The capacity of the air-compressor storage tanks maintains sufficient control pressure at a probable maximum bleed rate of 45 scfm for about 1 min. An emergency air compressor operating from the emergency generator supplies air to certain control valves essential for reactor containment and cooling. In case of total loss of control-air pressure, the valves which contain the reactor fail-close.

Table VII contains a detailed list of the valves, controls and instruments supplied by the emergency power systems.

5. Pump Failure

The most probable reason for failure of pumps is loss of power. The effect of individual pump failure is discussed in the following paragraphs.

Failure of the 10,000-gpm forced-convection pump or inadvertent closing of one of the valves in the forced-convection system stops forced-convection flow. Since the heat is not being carried out of the core by forced flow, the void content is increased, and the reactor power decreases before the fuel-element temperature rises to the point where "chugging," film boiling, and burnout occurs. Backup protection against this hazard is afforded by means of a low-forced-convection-flow alarm and scram.

If both reactor feed pumps fail, the reactor continues to operate, feedwater flow ceases, reactor inlet subcooling is reduced, reactor voids increase, the reactor automatically reduces power, and water is evaporated from the reactor. Alarms for low reactor-water level, low feedwater-pressure and high feedwater-storage-tank level notify the operator of these events. If these alarms are inoperative or unheeded, the low reactor-water-level scram shuts the reactor down, after decay heat from the reactor continues to evaporate water. The reactor water may still be maintained, at pressure, above core level, by manual operation of the batch-emergency-feed system. The capacity of this system is sufficient to handle the most severe decay heat conditions. Also the boron-injection system can be used to maintain reactor water level.

Failure of the condenser main-circulating pump leads to loss of condenser vacuum. A pump-failure alarm is energized and eventually a reactor scram occurs if condenser pressure exceeds +2 psig. Backup protection is given by the condenser pressure-relief valve set at 5 psig.

Failure of the well pump stops the flow of raw cooling water to the shield, ion-exchange system heat exchangers, and pump bearings. If water flow stops in the shield cooling system, the concrete shield can heat up to approximately 520°F at 40-Mw reactor power. It is probable that some spalling and crumbling of concrete in the shield may occur at this temperature. Though not hazardous, this situation is not desirable and a high shield-temperature alarm set at 200°F warns the operator. Other indications of cooling-water failure may come from the high ion-exchange-water-temperature alarm and the high circulating-pump bearing temperature alarm.

Failure of either or both condensate pumps causes alarms. Condensate pump failure causes the hotwell-water level to rise resulting in loss of condenser vacuum. If the operator takes no action, the condenser high-pressure scram eventually shuts down the reactor.

6. Instrument Failure

Instruments which are capable of scramming the reactor upon receipt of the proper signal also scram the reactor when electric power to the instrument fails. Similarly, most alarm circuits are designed to give alarms when the circuits do not function properly (see Table X). Failure of the high-voltage power supply to neutron detectors causes an alarm. The pressure-control system fail-closes to contain steam, water, and radioactivity within the pressure vessel.

7. Control Rod Failure

Experience with the control rod drives on EBWR has indicated that rod-sticking does occur. Even though the EBWR drives are being modified and improved for use on BORAX V, the possibility of malfunction of one or two rods or drives at some time cannot be ruled out. Prior to reactor startup each day, control rod drives and scram latches will be checked for proper operation and any rod control or drive which does not function properly will be repaired or replaced.

If not enough control rods can be inserted to shut down the reactor, or if the reactor has been experimentally or inadvertently loaded to some reactivity in excess of that available in the control rods for shutdown, the boron-addition system may be manually operated, remotely or locally, to shut down the reactor by injecting a boric-acid solution into the reactor water.

8. Change in Power Demand

Since the power produced by BORAX V is disposed of locally i.e., through either the water rheostat or atmospheric venting of steam, any change in power demand is normally made by the operators.

An increase in electrical load opens the turbine throttle valve. The turbine bypass valve closes, but, if the reactor power is not adjusted to compensate for the increased demand, the steam-line pressure is reduced. The reactor pressure should not be affected, however, because of the back-pressure-control valve in the main steam line which senses and controls reactor vessel pressure. Since opening of the vent-steam back-pressure-control valve is dependent on reactor pressure, it is not considered a load change.

A decrease in electrical load causes the turbine throttle valve to close and the turbine bypass valve to open. If the reactor operator does not adjust power, the capacity of the bypass system is exceeded and pressure increases in the steam line. In this case the excess steam is vented by the low pressure-relief valves which relieve at 388 and 400 psig. A secondary effect might be overloading of the condenser, with resultant loss of condenser vacuum and a scram. The reactor upstream from the main-steam, back-pressure-control valve is protected by the reactor-vessel safety valves.

If a decrease in steam flow is accomplished by sudden closing of the vent-steam, or main-steam, back-pressure-control valve and the reactor operator does not adjust power, a positive feedback is introduced. The result is an increase in power and pressure.

Protection against this hazard resides in the high reactor-pressure alarm, set at 625 psig, the two high power-level scrams, and the two reactor vessel pressure-relief valves, set at 650 and 675 psig.

9. Too Much or Too Little Water in Reactor

The reactor is protected by alarms and scrams against both high and low reactor water levels. In addition, a continuous reliable indication of water level is furnished by a level recorder and a TV-monitored gage glass.

a. Too Much Water

If the high water-level alarm and scram fail to operate, if the high water-level indication is ignored and if the reactor vessel becomes filled to the top with water, the reactor is less safe under fast excursion conditions, because expulsion of water from the core would be difficult, as is the case in pressurized water reactors. However, it is not sufficient just to fill the reactor vessel completely with water. To obtain an excursion some method must be found to add a large amount of reactivity suddenly. This is discussed in Section VII, B, 17, "Sabotage."

b. Loss of Water

Normal removal of decay heat from the boiling fuel elements immediately following a shutdown occurs by boiling of the coolant. The steam then flows through the superheater to the condenser or to the atmosphere. Water level is normally maintained by means of the feed pumps or, in an emergency, by means of the batch-emergency-feed system.

Gradual loss of water can take place while the reactor is at power, if the feed pumps fail, the operator takes no action and the low reactor-water-level scram fails. In this case the core is slowly uncovered and, because of loss of moderator, power is reduced until shutdown occurs. The uncovered portion of the boiling fuel elements and the superheater fuel elements are cooled for a while by steam, but as the water level nears the bottom of the core, the steam flow reduces and the core can eventually melt.

In the event of a rupture of the pressure vessel or its connecting piping below core level, the water is lost from the reactor vessel, the reactor shuts down immediately, but sufficient decay heat can melt the reactor.

10. Leaks from External System

The predominant radioactivity in the steam system, the forced-convection system, the reactor ion-exchange-system supply and the preheat system is due to the fast-neutron irradiation of O^{16} to form N^{16}. The short 7.4-sec half-life of N^{16} limits accumulation of this activity in the vicinity of a leak.

The forced-convection, reactor ion-exchange, and preheat systems contain N^{16} and other radioactive materials. These systems are located in the subreactor room, access shaft, pipe trenches, or the equipment pit, all of which are kept at a negative pressure and ventilated by the reactor-pit-exhaust system. Thus, if power is available to the pit-exhaust blowers, the radioactivity is confined below floor level in the reactor building and exhausted through AEC-type filters up a stack.

Since the density of steam is low compared to pressurized water, the amount of N^{16} carried over in the steam is correspondingly lower than the carryover in a pressurized-water system. A leaky steam line can release radioactivity into either the reactor building or the turbine building. Even if all the water in the reactor vessel flashes to atmospheric pressure, the integrated exposure from the N^{16} is not lethal for operating personnel in the buildings.

Leakage of condenser cooling water into the condenser increases the amount of impurities in the condensate and tends to increase the radioactivity of the reactor water and steam. However, the reactor ion-exchange system and the condensate filter and ion exchanger are designed to maintain water purity.

Area radioactivity monitors in both the reactor and turbine buildings sound an alarm when an undue amount of radioactivity is released.

11. Fuel Cladding Failure

The fuel cladding material, Type 304 stainless steel, is considered corrosion-resistant to water and steam at the boiling-fuel-rod surface temperature of 510°F maximum, and it also has excellent corrosion resistance to superheated steam at the maximum superheat-fuel-plate temperature of 1200°F. Stainless steel was chosen as the first cladding material for the boiling fuel elements to insure maximum reliability. Aluminum alloy X-8001, planned for future use as a cladding material on the boiling fuel rods, is also considered feasible even though it has a higher corrosion rate than does stainless steel. However, in the fabrication of thousands of fuel rods and plates, it cannot be guaranteed that all welds are perfect, that all bonds are sound, nor that cladding is completely free of defects. Fuel cladding failure may sometime occur.

Cladding failure on the stainless steel-UO_2 cermet superheat-fuel plates would result in only a minor amount of radioactivity release. Only a small amount of the dispersed fuel would be exposed by even a complete cladding failure.

An aggravated condition could occur if cladding failure or fuel element burnout coincided with an over-pressure accident which opens the pressure safety valves or occurred while venting steam. In either case fission products would be immediately released to the atmosphere. Protection against an excessive pressure accident is discussed in Section VII, B, 13, "Excessive Pressure." The area monitors, air monitors, and fission break monitors should warn personnel in the vicinity of the reactor.

Fortunately, the fuel element cladding rupture on BORAX IV has given first-hand experience on the effects of an oxide-rod cladding failure in a direct-cycle, boiling-water reactor. In the BORAX IV incident about one-third of the fuel assemblies developed small leaks in the ends of the elements. The fuel elements were made of uranium oxide-thorium oxide pellets, lead-bonded in aluminum tube-plates. Essentially, only fission products which were gaseous or had gaseous precursors were identified. Xe^{138}, Kr^{88} and Kr^{89} accounted for the major portion of radioactivity released from the fuel elements. Only trace amounts of products, such as Mo^{94}, with nongaseous precursors were detected in the reactor water, indicating these were retained in the ceramic fuel. The major contaminant released from the steam system was Cs^{139}. Long-lived radionuclides that were identified in trace amounts in the turbine and condenser after operation were Ba^{140}, La^{140}, Sr^{90}, Y^{90}, Sr^{89}, and Cs^{137}.

The major release of radioactivity to the atmosphere was via the condenser-air-ejector exhaust, even though the exhaust gases had passed through an AEC-type filter. A summary of radioactivity release is given in Table XIII.

Table XIII

LOCATION AND AMOUNT OF RADIOACTIVITY IN BORAX IV FUEL CLADDING RUPTURE

Reactor Power, Mw	Estimated Air Ejector Gas Activity, curies/min		Reactor Building Steam Line, mr/hr	Air Ejector Filter mr/hr	Main Door Turbine Building, mr/hr	Main Door Reactor Building, mr/hr	500 ft Downwind,* mrep/hr
	Xe^{138}	Kr^{88}					
2.4	3.45	0.7	500	30,000	20	20	120
6	8.64	1.8	500	>50,000	27	30	-

*Average on three high-volume air samples, representing approximately 7250 ft^3 of air sampled.

During operation the reactor and turbine buildings were entered for brief periods by personnel wearing protective clothing and independent breathing apparatus; no excessive exposure occurred. Within 24 hours after shutdown the radioactivity had returned to background levels in these buildings.

Since BORAX V is operated from a remote location a fuel element rupture is not considered hazardous. Fission-product monitors reading steam and reactor water activities give an alarm in case of cladding failure. Personnel in the reactor or turbine building are also warned of radioactivity release by local area monitor alarms.

12. Startup Accident

A low neutron background in a freshly loaded reactor can result in a dangerous situation, because knowledge of the multiplication or period is lacking. As is customary in startups of reactors or critical facilities, a strong neutron source is always used in BORAX V. A permanent, pre-irradiated Sb-Be source, maintained at strength by operation, is installed in the core. Both compensated and uncompensated ion chambers and counters are part of the nuclear instrumentation.

The rate of reactivity addition by control-rod withdrawal is limited to 0.05%/sec. With this rate of addition, the water begins boiling as soon as the fuel elements reach a temperature slightly higher than the saturation temperature of the water. The negative void and temperature coefficient automatically compensates for the reactivity added by rod withdrawal. The power-level-trip circuits prevent the reactor from reaching the burnout heat flux.

13. Excessive Pressure

Since an increase in pressure in a boiling water reactor can lead to a power increase and a still further increase in pressure, mechanisms for limiting the pressure are particularly important. In BORAX V, protection against excessive pressures is furnished by a high-pressure alarm and scram set at 625 and 640 psig, respectively, together with dual safety valves set at 650 and 670 psig. Opening of these valves will reduce the power level of the reactor while flashing occurs.

14. Improper Charging of Fuel

The effect of adding a boiling fuel assembly with 100 boiling fuel rods of about 5% enrichment in a minimum size boiling core at room temperature is an increase in reactivity of about 5% for a central location and of about 1% for a peripheral location. In a full-size core the effect is 0.2% for a peripheral location. The effect of adding a superheat fuel assembly which has a lower loading is less.

Loading always takes place with the reactor at atmospheric pressure and with the reactor well below critical. As set forth in Section V, "Reactor Operations," loadings are checked, by at least two qualified persons before any change, for anticipated effect, and during the change for proper placement of fuel in the core. In-core neutron detectors are always used during loading operations.

15. Earthquake

There exists some risk of seismic activity at the National Reactor Testing Station. The BORAX V reactor is located in a region which The Pacific Coast Uniform Building Code designates as a Zone 2 area. Therefore, the building structures and plant are designed for Zone 2.

The most serious consequences of a severe earthquake, if it should occur, might be to rupture the reactor vessel or piping and jam the control rods. In case water is lost from the reactor vessel during operation, the reactor shuts down, but might subsequently melt from decay heat. If control rods become jammed and the core is still immersed in water, the boron-addition system can probably be used to shut down the reactor. The concentration of fuel in the core of the reactor by an earthquake is unlikely; thus there is little danger of an increase in reactivity.

It is of interest to note that the recent major earthquake of August 19, 1959, centered near West Yellowstone, Montana, was severe enough to be felt at the NRTS. However, no damage was done to any reactor or installation.

16. Fire

The reactor, its building, and equipment are essentially fireproof. Hazardous amounts of combustible materials are not allowed in the building.

17. Sabotage

a. With Reactivity Additions

The fuel in the reactor can be made to melt by the sudden addition of an arbitrarily great amount of excess reactivity. This can be done by sudden injection of fissionable material or sudden withdrawal of a poison. As mentioned in Section VII, B, 14, "Improper Charging of Fuel," it is difficult suddenly to add a large amount of fuel to BORAX V. Sudden withdrawal of poison can be accomplished by removing control rods from the top of reactor. To accomplish this, the control-drive mechanisms and seal housings have to be removed from the bottom of the vessel and the control rods have to be disconnected from the extension shafts. As an alternative, the rods can be actuated from the subreactor room after disconnecting the control-drive mechanisms by some sudden means which overcomes reactor pressure. Any one of these methods of sabotage requires the collaboration of more than one person and seriously endangers the saboteurs.

b. With Explosives

The reactor can, of course, be badly damaged by explosives. For example, the pressure vessel or connecting piping can be ruptured, leading to loss of water and possible subsequent core meltdown. Concurrent with this can be an interruption of utility power and destruction of regular and emergency feedwater lines. Security control should discourage this method of sabotage.

c. With Maloperation

Sabotage could most simply be accomplished by opening the reactor-vessel-blowdown valve or the forced-convection-pump-drain valve and releasing the water from the vessel with a possible resulting core meltdown. Assuming the saboteur is one of the operating staff and has the necessary keys and knowledge, he can start up the reactor. Sabotage by means of normal control rod withdrawal alone is ineffective because of the limited rate of reactivity addition available. If the saboteur succeeds in bypassing the proper interlocks and scrams, a maximum cold-water accident can be set up by using the forced-convection system. If the reactor vessel safety valves are "gagged" an excessive pressure excursion can be set up.

18. Metal-water Reactions

The importance of a metal-water reaction is derived from the additional heat liberation possible during a reactor accident. Two types of reactor accidents are generally considered as being capable of producing the requisite conditions for metal-water reactions: One is the loss-of-coolant accident, which may result in core melting by decay heat after the reactor has shut itself down; the other is a power-excursion accident, in which an uncontrolled increase in neutron flux causes an exponential increase in power level, resulting in the possibilities of melting, vaporization of fuel elements, and an explosion. Of the two accidents mentioned, the power excursion is considered much more likely to initiate metal-water reactions, due to the higher temperatures and vaporization achieved.

Since the fuel material in BORAX V is UO_2, which does not react, the only metals available for reactions are the cladding materials. These are Type 304 stainless steel for the first cores, and possibly X-8001 aluminum for some future boiling core.

a. Stainless Steel-water Reactions

Experimental data on stainless steel-water reactions are extremely meager at this time. In fact the only known experiment was done by Aerojet General Corporation[18] in its explosion dynamometer. In this test, 7 g of molten, Type 303 stainless steel at a temperature of 2270°C was sprayed into about 50 in.^{3} of water. The particle size was 840 microns, peak pressure was 93 psig, rate of rise of pressure was 23,667 psi/sec, total impulse was 11 lb-sec, energy was 102 ft-lb and overall efficiency compared to the theoretically attainable reaction was 2.47%. By comparison, the zirconium-water reaction under similar conditions attained an efficiency of 7.7%. The ANL Chemical Engineering Division is preparing experiments on stainless steel-water reactions using the electrically surge-heated-wire method and the nuclear heating of test fuel elements in TREAT.

b. Aluminum-water Reactions

In contrast to the stainless steel-water tests, aluminum-water reaction experiments have been run by many organizations and by several different methods. A summary[19] of these aluminum-water tests follows:

(1) <u>In-pile Heating in MTR</u>

References: Phillips Petroleum Co.[20-22]
Westinghouse[23,24]

Results:
(a) Samples below melting point gave no reaction.
(b) Samples above melting point reacted.
(c) Sporadic explosions were obtained with samples above melting point.

(2) <u>Pouring or Spraying Molten Metal into Water</u>

References: Aerojet[18,25,26]
ALCOA[27-29]
ANL[30,31]
Mine Safety[32,33]

Results:
(a) ALCOA showed that explosions sometimes occurred when 50-lb batches of molten aluminum were discharged into water.

(b) ANL found no reactions using fine jets of molten metal.

(c) Aeorjet and Mine Safety Appliances found no reaction when molten aluminum was dropped into water without additional dispersion by a blasting cap.

(d) Aerojet, in their explosion dynamometer, obtained negligible reactions below 1200°C. Above 1200°C, evidence of strong reaction was obtained. Aluminum at 2070°C gave a stronger pulse than Zr, stainless steel, NaK, or black powder.

(3) <u>Dispersion of Molten Metal into Water by Explosive Charge</u>

References: Aerojet[19,25,26]

Results: There was very little reaction when the molten drops were comparatively large. However, when the metal was dispersed, the estimated reaction was 75% at a temperature of about 1565°C.

(4) Dispersion of Metal by Condenser Discharge

References: ANL[34]
Columbia University[35]
North American Aviation[36]

Results:
(a) ANL obtained complete reactions of aluminum and water by vaporizing the metal.

(b) NAA tests showed that significant reactions occur only when enough energy is used to melt the metal.

(c) Columbia University results indicated that aluminum is nonreactive at the melting point.

c. Hydrogen Reaction[37]

"As a result of metal-water reactions, hydrogen gas, H_2, is released. The hydrogen gas is generated under water where there is essentially no oxygen gas present. Therefore, it is assumed that any hydrogen explosions which occur will do so after explosions resulting from fission and/or chemical energy releases. Consequently, the fission and/or chemical energy explosions and hydrogen explosions are not simultaneous, but occur sequentially.

"A hydrogen explosion will occur only when H_2 is combined with air into an explosive mixture and is ignited. Although the hydrogen-air mixture immediately around the reactor shortly after the excursion may be within the explosion range, it will be accompanied by a large amount of water vapor. This water vapor will tend to reduce both the possibility and severity of explosion. The occurrence of an explosive mixture around the reactor will be a transient condition, and only a small fraction of the hydrogen would be contained in an explosive mixture with air at any particular instant. Therefore, while the potential energy of the hydrogen evolved from a maximum accident is nearly equal to the initial excursion energy, the probability of its detonation in a single explosion is remote. Any reaction which does occur would probably consist of small areas of inflammation and/or detonation surrounding the reactor."

d. Conclusions

The above tests on aluminum lead to the following general conclusions, some of which may be applied to stainless steel-water reactions:

(1) Particle Size and Shape

The completeness of the reaction depends on particle size and shape. The smaller particle sizes and the more irregular shapes give more reactive area and more complete reactions and higher energy release. Small particle size may be obtained by explosive dispersion and vaporization. Particle size and shape are also dependent upon the material.

(2) Temperature

In contrast to zirconium, the chemical reactivity of aluminum is nil at temperatures considerably greater than its melting point. At temperatures greater than 1170°C, however, the reactivity increases rapidly as temperature is increased. Above the boiling temperature of aluminum the reactions are complete.

In the one test at 2270°C (4120°F), stainless steel was about as powerful as zirconium - an occurrence which would not have been predicted from the thermochemical data.

(3) Overall Efficiency

The overall efficiency of the energy conversion process is the ratio of the mechnical work done to the theoretical chemical energy available from the complete reaction of the metal and water. The principal losses are (1) heat rejection of the thermodynamic cycle, (2) incomplete chemical reaction, (3) thermal losses due to heating the water and the metal parts and (4) mechanical losses.

(4) Explosive Characteristics

The damage from any metal-water reaction could be caused by (a) the pressure pulse resulting from the expansion of hydrogen and water vapor, (b) the shock wave which may be produced, and (c) if air or other oxidizing medium is present, by the secondary hydrogen explosion.

(5) BORAX V Analysis

In this preliminary hazards report, no analysis has been made of the excursion characteristics of any of the proposed BORAX V cores. Consequently, no nuclear energy release or temperature data is available on which to base an analysis of metal-water and hydrogen-oxygen reactions. It is expected that these analyses will be presented in the final hazards report. Also, by that time, more data on stainless steel-water reactions should be available from the experiments which are to be run by the ANL Chemical Engineering Division.

C. Site

The BORAX V facility is built at the National Reactor Testing Station, Idaho, on the site of the existing BORAX installation. As shown, in Fig. 39, the reactor building is located about one-half mile northwest of the control building and the EBR-I, ZPR-III, and AFSR area. This figure also shows the relative location of the several structures at the BORAX V site. Figure 40 is a map of the NRTS and adjacent areas, showing the other installations on the station. Table XIV shows the populated areas near BORAX V.

Table XIV

POPULATED AREAS NEAR BORAX V

Name	Estimated Population	Direction from BORAX V	Distance, Miles from BORAX V
On-site Areas			
Central Facilities	-	East Northeast	$3\frac{1}{2}$
Chemical Processing Plant	-	Northeast	5
MTR-ETR	-	North Northeast	5
Naval Reactor Facility	-	North Northeast	10
Off-site Areas			
Atomic City	200	Southeast	12
Arco	3,000	West Northwest	17
Howe	200	North	18
Terreton-Mudlake	300	Northeast	36
Idaho Falls	30,000	East	48
Blackfoot	7,500	Southeast	40

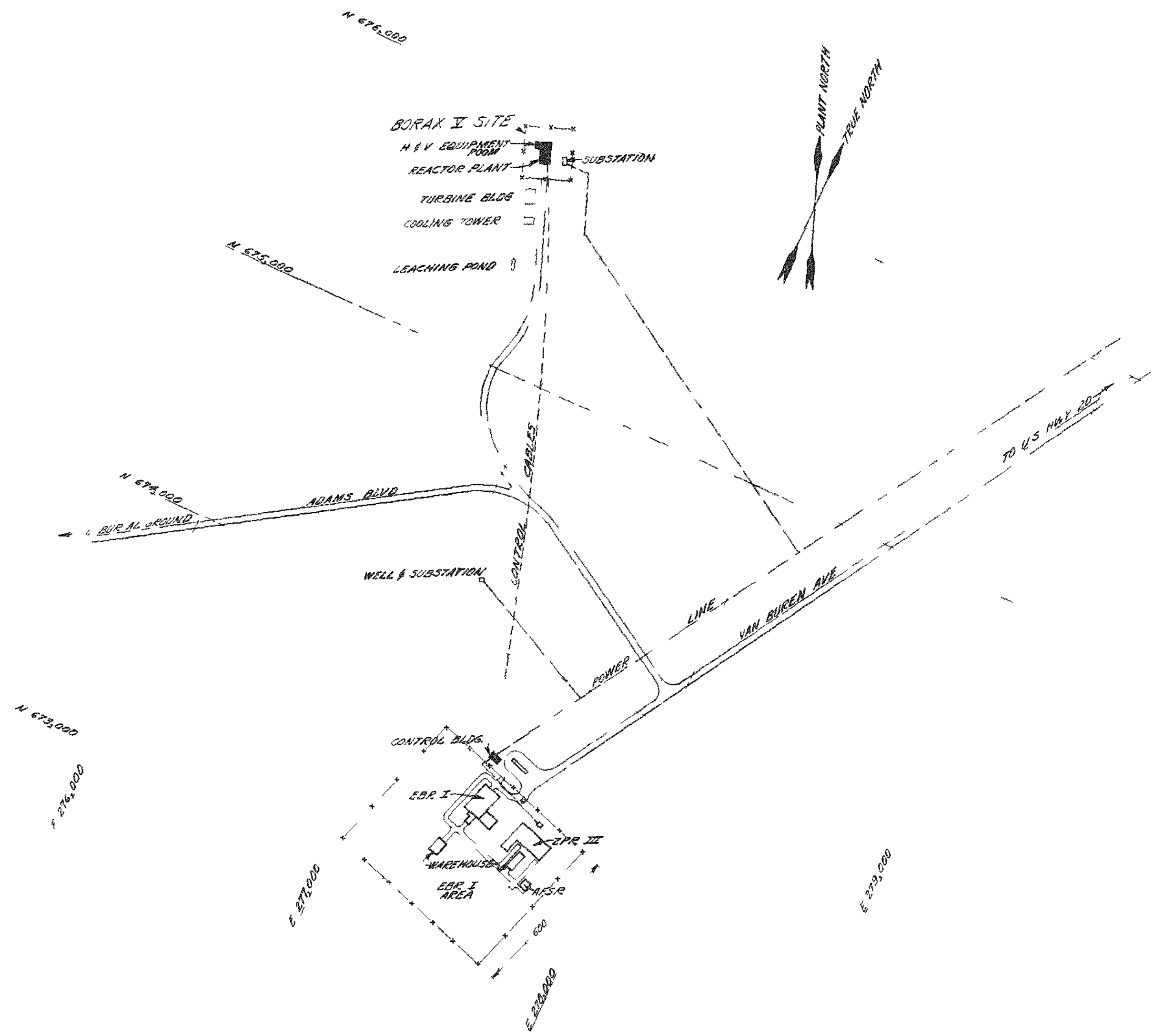

Fig. 39
Site Plan - BORAX V

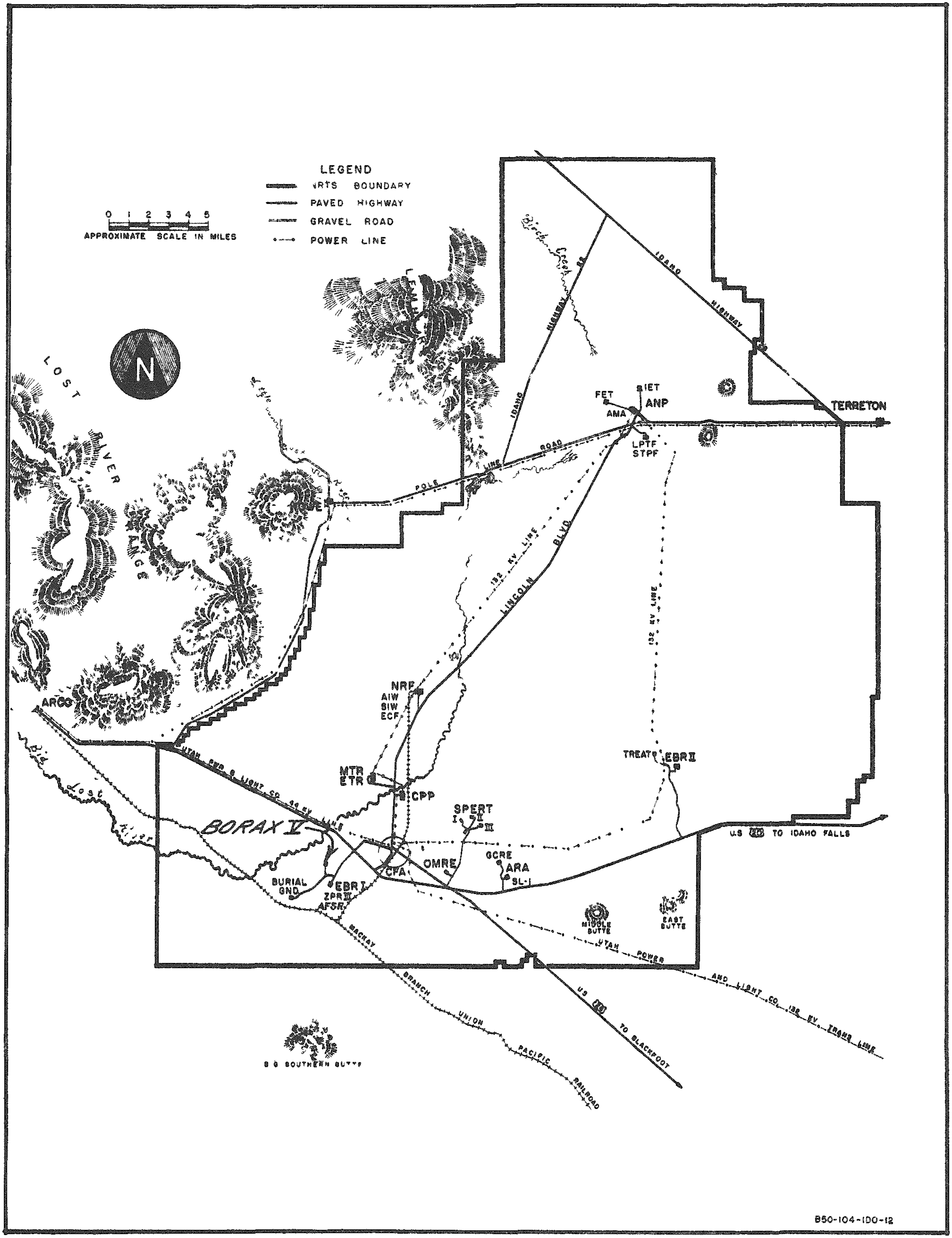

Fig. 40
National Reactor Testing Station

D. Evaluation of Radiation Hazards to the Surrounding Population

In order to obtain an estimate of the radiation danger to the surroundings accompanying a disaster with BORAX V, it is assumed that the reactor has undergone an excursion which destroys the integrity of the reactor system and releases a radioactive cloud. It is further assumed that the excursion follows steady operation at 20 Mw for an effectively infinite time. Under these conditions the fission products produced in the excursion have negligible effect on the surroundings when compared with the accumulated fission products.

The initial size and rate of rise of the cloud is determined by the total thermal energy generated during an incident. Since this thermal energy, which results from both fission and, possibly, a chemical reaction, may be only approximately estimated, the effects of variation are accounted for by assuming a range of cloud heights. As to cloud size, a point source is assumed, which diffuses outward according to weather conditions.

This point-source assumption is particularly pessimistic for short distances. For example, it leads to gamma doses at 800 meters (the distance from the reactor to the control building) at least 50% higher than doses from a ground-cloud assumption.

The weather conditions chosen are typical of those to be found at NRTS as determined by observations over the two-year period from September 1950 to August 1952.[44]

1. Dosage Calculations

Gamma and beta source strengths are both approximated by (Ref. 45, pp. 107, 157) the equation

$$Q = 2.3 \times 10^4 \; Pt^{-0.21} \text{ Mev/sec.}$$

If P is taken as 10^4 kw, representing 50% of fission products, we get

$$Q = 2.3 \times 10^{18} \; t^{-21} \text{ Mev/sec.}$$

Gamma doses are obtained from the Holland Nomograph. Total integrated dose (TID), or amount of airborne material to which a point on the ground may be exposed as the result of the contaminated cloud passing overhead, is given by

$$TID = \frac{2Q}{\pi C^2 \bar{u} D^{2-n}} \exp \frac{-h^2}{C^2 D^{2-n}} \quad ,$$

where

D = Downwind distance to observance, meters

h = height to center of cloud, m

$\bar{u}$ = mean wind speed, m/sec

n = Sutton's stability parameter

C = diffusion coefficient, $(\text{meter})^{n/2}$

t = decay time = $D/\bar{u}$.

Substituting the source term and including the following constants:

K, a conversion factor = 1 rep/(6.8×10^{10} Mev/m^3)

G, a geometry factor = (0.5) (0.64) = (0.32) .

(The 0.5 factor corrects for the fact that the beta flux at the surface of the skin is $\frac{1}{2}$ in free air. The other factor, 0.64, represents an average reduction factor for a man 1.8 meters tall, arising as a consequence of the ground effect on the free-air-radiation flux.) An expression for dose is obtained as

$$\text{Beta Dose (rep)} = (\text{TID})\ KG = \frac{6.89 \times 10^6}{\bar{u}^{0.79} C^2 D^{2.21-n}} \exp \frac{-h^2}{C^2 D^{2-n}} .$$

2. <u>Estimates of Radiation Hazard Downwind</u>

Values for a range of conditions are given in Table XV. Downwind distances are chosen to represent the hazard existing at the most critical populated on-site points listed in Table XIV.

The results indicate that under nocturnal conditions, the control building (and EBR-I, ZPR-III, etc.) could be within an exclusion radius for 300 r. This is true only under very adverse conditions. There are several mitigating circumstances to consider, however. The wind is in this direction less than 10% of the time. Further, there is ample time for personnel to escape.

Fumigation or washout will increase the doses, but the probability of an incident simultaneous with these conditions is extremely small.

Table XV

EXTERNAL β DOSE (REP) AND γ DOSE (ROENTGENS)
FROM AIRBORNE FISSION PRODUCT ACTIVITY

Distance Downwind		Daytime Conditions								Nocturnal Conditions			
		Average Wind Speed ($\bar{u}$)						Low Wind Speed ($\bar{u}$)					
		h = 10 $\bar{u}$ = 4 n = 0.20 C^2 = 0.088		h = 70 $\bar{u}$ = 4 n = 0.20 C^2 = 0.053		h = 500 $\bar{u}$ = 4 n = 0.20 C^2 = 0.027		h = 10 $\bar{u}$ = 1 n = 0.20 C^2 = 0.110		h = 10 $\bar{u}$ = 1 n = 0.50 C^2 = 0.0060		h = 70 $\bar{u}$ = 3 n = 0.50 C^2 = 0.0015	
Miles	Meters	β	γ	β	γ	β	γ	β	γ	β	γ	β	γ
0.5	800	38.4	38	37.2	34	<<1	0.18	92	85	6030	1000	<<1	200
3.5	5,630	0.77	0.85	1.3	2	-	0.55	1.82	2.6	438	190	4.3	35
5.0	8,045	0.39	0.5	0.64	0.85	-	0.6	0.93	1.4	133	140	9.5	29
10.0	16,100	0.09	12	0.16	0.25	-	0.23	0.22	0.1	74	65	29	22

Note. β doses should be divided by a factor of ten to allow for shielding effects of clothing.

E. Hydrology, Seismology, and Meteorology

1. Hydrology[44]

a. Soil and Subsurface Characteristics

The National Reactor Testing Station (NRTS) is located on a level plain at an average elevation of 4865 ft ranging from an elevation of 4788 to 4965 ft above sea level.

The surface of much of the plain is covered by waterborne and windborne top soil, under which there is a considerable depth of gravel, ranging in size from fine sand to 3 in. in diameter. At the several locations inspected to date, the gravel lies from approximately 1 to 50 ft under the top soil. Lava rock extends below this gravel layer to a considerable depth, ranging at least to the water table. The lava rock is honeycombed with openings of about $\frac{1}{8}$ in. in diameter. Frequently, large openings occur, and these range upwards to the size of tunnels, tubes, and caves.

The little surface drainage existing is toward the northeast, opposite to the main body of water flow. Normally, surface drainage is small due to the high porosity of the gravel overburden.

b. Drainage

The National Reactor Testing Station overlays a natural underground reservoir of water having an estimated lateral flow of not less than 500 ft^3/sec (323,136,000 gal/day).

The main sources of water for this reservoir are the streams which start in the mountains to the north and disappear into the porous soils of the NRTS area. These streams include Big Lost River, Little Lost River, and Birch Creek.

The path of water flow from the surface to the ground water level is unknown. However, it is expected that the drainage would be rapid. The flow would be very rapid through the gravel overburden, while the drainage pattern through the lava rock would be less rapid, but still very high as compared to flow through sands or clays. It is expected that the flow would be around, rather than through, the clay beds. Therefore, in case of a major accident with loss of a large volume of liquid wastes, the ground water would undoubtedly become contaminated in a very short time.

The estimated rate of flow of the main body of water through the lava is approximately one-half mile per year. Based on this estimate the contaminated water would reach the Snake River Canyon Springs and enter the Snake River in about 120 to 140 years, depending upon the exact location of the reactor plant within the Testing Station Area.

2. Seismology[44]

The NRTS site is located in a region which The Pacific Coast Uniform Building Code, 1949, designated as a Zone 2 area, as given by the Seismic Probability Map of the United States, published by the United States Coast and Geodetic Survey.

Quoting J. Stewart Williams:[46]

"Earthquake risk at this site (NRTS) is appreciable, but not great. Since isoseismal maps of principal earthquakes have been drawn, beginning in 1925, the isoseismals of only one earthquake reached Cerro Grande. (Cerro Grande is a stop on the Union Pacific Railroad located near the south boundary of the NRTS.) Prior to this time several earthquakes recorded for surrounding areas may have been felt at Cerro Grande. There is no record of a major earthquake originating close to Cerro Grande.

"However, Cerro Grande is surrounded by areas of comparatively high seismic activity. Furthermore, it lies in a region of geologically young faults, any of which must be considered potentially active. For these reasons earthquake risk at the NRTS site should not be dismissed from consideration in planning any structure to be built at the site.

"Cerro Grande is situated within 150 miles of several areas of pronounced earthquake activity. Any one of these might produce a shock stronger than it has yet produced with a corresponding greater intensity at Cerro Grande. The earthquake history of 100 years for this area is very short, from the geological point of view. An earthquake might occur any day that would alter substantially our ideas of the distribution of seismic activity in the area about the Snake River Plains.

"Earthquake risk in any area is relative to the type of structure to be built. Reinforced concrete buildings, well constructed in every way, with high factors of safety and incorporating features recommended by engineers acquainted with earthquake-proof design, stand less risk of being damaged. Such buildings, set on lava bedrock at Cerro Grande, certainly would be reasonably safe from earthquake damage.

"No traces of recent faults are known by the writer to cross the Snake River Plains. The chances, then, of displacement in the ground that would cut water supplies are small enough to be eliminated from consideration.

"In spite of the fact that a Zone 3 area exists both north and south of the Arco area, the distances are so great that Zone 2 has been considered completely safe."

3. Meteorology

The NRTS area is located south and west of the Continental Divide, on a high, gently rolling plain, surrounded by mountain ranges and skirted by the Snake River. Air masses reaching this area must pass over mountain barriers where a large share of their moisture is precipitated. As a result, relative humidity is normally very low, perhaps 20% on a summer afternoon. Average annual precipitation at the site is 7.5 in., mostly in the form of snow in the winter. The low humidity together with altitude results in intensive solar surface heating during the day and rapid radiation at night, giving large diurnal temperature variations, typically 30°F. The extreme temperature range for the site is considered to be -45 to 105°F.

Figure 41 is a wind rose at the 20-ft level at the Central Facilities Area for November, 1952 through December, 1956. The lengths of the bars represent the percentages of time that winds occurred from the given direction. There is little difference in seasonal wind behavior. Typically, the stronger prevailing southwest and west-southwest winds occur at or after the hottest part of the day, while the northeast and north-northeast winds tend to occur at night or early morning.

A more complete survey of meteorological data for the NRTS is given in ANL-5719.[44]

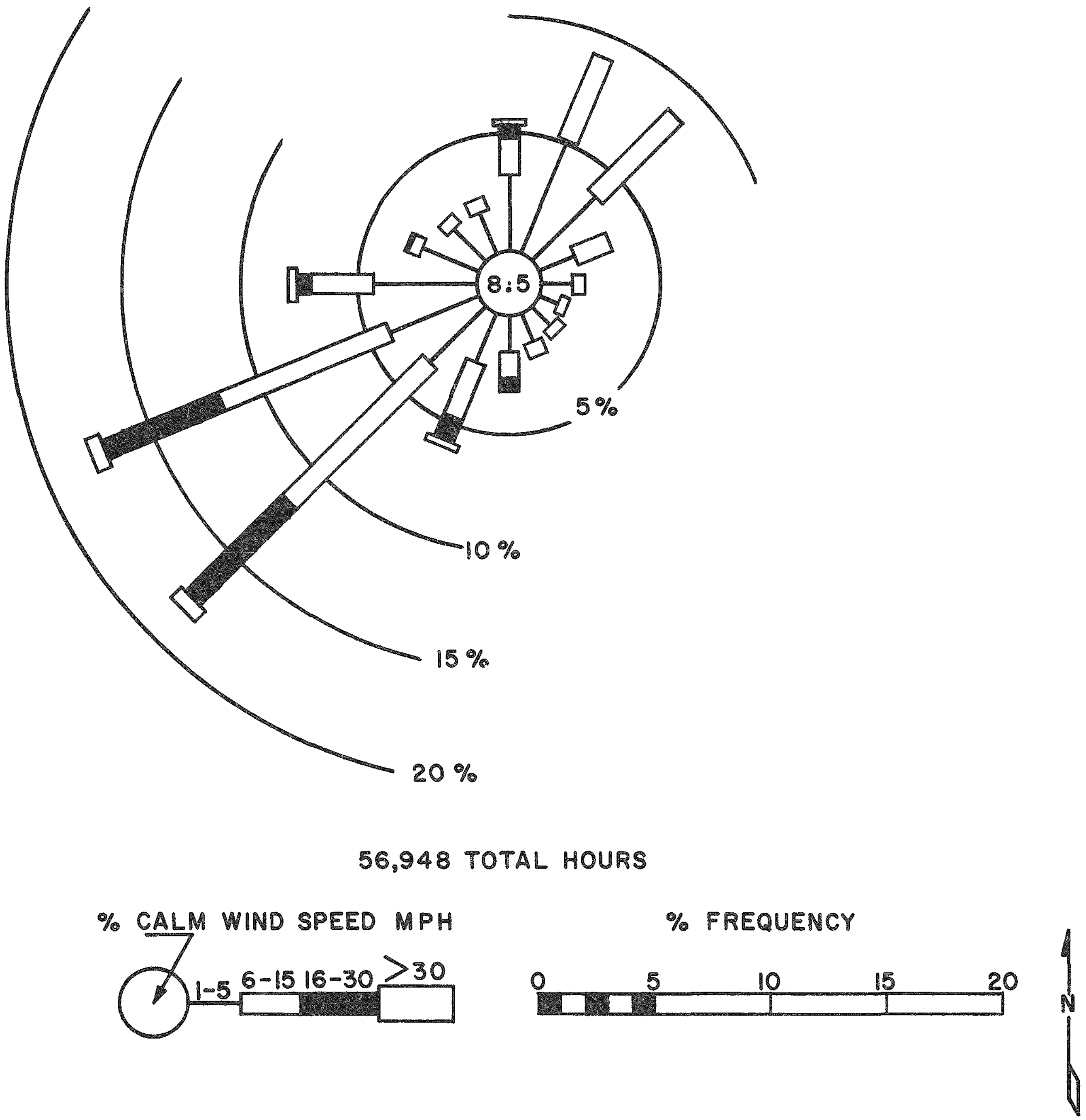

Fig. 41
Central Facilities 20-foot Level Wind Rose
1950 Through 1956

ACKNOWLEDGMENTS

Any report which covers the scope of the present preliminary design and hazards of the proposed BORAX V must necessarily owe a great deal to a great many persons. In addition to the contributions of colleagues who have brought the boiling-water reactor art to its present state of advancement, the editor of this report is indebted to the following persons: D. R. Greenwood, C. M. Ekelund, K. E. Tibbitts, and the Reactor Engineering Division Drafting Department for their tireless efforts in producing the illustrations herein; and to R. W. Adamson, E. D. Graham, W. J. Kann, C. C. Miles, W. P. Rosenthal, J. A. Thie, G. K. Whitham, and Norman Engineering Co., Beverly Hills, California, for advice and assistance in bringing this manuscript to its final form.

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