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MODIFICATION OF THE EXPERIMENTAL BOILING WATER REACTOR (EBWR) FOR HIGHER-POWER OPERATION

[Supplement to ANL-5607]

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DISCLAIMER

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MODIFICATION OF THE EXPERIMENTAL BOILING WATER REACTOR (EBWR) FOR HIGHER-POWER OPERATION

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1. INTRODUCTION*

This report describes alterations and additions made to the Experimental Boiling Water Reactor (EBWR) plant to permit operation at power levels up to 100 thermal megawatts. This report is a supplement to report ANL-5607,[1] which is a complete description of the plant as designed for operation at 20 thermal megawatts.

The Experimental Boiling Water Reactor, located at Argonne National Laboratory, was designed originally as a test reactor to prove the feasibility of a direct-cycle boiling water reactor plant. The original plant was designed for power output of 20 megawatts (thermal) or the equivalent of about 60,000 pounds per hour of steam at 600 psig. Numerous experiments covering many disciplines have been carried out at the EBWR facility since the first full-power run was achieved late in 1956. Sufficient data had been accumulated from operating the reactor at the design condition of 20 megawatts to attempt attainment of powers significantly in excess of the design level. Experiments at higher than design power indicated stable operation up to 65 megawatts (thermal) with the initial 4-foot-diameter core. In 1958 a short term operation of the plant at 61.7 megawatts (thermal) confirmed the prediction. The experimental approach to the higher powers consisted of oscillating the central control rod to obtain transfer functions and extrapolating the transfer functions sufficiently to perform a stepwise and stable increase in power. By extrapolating the experimental results to a 5-foot-diameter core which can be accommodated in the original grid plate and reactor vessel, operation of the reactor at 100 megawatts (thermal) appeared feasible. Therefore, the 20-megawatt-capacity plant was modified to increase the power output capacity to 100 megawatts, and a reboiler plant was added to supplement the existing turbine plant.

The reboiler plant is designed to dissipate 80 thermal megawatts of energy. The original turbine plant utilizes approximately 20 megawatts of thermal energy and the remaining energy generates process steam which is used to heat the Laboratory. The addition of the reboiler plant will permit the determination of the maximum power that can be realized from a reactor and vessel of this size. Higher power operation will also

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furnish information on core performance and permit faster fuel burnup for the study of fuel element life. The alterations and additions to the original EBWR system can also be used for experiments on high-density cores, superheat fuel elements, and liquid-poison control.

Many preparations were made prior to shutting down the reactor for modifications in the latter part of 1959. The five major items are listed below.

1. A program was initiated to develop and perfect tools and procedures for the modifications and additions to the reactor pressure vessel. Safety procedures were established and personnel was trained to cope with radiation problems. All machining and welding operations were performed, evaluated, and perfected on a vessel test section purchased from the vendor-fabricator of the pressure vessel.

2. A remotely operated underwater tool was designed and built specifically for cutting apart the highly radioactive control rods within the reactor vessel.

3. A new transfer coffin was built to accommodate a fuel assembly or an 8-foot section of a control rod. The motorized coffin was used to transfer fuel and control rods from the reactor to the storage well

4. A special fixture was designed and built to hold a guillotine saw so that the 6-inch pipe caps on the unused forced-circulation nozzles could be cut and removed without excessive radiation exposure to personnel. (Radiation levels due to accumulated products in the caps were as high as 100 roentgens per hour at 2 inches.) The special fixture permitted rapid mounting in the saw which was then operated remotely. The caps on the nozzles were removed and modified as described later.

5. A work platform for suspension in the reactor vessel was designed to support lead shielding, personnel, and equipment. Since work would be carried on at various elevations, the platform suspension rods were designed to be adjustable.

Before removing the control rods from the reactor, it was necessary to remove 114 fuel assemblies, 33 dummy fuel assemblies, and the antimony-beryllium neutron source. The lead coffin described as item 3 in the previous paragraph was used to make the transfer of these items to a water-filled storage well located adjacent to the reactor. The minimum time of transfer for a single assembly was limited to fifteen minutes. This limitation was imposed by the speed of the motorized fuel-handling tool and coffin.

The biological shield about the reactor vessel was originally built with concrete block and mortared joints. The contents of only one pipe tunnel were removed in order to accomplish the necessary work on the outside surface of the reactor vessel. This gave access to nozzles located on the reactor vessel and to piping located in the tunnel. Removal of the concrete blocks commenced as the last of the fuel elements was removed. Personnel was evacuated from the tunnel to prevent exposure to radiation whenever fuel or control rods were removed from the vessel. The tunnel was also evacuated during any period when the reactor was drained of water. With the reactor void of fuel, control rods, and water, the maximum radiation level in the empty tunnel was 400 milliroentgens per hour. This radiation level dropped to 25 milliroentgens per hour when the vessel was half filled with water.

The experiment program proposed for the higher-power-capacity system includes power runs to observe and analyze a number of parameters during approaches to high power. An essential and vital experiment is the determination of reactor stability. A known reactivity can be induced in the reactor by controlled oscillation of the center control rod. Stability of the core is then determined through transfer-function-measurement analysis. Pressure taps, probes, and thermocouples are located within the reactor vessel to obtain hydrodynamic and temperature measurements. Preliminary tests indicated that these pressure and temperature instruments function satisfactorily.

2. SUMMARY*

2.1 PLANT DESCRIPTION

2.1.1 General Description

The 100-megawatt-capacity reactor will operate (similar to the original plant) as a direct-cycle boiling water reactor with natural circulation. Light water is the moderator and coolant. Although the turbine plant was originally designed for use with heavy water, provisions for its use were not included in the reboiler plant design. Therefore, reboiler plant operation is limited to light water only.

The original turbine-generator and associated equipment in the turbine plant was kept intact except for minor modifications, and will be used in future operations to generate up to 5 megawatts of electrical energy. The turbine plant will operate in essentially the same manner as originally designed, and the reboiler plant will be operated in parallel to accept the increased heat output. Figure 2-1 is a flowsheet of the revised EBWR facility. A simplified control diagram is shown in Figure 3-28, and a simplified process instrument diagram is shown in Figure 3-26.

The 100 megawatts of thermal power generated at 600 psig in the reactor are dissipated by the two plants as follows: (1) 20 megawatts (thermal) are utilized by the turbine plant and (2) 80 megawatts (thermal) are absorbed by the reboiler plant. The reboiler plant is divided into three systems: the primary system, intermediate system, and secondary system. The primary system is operated at a pressure of 560 psig saturated, the intermediate system at 350 psig saturated, and the secondary system at 200 psig. The secondary system pressure is the same as the Laboratory steam system.

The reboiler plant can either supply steam to the Laboratory or dissipate heat to the air-cooled heat exchangers. During winter, operation of the reactor at 100 megawatts (thermal), approximately 67 megawatts of the 80 megawatts directed to the reboiler plant can be transferred to the Laboratory heating system. Of the remaining 13 megawatts about 10 megawatts are dissipated to the air-cooled heat exchangers and 3 megawatts are absorbed in the operation of associated primary-system equipment. If desired, the amount of heat distributed between the secondary reboiler and air-cooled equipment can be proportioned in accordance with Laboratory needs. During winter, sufficient steam can be generated to supply all Laboratory heating requirements. Any excess heat not required by the Laboratory heating system is absorbed in the air-cooled heat exchangers.

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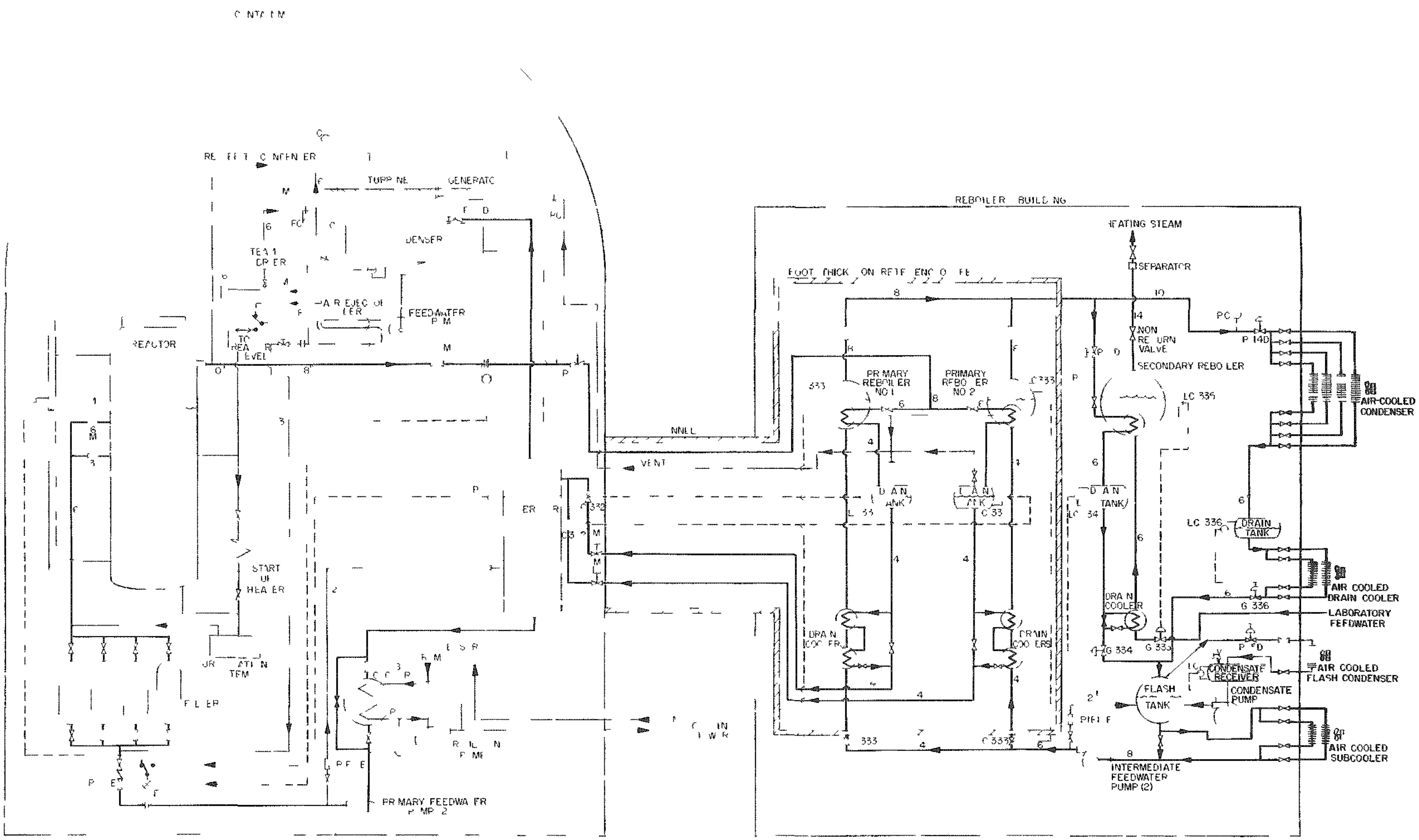

Figure 2-1

Flow diagram of EBWR 100-megawatt system
RE-8-31888-F

2.1.2 Primary System

The primary system is composed of the primary reboilers, primary drain coolers, deaerator, subcooler, reboiler-plant feedwater pump, and filters. The intermediate system consists of the primary reboilers, primary drain coolers, secondary reboiler, secondary drain cooler, flash tank, and intermediate feedwater pump. The secondary system consists of the secondary reboiler and secondary drain cooler.

Eighty megawatts of steam will be condensed in the tubes of the primary reboilers; the condensate is further cooled in the primary drain cooler and then discharged into the deaerator. In the deaerator the noncondensable gases generated in the reactor will be stripped from the primary water and routed to the condenser. To circumvent the possibility that the noncondensable gases, prior to discharge into the deaerator, may cause the primary system to become airbound, provisions are available to remove the gases from the reboiler exit header and the primary drain tank. These gases, as in the previous case, will be sent to the condenser.

The primary water upon leaving the deaerator is directed to the primary subcooler, and can be further cooled if desired. The reboiler-plant feedwater pumps return the primary water to the reactor through the filters. Before entering the last two filters, the primary stream from the new system is mixed with the condensate being pumped back from the hotwell.

Reactor pressure is controlled at 600 psig by the steam control valve, P-11D, which admits steam to the primary reboilers. Since this valve also compensates for any slight load variations of the turbine, it assumes the role of the turbine bypass valve used for this purpose in the original system. The amount of steam admitted to the primary reboilers is a function of reactor output and is unaffected by Laboratory demands.

Normally, reactor water level is maintained by an electrically operated control valve that regulates the feedwater from the reboiler plant. The feedwater regulator utilizes a conventional three-element flow controller that receives signals from reactor level, primary reboiler steam flow, and reboiler-plant feedwater flow. Turbine plant feedwater flow is controlled by hotwell level, turbine-plant feedwater flow, and steam flow to the turbine. Primary-drain-tank level is maintained by regulating the primary condensate flow to the deaerator.

Radiolytic dissociation of water in the reactor results in the formation of hydrogen and oxygen. Separation of these gases from the liquid at the interface is followed by carryover in the effluent steam. Under equilibrium conditions the carryover results in essentially a stoichiometric mixture of hydrogen and oxygen in the exit steam. The rate of formation of the combined gases thus generated amounts to about 1 scfm per

20 megawatts of reactor power, or about 5 scfm per 100 megawatts. These gases are removed by flashing the condensate from the primary drain coolers into the deaerator. The extracted noncondensable gases and flashed steam in the deaerator are then directed to the condenser. Also, each primary drain tank and reboiler exit-header box are provided with bleed orifices to vent any accumulated gases at these points. This provision precludes the possibility that these units will become airbound. Noncondensable gases are removed from the condenser by the air ejector and directed to the recombiner system where the hydrogen and oxygen are recombined. The gases are next sent through the cooler and exhausted through the building stack. If desired, the recombiner may be bypassed. The recombiner was developed and installed in the original EBWR facility for possible future operation with heavy water. A recombiner is not necessary in a light-water system.

2.1.3 Intermediate System

The intermediate steam generated in the primary reboiler is directed to the secondary reboiler where it is condensed in the tubes. Steam admission to the secondary reboiler is controlled by secondary reboiler pressure. Consequently, intermediate steam pressure is controlled by the pressure control valve on the line to the air-cooled heat exchangers. This is somewhat analogous to the primary-system pressure control. However, in this case a predetermined amount of steam will be bypassed to the air-cooled heat exchangers at all times.

The intermediate condensate is then directed to the secondary drain cooler where it is further cooled before being discharged to the flash tank. Vapor caused by flashing in the flash tank is condensed by means of the air-cooled flash condenser which returns the condensate formed to the flash tank. The intermediate stream is then pumped from the flash tank to the primary drain cooler and primary reboiler by means of the intermediate pump. Water level in the primary reboiler shell is maintained by the intermediate-feedwater control valve. If further cooling is required after the flash tank, an air-cooled subcooler is provided to accomplish this.

Steam admission to the secondary reboiler is regulated by an electrically driven valve and is a function of the condensing rate within the secondary reboiler. Condensing rate is established by choice of condensing pressure which fixes the temperature differential across the heat transfer surface. In summary, the secondary reboiler is base loaded and essentially generates a constant supply of steam to the Laboratory. Changes in Laboratory demand are compensated for by the conventional Laboratory power plant.

Since intermediate steam admission to the secondary reboiler is fixed by operator's choice, the intermediate-steam pressure is controlled by the electrically driven regulating valve which admits steam to the air-cooled condensers. Consequently, slight fluctuations in the primary system or Laboratory system are offset by regulating the steam admission to the air-cooled heat exchangers. This arrangement, coupled with the conventional power plant regulation, prevents Laboratory steam demands from being reflected to the reactor.

If necessary during winter operation, the air-cooled heat exchangers can accept the entire load of 77 megawatts, provided the ambient air temperature is below 65°F. Hence, in winter, continuous operation of the EBWR facility is assured. Although the combined heat-exchanger capacity is sufficient to operate the reactor at 100 megawatts during the summer, the air-cooled equipment cannot singularly accept the entire 77-megawatt load. At an ambient air temperature of 95°F, the air-cooled heat exchangers are designed to remove a maximum of 67 megawatts; at lower air temperatures capacity increases. Normally, the secondary reboiler is supplied with 10 megawatts during the summer which, when coupled with the air-cooled heat exchangers, accounts for 77 megawatts. If the secondary reboiler is inoperative during summer, reactor power must be correspondingly reduced commensurate with ambient air temperatures.

2.1.4 Safety Systems

For added containment safety, the primary piping and equipment located outside of the containment vessel are housed in a relatively gas-tight concrete enclosure within the reboiler building. The enclosure contains the primary reboilers, drain tanks, primary drain coolers and associated primary piping. The four walls and ceiling of the enclosure are of reinforced concrete, one foot in thickness. The enclosure, in addition to confining possible contamination, also affords limited radiation shielding.

Electrical and piping penetrations through the containment vessel are contained in sealed sleeve and expansion joints so that the gas-tight integrity of the containment vessel is maintained. Controlled ventilation of the tunnel and enclosure is maintained and the exhaust air is monitored for activity. If high activity is detected in the exhaust air, the reboiler building is automatically isolated from the containment vessel and the reactor is shut down.

To prevent contamination of the Laboratory system, the intermediate system was incorporated as a means of isolating the primary system from the laboratory system. In addition, radiation monitors are installed on the intermediate system to detect internal leakage between the primary and intermediate systems. Because of the welded tube and tube sheet design of the primary heat exchangers and the system of radioactivity monitoring, the possibility of contaminating the Laboratory system is virtually nil.

If loss of electrical power should occur, the steam control valve on the primary steam line to the reboiler building closes automatically. Operation of the emergency cooler is initiated manually unless there is also a loss in 125-volt d-c power, in which case the emergency cooler is initiated automatically.

The first and most important method of preventing overpressure is by the automatic insertions of control rods when reactor pressure exceeds 640 psig. In the event of a malfunction of the normal heat removal equipment, a system of pop-safety valves is provided to protect the primary equipment. Each pop-safety valve on the primary system and on the shell of the primary reboilers is equipped with a microswitch device to detect opening and to initiate reactor shutdown.

2.1.5 Material

With minor exceptions, all additional equipment and piping for the primary system is either all 304 stainless steel or clad with stainless steel. The intermediate and secondary system is fabricated on plain carbon steel with the exception of the secondary reboiler and secondary drain-cooler tubes which are 80 Cu - 20 Ni. The air-cooled equipment is also constructed of plain carbon steel, except for the finned tubes which are of Admiralty metal

2.1.6 Water Treatment

Water treatment for the primary and secondary reboilers consists of phosphate and sulfite chemical addition. The basic treatment consists of a coordinated phosphate-pH for control of alkalinity and the use of blowdown to control total solids. In addition, to prevent any possibility of chloride stress corrosion of the stainless steel tubes in the primary reboilers, demineralized water is used for intermediate-system makeup.

2.2 PLANT MODIFICATIONS

2.2.1 Reactor Vessel Alterations

Because of the increased steam flow requirements, the 6-inch-diameter steam nozzle was replaced by a 10-inch-diameter nozzle to prevent excessive steam velocity, erosion and to minimize pressure drop. A 1-inch drain line was added to the lowest point of the 10-inch pipe line. This point is located within the concrete-block biological shield and drains directly to the steam drier. The 6-inch toroidal-shaped steam collecting pipe was removed and a steam collecting duct was installed in the uppermost section of the vessel. A steam duct extension can be added to raise this opening another foot if required. The duct is located within the diameter of the shield-ring lugs. The duct does not interfere with removal of core shroud, risers, or fuel. Operation with the higher water level necessitated the new steam collector. The extent of the internal modification is shown in Figure 2-2.

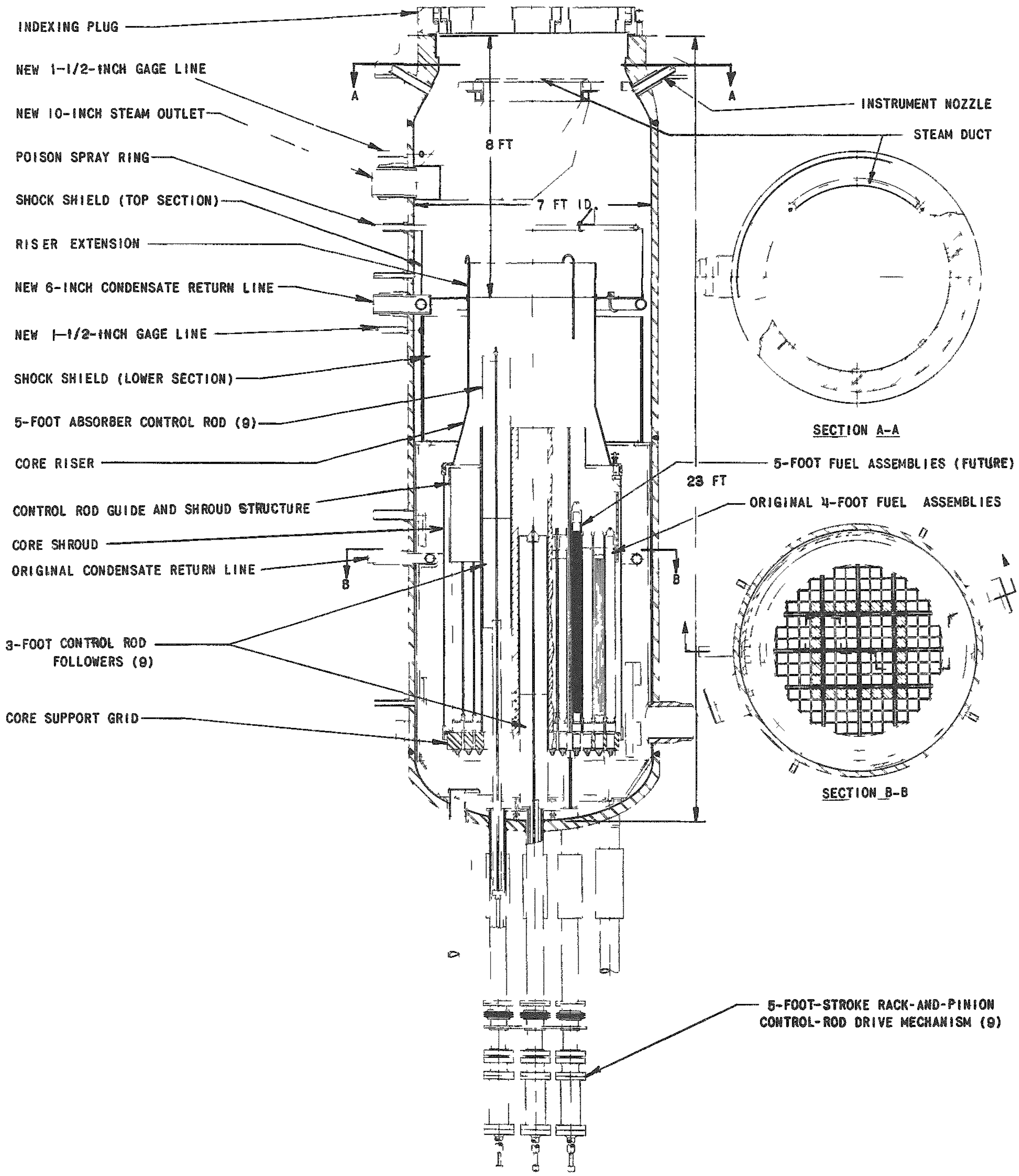

Figure 2-2

Schematic drawing of reactor showing modifications for conversion of EBWR to 100-megawatt system
RE-6-31887-C

A new 6-inch feedwater nozzle with a 4-inch-diameter toroidal distribution ring is installed just below the top of the new riser. The old feedwater ring was not altered and can be utilized for experiments in the modified system. The new distribution ring provides injection of feedwater at the top of the downcomer where it is most effective in collapsing entrained steam bubbles in the recirculating water, thus securing the maximum driving force for natural circulation. Alternate spray holes are located at 30 degrees and 60 degrees from the bottom of the ring to distribute the feedwater for better over-all quenching of entrained steam.

An additional 2-inch instrument nozzle was installed in the northeast quadrant of the reactor-vessel-closure ring forging. This nozzle will be utilized for instrumentation leads for observing and studying hydrodynamic changes during approach to high power levels. The nozzle extends through the forging perpendicular to the inner surface and emerges on the main floor area. Two other nozzles were previously welded into the ring forging to accommodate a superheat experiment. All three nozzles are available for instrumentation and will be utilized during initial operations at elevated powers.

An extension to the water column located external to the biological shield necessitated two additional vessel penetrations. Therefore, two $1\frac{1}{2}$-inch nozzles were installed for indicating and measuring water level at the higher operating levels. These penetrations will be used in determining the location of the interfaces between the water and mixture (steam and water) as well as between the mixture and steam.

A 5-foot-diameter shroud encompasses the core from the grid plate to the top of the control-rod-guide shroud. Attached to the core shroud by means of three bayonet-type locking lugs is a removable 4-foot-high riser. Machined metal surfaces seal the riser to the shroud. Increased riser height is readily obtained by adding a 1-foot extension to the riser. The riser and riser extension are removable for access to the outer rows of fuel assemblies.

With the core shroud and riser combination, sufficient recirculation should be made available for removing the heat from the core at the elevated power levels. The riser produces a greater differential head between the low density steam-water mixture leaving the core and the water in the downcomer space. The conical reduction in the core exit effects a corresponding increase in downcomer area and decrease in velocity which enhances disengagement of steam from the recirculating water.

The original shock shield was removed to provide an area for the new feedwater nozzle. Since the shock shield required alterations and could not be decontaminated below a radioactivity of 2.5 roentgens, a new, two-part shock shield without louvers was installed.

Three of the four 6-inch forced-circulation inlet nozzles were altered to provide a means for flushing. The original nozzles were natural pockets for collecting particulate matter which reached radiation levels as high as 100 roentgens per hour at 1 inch. New 6-inch caps were installed with a smooth transition to the 1-inch drain outlet used for flushing. One-inch valves replace the $\frac{1}{2}$-inch instrument valves originally used on the drains of the forced-circulation nozzles. The drains are connected by manifolds to a blowdown system which will facilitate cleanup if it is required. The fourth nozzle is connected to the purification system and is kept clean by the flow to the reactor purification system.

2.2.2 Core and Fuel

The core loading arrangement has been expanded by the addition of 32 new "spike" fuel elements. The "spike" concept was utilized for the following reasons:

1. A minimum requirement for new fuel.
2. A maximum utilization of the existing Core I elements.
3. Eventual enlargement of the core to 5 feet in diameter and 5 feet in height. Because of hydrodynamic considerations, the present fuel elements have been limited to a 4-foot active length.

The Core IA loading, as expanded to a 5-foot equivalent diameter, consists of the following types of fuel elements:

50	enriched, thick-plate type (used)
55	enriched, thin-plate type (used)
8	natural, thick-plate type (new)
1	natural, thin-plate type (new)
1	stainless steel dummy with source
1	stainless steel dummy for test samples
32	highly enriched, dispersion, rod type (new)
148	total core locations

The natural-uranium fuel elements will gradually be replaced by individual experimental assemblies as they become available.

The original fuel grid plate and control-rod-guide shroud are utilized in containing a new core of approximately 5 feet in diameter and 4 feet in active length. A complete description of the original fuel assemblies can be found in report ANL-5607.[1]

The spike fuel assembly contains 49 rods having UO_2 of greater than 90 percent enrichment dispersed in ZrO_2-CaO_2 matrix and clad with Zircaloy-2. The fuel rod assembly is reversible, that is, symmetrical end for end, and slides into a fuel element frame. The reversible feature allows the ultimate full use of the spike fuel. Twenty-eight spikes are located in a square pattern surrounding the 36 central four-foot-long fuel elements. The amount of reactivity produced by the U^{235} in the spiked region will be determined experimentally during pre-startup operations. Should less than a full loading be used, stainless steel dummy fuel assemblies will fill the vacant holes in the grid.

2.2.3 Control Rods and Drives

To provide adequate control for 5-foot fuel assemblies (contemplated for future installation), all control rods were removed from the reactor and nine new control rods having 5-foot absorber lengths with 3-foot Zircaloy-2 followers were installed. The new control rods have the shape of a cruciform $\frac{1}{4}$ inch thick by 10 inches wide. The absorber sections are made of 304 stainless steel containing 2 percent boron by weight. The Zircaloy-2 follower section is welded to the poison section. A 3-inch cruciform is attached to the follower and absorber with a $\frac{1}{2}$-inch bar and a threaded handling fitting to complete the control rod assembly. This bolting feature permits the rod to be disassembled for ease of handling during removal from the core.

Additional control can be obtained by adding boron-stainless steel strips to the spike fuel assemblies. Initial loading experiments will determine the exact worth of the rods as well as the number and location of the spikes and burnable poison strips.

Additional emergency shutdown control is available with the high-pressure boric acid system. The volume of concentrated boric acid solution has been increased to handle the greater dilution brought about by the increased water volume in the reactor vessel and system. Manual, remote injection of the poison solution through the 3-inch feedwater line is possible at the reactor console by means of a handwheel on the wall of the control room, or by manually actuating a lever in the containment vessel.

The low-pressure boric acid system is still available. Boric acid from this system is automatically injected when both the reactor pressure is less than 20 psig and the water level in the reactor drops to within 6 inches of the fuel assemblies. The boric acid solution is introduced into the top of the vessel through a distribution ring with nozzles directed into the downcomer. The core riser does not allow the solution to be injected directly into the fuel. The primary purpose of the spray is to remove some decay heat and to maintain subcriticality with water addition.

The 5-foot-long control rods installed to accommodate the possible use of 5-foot active-length fuel assemblies required drive units capable of lifting the rods through a full 5-foot stroke. The 4-foot external lead-screw and nut-type drives were not readily adaptable to longer strokes because of column limitations. In addition, considerable difficulty was experienced with dirt collecting in the seals.

As a result, the nine new mechanisms incorporate a rack-and-pinion drive for translating the rotating drive motion into a linear motion of the control rod. The mechanisms engage the new control-rod-extension fittings and are bolted to the vessel nozzles in the subreactor room in the same manner as the original drives. Since the rack-and-pinion drives are enclosed and under balanced reactor pressure, the rods drop by force of gravity. The weight of the rod is sufficient to drop 56 inches by free-fall into the core in about 1.35 seconds. The last 4 inches of stroke is slower because of a dashpot action; therefore, the full 60-inch travel requires about 1.5 seconds.

A one-way clutch is provided to assist rod insertion in case the rod should not drop into place. All rods can be released for simultaneous insertion but only one rod at a time can be withdrawn from the core. Withdrawal is at the rate of 28 inches per minute or, in terms of reactivity increase, at an over-all average of about 0.015 percent reactivity per second.

A breakdown labyrinth-type seal around the drive shaft reduces the reactor pressure to atmospheric.

2.2.4 Chemical Control

Chemical control in EBWR 100-megawatt system is being used since it is not feasible to fabricate control rods of greater worth at this time. Since the full nine-rod complement of control rods in EBWR will be required for holddown of Core IA in the cold, clean reactor, the primary purpose of chemical control with boric acid is to provide for an eight-rod shutdown which is required for rod-drop tests, maintenance, and rod failure. Introduction of boric acid also provides an additional margin of safety over and above the nine-rod shutdown. A secondary purpose is for the full evaluation of soluble-poison shim control of the operating boiling water reactor as a possible improvement over existing control methods. Future reactor design utilizing the soluble-poison shim-control concept could improve nuclear power economy.

Boric acid must be present in the reactor water at concentrations up to 0.014 pounds per gallon during all periods when the reactor is not operating and the reactor temperature is less than 392°F. Boric acid is removed as the reactor power is increased and added as the reactor is shut down. Additions must also be made during filling for hydrostatic testing of the reactor system. Addition or removal will be made as necessary during reactor operation with soluble-poison shim control.

Addition of boric acid to the reactor is accomplished by using positive displacement metering pumps. A nearly saturated aqueous solution of boric acid is used. Strongly regenerated, high-capacity, anion-type ion exchange resin is used for removal of the boric acid from the reactor water. The facilities for removal are a part of the high-pressure purification system. The concentration of boric acid in the reactor water is monitored with two instruments; both are based on the neutron attenuation properties of the soluble poison.

During shim control studies the concentration of boric acid in the reactor water will be manually controlled. One or more control rods will be used for fine reactor control.

3. OVER-ALL SYSTEM MODIFICATIONS AND ADDITIONS

3.1 TURBINE PLANT*

Although the turbine plant was kept intact, many additions and modifications associated with the reboiler plant were made within the existing facility. Five major items installed in the containment vessel to service reboiler plant requirements and higher-power operation are:

1. Two additional full-flow feedwater filters
2. One deaerator
3. One subcooler
4. Two additional reboiler-plant feedwater pumps
5. One large-capacity instrument air compressor

A 16-foot control panel was installed in the control room to accommodate the control room requirements of the reboiler plant and controls for operating the newly installed equipment for 100-megawatt operation.

Modifications and installation of additional piping, relief valves, electrical and instrument wiring, and the penetrations of the containment building for piping and cables were required to complete the modifications within the turbine plant.

Since design manual ANL-5607(1) was written, many modifications have been made to the turbine plant. These changes along with the additions and modifications required for the new reboiler plant have resulted in major changes in many of the flowsheets shown in ANL-5607. Therefore, the flowsheets listed in Table 3-1 were revised and incorporated in the present report to supersede the drawings in the original design manual.

Table 3-1. Drawing Modifications

Original	Revised	Drawing title
Figure 6	3-1	Diagram of main steam and feedwater piping in turbine plant
Figure 11	3-2	Diagram of reactor auxiliaries
Figure 10	3-3	Diagram of circulating water system
Figure 7	3-4	Diagram of make-up water system
Figure 9	3-5	Diagram of vents and drains

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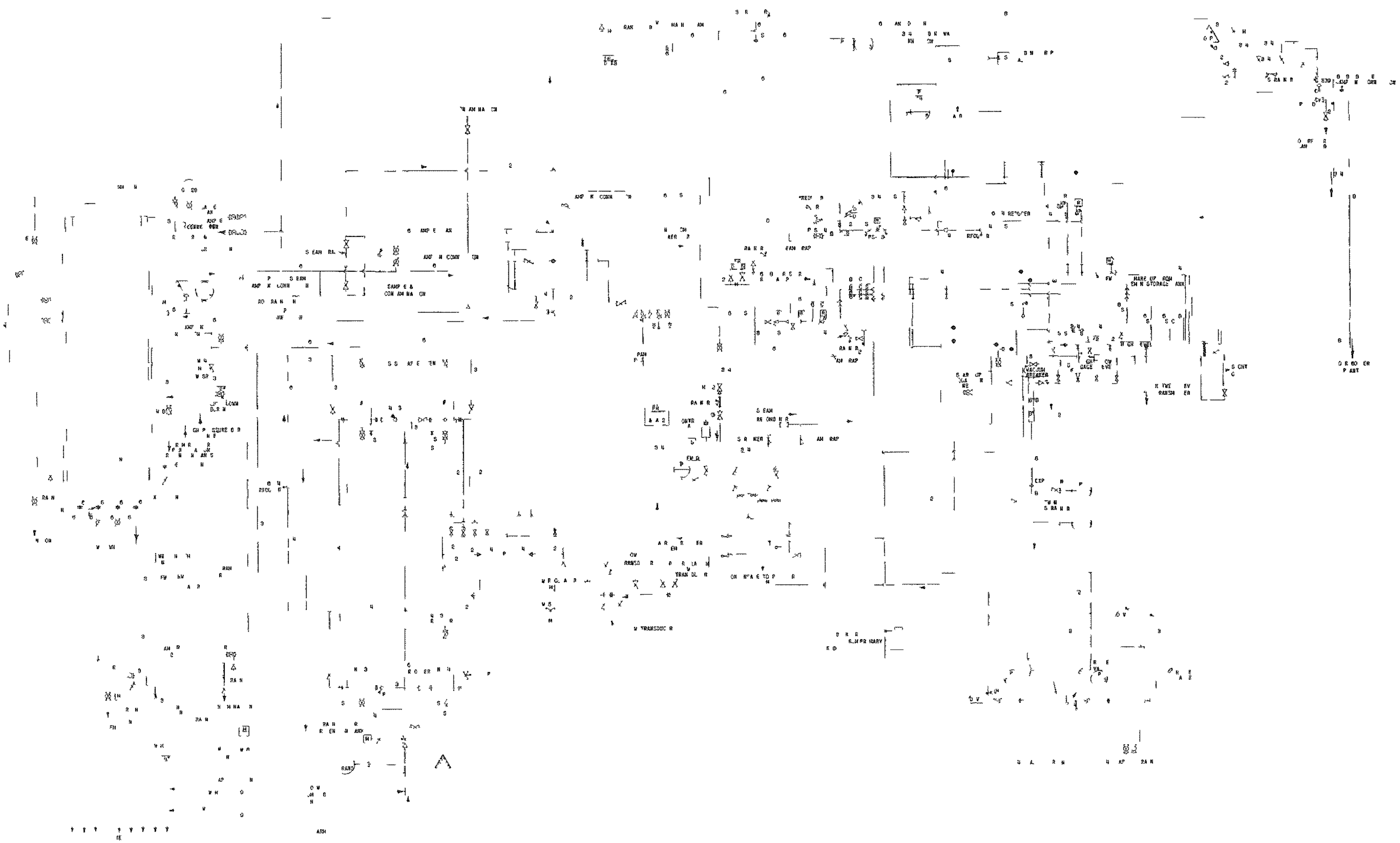

Figure 3-1

Diagram of main steam and feedwater piping in turbine plant
RE-8-33203-F

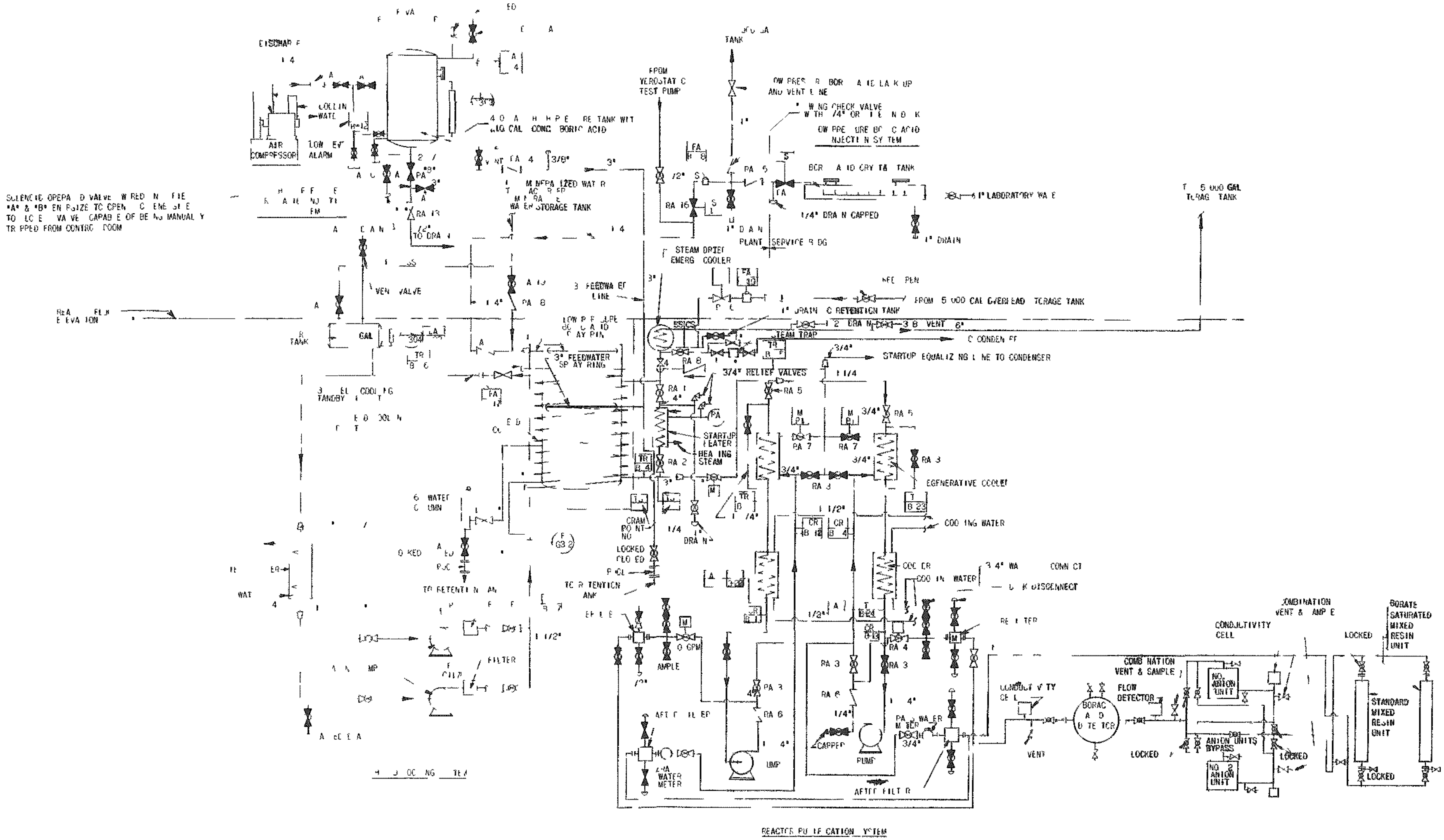

Figure 3-2

Diagram of reactor auxiliaries
RE-8-20141-D

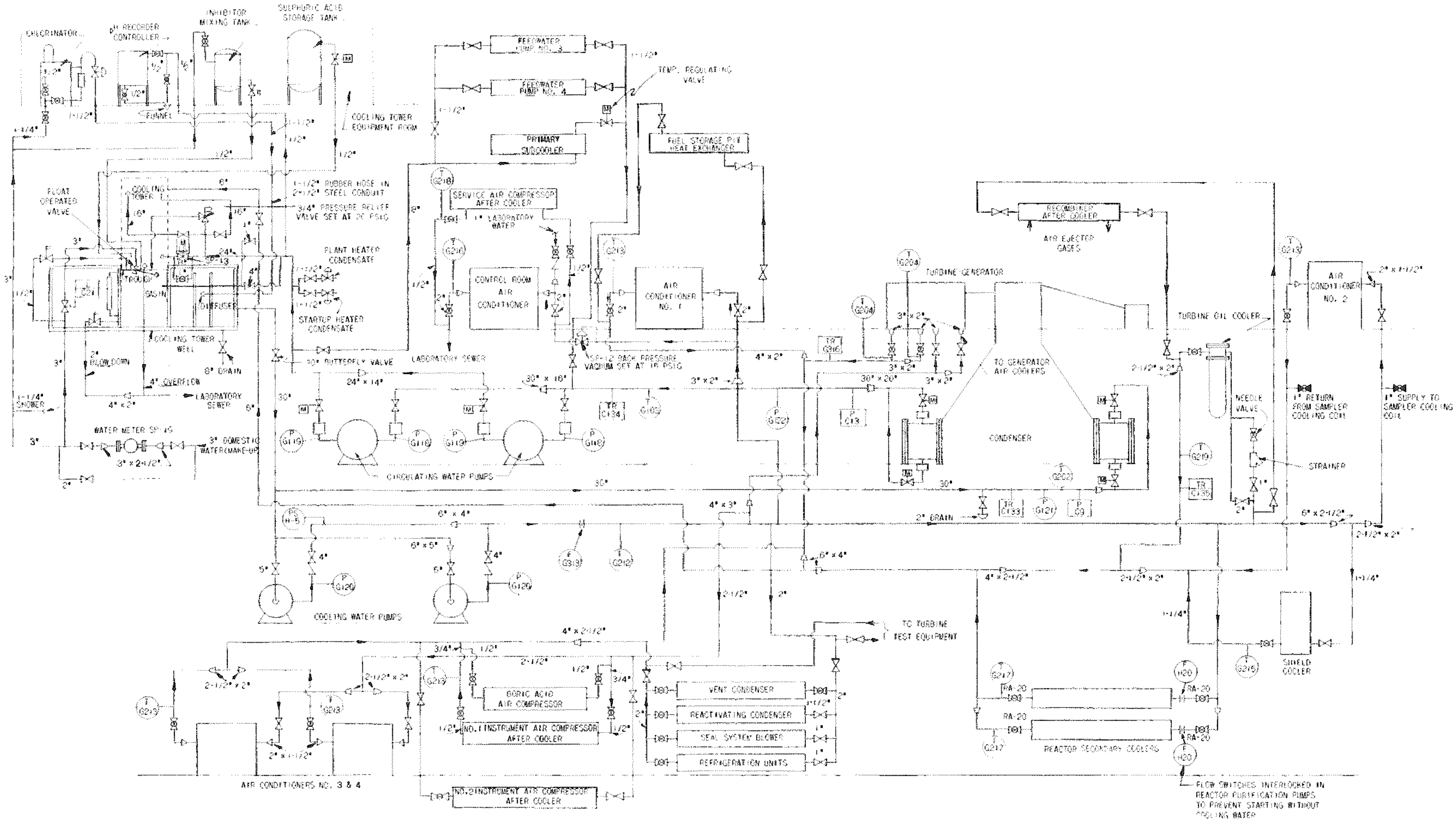

Figure 3-3

Diagram of circulating water system
RE-8-20140-D

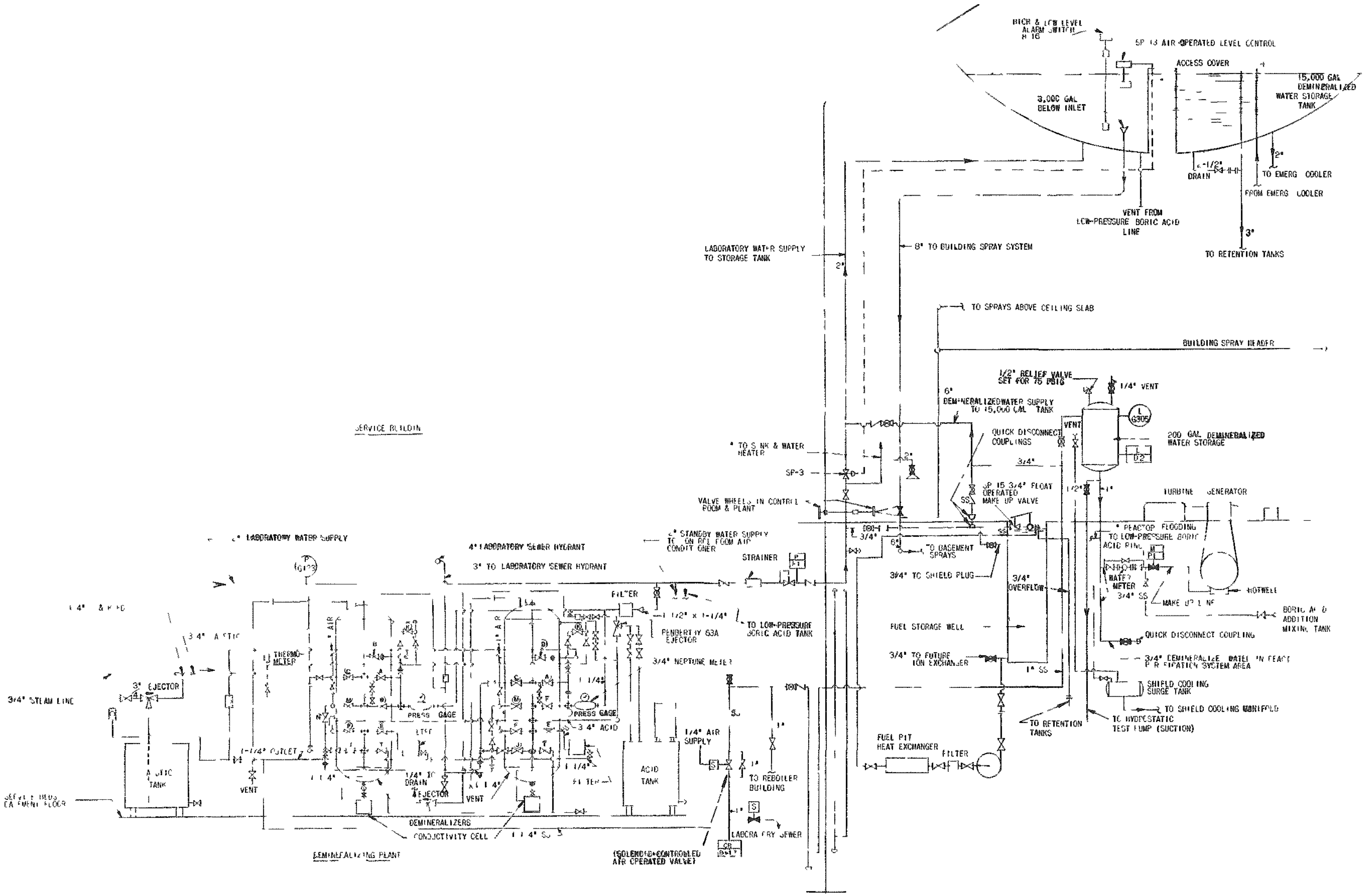

Figure 3-4

Diagram of make-up water piping
RE-8-20138-D

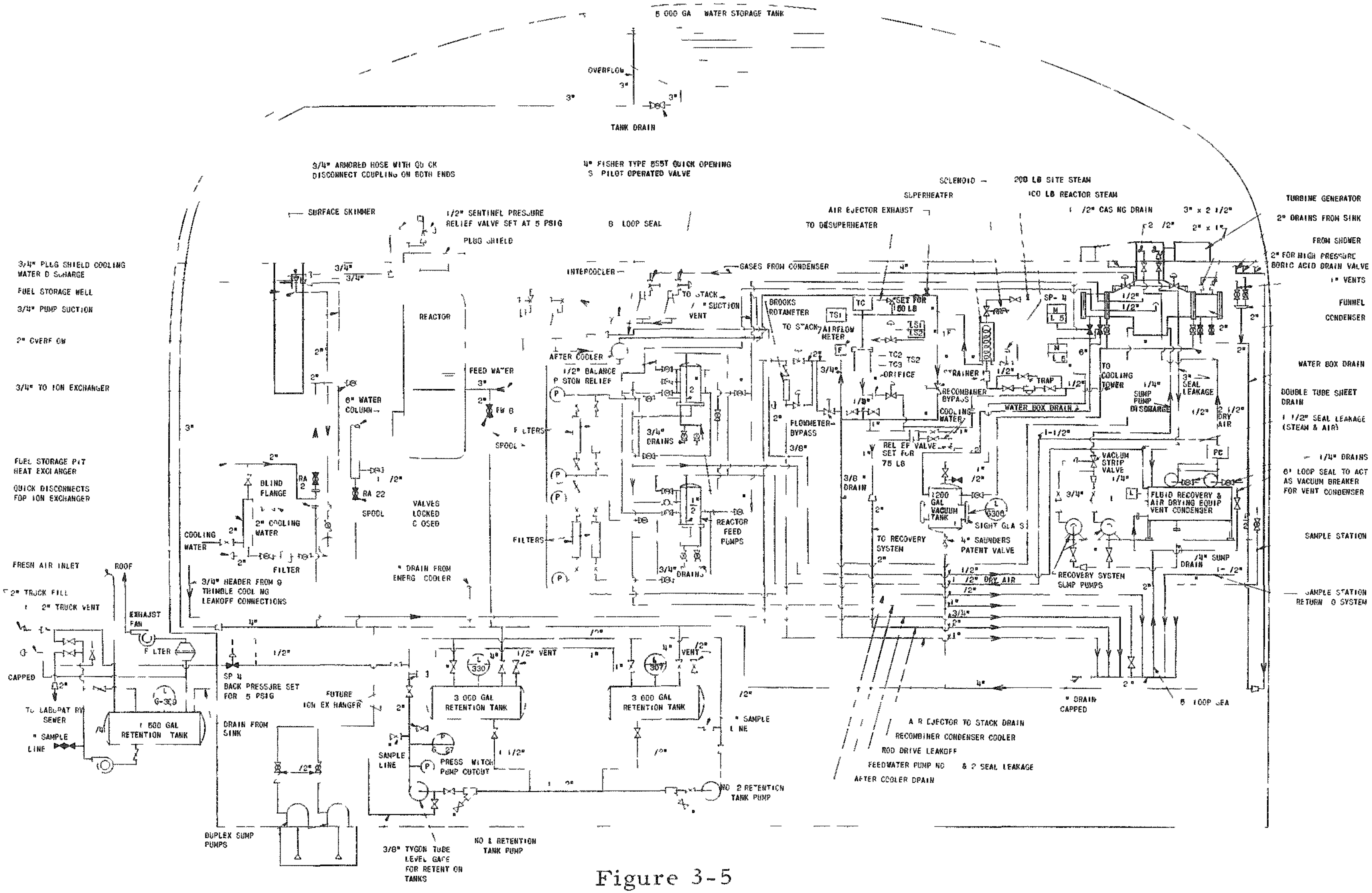

Figure 3-5

Diagram of vents and drains
RE-8-33110-D

3.2 REBOILER PLANT*

3.2.1 General Description

The 100 megawatts of thermal power generated in the reactor is dissipated by two systems: (1) the turbine-generator system utilizes 20 megawatts and (2) the new heat-dissipating system absorbs 80 megawatts. The heat-dissipating equipment supplies steam to the Laboratory heat-distribution system or to the air-cooled heat exchangers. An intermediate loop is incorporated to prevent any contaminated steam from escaping into the heating system. Primary steam condensed in the primary reboiler generates steam in the intermediate loop. Intermediate steam is condensed in the secondary reboiler and/or in the air-cooled condenser. Secondary steam is generated in the secondary reboiler and directed to the Laboratory heating system.

In the revised plant the turbine-generator system is unchanged and is used to generate up to 5 megawatts of electric power in the same manner as in the past. There are no significant mechanical or equipment changes in the turbine-generator system, except an operating change in the turbine bypass valve.

During operation of the reactor at 100 megawatts, the additional heat removal equipment is capable of transferring approximately 67 megawatts (227,000 lbs/hr at 200 psig) to the Laboratory system. At a reactor output of 100 megawatts, about 10 megawatts of the remaining 13 megawatts is dissipated to the air-cooled equipment, and 3 megawatts are lost to the primary deaerator and intermediate flash-tank systems. If desired, the amount of heat distributed between the secondary reboiler and air-cooled heat exchangers can be proportioned in accordance with need. During the winter sufficient steam is generated to supply all Laboratory heating requirements. Any excess heat not required by the Laboratory heating system during the winter is absorbed in the air-cooled equipment. In an emergency during the winter, the air-cooled heat exchangers can sustain the entire load of 77 megawatts, provided the ambient air temperature is below 65°F. Hence, in winter continuous operation of the EBWR facility is assured whether or not the secondary heat exchangers are operative. Although the combined heat exchanger capacity is available to operate the reactor at 100 megawatts during the summer, the air-cooled equipment is not designed to singularly accept the entire 77-megawatt load. At an ambient air temperature of 95°F, the air-cooled exchangers remove a maximum of 67 megawatts; at lower air temperatures capacity increases. Normally, the secondary exchangers are supplied with 10 megawatts during summer, which, when added to the 67 megawatts of the air-cooled exchangers, accounts for 77 megawatts. If the secondary exchangers are inoperative during the summer, reactor power must be correspondingly reduced commensurate with ambient air temperature.

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The new heat-dissipating system is divided into three parts: the primary system, intermediate system, and secondary system. The primary system is operated at 560 psig saturated, the intermediate system at 350 psig saturated, and the secondary system at 200 psig saturated. The operating pressure of the secondary system, 200 psig saturated, is also the pressure of the existing Laboratory steam system.

The primary-steam-plant system consists of the following major components:

1. Two primary reboilers (tube side)
2. Two primary drain tanks
3. Four primary drain coolers (tube side)
4. One deaerator
5. One subcooler (shell side)
6. Two reactor feed pumps (one for standby)
7. Four filters
8. Connecting pipe
9. Shut-off and control valves
10. Relief valves
11. Sampling lines and valves

The intermediate system consists of the following major components:

1. Two primary reboilers (shell side)
2. Four primary drain coolers (shell side)
3. One secondary reboiler (tube side)
4. One secondary drain tank
5. One secondary drain cooler (tube side)
6. Air-cooled condenser
7. Condenser drain tank
8. Air-cooled drain cooler
9. Flash tank
10. Air-cooled flash condenser and condensate return unit
11. Air-cooled subcooler
12. Two intermediate feedwater pumps

13. Intermediate chemical-treatment system
14. Blowdown tank
15. Connecting pipe
16. Shut-off and control valves
17. Relief valves

The secondary system consists of the following major components:

1. One secondary reboiler (shell side)
2. One secondary drain cooler (shell side)
3. Secondary-feedwater chemical treatment system
4. Two steam separators
5. Connecting pipe
6. Shut-off and control valves
7. Relief valves

It was necessary to locate most of the new items of equipment outside the containment shell because of internal space limitations. The primary feedwater pumps, deaerator, subcooler, two new filters, and associated appurtenances are located in the containment shell. The primary heat exchangers, secondary heat exchangers, flash tank, blowdown tank, drain tanks, and intermediate pumps are all housed in the reboiler building. The air-cooled intermediate heat exchangers are located out-of-doors next to the reboiler building. For safety purposes, the primary heat exchangers and primary piping that connect the exchangers with the reactor system are housed within a relatively gas-tight concrete enclosure inside the reboiler building. The piping from the shell is contained in a concrete tunnel which connects the two buildings. Three walls of the concrete enclosure are one foot in thickness; the fourth wall consists of removable concrete blocks to facilitate removal of equipment. The ceiling of the enclosure is also of reinforced concrete one foot in thickness. This enclosure, in addition to confining possible contamination, also affords some radiation shielding. Additional radiation shielding can be added if required. Footings have been prepared to accept the load of 12 inches of additional concrete in the walls and roof. Figure 3-6 is a schematic drawing of the equipment layout in the reboiler building.

All electric cables and pipes leading from the containment vessel are sealed to withstand an internal building test-pressure of 15 psig. Five penetrations through the shell accommodate the primary-system piping to the adjacent reboiler building. The five penetrations consist of one 8-inch steam line, one 1-inch vent line, two 4-inch condensate lines, and one

$\frac{3}{4}$-inch control-air line. Air-operated control valves are installed on the steam line and condensate lines just inside the wall of the shell. In the control-air line a solenoid valve inside the shell closes on plant shutdown. One back-pressure regulating valve is installed in the vent line.

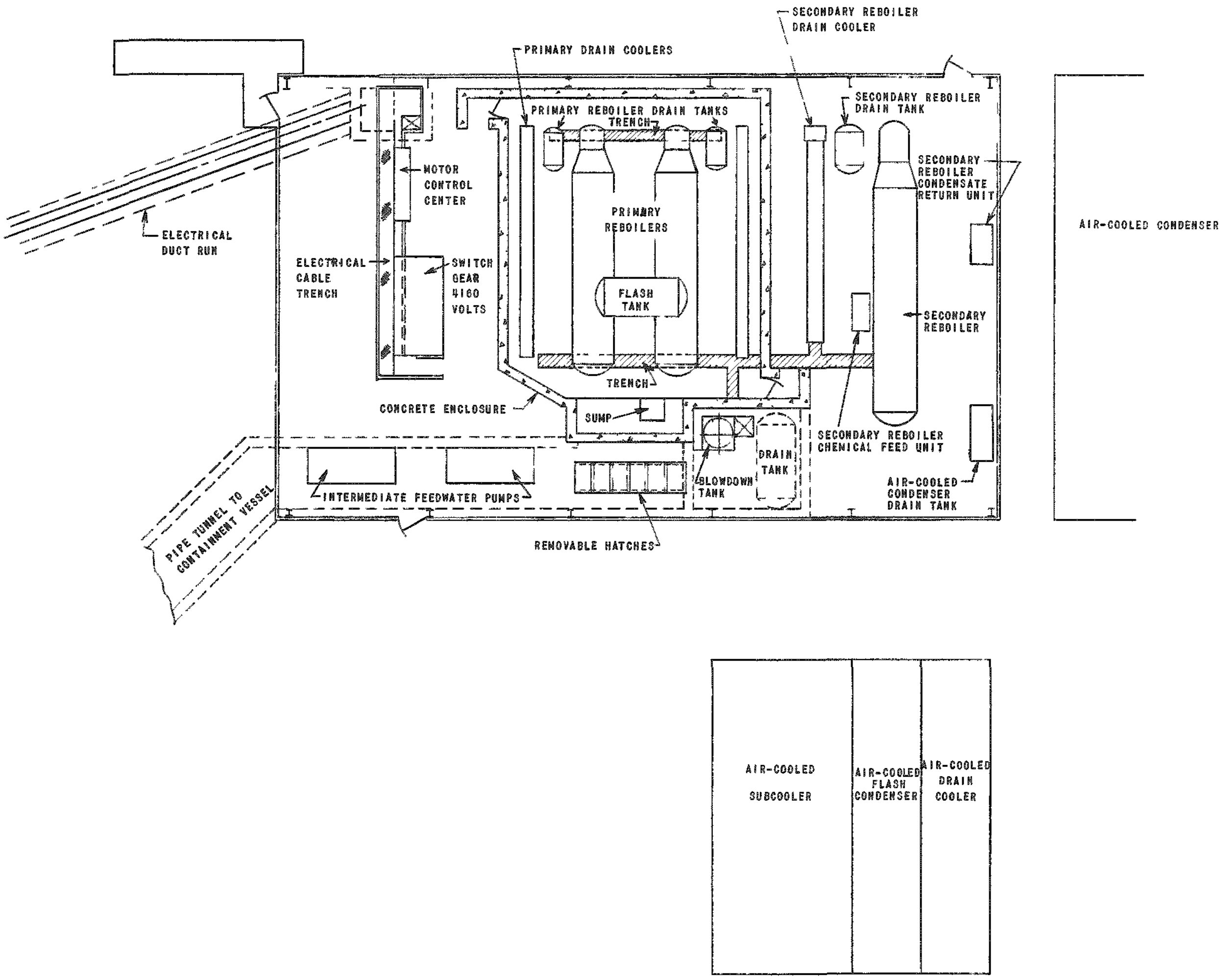

Figure 3-6

EBWR reboiler building layout
RE-8-23998-C

Previously, a trip of the radioactivity monitor of the containment-vessel vent stack closed the inlet and exhaust air ducts in addition to shutting down the reactor. This system has been modified to include automatic closure of the air-operated steam control valve, P-11D and its bypass in the trip circuit of the containment-vessel stack monitor. However, manual override of the automatic closure is incorporated in the modified circuit to allow the operator to dissipate heat to the reboilers despite high radioactivity in the containment vessel. The manual override is accomplished at the control panel.

The radioactivity monitoring system has been expanded to include the primary reboiler enclosure and connecting tunnel. The enclosure air intake is located at the end of the tunnel adjacent to the containment shell. Consequently, airborne radioactivity originating in the tunnel is swept into the enclosure. Effluent air discharged from the enclosure is monitored before being vented through the stack located in the reboiler building. If radioactivity is detected in the reboiler building enclosure, one air-operated valve in all lines (except the vent line) and the steam-control-valve bypass are automatically closed, and the reactor is shutdown. The direct-operating, back-pressure regulating valve in the vent line from the drain tanks and reboilers will normally be open during operation to allow vent gases to flow to the condenser. However, once the upstream pressure drops below 50 psig the valve will close and the systems will be isolated. All of the air-operated control valves in the steam and condensate lines interconnecting the two buildings are backed up by manually operated motorized valves which, in the event of power failure, can be powered by the emergency power supply. The operator has control of all valves at all times, and can close or open them as necessary. The sequential action that automatically occurs simply provides an immediate remedy and safe starting point for the plant operators.

3.2.2 Primary Steam System

The main steam and feedwater piping is shown in Figure 3-7. The reactor generates 100 megawatts of steam at 600 psig saturated, of which 80 megawatts is directed to the primary reboilers through a steam back-pressure control valve. This action results in a reduction of steam pressure to approximately 560 psig. Pressure within the reactor is controlled by the new 8-inch, air-operated steam control valve, P-11D, which maintains 600 psig on the reactor (see Figure 3-29). In this respect, the new steam-pressure control valve assumes the role of the turbine bypass valve previously used for this purpose. During actual operation of the turbine-generator and the new steam plant, the turbine bypass valve reference is set above control valve P-11D reference. Therefore, should the load drop out on the steam plant, the turbine bypass valve can dump excess steam to the condenser until either the reactor over-pressure interlock initiates reactor shutdown at 640 psig or some other affected device shuts down the reactor. Also, should the turbine load be dropped out, any excess steam that cannot be condensed by the reboilers without a pressure rise will be dumped to the condenser by the turbine bypass valve.

The amount of steam admitted to the primary reboilers is a function of the reactor output, and consequently, demand changes in the Laboratory steam system are not reflected back to the reactor. The method for accomplishing reactor independence from Laboratory demands is explained in Section 3.2.5, which covers the intermediate system.

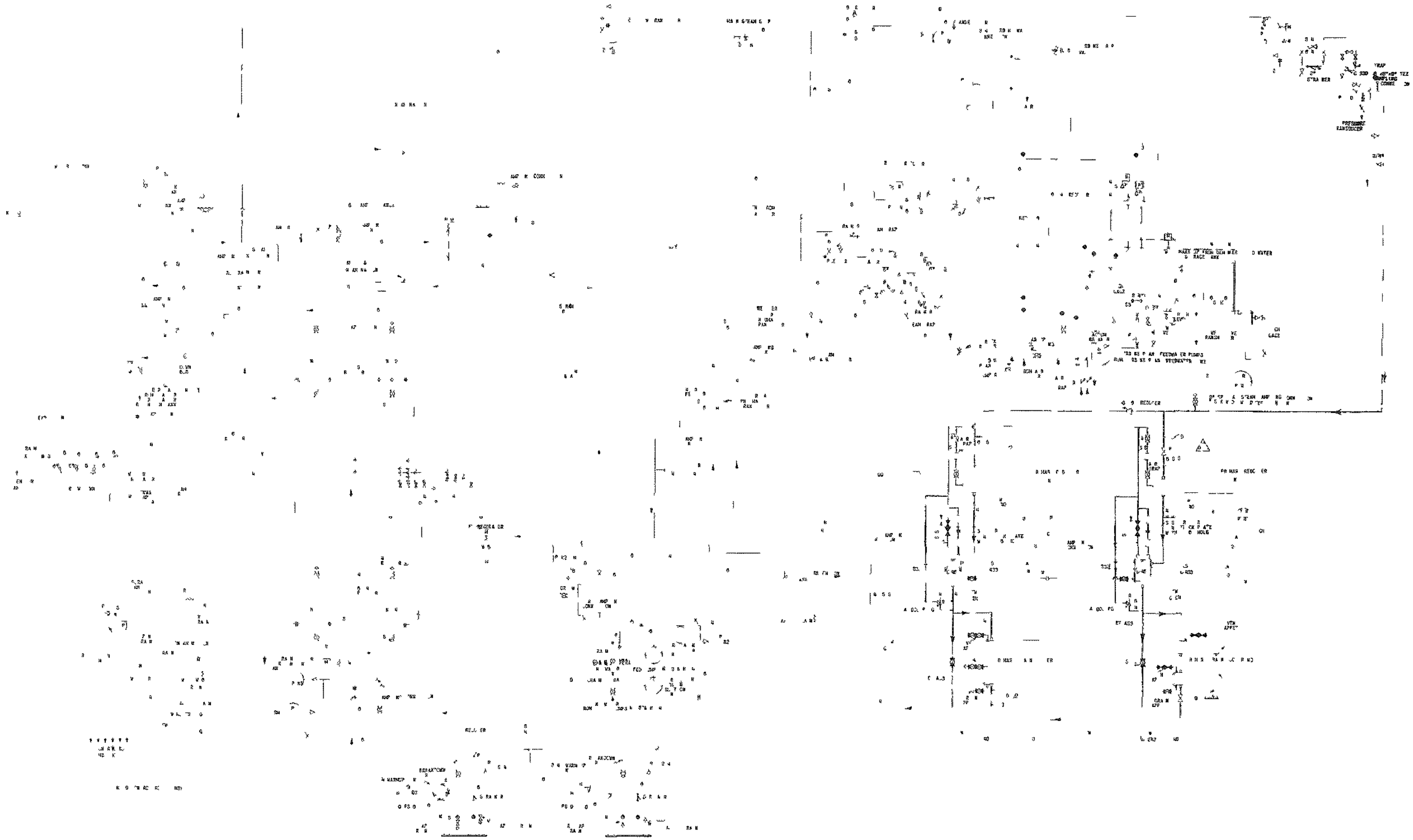

Figure 3-7

Diagram of main steam and feedwater piping in reboiler plant
RE-8-33224-F

Primary steam is condensed in the two primary reboilers operating in parallel. Each reboiler can handle one-half of the primary system requirements. Primary steam condensed on the tube side of the reboilers regenerates steam in the shell side for the intermediate system.

The two primary reboilers are each designed to condense 150,000 pounds per hour of steam at 560 psig and 482°F. In so doing, 140,500 pounds per hour of saturated steam at 350 psig are regenerated in each shell. At the flow rates and temperatures given above, 32.5 megawatts of heat are exchanged in each reboiler for a total of 65.1 megawatts. The quality of the intermediate steam leaving the reboiler shell is about 98 percent, this quality being accomplished by the incorporation of a dry pipe within the reboiler shell. Additional moisture-separating devices are not installed on the outlet of the primary reboilers since the 98 percent quality is acceptable.

Since the intermediate steam pressure is normally set at 350 psig and the shell water level is constant, the condensing pressure in the reboiler tubes is a function of the primary steam flow. The amount of heat transferred is directly related to the steam admitted to the reboilers. The temperature differential between the two fluids is inherently self-adjusting and fixed for any specific steam flow. Increase in the primary steam flow results in higher condensing pressure. Higher condensing pressure increases the heat transfer rate. Therefore, at partial loads the reboilers are self-adjusting to meet the conditions. Primary steam condensing pressures can vary, according to load, from slightly over 350 psig to approximately 560 psig, if necessary.

The tube side of the primary reboilers is designed for 800 psig and 520°F. Tubes are fabricated of type-304 stainless steel and all other surfaces in contact with primary fluid are either clad with or constructed of type-304 stainless steel. The shell side is designed for 400 psig and 448°F. The shell is constructed of carbon steel as are all other parts, except the tubes, in contact with the intermediate fluid. Each primary reboiler shell is provided with a chemical-feed inlet nozzle for the addition of: (1) trisodium phosphate to maintain the desired pH and to prevent scale formation and (2) sodium sulfite to scavenge dissolved oxygen. Piping is arranged for independent chemical treatment of either reboiler by utilizing the pressure drop over the associated feedwater valve for the introduction of the chemical solutions. Also, the blowdown connection is located approximately 6 inches below the normal water level in the shell, since solids concentration is generally greatest near the top.

The reboilers conform to all design requirements of Section VIII, Unfired Pressure Vessels, of the ASME Boiler and Pressure Vessel Code. Two safety valves are mounted on each of the reboiler shells to prevent overpressure. The safety valves are set to operate at 400 and 410 psig with discharge rates of 122,500 and 85,350 pounds per hour of steam respectively. Both valves exhaust directly to the atmosphere outside the reboiler house. The tube side exit header is provided with a vent to continuously remove a portion of the noncondensable gases generated in the reactor by dissociation of water. Vent gases are piped through a 1-inch line to the condenser hotwell.

The water level in each reboiler shell is separately maintained by its level controller, LC-333, and associated air-operated control valve G-333. Feedwater flow from the intermediate feedwater pumps is proportional to the steaming rate within the reboiler shell. Other controls for the reboilers consist of sight glasses and high and low level alarms.

The two primary drain tanks are constructed of type-304 stainless steel, and each has a total capacity of 150 gallons. Each vessel is designed for 800 psig and 520°F in conformance with Section VIII of the ASME Code.

The primary steam condensed in the tubes of the primary reboilers accumulates in the primary-reboiler drain tanks and then flows through the primary drain coolers to the deaerator. The level of the condensate accumulated in the drain tank is maintained by controlling the flow to the deaerator. A float cage senses the level in the drain tank and, through liquid level controller LC-332, actuates control valve G-332, which regulates flow to the deaerator. Liquid level is maintained independently in each vessel during operation by the level controller and associated air-operated control valve. Drain tank effluent is directed through a 4-inch line to two primary drain coolers connected in series. Each reboiler drain tank and two drain coolers comprise one-half of a parallel system. The effluent from each of the drain coolers is conducted in a 4-inch line to the deaerator through control valves G-332.

Each drain tank and reboiler exit header are provided with bleed orifices to vent a portion of the noncondensable gases and prevent any accumulation; this provision precludes any possibility of the drain tank and reboiler becoming airbound. Vent gases are directed through a 1-inch line to the condenser.

The combined capacity of the drain coolers is sufficient to subcool 264,400 pounds per hour of primary condensate from 482° to 231°F, and raise the temperature of 247,000 pounds per hour of intermediate feedwater from 160° to 435.7°F. The primary condensate flows through the tube side of the drain coolers and provides the heat necessary to raise the temperature of the intermediate feedwater on the shell side to the saturation point. At the flow rates and temperatures given above, 20.68 megawatts of heat are exchanged.

The tubes and all surfaces of the drain coolers exposed to primary condensate are stainless steel; the shell is fabricated of carbon steel. The tube side is designed for 800 psig and 520°F; the shell side is designed for 400 psig and 448°F. Construction complies with Section VIII of the ASME Code where applicable. In cases not covered by ASME Code, the design is in accordance with Tubular Exchanger Manufacturers Assn. (TEMA) standards.

Radiolytic dissociation of water in the reactor results in the formation of hydrogen and oxygen. Separation of these gases at the liquid interface is followed by carryover in the effluent steam. Under equilibrium conditions the carryover results in essentially a stoichiometric mixture of hydrogen and oxygen in the exit steam. The rate of the combined gases thus generated amounts to about 1 scfm per 20 megawatts of reactor power. Therefore, at 100-megawatt operation about 5 scfm of hydrogen and oxygen are liberated. The distribution of these gases is directly proportional to the division of steam to the turbine and reboiler. At the design flow of 301,000 pounds per hour to the reboiler and 60,600 pounds per hour to the turbine, the combined gases enter the primary reboilers at a rate of 4.17 scfm. From the molecular ratio of hydrogen and oxygen in water, the oxygen rate then becomes 1.39 scfm. The condensate on this basis contains about 17.2 cc (STP) oxygen per liter, or 24.6 ppm. Although the deaerator is designed to reduce the oxygen concentration to 0.1 cc (STP) per liter (or 0.143 ppm) at the aforementioned maximum flow, it is not the only stage where noncondensable gases are removed.

As mentioned above, discharge to the primary deaerator is regulated by the two air-operated control valves (G-332). Condensate leaves the primary drain coolers at a pressure of approximately 525 psig and at temperatures varying from 321° to 340°F depending upon the mode of operation. Flashing of condensate to the low pressure in the deaerator results in further cooling and the removal of noncondensable gases. Flashed steam and noncondensable gases are piped to the condenser through a 12-inch carbon steel line. Noncondensable gases are removed from the condenser by the air ejector and radioactive gas disposal system.

The design of the deaerator is based on a maximum flow of 301,000 pounds per hour flashing from a condensate temperature of 339°F to a deaerator pressure of 90 psia. At these conditions 6570 pounds per hour of condensate is flashed which, with the extracted noncondensable gases, is sent to the condenser. Normally, the temperature of the condensate is between 230° and 235°F, and the deaerator is operated at 10 psia. At this condition about 10,450 pounds per hour of vapor is formed.

Deaerator pressure is varied in accordance with the mode of operation throughout a range of 10 to 90 psia. Pressure within the deaerator is maintained by the 10-inch regulating valve, P-12D, which is electrically driven and actuated by the pressure controller, P-12. Depending upon operation, the amount of condensate flashed varies from 2 to 4 percent. (See Figure 3-32.)

The tray-type deaerator is a vertical vessel constructed entirely of type-304 stainless steel and designed in accordance with Section VIII of the ASTM Code. The design pressure and temperature are 100 psig and 338°F respectively. The vessel is also designed to withstand an external pressure of 15 psig at 100°F. The vessel has a storage capacity of 1000 gallons. Gage glasses are provided for direct visual observation of the liquid level. Nozzles are provided for draining and venting. The vessel is provided with a liquid-level indicating-recorder controller (P-21). Also, to detect flooding or low level, a Magnetrol liquid-level switch is mounted on the vessel for annunciation of these conditions and tripping of the pump-motor circuit breaker.

After leaving the deaerator, the condensate can be cooled further in the primary subcooler. The temperature of the condensate leaving the subcooler is maintained by control valve P-42C, which regulates the flow of cooling water to the subcooler in accordance with the signal received from a temperature controller, TIC-P-42 (see Figure 3-35). The cooling-water tower, located 150 feet south of the containment vessel, is used as the heat sink for the removal of heat from the subcooler. Cooling water at a temperature of 95°F is obtained from the discharge side of the condenser circulating pumps. Cooling water return from the subcooler is directed back to the suction side of the pumps. Approximately 25-psig water pressure is available to circulate cooling water through the subcooler.

The primary condensate is directed through or bypassed around the subcooler in accordance with the mode of operation. The subcooler is designed to remove a maximum of 5.1 megawatts with a condensate flow of 235,570 pounds per hour. At this flow rate the temperature of the condensate is decreased from 193.2° to 120°F. Approximately 1000 of the 13,000 gallons per minute of the cooling water circulated through the condenser is used at the design condition.

The condensate flows through the shell side of the subcooler, whereas cooling water passes through the tube side. The tubes and shell are designed for 100 psig and 338°F in conformance with Section VIII of the ASME Code. The subcooler shell can also withstand an external pressure of 15 psi. All surfaces in contact with the primary water, such as tubes, shell, and tube sheets, are constructed of type-304 stainless steel. The header is fabricated of carbon steel.

Two reactor feedwater pumps, of which one serves as a standby, are installed in the primary circuit immediately following the subcooler.

Each pump is designed to furnish 645 gpm at a differential head of 1780 feet across the pump and a net positive suction head (NPSH) of 30 feet. Both pumps are of the horizontal, centrifugal, split-case type, and casings are constructed of 11-13 percent chromium-steel alloy. Prior to installation, nozzles were safe-ended for welding to type-304 stainless steel pipe.

The casing design pressure is 900 psig at 320°F, and design of the pumps is in accordance with applicable ASME Code. Each pump consists of six stages arranged with a crossover after three stages to give a balance shaft essentially free of end thrust. The drive unit of each pump is a 3550-rpm 350-horsepower electric motor.

The pump is designed to give sufficient head to assure adequate flow of feedwater to the reactor even in the event of overpressure in the vessel. The differential shut-off head of the pump is 2220 feet of the water being pumped which, under the worst condition of 320°F feedwater temperature, is equivalent to 873 psi. At a feedwater temperature of 120°F the 2220-foot head is equivalent to 952 psi. This shut-off head permits pumping of feedwater well past the opening pressure of the last relief valve, which is set at 775 psig.

Both pumps are provided with a strainer, a warm-up orifice, and a minimum-flow orifice. Minimum flow for each pump is 20 gpm, and this amount is maintained by the minimum-flow orifice to prevent damage to the pump should flow be interrupted. The minimum-flow orifice discharges to the deaerator. Mechanical seals are employed on the pump to limit shaft leakage. In the event of failure of the mechanical seal, an inboard throttle bushing, or disaster bushing, is provided to restrict leakage to a nominal rate, sufficiently low to allow the pump to continue operation without initiating emergency shutdown. An outboard bushing is also provided to allow for the collection of leakage within a chamber from which it is piped to the floor drain. The floor drains are connected to retention tanks. The escape of an excessive amount of contaminated water is thus prevented by this arrangement.

Reactor water level is maintained by the new 4-inch electrically operated feedwater control valve, P-21E. The new feedwater regulator, P-21E, utilizes a conventional three-element flow control that receives signals from reactor level, reboiler steam flow, and feedwater-flow controllers (see Figure 3-34). Feedwater flow from the turbine-generator plant is controlled by hotwell level, feedwater flow, and turbine-steam flow; regulation is obtained by the feedwater regulator (FW-reg). In the initial 20-megawatt plant the feedwater was controlled by reactor level, feedwater flow, and steam flow; this system will be retained for startup of the turbine-generator plant. During startup of the new steam plant, control valve P-21E will be controlled by deaerator level, reboiler-steam flow, and the feedwater-flow controller. Once the new steam plant is on-stream, provisions are available for switching reactor-level control from the feedwater regulator to control valve P-21E and the feedwater regulator to hotwell level.

Discharge from the new feedwater pumps is mixed with the discharge from the original pumps prior to passage through the filters. Two filters were added to the primary circuit to supplement the two filters in the original plant. The new filters are operated in parallel with the original filters and perform the function of removing particles transported by the primary water. The new filters have the same flow capacity as the original filters; each filter is designed for 180 gpm at 120°F. All filters will operate under the same conditions. Provisions are incorporated in the piping and valving to operate any number desired. Instrumentation is also provided to indicate flow through each filter in order to establish whether fouling is occurring.

The new filters are of the vertical type with a removable cotton-filter cartridge. The filter material is capable of removing particles in sizes of 2 microns or larger. The clean-flow pressure drop is approximately 5 psi at 180-gpm design flow rate and 120°F. Each filter is designed for 900 psig and 500°F in accordance with Section VIII of the ASME Code. The vessels are constructed entirely of type-304 stainless steel, except the holddown rings which are of carbon steel. All surfaces in contact with primary water are of stainless steel.

3.2.3 Pressure Relief System

The first and most important method of preventing excessive overpressure in the primary system is the automatic insertion of control rods when reactor pressure exceeds 640 psig. In the event of malfunction of the normal heat-removing equipment and failure of the reactor shutdown mechanism and/or emergency shutdown, a system of pressure relief is provided to protect the primary system. This system consists of one steam-powered pressure-regulator valve and four pop-safety valves located on the reactor steam line.

The 4-inch by 8-inch Foster steam-powered pressure regulator is automatic, featuring an adjustable set point and proportional band which must be adjusted locally. This valve is set to open at 650 psig. The pop-safety valves are preset and only serve to protect the reactor system from exceeding design pressure. The pressure settings of these four safety valves are 700, 725, 750, and 775 psig. All valves are discharged to the condenser. Two rupture disks (one new) in the condenser shell will burst if steam pressure exceeds 20 psig and permit steam to be released inside the containment vessel. Each rupture disk is sized to discharge 180,000 pounds per hour of steam.

The safety valves that are set at 700 and 725 psig were used on the original system and are adequately described in report ANL-5607.(1) Each valve has a rated capacity of 65,000 pounds per hour of steam. The remaining two valves each have a rated capacity of 160,000 pounds per hour steam at 90 percent of full flow at 800 psig in accordance with Section I of the ASME Code. Both of the larger-capacity valves have a 4-inch inlet and a 6-inch outlet, and both are the Crosby-Ashton type, 4 inches by 6 inches, style JO-36 Special. As in the case of the two original safety valves, each of the added valves for 100-megawatt operation are provided with a microswitch device to detect valve opening and initiate reactor shutdown. The control rods are interlocked with the microswitch device of each valve. Opening of any safety valve will initiate shutdown of the reactor. The control rods are backed up by the boric acid system. High-pressure injection of the boric acid is completed in about 20 seconds after manual initiation thereby guaranteeing permanent shutdown before the temporary relief afforded by the expulsion of steam has ended. For example, the full flow for all safety valves, discounting the regulator, is approximately 450,000 pounds per hour or about 2500 pounds in 20 seconds. This amount of water flashed will not result in uncovering the core within the reactor vessel.

The intermediate piping that connects each reboiler to the corresponding drain coolers contains no valves which can isolate the drain coolers from the reboiler. Hence, in accordance with Section VIII (UG-132) of the ASME Code, the two vessels can be considered as one shell, and safety valves on the reboiler are sufficient. These safety valves can handle any incident in the drain coolers, provided the connecting lines are free of isolating devices. In the event of internal tube failure in the drain cooler, it is impossible for the vessel to become overpressured by the primary system since the steam space in the reboiler offers protection as a buffer. In an incident of this type pressure cannot be relieved through the inlet-feedwater line because a check valve on the intermediate-pump discharge line will prevent backflow.

3.2.4 Shutdown Cooling

The decay heat which must be removed from the reactor decreases from a maximum of 6 percent of operating power immediately after shutdown to about 1 percent after seven hours (see Figure 3-42). During normal shutdown the decay heat is removed by generating steam in the reactor vessel. The steam generated is rejected to the condenser through the steam bypass valve until the reactor temperature reaches about 260°F. Thereafter, the secondary coolers of the purification system are used for further cooling of the reactor water. The circulating pumps are kept in operation to supply cooling water to the main condenser to condense any steam that enters the condenser from the steam traps in the steam lines.

In the event of a shutdown caused by loss of electric power, valves G-339 and P-11D automatically close, and the emergency-shutdown cooler, which is located within the original steam drier, is used to remove decay heat. Operation of the emergency cooler is initiated manually from the control room or automatically upon loss of instrument air or 125-volt d-c power. (The emergency cooler is designed to remove 1000 kilowatts at a steam pressure of 600 psig.) However, the cooling coils within the original steam drier can dissipate the integrated decay heat resulting from 100 megawatt operation of the core without allowing the pressure, unrelieved, to exceed the primary system 800-psig design. As a safety measure, the reactor is equipped with pop-safety valves and a 4-inch by 8-inch steam-powered regulating valve (Foster relief valve on the original installation). The regulating valve limits the reactor to a preset pressure of 650 psig. During emergency shutdown caused by power failure, reactor pressure will rise until 610 psig is reached; thereafter, steam will be discharged through the turbine bypass valve to the condenser until such time that the emergency cooler capacity equals the steam formation at 610 psig. If the turbine bypass valve is unable to handle the steam load, the Foster relief valve will limit the reactor pressure to 650 psig. If the turbine bypass valve is inoperative, calculations indicate that during the time interval the regulating valve is open approximately 1200 pounds of reactor water will be flashed to the condenser. The amount of water flashed results in a 7.5-inch decrease in reactor water level but does not result in uncovering the core. Even though the condenser is not supplied by the circulating water pumps at this time, sufficient cooling is available by natural circulation in the condenser to absorb the 1200 pounds of steam discharged over the estimated 1-hour interval involved.

Emergency-shutdown cooling relies upon natural circulation of water from the 15,000-gallon storage tank located in the overhead dome of the containment vessel. Reactor steam condenses outside of the cooling coils, and the condensate flows back to the reactor by gravity. The steam-water mixture formed within the tubes is directed back to the storage tank. The difference in density between the cooling water supply and steam-water mixture establishes natural circulation between the emergency cooler and storage tank. Main steam valves PS-1 and P-11D in the line to the reboilers are closed during operation of the cooler to minimize loss of reactor water. Laboratory water for makeup to the storage tank is supplied and regulated by a float valve. Approximately 80 to 100 gallons per minute can be provided by the Laboratory line, which more than compensates for steam escaping from the storage tank.

The high-pressure boric acid injection system is a standby system designed to reduce the reactor core reactivity below criticality should the control rods fail to operate properly under shutdown conditions. The original automatic injection feature has been eliminated in favor of manual operation for the 100-megawatt system since: (1) rapid boric acid injection is not necessary (the high-pressure system is a backup safety measure to be used only after consideration of conditions in the reactor), (2) at equilibrium-controlled operation, insertion of one rod is sufficient to override the void-controlled reactivity and reduce power, and (3) manual operation prevents inadvertent injection of boric acid because of malfunction or misuse of the control interlock system.

The volume of concentrated boric acid solution is able to hold down more than 13 percent reactivity. Manual injection of the solution is accomplished by pushbutton control at the reactor console, handwheel operation in the control room, or lever operation in the power plant building.

With minor exceptions, all additional equipment and piping for the primary-heat-dissipating equipment is stainless steel. The stainless steel surface in contact with the primary fluid imposes only a minor additional burden on the water purification system. Therefore, more capacity was not added to the reactor water purification system since it is adequate for the 100-megawatt operation.

3.2.5 Intermediate and Secondary Steam Systems

The intermediate steam generated in the primary reboilers is directed to either the secondary reboiler or the air-cooled condenser, or both, depending upon mode of operation. Steam admission to the secondary reboiler is regulated by the 12-inch electrically driven valve, P-13D, and is a function of condensing rate within the reboiler (see Figure 3-31). However, condensing rate is established by choice of condensing pressure, which fixes the temperature differential across the heat transfer surface. Once the condensing pressure is selected, shell pressure being maintained at 200 psig, the amount of intermediate steam admitted to the secondary reboiler tubes is fixed by heat transfer area and coefficient of heat transfer. To summarize, the secondary reboiler essentially generates a constant supply of steam to the Laboratory. From the standpoint of the Laboratory heating system, the EBWR plant is the base load, and the conventional power plant compensates for changes in demand. Figure 3-8 is a diagram of the intermediate and secondary steam systems.

Since the secondary-reboiler steam generation is fixed by the operator, the intermediate system must be provided with at least one degree of freedom to allow for normal pressure control in the event of primary system or Laboratory system fluctuations. Consequently, intermediate-steam pressure is regulated by the 10-inch electrically driven control valve, P-14D, which admits steam to the air-cooled condenser. In this respect the arrangement is analogous to the primary-system pressure control (see Figure 3-30).

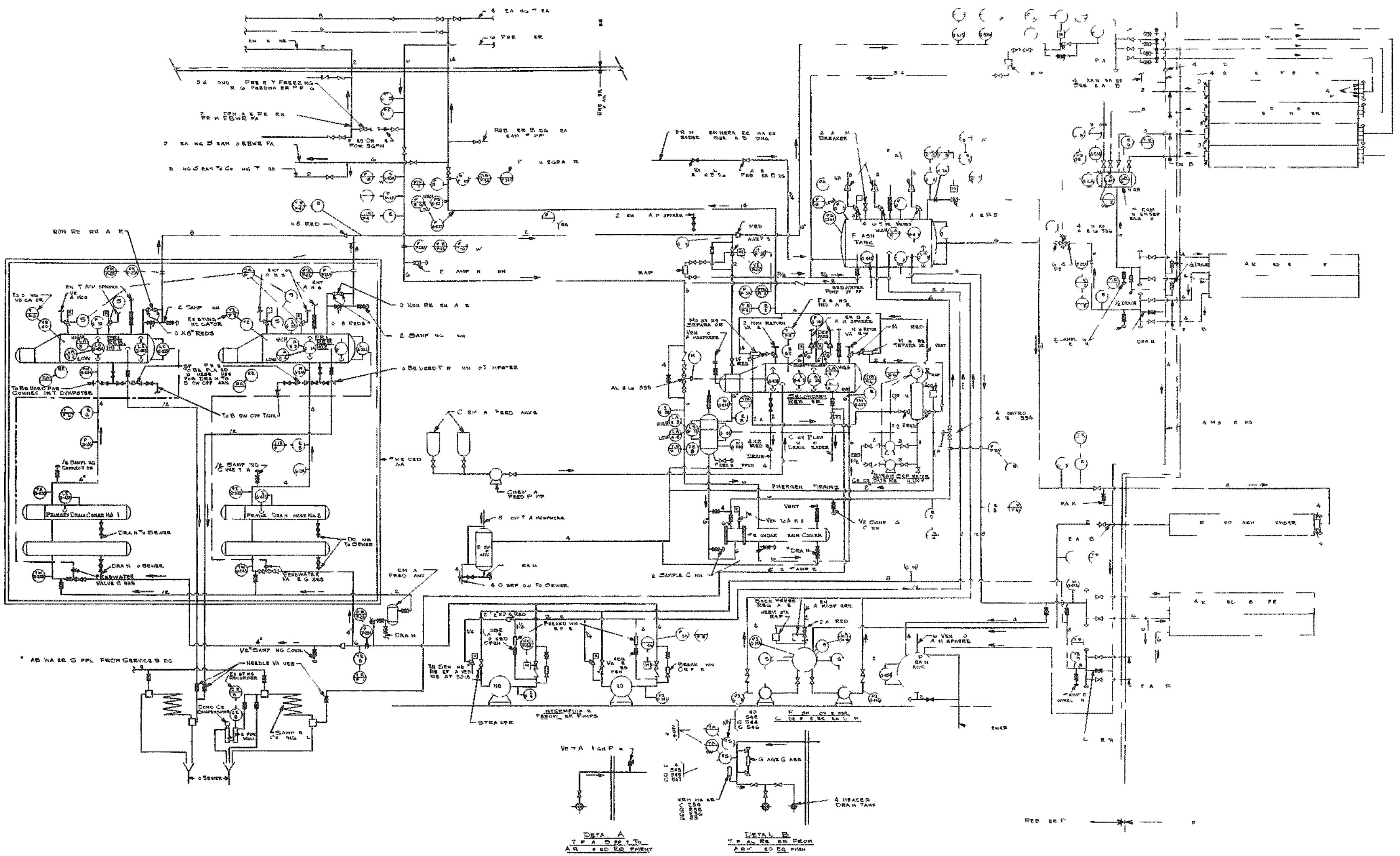

Figure 3-8

Diagram of intermediate and secondary steam systems
RE-8-32768-F

Intermediate condensate formed in the secondary reboiler is discharged into the secondary drain tanks from which it is directed to the tubes of the secondary drain cooler where further cooling is accomplished. Liquid interface is maintained in the secondary drain tanks by level controller LC-334, and flow to the drain cooler is regulated by air-operated control valve G-334 located on the discharge line of the drain cooler. The intermediate condensate is flashed over valve G-334 to the lower pressure maintained in the flash tank. Steam formed by flashing is routed to the air-cooled flash condenser where it is condensed, collected in the condensate tank, and returned to the flash tank by the condensate pump. Pressure in the flash tank is regulated by the 8-inch electrically driven valve, P-15D, which admits steam to the flash condenser (see Figure 3-33).

Steam admitted to the air-cooled condenser by control valve P-14D is condensed and flows to the drain tank where liquid interface is maintained. Effluent from the drain tank is further cooled in the air-cooled drain cooler from which it is directed to the flash tank. Discharge from the air-cooled drain cooler is regulated by control valve G-336 which is controlled by level controller LC-336 in the drain tank.

The intermediate condensate in the flash tank is pumped to the primary-drain-cooler shell by the intermediate pump and then to the reboiler shell where the cycle is completed. Should additional cooling be required after the flash tank, all or a portion of the condensate can be directed through the air-cooled subcooler prior to the intermediate pump.

Intermediate-feedwater flow to the primary drain coolers is regulated by the two 3-inch air-operated valves, G-333, and controlled by the level controller, LC-333, in the primary-reboiler shells. Since liquid levels are controlled only in the primary reboiler shell, secondary reboiler drain tank, and air-cooled condenser drain tank, changes in the intermediate system inventory are evidenced by changes in flash tank level which is detected by high-low alarms. Makeup water for the intermediate system is obtained from the demineralizers in the service building and is manually introduced as required. Primary-heat-exchanger tube rupture or excessive leakage of primary fluid into the intermediate system is manifested as a change in inventory, and a rise in flash tank level is caused. Should this condition arise, the reactor must be shut down and the cause of leakage determined. The flash tank is drained to the 1000-gallon drain tank through a 4-inch drain line controlled by the 4-inch drain valve, which must be operated manually.

As mentioned previously, Laboratory changes in demand are prevented from being reflected back to the reactor. Although changes in Laboratory demands will affect the operation of the secondary reboiler to some extent, moderate variations are damped completely by the operation of the flash tank and air-cooled flash condenser. At a fixed pressure, the flash tank assures constant feedwater temperature to the air-cooled subcooler, and the flash condenser automatically rectifies slight variations in reboiler heat transfer or condensate temperatures. Therefore, the flash tank system acts as a buffer which can compensate for slight changes in the preceding equipment. Should this system fail to fully alleviate any slight abnormal condition resulting in intermediate-condensate temperature changes to the primary drain cooler, the deaerator system will provide additional buffering to the primary system.

Effects inadvertently imposed on the secondary reboiler by changes in Laboratory demand, such as pressure fluctuations resulting in reboiler temperature differential (ΔT) changes, are only partially compensated for by control valves P-13D and P-14D. For example, should the Laboratory steam pressure drop slightly, resulting in an increase in the secondary reboiler temperature differential, control valve P-13D will admit more steam to the reboiler to maintain condensing pressure. Control valve P-14D will close correspondingly to maintain intermediate system pressure. In this way the secondary reboiler partially compensates for changes in load demand. However, the normal Laboratory pressure cannot be restored completely by this method; complete restoration would bring the reboiler steam generation back to the lower initial rate since the original temperature differential would prevail once again. Briefly, this means that as the Laboratory pressure begins to recover, the steam generation decreases, a relation not conducive to complete restoration of pressure by the EBWR system. Hence, Laboratory system steam pressure is maintained at the conventional power house. As an additional buffer, the air-cooled equipment will be operated at all times so that steam will be available to partially offset Laboratory system changes. Also, if the opposite situation occurs in which the Laboratory-system pressure rises, continuous operation of the air-cooled heat exchangers will keep them in a standby condition to accept steam.

The secondary reboiler is designed to condense a maximum of 241,000 pounds per hour of steam at 310 psig and 424.7°F at a mean temperature differential (corrected) of 36.7°F. In so doing, 227,000 pounds per hour of saturated steam is generated at a shell pressure of 200 psig. Heat transfer rating is in accordance with TEMA standards. The steam as provided to the Laboratory has a quality of 99.75 percent, this being accomplished by passing the steam through two horizontal line-type separators. Two 14-inch steam nozzles, which are manifolded before entry into the Laboratory system, are installed on the reboiler shell. Prior to manifolding, each steam line consists of a 12-inch, flanged, nonreturn valve which is followed by the line separator.

The tube side of the secondary reboiler is designed for 400 psig and 448°F, whereas the shell side is designed for 250 psig and 406°F. The reboiler design conforms to the requirements of Section VIII of the ASME Code where applicable. In cases where ASME Code does not apply, the design is in accordance with TEMA standards.

Secondary reboiler tubes are fabricated of 80 Cu - 20 Ni (SB-111 type A), and the remainder of the reboiler is constructed entirely of steel. The reboiler is of the conventional U-tube type with the tubes rolled in. The secondary-reboiler shell is provided with a chemical-feed inlet nozzle for the addition of trisodium phosphate to obtain the desired pH to prevent scale deposition, and sodium sulfite to scavenge dissolved oxygen. A blowdown connection is located at the bottom of the reboiler shell to remove the sludge that is expected to form.

Secondary-reboiler-level control is maintained by level controller LC-335 and regulated by the 4-inch air-operated feedwater control valve G-335. The reboiler is also equipped with a high-low level alarm switch and a Jerguson sight-glass level indicator. Two 4-inch by 8-inch safety valves installed on the reboiler shell discharge outside the reboiler building in the event of overpressure.

The secondary drain cooler is designed to subcool 241,000 pounds per hour of intermediate condensate from 424.7° to 289.8°F, and raise the temperature of 227,000 pounds per hour of Laboratory feedwater from 230° to 376.5°F. Normally, the Laboratory feedwater temperature leaving the secondary drain cooler is 388°F, which corresponds to the saturation temperature of the secondary reboiler shell.

The tube side of the drain cooler is designed for 400 psig and 448°F, and the shell side is designed for 250 psig and 406°F. The drain cooler design conforms to Section VIII of the ASME Code where applicable. In cases where ASME Code does not apply, the design is in accordance with TEMA standards. The drain cooler tubes are fabricated of 80 Cu - 20 Ni (SB-111 type A), and the remainder of the cooler is constructed of steel.

Because of the critical water supply shortage at the Laboratory, dissipation of heat during summer operation is provided by air-cooled heat exchangers. Although this form of heat sink is adequate, a better method for winter operation is desirable. The installation of this type of unit for heat dissipation is unique for this section of the country because of freezing difficulties that can be encountered. However, protective features, which reduce the probability of freezing, have been incorporated in the installation.

Four air-cooled heat exchanger units are employed to dissipate heat to the atmosphere. These heat exchangers are located in two groups. The steam condenser constitutes one group; the drain cooler, flash condenser, and subcooler comprise the second group. The air-cooled exchangers are of standard design for this type of equipment, incorporating finned tubes and motor-driven fans. The units are fabricated of steel, except for the tubes, which are of Admiralty metal (SB-111 type C). Aluminum fins are mechanically bonded to the tubes. Each unit consists of vertical shaft fans which force air upward across the horizontal tube bundles. Each group of units is enclosed on all four sides by manually adjustable shutters. In the group containing three units, the units are isolated from each other by solid panel partitions. A 4-foot vertical panel is also attached along the periphery of the top deck of each group. Doors are provided in the shutters for admission to the area beneath the fans.

The shutters and top vertical panels offer protection against freezing during winter operation. The adjustable shutters allow for regulation of airflow, if need be, or can close entirely. The top vertical panels will provide some additional protection by partially alleviating the effects of natural winds. One fan in each unit is designed with variable pitch which provides stepless airflow regulation or reversal. The change in blade angle of the variable pitch fan is executed by means of a pneumatic operator, controllable from the control panel. The remaining fans consist of manually adjustable blades. If required during winter operation, the shutters on any unit can be closed completely and the variable pitch fans set for reverse flow while the remaining fans are operated in the normal fashion. In this way, air regulation is obtained and some of the warm effluent air is recirculated. For added protection drain valves are provided in the reboiler building for draining each unit during periods of shutdown.

The air-cooled condenser unit consists of eight equal-size, horizontal, single-pass heat exchanger sections arranged for parallel flow. The entire unit is capable of condensing 245,400 pounds per hour of steam at 310 psig with an ambient air temperature of 95°F, this being equivalent to 57.6 megawatts of heat. The airflow at this rate of heat transfer is 3.4×10^6 pounds per hour with an exit temperature of 336°F. Four 12-foot-diameter fans with six blades are mounted below the unit to furnish the airflow. The condenser is designed for 400 psig and 448°F and adheres to Section VIII of the ASME Code. One exception to the code was made in regard to hydrostatic test pressure. The ASME Code requires a test pressure of $1\frac{1}{2}$ times design pressure, or 600 psig, for this type of construction. However, for reasons of safety, the unit was subjected to a test pressure of approximately 900 psig.

The air-cooled drain cooler unit is composed of two horizontal, four-pass, heat exchanger sections. The entire unit is capable of cooling 245,400 pounds per hour of intermediate condensate from 424.7° to 291.7°F with an ambient air temperature of 95°F, this being equivalent to 10.1 megawatts of heat. The airflow necessary to accomplish the heat transfer is 0.72×10^6 pounds per hour at an exit temperature of 295°F. Three 7-foot-diameter fans with six blades maintain the airflow. The drain cooler is designed for 400 psig and 448°F. The design conforms to

Section VIII of the ASME Code except for test pressure, which like the condenser was 900 psig.

The flash condenser consists of one horizontal, single-pass heat exchanger section. A condensing rate of 4930 pounds per hour at 19.8 psig is obtainable in the unit at an ambient air temperature of 95°F, this being equivalent to 1.36 megawatts of heat. Approximately 0.6×10^6 pounds per hour of air are required at this condensing rate for an effluent air temperature of 127°F. Three 7-foot-diameter fans with four blades are available. The flash condenser is designed for 100 psig and 338°F in compliance with Section VIII of the ASME Code.

The air-cooled subcooler unit consists of two equal-size, horizontal, four-pass heat exchanger sections. The entire unit is capable of cooling 246,600 pounds per hour of condensate from 258.5° to 160°F at an ambient air temperature of 95°F. For this heat dissipation of approximately 7.12 megawatts, an airflow of 1.2×10^6 pounds per hour is required for an exit temperature of 180°F. Two 12-foot-diameter fans with four blades are installed below the heat exchanger deck. The subcooler is designed for 100 psig and 338°F in accordance with Section VIII of the ASME Code.

Two intermediate feedwater pumps, one of which is a standby, feed the flash tank condensate to the shell-side primary drain cooler. Each pump is capable of delivering 600 gpm of water with a differential head of 1040 feet across the inlet and outlet connections of the pump. Both pumps, which are of conventional design for this type of service, are of the horizontal, centrifugal, split-case type. Each pump, consisting of two stages, is driven by a 250-horsepower squirrel-cage induction motor. Casings are of carbon steels and internals are generally of 11-12 percent chromium steel.

The pumps are designed to develop sufficient head to assure adequate flow to the primary reboiler up to design shell pressures of 400 psig. The differential-shutoff head of each pump is 1160 feet, which at the highest anticipated operating temperature of the intermediate feedwater is equivalent to 468 psi. With a flash tank pressure of 33 psig and a static head of 15 feet added to the differential head, the total discharge head at shutoff is 507 psig. At lower feedwater temperatures the total discharge head at shutoff is greater. During certain operating conditions in which the subcooler is not used, the maximum available NPSH, furnished to the pump, neglecting minor friction losses, is slightly less than 15 feet. However, should pump cavitation be encountered, the available NPSH can be increased by subcooling the condensate. In the range of intermediate-feedwater temperatures contemplated, 160° to 278°F, subcooling the condensate 5°F results in an increase in available NPSH of approximately 7 to 9.5 feet respectively.

Both pumps are provided with strainers, warmup orifice, and minimum-flow orifice. Minimum flow required to prevent damage to the pump is 45 gpm, and the minimum-flow orifice maintains this amount at all times. The warmup orifice is arranged to permit 5 gpm of operating pump discharge to flow in reverse direction through the standby pump, thereby keeping it warm and ready for immediate service. Conventional packing is provided for shaft sealing, and a lantern gland for leak-off is located in the packing chamber.

3.3 CONTROL ROOM*

In the conversion of EBWR to the 100-megawatt system, the philosophy that all of the important system operating parameters be controlled from the control room was incorporated in the system control design. To accomplish this, a new section was added to the panel board and some of the original panel board was modified.

Figure 3-9 is a plan view of the control room after modification. Panel 12, the new section, contains the controls and instrumentation for the reboiler building and air-cooled condensers and heat exchangers.

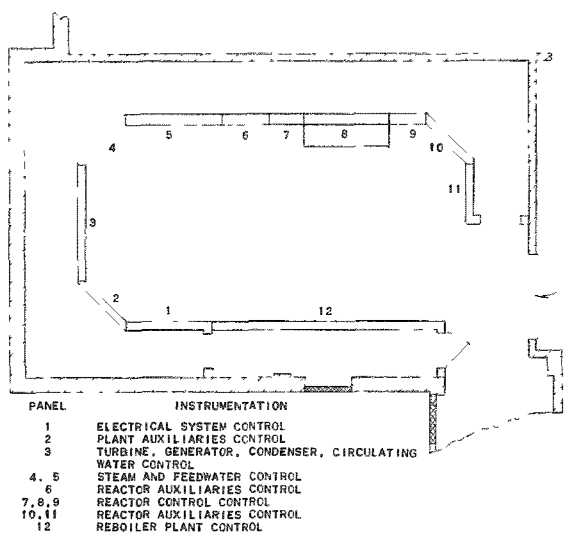

Figure 3-9

Plan view of EBWR control room
RE-6-19004-A

The addition of panel 12 prevented unobstructed, free passage of personnel through the original emergency-exit door. To provide direct exit, a second doorway was cut in the wall between the containment vessel and the control room.

Past operating experience has indicated that some changes should be made in the original panel board. The changes that were incorporated in the final design are reported in this publication.

Figure 3-10 is a layout of the electrical system, panel 1. Details 45, 46, 47, and 48 are additions to panel 1. Details 45 and 46 duplicate detail 5 of Figure 3-13 and detail 13 of Figure 3-12 respectively, the turbine-inlet-valve position indicator and turbine speed indicator. Operating experience has shown that these two instruments should be located on the same panel as detail 29, the generator-speed-governor control station.

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*A. Hirsch

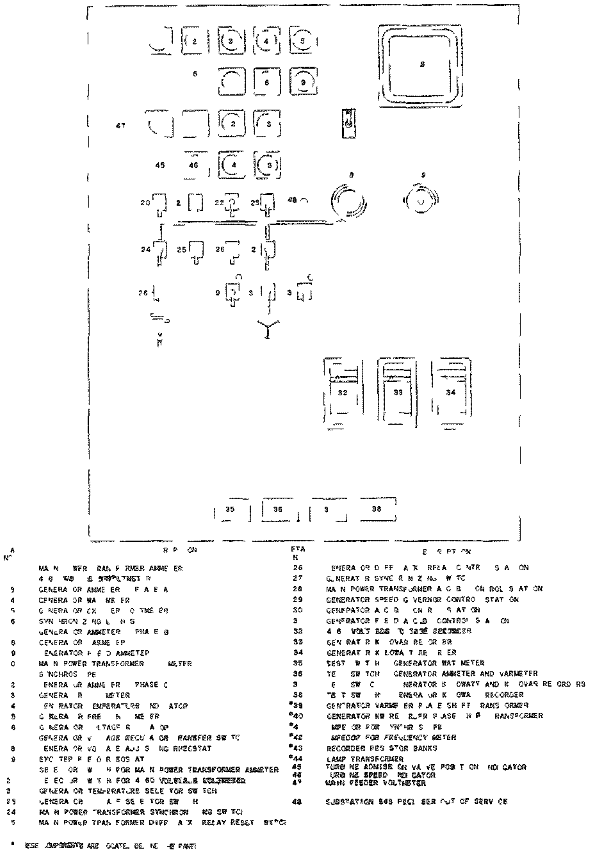


Figure 3-10

Electrical system, panel 1
RE-2-17811-C

Detail 47 is a voltmeter that reads the voltage of the 13 kilovolt-ampere main feeder. Detail 48 indicates out-of-service condition of the recloser at substation 543. The recloser is interlocked with air circuit breakers of the 13.2-kilovolt/4160-volt main power transformer and generator. When the generator is tied to the feeder, damage could be caused by a reclosure of substation 543 without synchronizing the Laboratory system with the utility system.

Figure 3-11 is the layout of the plant auxiliaries control, panel 2. Addition and modification of details 9, 12, 13, and 14 was necessary to provide control of the recombiner system.

Figure 3-12 is the layout of the turbine, generator, condenser, and circulating water control, panel 3. No changes were made on this panel other than the description of detail 2.

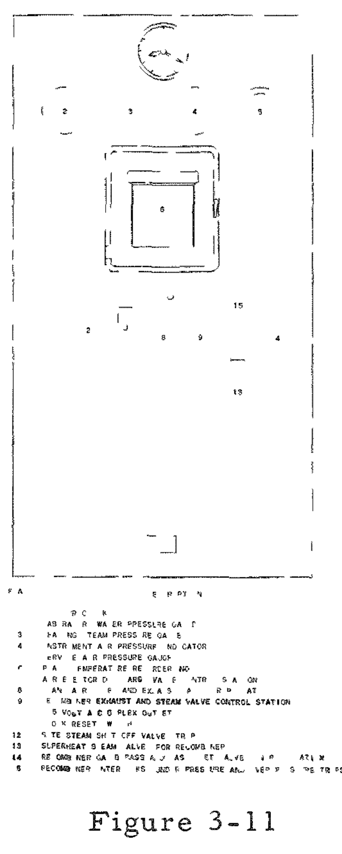
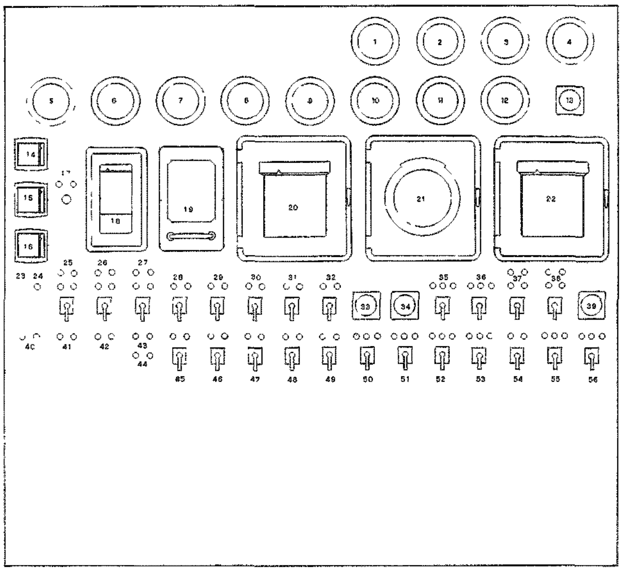


Figure 3-11

Plant auxiliaries control, panel 2
RE-2-17812-C



DETAIL NO	DESCRIPTION	DETAIL NO	DESCRIPTION
1	CONDENSER INLET PRESSURE INDICATOR	30	CONDENSER WATER BOX DRAIN CONTROL STATION (NORTH)
2	CONDENSER OUTLET PRESSURE INDICATOR	31	DESUPERHEATER SPRAY VALVE CONTROL STATION
3	GOVERNOR OIL PRESSURE INDICATOR	32	CIRCULATING WATER PUMP NO 1 DISCHARGE VALVE CONTROL STATION
4	BEARING OIL PRESSURE INDICATOR	33	CIRCULATING WATER PUMP NO 1 MOTOR AMMETER
5	G AND EXHAUST PRESSURE INDICATOR	34	CIRCULATING WATER PUMP NO 2 MOTOR AMMETER
6	DRY AIR PRESSURE INDICATOR	35	COOLING TOWER FAN NO 1 CONTROL STATION
7	GLAND STEAM PRESSURE INDICATOR	36	COOLING WATER PUMP NO 1 CONTROL STATION
8	COOLING TOWER LEVEL INDICATOR	37	AIR EJECTOR NO 1 CONTROL STATION
9	THROTTLE STEAM PRESSURE INDICATOR	38	AIR EJECTOR NO 2 CONTROL STATION
10	FIRST NOZZLE STEAM PRESSURE INDICATOR	39	TURBINE TURNING GEAR MOTOR AMMETER
11	RA EA INLET STEAM PRESSURE INDICATOR	40	RECOVERED AIR REACTIVATING BLOWER INDICATING LIGHTS
12	TURBINE EXHAUST PRESSURE INDICATOR	41	RECOVERED AIR REACTIVATING HEATER INDICATING LIGHTS
13	TURBINE SPEED INDICATOR	42	MAKE-UP AIR REACTIVATING BLOWER INDICATING LIGHTS
14	MAKE UP AIR FLOW RECORDER	43	MAKE-UP AIR REACTIVATING HEATER NO 1 INDICATING LIGHTS
15	RECOVERED AIR FLOW RECORDER	44	MAKE-UP AIR REACTIVATING HEATER NO 2 INDICATING LIGHTS
16	AIR EJECTOR AIR FLOW RECORDER	45	CONDENSER INLET VALVE CONTROL STATION (SOUTH)
17	TURBINE TRIP PUSH BUTTON	46	CONDENSER OUTLET VALVE CONTROL STATION (SOUTH)
18	TURBINE VIBRATION RECORDER	47	CONDENSER WATER BOX DRAIN CONTROL STATION (SOUTH)
19	TURBINE VIBRATION RECORDER POWER UNIT	48	COOLING TOWER BYPASS CONTROL STATION
20	PLANT TEMPERATURES RECORDER NO 2	49	CIRCULATING WATER PUMP NO 3 DISCHARGE VALVE CONTROL STATION
21	CONDENSER ABSOLUTE PRESSURE RECORDER	50	CIRCULATING WATER PUMP NO 1 CONTROL STATION
22	BEARING TEMPERATURE RECORDER	51	CIRCULATING WATER PUMP NO 2 CONTROL STATION
23	VAPOR RECOVERY CYCLE NO 1 INDICATING LIGHT	52	COOLING TOWER FAN NO 2 CONTROL STATION
24	VAPOR RECOVERY CYCLE NO 2 INDICATING LIGHT	53	COOLING WATER PUMP NO 2 CONTROL STATION
25	FREON COMPRESSOR CONTROL STATION	54	TURBINE GLAND STEAM CONTROL STATION
26	SEAL SYSTEM BLOWER CONTROL STATION	55	TURBINE OIL PUMP CONTROL STATION
27	RECOVERY SYSTEM SUMP PUMP CONTROL STATION	56	TURBINE TURNING GEAR CONTROL STATION
28	CONDENSER INLET VALVE CONTROL STATION (NORTH)		
29	CONDENSER OUTLET VALVE CONTROL STATION (NORTH)		

Figure 3-12

Turbine, generator, condenser, and circulating water control, panel 3
RE-2-17813-C

Figure 3-13 is the layout of the steam control, panel 4. The only change is in the description of detail 6, turbine-steam stop-valve control station. Since the revised system incorporates two steam stop valves, the new title of detail 6 in the figure shows that this stop valve controls steam to the turbine.

Feedwater control, panel 5, shown in Figure 3-14, is unmodified in the new system. Items 11 and 12 were disconnected and are now spare.

Figure 3-15, reactor auxiliaries control, panel 6, shows added details 5 and 6. Motor-operated discharge valves were added to No. 1 and No. 2 feedwater pumps. Details 5 and 6 are the control stations for these valves.

Many changes were made in reactor control, panels 7, 8, and 9. Figure 3-16 shows the present layout of the panels which reflects the various changes and additions.

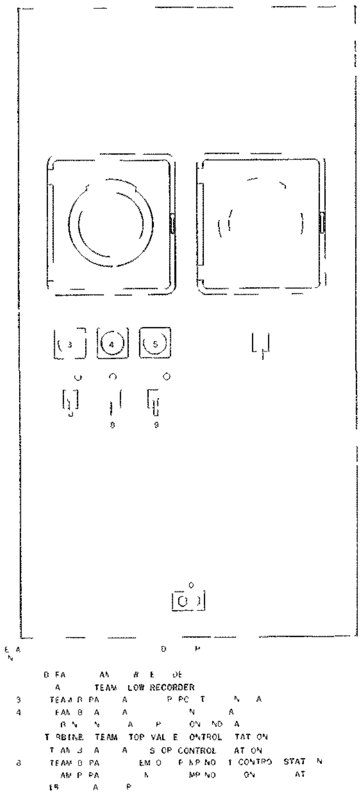


Figure 3-13

Steam control, panel 4
RE-2-17814-C

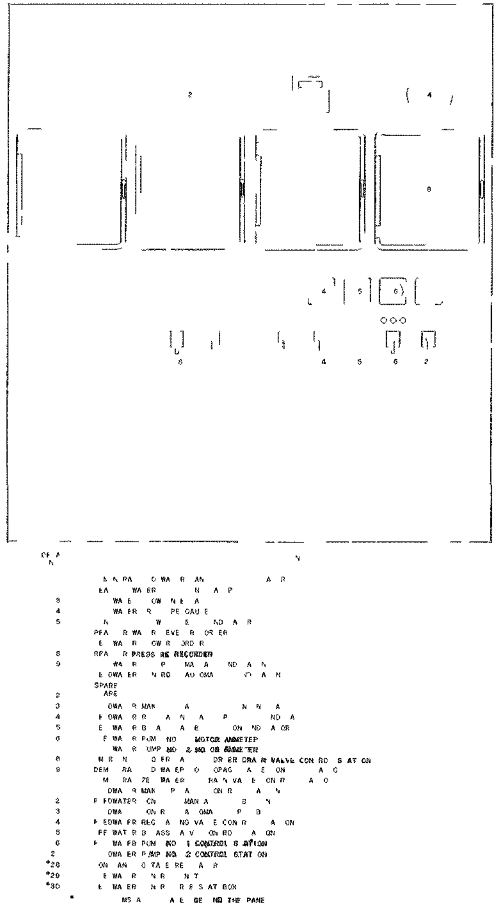


Figure 3-14

Feedwater control, panel 5
RE-2-17815-C

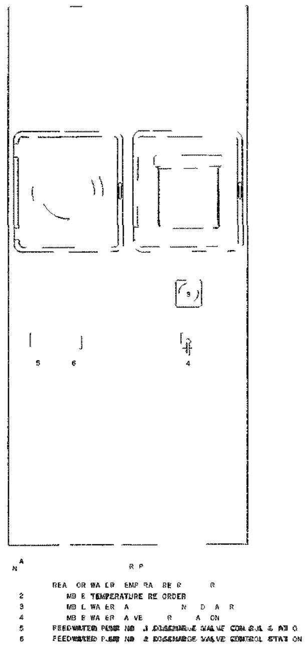
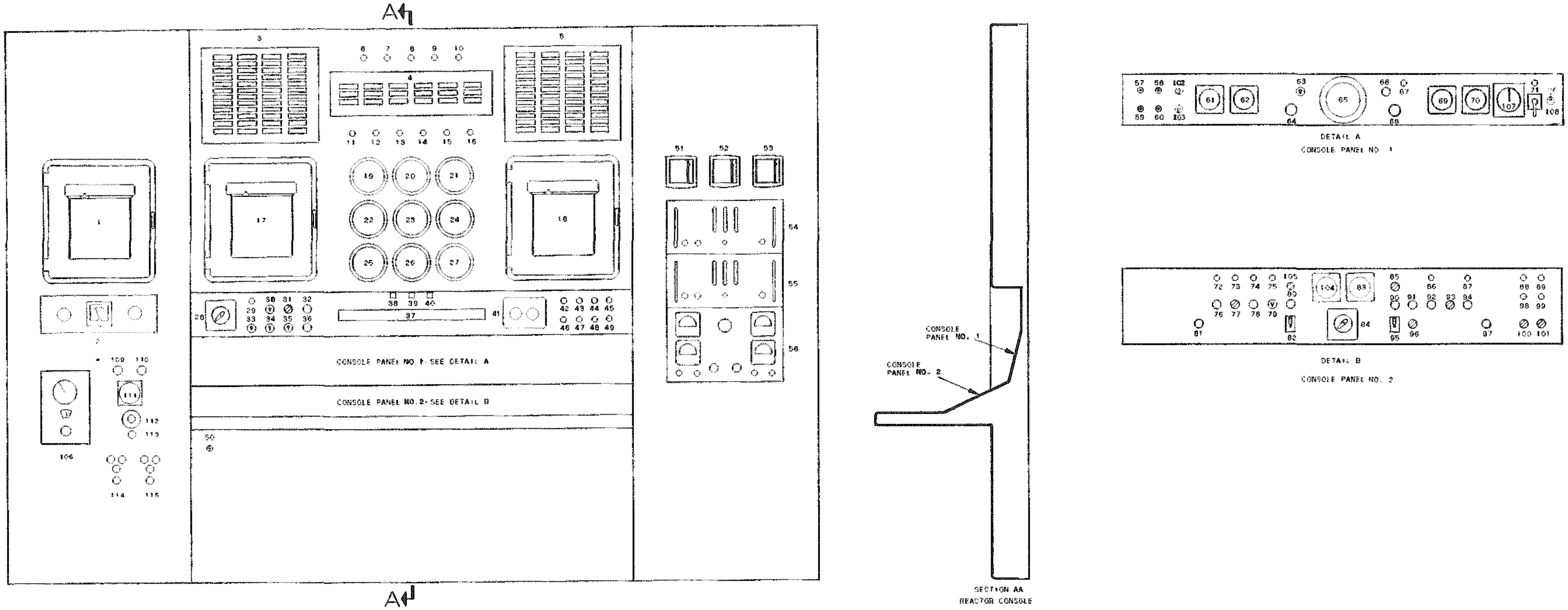


Figure 3-15

Reactor auxiliaries
control, panel 6
RE-2-17816-C



DETAIL NO.	DESCRIPTION
1	RADIATION MONITORING RECORDER
2	LINEAR FLUX LEVEL MICRO-MICRO AMMETER
3	PLANT ALARM ANNUNCIATOR NO. 1 - 48 POINT
4	SHUTDOWN ANNUNCIATOR - 18 POINT
5	PLANT ALARM ANNUNCIATOR NO. 2 - 48 POINT
6	HIGH-PRESSURE BORON SYSTEM DISCHARGE VALVE "OPEN" INDICATING LIGHT
7	HIGH-PRESSURE BORON SYSTEM DRAIN VALVE "CLOSED" INDICATING LIGHT
8	HIGH-PRESSURE BORON SYSTEM SHUT-OFF VALVE "OPEN" INDICATING LIGHT
9	LOW-PRESSURE BORON SYSTEM INLET VALVE "OPEN" INDICATING LIGHT
10	LOW-PRESSURE BORON SYSTEM DISCHARGE VALVE "OPEN" INDICATING LIGHT
11	HIGH-FLUX TRIP NO. 1 RESET PUSHBUTTON
12	HIGH-FLUX TRIP NO. 2 RESET PUSHBUTTON
13	HIGH-FLUX TRIP NO. 3 RESET PUSHBUTTON
14	PERIOD SHUTDOWN RESET PUSHBUTTON
15	REACTOR PRESSURE RATE-OF-CHANGE TRIP RESET PUSHBUTTON
16	REACTOR WATER TEMPERATURE TRIP RESET PUSHBUTTON
17	LOG-FLUX LEVEL RECORDER
18	LINEAR FLUX LEVEL RECORDER
19	ROD POSITION INDICATOR - SHIM ROD NO. 4
20	ROD POSITION INDICATOR - SHIM ROD NO. 5
21	ROD POSITION INDICATOR - SHIM ROD NO. 6
22	ROD POSITION INDICATOR - SHIM ROD NO. 3
23	ROD POSITION INDICATOR - CENTER ROD
24	ROD POSITION INDICATOR - SHIM ROD NO. 7
25	ROD POSITION INDICATOR - SHIM ROD NO. 2
26	ROD POSITION INDICATOR - SHIM ROD NO. 1
27	ROD POSITION INDICATOR - SHIM ROD NO. 8
28	HIGH-FLUX TRIP TEST SELECTOR SWITCH
29	HIGH-FLUX TRIP TEST LIGHT
30	TURBINE THROTTLE TRIP BYPASS KEY SWITCH
31	GALVANOMETER LIGHT SWITCH
32	PLANT ANNUNCIATOR TEST PUSHBUTTON
33	HIGH-FLUX TRIP TEST KEY SWITCH
34	CONDENSER HIGH-PRESSURE TRIP BYPASS KEY SWITCH
35	PERIOD TRIP BYPASS KEY SWITCH
36	SHUTDOWN ANNUNCIATOR TEST PUSHBUTTON
37	GALVANOMETER SCALE
38	ELECTRONIC HIGH-FLUX TRIP NO. 1 LIGHT
39	ELECTRONIC HIGH-FLUX TRIP NO. 2 LIGHT
40	ELECTRONIC HIGH-FLUX TRIP NO. 3 LIGHT
41	GALVANOMETER RANGE SELECTOR
42	MAIN ENTRANCE WEST DOOR OPEN INDICATOR LIGHT
43	REACTOR BUILDING FREIGHT DOOR OPEN INDICATOR LIGHT
44	CONTROL ROOM EMERGENCY EXIT OPEN INDICATOR LIGHT
45	PLANT EMERGENCY ESCAPE OUTSIDE DOOR OPEN INDICATOR LIGHT
46	SERVICE BLDG. NORTH ENTRANCE OPEN INDICATOR LIGHT
47	SERVICE BLDG. FREIGHT DOOR OPEN INDICATOR LIGHT
48	PLANT AIR LOCK INSIDE DOOR OPEN INDICATOR LIGHT
49	PLANT AIR LOCK OUTSIDE DOOR OPEN INDICATOR LIGHT
50	RECORDER JACK FOR GALVANOMETER CHAMBER CURRENT
51	HIGH-FLUX TRIP RECORDER NO. 1
52	HIGH-FLUX TRIP RECORDER NO. 2
53	HIGH-FLUX TRIP RECORDER NO. 3
54	SCALER FOR STARTUP COUNTER NO. 1
55	SCALER FOR STARTUP COUNTER NO. 2
56	REMOTE CONTROL FOR WIDE RANGE CHAMBER CURRENT AMPS
57	INTERCOM JACK - LINE NO. 1
58	INTERCOM JACK - LINE NO. 2
59	INTERCOM JACK - LINE NO. 3
60	INTERCOM JACK - LINE NO. 4
61	REACTOR PERIOD METER
62	FEEDWATER CONTROL VALVE POSITION INDICATOR
63	LOW-PRESSURE BORON INJECTION KEY SWITCH
64	PLANT SHUTDOWN PUSHBUTTON
65	REACTOR PRESSURE INDICATOR
66	HIGH-PRESSURE BORON INJECTION PUSHBUTTON
67	HIGH-PRESSURE BORON INJECTION "AUTOMATIC" INDICATOR LIGHT (DISCONNECTED)
68	REACTOR SHUTDOWN PUSHBUTTON
69	STEAM BYPASS VALVE POSITION INDICATOR
70	STEAM BYPASS VALVE STOP POSITION INDICATOR
71	REACTOR START-UP HEATER CONTROL STATION
72	FEEDWATER CONTROL "MANUAL" INDICATOR LIGHT
73	FEEDWATER REGULATOR VALVE "POWER ON" INDICATOR LIGHT
74	FEEDWATER CONTROL "AUTOMATIC" INDICATOR LIGHT
75	CONTROL POWER "ON" INDICATOR LIGHT
76	FEEDWATER CONTROL "MANUAL" PUSHBUTTON
77	FEEDWATER REGULATING VALVE CONTROL STATION
78	FEEDWATER CONTROL "AUTOMATIC" PUSHBUTTON
79	CONTROL POWER KEY SWITCH
80	CENTER ROD DROP TEST PUSHBUTTON
81	ANNUNCIATOR BELL RESET PUSHBUTTON
82	CENTER ROD CONTROL SWITCH
83	REACTOR PRESSURE REFERENCE SETTING INDICATOR - BYPASS
84	SHIM ROD SELECTOR SWITCH
85	REACTOR PRESSURE REFERENCE CONTROL STATION - BYPASS
86	STEAM BYPASS CONTROL "AUTOMATIC" INDICATOR LIGHT
87	STEAM BYPASS CONTROL "MANUAL" INDICATOR LIGHT
88	STARTUP COUNTER NO. 2 A-C POWER "ON" INDICATOR LIGHT
89	STARTUP COUNTER NO. 2 HIGH VOLTAGE "ON" INDICATOR LIGHT
90	SHIM ROD DROP TEST PUSHBUTTON
91	SHIM ROD SEQUENTIAL "OUT" PUSHBUTTON (DISCONNECTED)
92	STEAM BYPASS CONTROL "AUTOMATIC" PUSHBUTTON
93	STEAM BYPASS VALVE STOP CONTROL STATION
94	STEAM BYPASS CONTROL "MANUAL" PUSHBUTTON
95	SHIM ROD CONTROL SWITCH
96	SHIM ROD "SIMULTANEOUS LOWER" CONTROL STATION
97	SHUTDOWN ANNUNCIATOR RESET PUSHBUTTON
98	STARTUP COUNTER NO. 1 A-C POWER "ON" INDICATOR LIGHT
99	STARTUP COUNTER NO. 1 HIGH VOLTAGE "ON" INDICATOR LIGHT
100	STARTUP COUNTERS A-C POWER CONTROL STATION
101	STARTUP COUNTER HIGH VOLTAGE CONTROL STATION
102	REACTOR LOW TEMP. TRIP BYPASS KEY SWITCH
103	REBOILER STEAM VALVE (P-110) TRIP BYPASS KEY SWITCH
104	REACTOR PRESSURE REFERENCE SETTING INDICATOR-REBOILER
105	REACTOR PRESSURE REFERENCE CONTROL STATION-REBOILER
106	REACTOR PRESSURE PRECISION INDICATOR
107	ROD DROP-PERIOD-COUNTER AUTOMATIC TIMER
108	ROD DROP-PERIOD-COUNTER AUTOMATIC TIMER SELECTOR SWITCH
109	BORIC ACID CONCENTRATION LOW LIGHT
110	BORIC ACID CONCENTRATION HIGH LIGHT
111	BORIC ACID CONCENTRATION METER
112	BORIC ACID CONCENTRATION ALARM PRESET SWITCH
113	BORIC ACID CONCENTRATION ALARM RESET
114	BORIC ACID PUMP NO. 1 CONTROL STATION AND INDICATOR LIGHTS
115	BORIC ACID PUMP NO. 2 CONTROL STATION AND INDICATOR LIGHTS

NOTE SEE TEXT FOR DETAILED LISTING OF ALARM ANNUNCIATOR POINTS

Figure 3-16

Reactor control, panels 7, 8, and 9
RE-2-17817-D

Detail 2 of panel 7 was changed to a Keithley micromicroammeter. The time interval meter was replaced by a new timer. Added detail 106 is a precision pressure indicator that is readable and accurate to about 0.5 psi. Details 109 through 115 are associated with boric acid control.

Detail 102, shown on console panel 1 in detail A of Figure 3-16, is a reactor-low-temperature trip bypass key switch. This bypass is necessary for fuel loading, cold critical experiments, and slow reactor startups. Detail 103 is another bypass key switch needed for various modes of operation: the reboiler-steam-valve (P-11D) trip bypass key switch. Detail 107 is the rod-drop-period-counter automatic timer and detail 108 is the selector switch for setting the parameter to be timed.

On console panel 2 in detail B of Figure 3-16, detail 83 was moved to the right. Detail 104, reactor-pressure-reference setting indicator for the control valve to the reboiler has been placed next to detail 83. Detail 105 is the reactor-pressure-reference control station for the reboiler. When the reboiler plant is in operation, reactor pressure reference is controlled by these details.

Many changes were made on the plant annunciators and shutdown annunciator. The legends of Figures 3-17, 3-18, and 3-19 give the title of each annunciator point.

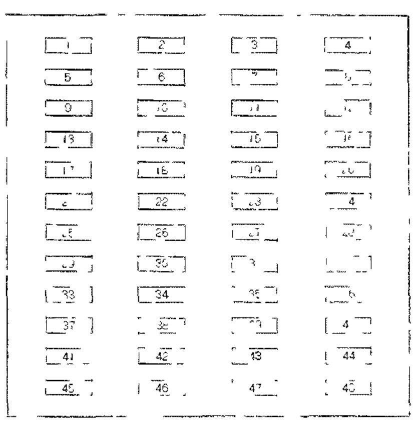


Figure 3-17

Plant alarm annunciator No. 1, panel 8 RE-6-19007-C

WINDOW NO.	ALARM TITLE	ALARM SETTING
	RADIATION MONITORING (HIGH)	40% SCALE
2	HIGH-PRESSURE BORIC ACID SYSTEM PRESSURE (LOW)	1400 PSI
3	TURBINE EXHAUST TEMPERATURE (HIGH)	250°F
4	REACTOR VESSEL GASKET LEAK	-
5	SHIELD COOLING WATER FLOW (LOW)	1.0 GPM
6	HIGH-PRESSURE BORIC ACID TANK LEVEL (LOW)	2 IN.
7	TURBINE OIL PRESSURE (LOW)	[illegible] PSIG
8	LIQUID RECOVERY SYSTEM DRYER SECTION ABNORMAL	-
9	NUCLEAR INSTRUMENTATION POWER FAILURE	-
10	HIGH-PRESSURE BORIC ACID SYSTEM CONTROL POWER FAILURE	-
11	TURBINE OIL TANK LEVEL (LOW OR HIGH)	±2 IN.
12	LIQUID RECOVERY SYSTEM VENT CONDENSER TEMPERATURE (HIGH)	50°F
13	REACTOR STARTUP HEATER PRESSURE (HIGH)	350 PSI
14	LOW-PRESSURE BORIC ACID SYSTEM CONTROL POWER FAILURE	-
15	TURBINE BEARING VIBRATION (EXCESSIVE)	1 MIL
16	LIQUID RECOVERY SYSTEM MAKE-UP AIR COOLER TEMPERATURE (HIGH)	50°F
17	REACTOR STARTUP HEATER VALVES (ABNORMAL)	-
18	LOW-PRESSURE BORIC ACID (LOW)	-
19	TURBINE BEARING TEMPERATURE (HIGH)	150°F
20	LIQUID RECOVERY SYSTEM SUMP TANK LEVEL (HIGH)	-
21	REACTOR WATER TEMPERATURE (LOW)	325°
22	WATER- [illegible] EXPANDERS TEMPERATURE (HIGH)	[illegible]
23	GENERATOR WATER LEAK	-
24	LIQUID RECOVERY SYSTEM LOSS OF VACUUM	2 IN. H_2O VAC.
25	REACTOR WATER LEVEL (HIGH)	5 IN.
26	HOTWELL LEVEL (LOW)	-2 IN.
27	CONDENSER PRESSURE (HIGH)	4 IN. Hg ABS
28	LIQUID RECOVERY SYSTEM DRY AIR PRESSURE (LOW)	+2 IN. H_2O
29	REACTOR WATER LEVEL (LOW)	-5 IN.
30	HOTWELL LEVEL (HIGH)	+9 IN.
31	COOLING WATER PUMP HEADER PRESSURE (LOW)	40 PSI
32	LIQUID RECOVERY SYSTEM DRYER AIR FLOW (HIGH)	100% SCALE
33	TURBINE STEAM STOP VALVE (CLOSED)	-
34	STEAM BYPASS SYSTEM OIL PRESSURE (LOW)	800 PSI
35	FISSION PRODUCT MONITOR (HIGH)	-
36	LIQUID RECOVERY SYSTEM MAKE-UP AIR FLOW (HIGH)	100% SCALE
37	REACTOR PRESSURE (HIGH)	615 PSIG
38	STEAM BYPASS SYSTEM [illegible] LEVEL (LOW)	0%
39	HOTWELL WATER CONDUCTIVITY (HIGH)	2 μMHOS
40	AIR EJECTOR AIR COUNT (HIGH)	100% SCALE
41	REACTOR PRESSURE (LOW)	550 PSIG
42	PURIFICATION SYSTEM FLOW (LOW)	-
43	ION EXCHANGE WATER CONDUCTIVITY (HIGH)	2 μMHOS
44	CONTROL BORIC ACID CONCENTRATION (LOW)	-
45	SHIELD COOLING SURGE TANK LEVEL (LOW)	1/2 FULL
46	RECOMBINER BYPASSED	0
47	CONTROL ROD THIMBLE TEMPERATURE (HIGH)	150°F
48	DRIER DRAIN VALVE (OPEN)	-

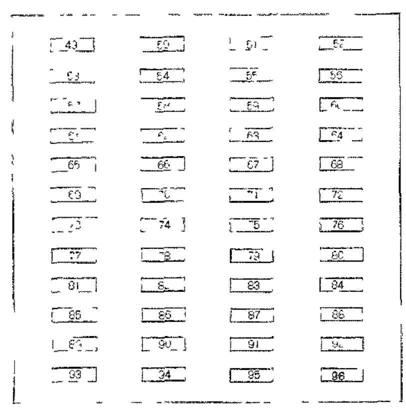
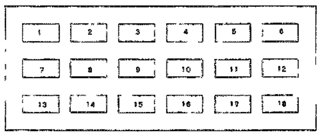


Figure 3-18

Plant alarm annunciator No. 2, panel 8 RE-6-19007-C

WINDOW NO.	ALARM TITLE	ALARM SETTING
49	MAIN POWER TRANSFORMER ACB AUTO TRIP	-
50	4160 VOLT BUS VOLTAGE (HIGH)	28 V
51	GENERATOR ACB AUTO TRIP	-
52	200-KW LABORATORY LOAD	-
53	MAIN POWER TRANSFORMER OIL TEMPERATURE (HIGH)	50°C
54	4160 VOLT BUS VOLTAGE (LOW)	102%V
55	GENERATOR STATOR TEMPERATURE (HIGH)	60°C
56	MOTOR CONTROL CENTER NO. 1 - 440-VOLT ACB AUTO TRIP	-
57	MAIN POWER TRANSFORMER OIL LEVEL (LOW)	-
58	MAIN POWER TRANSFORMER DIFFERENTIAL VOLTAGE	-
59	GENERATOR DIFFERENTIAL VOLTAGE	-
60	MOTOR CONTROL CENTER NO. 2, NO. 3 - 440-VOLT ACB AUTO TRIP	-
61	AUXILIARY POWER TRANSFORMER - 4160-VOLT ACB AUTO TRIP	-
62	RESERVE AUXILIARY TRANSFORMER - 440-VOLT ACB AUTO TRIP	-
63	DIESEL-GENERATOR SET (FAILURE TO START)	-
64	BATTERY M-G SET AUTO TRIP	-
65	AUXILIARY POWER TRANSFORMER - 440 VOLT ACB AUTO TRIP	-
66	RESERVE AUXILIARY TRANSFORMER OIL TEMPERATURE (HIGH)	80°C
67	DIESEL-GENERATOR SET WATER TEMPERATURE (HIGH)	200°F
68	LIGHTING AUTO SWITCH OPERATION	-
69	AUXILIARY POWER TRANSFORMER TEMPERATURE (HIGH)	90°C
70	RESERVE AUXILIARY TRANSFORMER OIL LEVEL (LOW)	-
71	DIESEL-GENERATOR SET (OIL PRESSURE LOW)	10S
72	125-VOLT D-C BUS GROUND	-
73	EMERGENCY ESCAPE LOCK (INSIDE DOOR OPEN)	-
74	CONTROL AIR PRESSURE (LOW)	80 PSIG
75	15,000-GAL WATER TANK LEVEL (HIGH)	+2 IN.
76	ELEVATOR EMERGENCY	-
77	BOTH AIRLOCK DOORS OPEN	-
78	SERVICE AIR PRESSURE (LOW)	80 PSIG
79	15,000-GAL WATER TANK LEVEL (LOW)	-2 IN.
80	AIR LOCK EMERGENCY	-
81	EMERGENCY COOLING WATER FLOW	-
82	RECIRCULATION WATER FLOW (LOW)	60 GPM, 100 AMPS TO PUMP
83	DEMINERALIZED WATER TANK LEVEL (LOW)	-
84	PLANT TEMPERATURE NO. 2 (HIGH)	-
85	REACTOR FEEDWATER PUMP NO. 1 - 440-VOLT ACB AUTO TRIP	-
86	CIRCULATING WATER PUMP NO. 1 - 440-VOLT ACB AUTO TRIP	-
87	RETENTION TANK NO. 2 WATER LEVEL (HIGH)	5 FT
88	STEAM DRYER EMERGENCY COOLER WATER LEVEL (HIGH)	-
89	REACTOR FEEDWATER PUMP NO. 2 - 440-VOLT ACB AUTO TRIP	-
90	CIRCULATING WATER PUMP NO. 2 - 440-VOLT ACB AUTO TRIP	-
91	RETENTION TANK NO. [illegible] WATER LEVEL (HIGH)	5 FT
92	SERVICE BUILDING RETENTION TANK WATER LEVEL (HIGH)	4 FT
93	COOLING TOWER pH	6.2-7.0
94	COOLING TOWER PHOSPHATE TANK (EMPTY)	PHOSPHATE NOT BEING ADDED TO COOLING TOWER
95	FEEDWATER REG. POWER OFF & SHIELD BLOWER OFF	-
96	NONE	-



WINDOW NO	ALARM TITLE
1	REACTOR WATER LEVEL (HIGH)
2	CONDENSER PRESSURE (HIGH)
3	TURBINE THROTTLE TRIP
4	REBOILER PLANT RADIATION MONITORING RATE (HIGH)
5	HIGH-PRESSURE BORIC ACID INJECTION VALVE (OPEN)
6	REACTOR FEEDWATER PUMPS 3 & 4 ACB AUTO TRIP
7	REACTOR WATER LEVEL (LOW)
8	PRESSURE RELIEF VALVE NO. 1 (OPEN)
9	PRIMARY-SYSTEM PRESSURE SAFETY VALVES (OPEN)
10	INTERMEDIATE-SYSTEM PRESSURE SAFETY VALVES (OPEN)
11	REACTOR STEAM PRESSURE (HIGH)
12	CONTAINMENT SHELL RADIATION MONITORING RATE (HIGH)
13	HIGH-FLUX TRIP NO. 1
14	HIGH-FLUX TRIP NO. 2
15	HIGH-FLUX TRIP NO. 3
16	REACTOR PERIOD TRIP
17	REBOILER STEAM VALVE TRIP (P-11D)
18	REACTOR WATER TEMPERATURE (LOW)

Figure 3-19

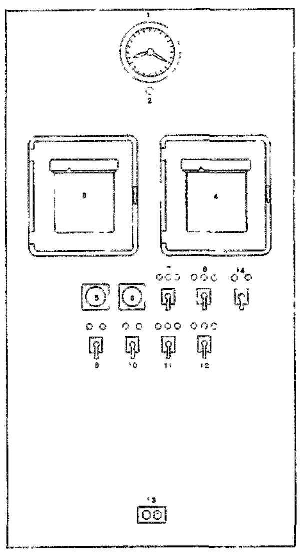
Shutdown annunciator, panel 8
RE-6-19005-A

Figure 3-20, the layout of reactor auxiliaries control, panel 10, shows a new control station. Detail 14 is the purification-system stop-valve control station added during the modification to control a motor-operated valve installed in the purification line.

The reactor auxiliaries control, panel 11, shown in Figure 3-21, was not changed during the modification.

Figure 3-22 is the layout and legend of the reboiler plant control, panel 12. The total panel is new to the control room and controls the reboiler plant and air-cooled condensers and heat exchangers. The legend conveys the purpose of each detail.

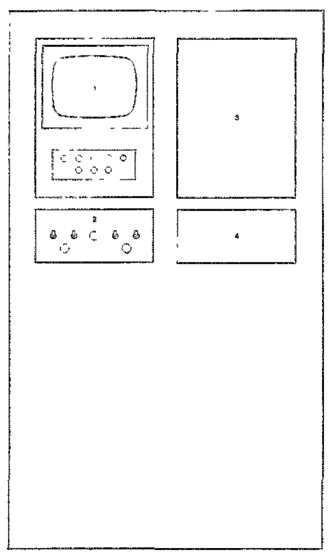
Figure 3-23 shows the annunciator points of plant annunciator 3, panel 12.



DETAIL NO.	DESCRIPTION
1	24-HOUR CLOCK
2	CLOCK RESET SWITCH
3	PLANT TEMP. RECORDER NO. 5
4	WATER CONDUCTIVITY RECORDER
5	VALVE POSITION INDICATOR ION EXCHANGE SYSTEM NO. 1
6	VALVE POSITION INDICATOR ION EXCHANGE SYSTEM NO. 2
7	ION EXCHANGE SYSTEM NO. 1 PUMP CONTROL STATION
8	SHIELD COOLING SYSTEM NO. 1 PUMP CONTROL STATION
9	ION EXCHANGE SYSTEM NO. 1 VALVE CONTROL STATION
10	ION EXCHANGE SYSTEM NO. 2 VALVE CONTROL STATION
11	ION EXCHANGE SYSTEM NO. 2 PUMP CONTROL STATION
12	SHIELD COOLING SYSTEM NO. 2 PUMP CONTROL STATION
13	115-VOLT A C UTILITY OUTLET
14	PURIFICATION SYSTEM STOP VALVE CONTROL STATION

Figure 3-20

Reactor auxiliaries control, panel 10
RE-2-17818-C



DETAIL NO	DESCRIPTION
1	CLOSED CIRCUIT TELEVISION MONITOR
2	TELEVISION CAMERA REMOTE CONTROL
3	BLANK PANEL FOR FUTURE TELEVISION
4	BLANK PANEL FOR FUTURE TELEVISION
*5	115 VOLT A-C SUPPLY NO. 1 CIRCUIT BREAKER
*6	115-VOLT A-C SUPPLY NO 2 CIRCUIT BREAKER
*7	125-VOLT D-C SUPPLY NO. 1 CIRCUIT BREAKER
*8	25-VOLT D C SUPPLY NO 2 CIRCUIT BREAKER

* THESE COMPONENTS ARE LOCATED BEHIND THE PANEL

Figure 3-21

Reactor auxiliaries control, panel 11
RE-2-17819-C

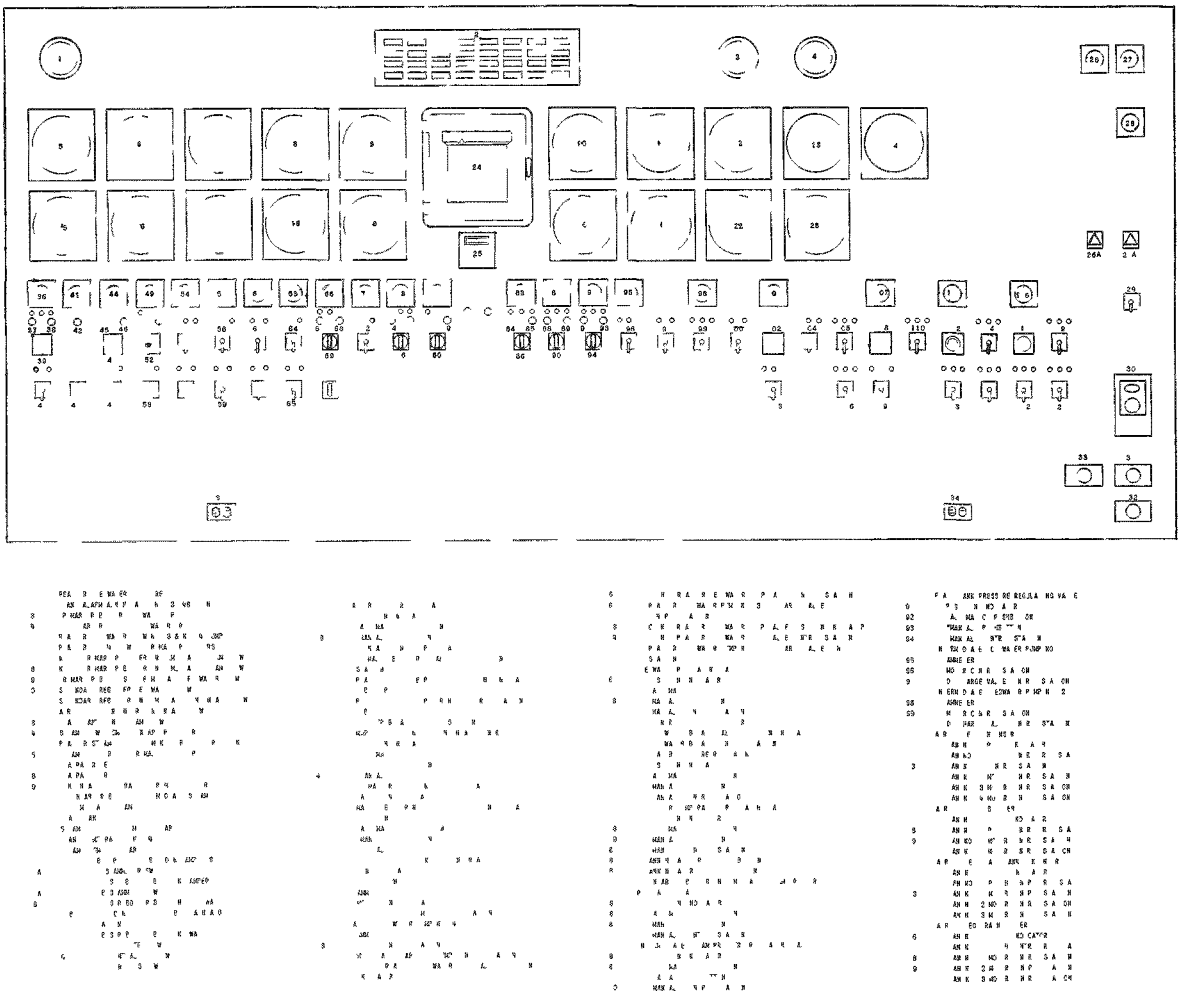


Figure 3-22

Reboiler plant control, panel 12
RE-2-33190-E

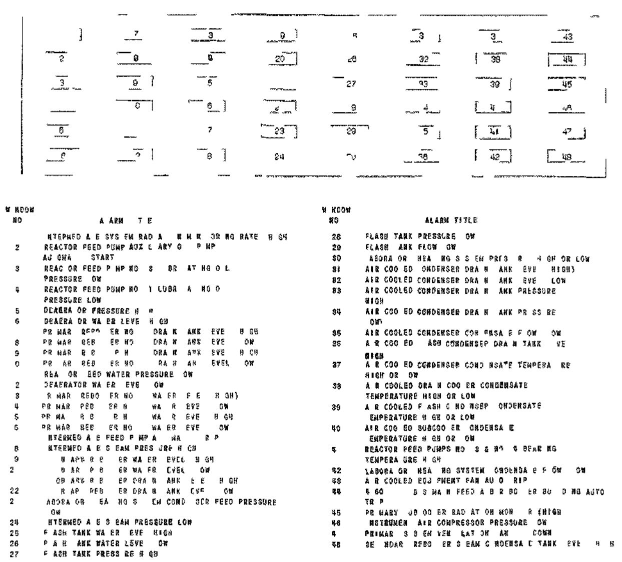


Figure 3-23

Plant alarm annunciator
No. 3, panel 12
RE-6-33125-C

3.4 ELECTRICAL DISTRIBUTION*

The operational characteristics of the turbine-generator plant remain as described in ANL-5607.[1] Modifications to the original plant are negligible and new equipment has been added only where additional electric power is necessary.

Past experience had shown that operation of some auxiliaries from the same transformer that supplies the panel board was not advisable. A 15-kilovolt-ampere, 3-phase, 440/120-208-volt transformer has been installed to supply the panel board only. In case of failure of this transformer, it is possible to switch to the original 30-kilovolt-ampere transformer.

Power for motor-operated valves, control valves, and boric acid addition pumps inside the containment vessel is supplied through an added circuit breaker box. Power is supplied to the box by the 30-kilovolt-ampere, 3-phase, 60-cycle, 440/120-208-volt transformer. This supply is on the emergency system using the diesel generator.

Equipment has been added to supply the power necessary for the reboiler plant. Refer to Figure 3-24 for a simplified line diagram of the EBWR electrical system.

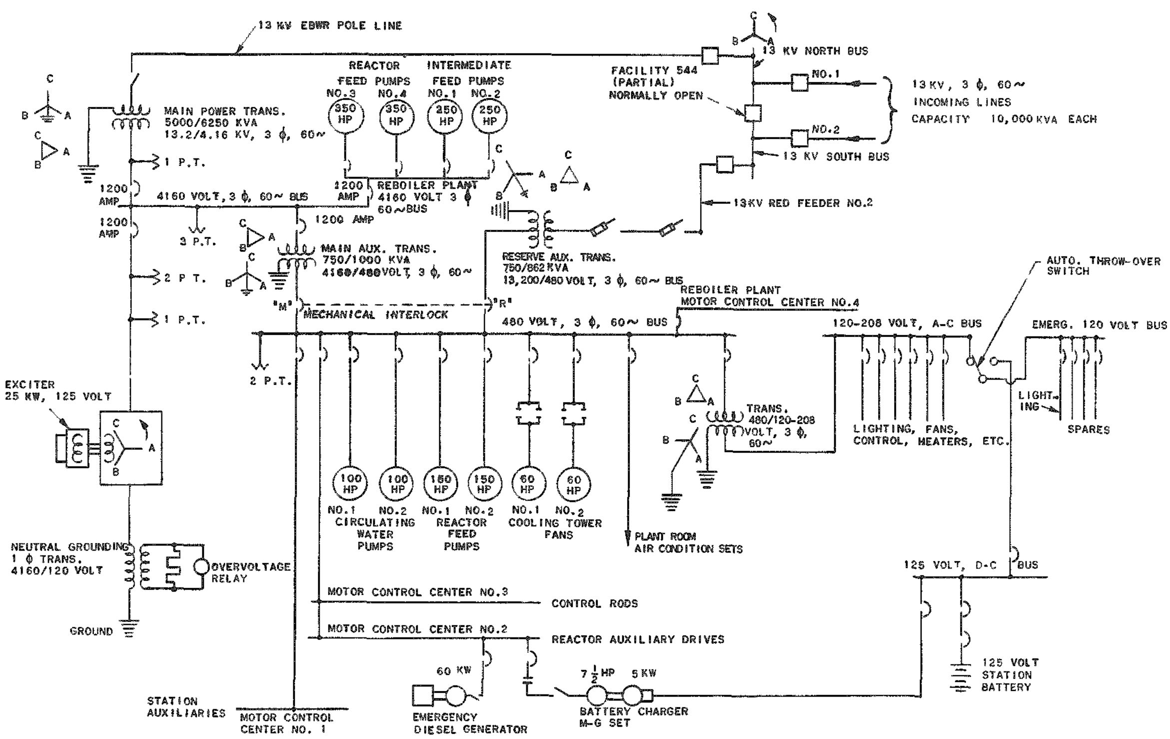


Figure 3-24

Simplified line diagram of EBWR electrical system
RE-8-17047-B

*A. Hirsch

A fifth unit was added to the outdoor 4160-volt bus substation. The unit contains a 1200-ampere, 3-pole single-throw air circuit breaker. Power is fed to the 4160-volt, 60-cycle, 3-phase bus of the reboiler plant. Four 1200-ampere indoor switchgear units in the reboiler building control power to the two 350-horsepower reactor feedpumps (No. 3 and No. 4) and the two 250-horsepower intermediate feedpumps (No. 1 and No. 2).

An air circuit breaker in the 440-volt switchgear in the service building supplies 440-volt, 60-cycle, 3-phase power to the bus in motor control center No. 4 of the reboiler building. This breaker was a spare in the original installation.

Motor control center No. 4 contains combination starter breakers and air circuit breakers for the twelve fans used on the air-cooled heat exchangers as well as combination units of this type for four condensate pumps and the primary ventilation system fan for the primary-reboiler enclosure and connecting tunnel. Air circuit breakers for the 30-kilovolt-ampere 440/120-208-volt transfer and the 440-volt, 3-phase utility outlets are also contained in motor control center No. 4.

The 30-kilowatt-ampere transformer supplies power to the auxiliaries in the reboiler building. This includes motor-operated valves, control valves, radiation monitors, lights, and utility outlets.

3.5 INSTRUMENTATION*

3.5.1 Nuclear

A design study of the EBWR 100-megawatt system revealed no significant excess demands on the nuclear instrumentation of the 20-megawatt system. Except for a few minor changes, the original nuclear instrumentation was incorporated in the 100-megawatt design. Figure 3-25 shows the ranges covered by the various instruments used to measure the neutron level of the reactor. To prevent saturation of the neutron detectors, the detectors were moved back in the tangential instrument holes to give approximately 20 microamperes at 100-megawatt operation.

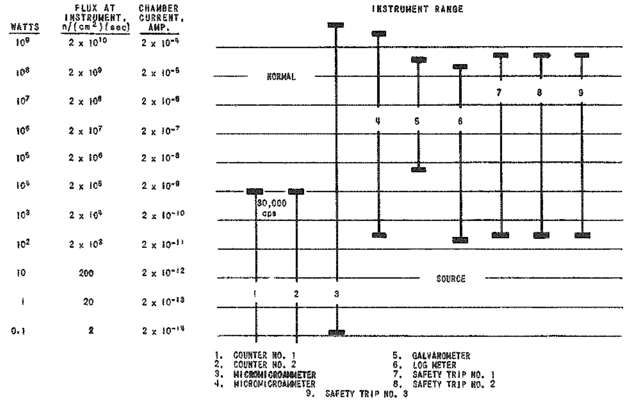


Figure 3-25

Ranges of nuclear instrumentation
RE-8-30197-A

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*A. Hirsch

Prior to 100-megawatt conversion, the vibrating-reed capacitance electrometer of channel No. 3 had been replaced by a micromicroammeter of greater range than the electrometer. This instrument can span the range from source level through the level of 100-megawatt operation. The minimum sensitivity is 10^{-3} amperes full scale.

The trip levels of the three safety-trip channels are set for trip-out when the flux exceeds a percentage above the desired flux. Since the heat load on the reactor is determined by the experiment in progress, there is a maximum power for various modes of operation. The magnetic trips and the electronic trips are set 30 percent and 25 percent respectively above the desired level.

3.5.2 Radiation Monitoring

Radiation monitoring within the containment vessel remains unchanged from the original system. Adjustable trip level and decade switching on the original monitors accommodates any increase in background level. The stack monitor will reflect a considerable increase in radiation level at 100-megawatt operation; an increase by a factor of 50 is the expected maximum, which is within allowable limits. High stack activity will initiate shutdown of the reactor and close the valves to the primary system external to the containment vessel.

The tunnel and shielding cubicle, built around the primary system external to the containment vessel, is ventilated through the roof of the reboiler building. A monitor, similar to the original stack monitor, is placed in the vent system. The activity level will normally be very low. The trip level will be set about a factor of 10 above normal level. Should the monitor detect a high radiation level due to steam or condensate leakage, the trip circuit will actuate the shutdown annunciator and initiate shutdown of the reactor. Shutdown of the reactor will automatically close steam valves P-11D and G-339 and isolate the primary reboilers from the reactor.

Five scintillation-type gamma monitors detect leakage from the primary system into the intermediate system or the circulating water system. Each instrument consists of a scintillation-detector head, a count-rate meter, and a trip circuit for annunciator alarm.

One of the scintillation detectors is placed adjacent to the side of each primary reboiler. Under steady-state nonleaking operation, the activity level will depend on power level and should become constant. The intermediate water in the reboiler will act as a shield to the detector. If a leak develops, the activity level of the intermediate water will rise integrating the activity that leaks through. The effective shielding of the water will lessen as the activity becomes diffused into the water and closer to the detector and the monitor count rate will rise.

The third detector, placed outside the cell next to the intermediate steam line, detects radioactivity carried over by the intermediate steam. The fourth detector is placed next to the drain tank of the secondary reboiler. The secondary-reboiler drain tank is the place where condensate of the intermediate system is first held up, which should enhance accurate measurement of the radiation concentration of the condensate.

The fifth detector is located next to the circulating water outlet of the subcooler. This detector is used primarily to assure that the activity of the circulating water is low, rather than a means of leak detection.

The four intermediate-system detector trip circuits are connected in series and are annunciated on window 1 of plant annunciator 3. The subcooler detector trip circuit is connected to window 45 (see Figure 3-23).

3.5.3 Turbine Plant

The turbine plant is basically the same plant that existed before modification. The major change is the modification of the water column and is described in section 3.5.3.3. Figure 3-26 shows the sensing locations of the process instrumentation.

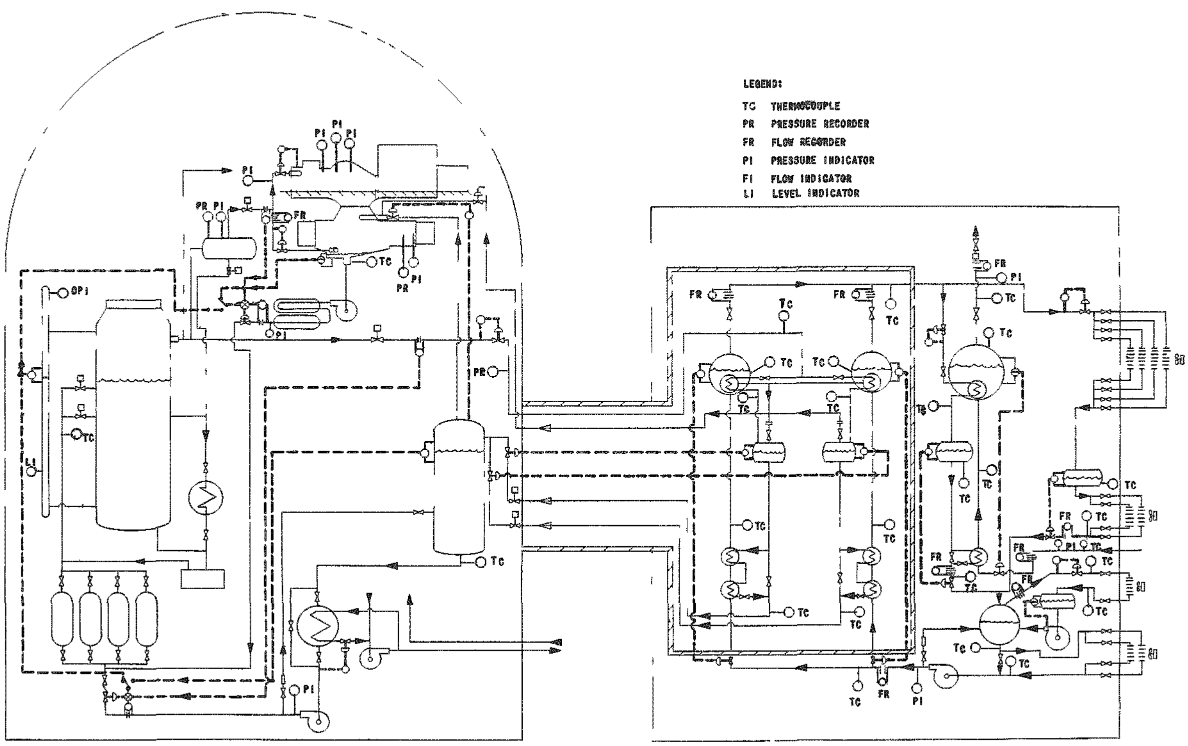

Figure 3-26

Simplified process instrument diagram
RE-1-33189-B

3.5.3.1 Pressure

1. Tests on EBWR had shown that the process-type pressure instruments did not have sufficient accuracy nor sensitivity for the experimental work of EBWR. A 0 to 1000-psig Norwood Electrosyn transducer with a Norwood Electrosyn precision indicator as the read-out device (Figure 3-16) was installed. This transducer senses reactor pressure through a tap in the top of the water column. Suitable lines and valves are provided so that calibration with a dead weight tester is possible.

2. A differential pressure gage was installed to read the pressure across the feedwater filters. As the filters load up, the pressure will rise across the filters. An abnormally high pressure could rupture the filters.

3.5.3.2 Temperature

1. The addition of a 6-inch feedwater ring to the reactor caused separation of the feedwater and purification water returning to the reactor. The purification water always returns through the 3-inch feedwater ring. A new thermocouple was installed in the 6-inch line.

3.5.3.3 Level

The following revisions and additions were made in the turbine plant for level determinations.

1. The water column was extended to elevation 724 feet $9\frac{3}{8}$ inches.

2. Three new cross-arms were installed between the water column and reactor vessel. These are at elevations 719 feet $3\frac{3}{8}$ inches, 724 feet $5\frac{3}{8}$ inches, and from the top of the water column to the reactor vent line. This arrangement allows better circulation of water between the column and vessel when operating at high water levels. The increased circulation will heat the column and reduce the cold water leg error in the sight glass. Figure 3-27 is a diagram of the water column instrumentation locations.

3. A new sight glass is installed to show reactor water level up to elevation 723 feet $1\frac{1}{4}$ inches; these levels are viewed through the television camera.

4. Three new Magnitrol water level instruments were added to the reactor water column: one for high water level measurement and two for low water level measurement. The new high-level instrument is at elevation 723 feet and is used when operating with the core riser. The two new low-level Magnitrol instruments are at elevations 720 feet 3 inches and 721 feet 3 inches and are used with the core riser and riser extension respectively. The configuration of the core riser and riser extension will determine which low-level instrument will be connected into the circuit. Limit-type switches for annunciator alarms were added to the three Magnitrol level devices installed on the water column. Alarms are now available as shown in Figure 3-27.

5. The water column taps of the level transducers have been moved to include a 96-inch span between elevations 715 feet 0 inch and 723 feet 0 inch. This allows full range of possible operating water levels without change of instruments.

6. The television camera has been mounted on a movable platform. The platform moves vertically, allowing perpendicular scan of the extended sight glass.

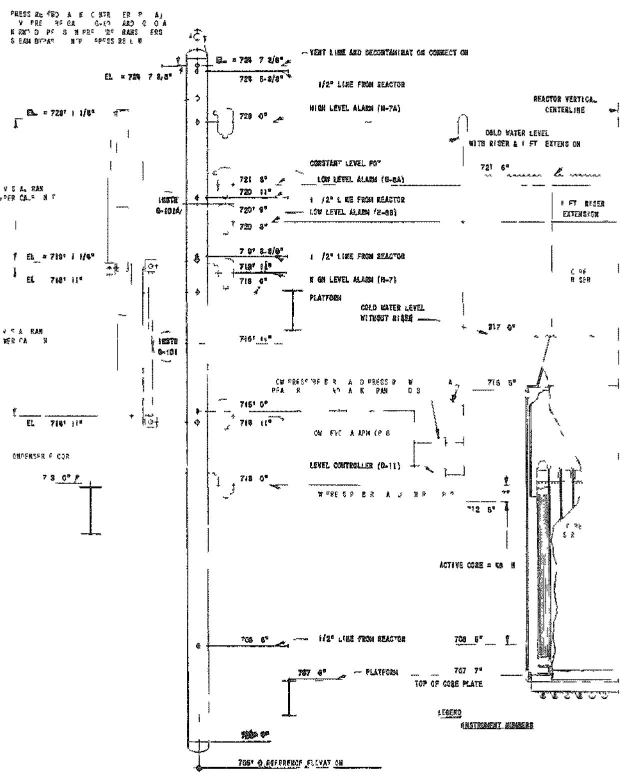

Figure 3-27

Water column instrument locations
RE-6-32769-C

3.5.3.4 Flow

An orifice was added to the outlet of each feedwater filter. The pressure lines from these orifices are connected to a manifold so that they can be read on a common differential pressure gage. This will make possible the reading of flow through each filter. As the filters load up, the flow will drip.

3.5.3.5 Conductivity

Revision of the purification system caused some changes in the conductivity measurement points to be made. Cells are placed to read:

1. Ion Exchanger Inlet
2. Anion Exchanger No. 1 Outlet
3. Anion Exchanger No. 2 Outlet
4. Ion Exchanger Outlet

3.5.4 Reboiler Plant

Modifications in the EBWR control system to accommodate parametric measurement-control instrumentation of the reboiler plant are delineated in the following paragraphs. Table 3-2 lists the reboiler instrumentation of the 100-megawatt-capacity system.

3.5.4.1 Pressure

1. The reactor-pressure-measurement system includes a 250 to 650-psig pressure transducer for sensing reactor pressure. The transducer is connected to a tap at the top of the water column. The output signal of the transducer is compared to a pressure reference signal from the control panel and the difference is displayed on a ±50-psi, reactor-steam-pressure minus pressure-reference recorder. A slidewire in the recorder supplies a signal to the reactor-pressure controller. This system is labeled P-11 in Table 3-2.

2. The reactor feedwater pressure from No. 1 and No. 2 feedwater pumps is measured by a 0 to 1000-psig pressure transducer. The transducer is tapped into the feedwater line between the reboiler-plant reactor feedpumps and control valve (P-21E). The pressure is displayed on an indicator on the control panel. This system is labeled P-2.

3. To measure the steam pressure to primary reboilers, a 0 to 1000-psig transducer was tapped into the reactor steam line leaving the containment vessel. The tap is downstream of

pressure control valve P-11D. The pressure reading will be the sum of the condensing pressure in the primary reboilers plus the pressure drop over the piping. The system is designated P-16.

4. Deaerator pressure is measured by a pressure transducer that ranges from -15 to +100 psig. The pressure is presented at the control board on an indicator-recorder-controller instrument. The system is designated P-12.

5. Intermediate steam pressure is measured by a 0 to 500-psig transducer connected to the intermediate steam line. The transducer transmits a signal to an indicator-recorder-controller on the panel. The system is designated P-14.

6. The condensing pressure of the secondary-reboiler intermediate steam is measured by a 0 to 500-psig transducer. An indicating-recording-controlling instrument on the panel receives the signal. The system is designated P-13.

7. Secondary-reboiler feedwater pressure is measured by a 0 to 500-psig pressure transducer that was tapped into the 6-inch feedwater line from the Laboratory power plant to transmit a signal to an indicator on the panel. This system is designated P-4.

8. Primary-reboiler feedwater pressure is transmitted by a 0 to 500-psig transducer tapped into the line between the intermediate feedwater pumps and the regulating valve. An indicator on the panel receives the signal. The system is designated P-5.

9. The steam pressure from the secondary reboilers is measured by a 0 to 300-psig transducer tapped into the 12-inch line between the nonreturn valve on top of the reboiler and the shut-off valve. The transducer will measure the reboiler-secondary-side pressure minus the drop across the nonreturn valve. When the reboiler is not in operation, the transducer will measure Laboratory steam pressure. The system is designated P-17.

10. Flash tank pressure is measured by a 0 to 50-psig transducer tapped into the tank. The signal is read at the panel on an indicator-controller instrument. The system is designated P-15.

3.5.4.2 Temperature

1. The temperature of the condensate from the subcooler is measured by an iron-constantan thermocouple. The indicating-recording-controlling instrument on the panel supplies a signal

to the circulating-water valve to control the temperature. The system is designated P-42.

2. Plant temperature No. 4 is measured and recorded by a 24 point indicating-recording instrument. All thermocouples are iron-constantan and the range is double scale, 0 to 600°F. The instrument and thermocouples are designated P-41.

   The points are:

   a. Steam to primary reboiler
   b. Condensate from primary reboiler No. 1
   c. Condensate from primary drain cooler No. 1
   d. Condensate from deaerator
   e. Feedwater to primary drain coolers
   f. Feedwater to primary reboiler No. 1
   g. Intermediate steam
   h. Condensate from secondary reboiler
   i. Condensate from secondary drain cooler
   j. Feedwater to secondary drain cooler
   k. Feedwater to secondary reboiler
   l. Heating steam
   m. Condensate from primary reboiler No. 2
   n. Condensate from primary drain cooler No. 2
   o. Feedwater to primary reboiler No. 2
   p. Primary reboiler No. 1 steam temperature
   q. Primary reboiler No. 2 steam temperature
   r. Secondary reboiler steam temperature
   s. Condensate from air-cooled condenser
   t. Condensate from air-cooled drain cooler
   u. Condensate from flash tank
   v. Condensate from air-cooled subcooler
   w. Flash tank vent steam
   x. Condensate from air-cooled flash condenser

3.5.4.3 Level

1. Deaerator level is measured by a transducer with a span of 0 to 100 inches of water and instrumented with an indicator-recorder-controller. The system is designated P-21.

3.5.4.4 Flow

1. Reactor steam flow to the primary reboilers is measured by a flow nozzle and transducer. The nozzle develops a differential pressure of 100-inch water column at a flow of 400,000 pounds per hour. The transducer extracts the square root of the water column head and transmits the signal to the panel board. The indicator-recorder-controller receives the signal. This system is designated P-22. A retransmitted signal from the instrument is used in the three-parameter feedwater control system.

2. Flow through reactor feedwater pumps 3 and 4 is measured by a flow nozzle that develops a differential pressure of 100-inch water column for flow of 400,000 pounds per hour. The indicator-recorder-controller instrument develops a signal proportional to flow rate for the reactor power integrator. The system designation is P-23.

3. Intermediate steam flow through primary reboiler No. 1 is measured by a nozzle that develops a differential pressure of 100-inch water column for 200,000 pounds per hour of steam flow and a transducer that extracts the square root of the water column head and transmits to an indicator-recorder. This system is designated P-25.

4. Intermediate steam flow through primary reboiler No. 2 is measured as in item 3 above. The designation is P-26.

5. Intermediate feedwater flow through the primary reboilers is measured by a nozzle that develops a differential pressure of 100-inch water column for 400,000 pounds per hour of feedwater flow and a transducer that extracts the square root of the water column head and transmits to an indicating-recording instrument in the control panel. The system designation is P-28.

6. Feedwater flow through the secondary reboiler is measured as in item 5 above. The system is designated P-29.

7. Intermediate condensate flow through the secondary reboiler is measured as in item 5 above. The system designation is P-31.

8. Air-cooled condenser condensate flow is measured by a nozzle that develops a 100-inch water column differential pressure for 300,000 pounds per hour of flow. A transducer extracts the square root of the water column head and transmits to an indicating-recording instrument. The system designation is P-32.

9. Flash-tank-vent steam-flow measurement includes an orifice plate that develops a differential pressure of 100-inch water column for a full flow of 15,000 pounds per hour. The flow is indicated and recorded. The system is designated P-33.

10. Steam flow from the secondary reboiler is measured by a nozzle that develops a differential pressure of 100-inch water column at full scale flow of 400,000 pounds per hour. The flow is indicated and recorded. The system is designated P-34.

3.5.4.5 Conductivity

The original conductivity instrument has been modified from a 6-point to a 12-point recorder. A conductivity cell is installed in the continuous blowdown line from the secondary reboiler. The conductivity prints out on points 6 and 12 of the recorder.

3.5.4.6 Position Indication

It is necessary or desirable to know the position of various valves and other devices. A standard circuit is used for position indication. Valves and fan pitch devices having position indication in the control room are:

1. Reactor-pressure regulating valve P-11D
2. Primary reboiler No. 1 condensate control valve G-332
3. Primary reboiler No. 2 condensate control valve G-332
4. Reactor feedwater 3-inch stop valve FDW3
5. Reactor feedwater 6-inch stop valve PS-12
6. Feedwater pump No. 3 and No. 4 regulating valve P-21D
7. Deaerator-pressure regulating valve P-12D
8. Subcooler-temperature regulating valve P-42C
9. Secondary-reboiler intermediate-pressure regulating valve P-13D
10. Intermediate-steam-pressure regulating valve P-14D
11. Flash-tank-pressure regulating valve P-15D
12. Air-cooled condenser fan No. 1
13. Air-cooled subcooler fan No. 1
14. Air-cooled flash tank condenser fan No. 1
15. Air-cooled drain cooler fan No. 1

Table 3-2 Reboiler Instrumentation

System designation	Title	Parameter: Pressure	Level	Temp	Flow	Range	Panel instrument: Initiate	Record	Control	Fig 3-? Detail	Transducer range	Nozzle: Size in	Material	Working pressure	Remarks
P-2	Reactor feedwater pressure	x				1000 psig	x			1	0-1000 psig				
P-4	Secondary reboiler feedwater pressure	x				0-[illegible]00 psig	x			4	0-400 psig				
P-5	Primary reboiler feedwater pressure	x				0-500 psig	x			3	0-500 psig				
P-11	Reactor steam pressure minus P_0	x				50 psi	x		x	15	50-650 psig				P_0 is a bucking voltage
P-12	Deaerator pressure	x				15 to 100 psig	x	x	x	18	15 to 100 psig				Instrument has adjustable set point for control
P-13	Secondary reboiler intermediate steam pressure	x				0-500 psig	x	x	x	20	0-500 psig				Instrument has adjustable set point for control
P-14	Intermediate steam pressure	x				0-500 psig	x	x	x	21	0-500 psig				Instrument has adjustable set point for control
P-15	Flash tank pressure	x				0-50 psig	x		x	22	0-50 psig				Instrument has adjustable set point for control
P-16	Steam pressure to primary reboilers	x				0-1000 psig	x	x		16	0-1000 psig				
P-17	Steam pressure from secondary reboiler	x				0-300 psig	x	x		23	0-300 psig				
P-21	Deaerator level		x			0-100 in	x	x	x	17	0-100 in W C				Instrument has adjustable set point for control
P-22	Reactor steam flow to primary reboilers				x	0-400 000 lb/hr	x	x	x	6	0-100 in W C $\sqrt{}$[a]	8	SS	600 psig	Control slide wire used in 3-way feedwater control
P-23	Reactor feedwater flow No 3 and No 4 pump				x	0-400 000 lb/hr	x	x	x	5	0-100 in W C $\sqrt{}$	6	SS	650 psig	Slide wires for 3-way feedwater control and integrator
P-25	Primary reboiler No 1 intermediate steam flow				x	0-200 000 lb/hr	x	x		7	0-100 in W C $\sqrt{}$	8	SS	350 psig	
P-26	Primary reboiler No 2 intermediate steam flow				x	0-200 000 lb/hr	x	x		8	0-100 in W C $\sqrt{}$	8	SS	350 psig	
P-28	Primary reboiler intermediate feedwater flow				x	0-400 000 lb/hr	x	x		9	0-100 in W C $\sqrt{}$	6	SS	[illegible]00 psig	Slide wire for integrator
P-29	Secondary-reboiler feedwater flow				x	0-400 000 lb/hr	x	x		10	0-100 in W C $\sqrt{}$	6	SS	250 psig	Limit switch for low flow
P-31	Secondary reboiler intermediate condensate flow				x	0-400 000 lb/hr	x	x		11	0-100 in W C $\sqrt{}$	6	SS	325 psig	
P-32	Air cooled condenser condensate flow				x	0-300 000 lb/hr	x	x		12	0-100 in W C $\sqrt{}$	6	SS	325 psig	Limit switch for low flow
P-33	Flash-tank vent steam flow				x	0-15 000 lb/hr	x	x		13	0-100 in W C $\sqrt{}$	8 orifice	SS	34 psia	Limit switch for low flow
P-34	Steam flow from secondary reboiler				x	0-400 000 lb/hr	x	x		14	0-100 in W C $\sqrt{}$	14	SS	200 psig	Slide wire for integrator
P-41	Plant temperature No 4			x		0-600°F	x	x		24	I C couple[b]				24 point divided chart
P-42	Condensate temperature from subcooler			x		0-400°F	x	x	x	19	I C couple				

[a] $\sqrt{}$ denotes transducer extracts square root of water column to give signal proportional to flow

[b] I C couple = iron constantan thermocouple

3.6 PLANT CONTROL*

3.6.1 Mode of Operation

EBWR over-all control-system design is based on four main operating objectives: (1) the reactor is operated at the power level required by the experimental work in progress, (2) the power produced is supplied to the Laboratory as useful electric power and heating steam, (3) during cold weather operation, some of the intermediate steam is used to protect the air-cooled equipment against freeze-up and maintain the equipment in a standby condition to assume load if necessary and (4) power in excess of Laboratory requirements is dissipated by the air-cooled equipment.

From the standpoint of the Laboratory heating system, the EBWR plant is base loaded, and the Laboratory coal-fired boilers assume regulation of Laboratory system steam pressure. Changes in the Laboratory steam system are not reflected back to the reactor.

The reactor and turbine plant can be operated, as described in report ANL-5607,[1] without operating the reboiler plant. However, operation of the reboiler plant is not possible unless the turbine plant is in operation. The turbine is not required to accept steam, but the condenser must be in operation. The condenser is the low-pressure sink necessary for the deaerator to operate. Noncondensable gases from the deaerator will pass to the condenser and be removed by the air ejectors. Cooling of the air ejectors is accomplished with condensate flow from the condenser and therefore requires operation of No. 1 or No. 2 feed pump. To maintain water inventories in the reactor and deaerator, the condenser condensate must be returned to the reactor.

Auxiliary equipment in the reboiler plant requires considerable electrical power (possibly around 1 megawatt when dissipating at full capacity through the air-cooled equipment). It is desirable, but not required, that the turbine plant be operating and producing electric power.

Normal operation of the turbine plant is described in report ANL-5607 and is not repeated in this publication. However, operation of the reboiler plant that affects or modifies the operation of the turbine plant is included.

Operation of the turbine-bypass system is explained in report ANL-5607. This section refers only to operation with the reboilers in operation.

Reference to the simplified control diagram shown in Figure 3-28 will show that reactor pressure is controlled by an upstream pressure-regulator valve, P-11D. This valve is operated in the same fashion as the

*A. Hirsch

original steam bypass valve to maintain reactor pressure. During normal operation, the original bypass valve is closed and P-11D operates, with reset action, to maintain the pressure at the desired level. The pressure reference regulator for the bypass valve is set slightly above the desired pressure. In case of a pressure rise at a rate greater than P-11D can control, the bypass valve will open and aid in holding the pressure down. As reset action allows P-11D to open further, the bypass valve will again close and remain in standby condition.

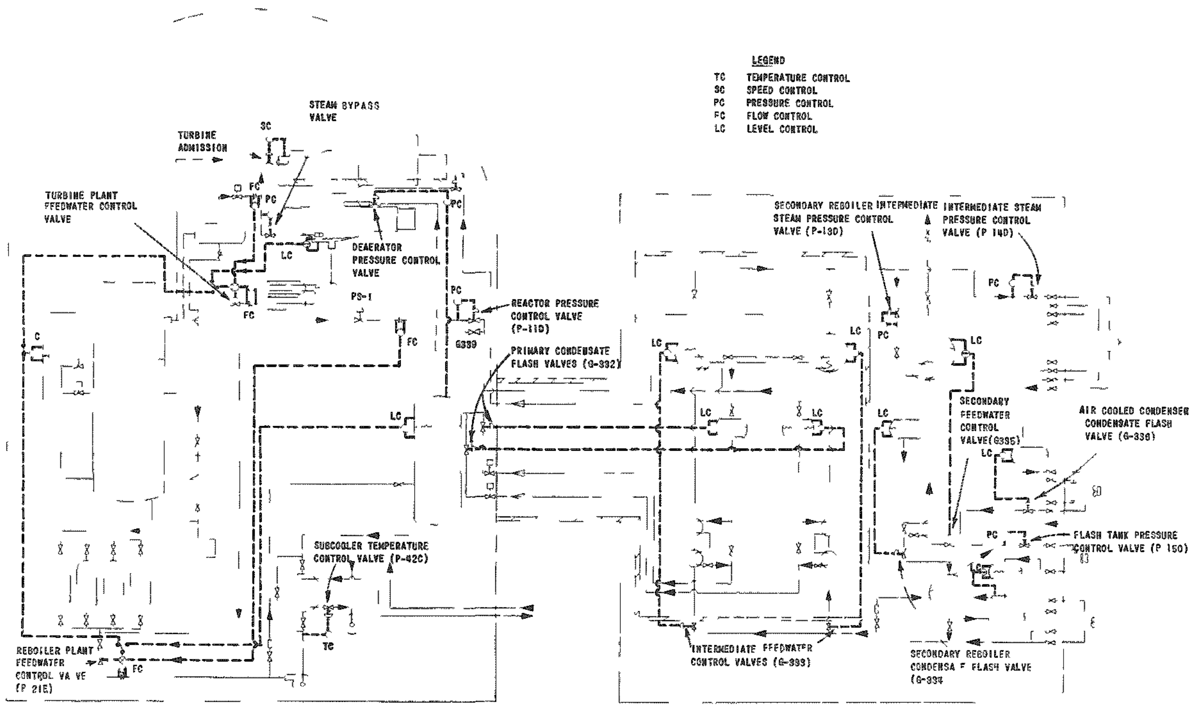

Figure 3-28

Simplified control diagram
RE-8-33158-B

Increased steam flow into the primary reboiler tubes will raise the pressure, and the temperature difference across the heat transfer walls will rise. Thus, more heat will be transferred to the intermediate shell side. The increased steam flow will produce a corresponding increase in condensate flow and the drain coolers will heat more intermediate feedwater.

Flashing the condensate into the deaerator will cool the condensate to the saturation temperature that corresponds to the deaerator pressure. The deaerator pressure will be maintained by an upstream pressure regulating valve, P-12D, which discharges into the condenser.

Further cooling of the condensate is accomplished in the subcooler. The temperature of the condensate leaving the subcooler is regulated by P-42C which controls the flow of circulating water through the tubes of the subcooler.

Since the reactor operates at a constant pressure and receives feedwater at a constant temperature, the flow is a direct function of reactor power.

When power delivered to the primary heat exchangers exceeds the power delivered to the Laboratory heating system and the power necessary to protect the air-cooled equipment, the excess heat is dumped through the air-cooled equipment. Valve P-14D will admit the excess steam to the air-cooled condenser to maintain the set pressure in the intermediate system.

When the power delivered to the primary heat exchangers is less than the power delivered to the Laboratory heating system and the power necessary to protect the air-cooled equipment, the intermediate pressure regulator, P-14D, closes as the intermediate pressure drops. Valve P-13D opens to maintain secondary reboiler intermediate pressure. The intermediate pressure will drop until the temperature difference across the reboilers is proper for the amount of heat available. Should the need arise to keep the air-cooled equipment hot, it will be necessary to open P-14D to a fixed position. The pressure in the intermediate loop will again adjust itself for the proper temperature difference.

The amount of heat transferred will be a function of the secondary pressure and the tube condensing pressure. The condensing pressure is controlled by valve P-13D, a down-stream pressure regulator.

3.6.2 Reactor

3.6.2.1 Control Rods and Rod Drives

New control rods and rod drives are installed in the 100-megawatt-capacity reactor. Limit switches and electrically operated position indicators for the new rod drives operate in the same manner as the previous control rod drives except for two modifications: (1) limit switches LS-18 through LS-88 and the associated high-pressure boric acid automatic injection system are disconnected and (2) since no latching device is used, limit switches LS-17 through LS-97 are replaced by a current relay in series with the clutch. Except for the high-pressure boric acid automatic injection system, there is no difference in control sequence and operation between the new and the original control rods and drive systems.

Latch magnets are not used on the new rod drives; instead, electric clutches engage and disengage the rod drives from the drive motors. Current relays in series with the clutch coils pull in when the clutch is energized. A 90-volt d-c source supplies current to each of the nine clutches. Operation of the rods from the control console remains the same as for the original rod drives.

3.6.2.2 Shutdown Parameters

Inclusion of the reboiler plant in the 100-megawatt-capacity EBWR design required that the shutdown system be expanded to accommodate additional shutdown parameters. The number of additional shutdown circuits exceeded the number of spare circuits provided in the original shutdown annunciator. To accommodate the required number of circuits, auxiliary circuits that activate a single annunciator window were installed in the shutdown system. The auxiliary circuits have indicators to specify the circuit causing shutdown. Figure 3-19 shows annunciator window assignments. The parameters and conditions of alarm of the new shutdown circuits are as follows:

1. Reactor water level, high. A second high-level Magnetrol instrument was added to the water column (see Section 3.5.3.3). Choice of high level Magnetrol in operation allows limit level at 78 inches above the top of a 4-foot core or 24 inches above the top of the 5-foot core riser.

2. Condenser pressure, high (greater than 20 inches Hg absolute). Keyswitch 100 bypasses this interlock and allows the reactor to be operated before normal vacuum is established in the condenser. Under these conditions, reactor steam can be used to establish the vacuum by operating the steam-powered air ejectors.

3. Turbine throttle trip, closed. Loss of electric load automatically closes the turbine throttle valve through an interlock in the main generator breaker. The reactor will be shut down automatically by tripping of this valve to the closed position. Keyswitch 85 can be closed to bypass this interlock.

4. Reboiler-plant-radiation monitoring rate, high. Alarm level is set at about a factor of 10 above normal operating background (see Section 3.5.2).

5. High-pressure boric acid injection valve, open. Opening the valve by any means will alarm.

6. Reactor-feedwater-pump 3 and 4 air circuit breakers, automatic trip. Automatic trip-out of the pumps causes the reactor water level to drop rapidly if the reactor is at high power. To prevent a low water level condition, due to automatic trip-out of the pumps, the reactor shuts down automatically.

7. Reactor water level, low. Two additional low-level Magnetrol instruments were added to the reactor water column (see Section 3.5.3.3). Depending on core riser height, one of the three low-level instruments may be put into operation. The lower limits then are: (1) water level less than 30 inches above the top of the 4-foot core, (2) less than the height of the 4-foot core riser, and (3) less than the height of the 5-foot core riser.

8. Pressure relief valve No. 1 (Foster valve), open. In report ANL-5607 this valve is referred to as the auxiliary steam bypass valve. It is set to open at about 640 psig and proportionally banded for full-open at 695 psig. The valve is a steam-operated, back-pressure-type regulator. The shutdown circuit is actuated whenever the valve unseats.

9. Primary-system-pressure safety valves, open. Four pop-safety valves now protect the pressure vessel. Two pop valves used on the original plant and two new pop valves are set to open at 700, 725, 750, and 775 psig in rising order. A fifth pop valve on the deaerator will open at 100 psig. Opening of any one of these primary-system valves will shut down the reactor. An auxiliary circuit indicates the open valve.

10. Intermediate-system-pressure safety valves, open. Two pop-safety valves protect each of the primary reboilers on the intermediate side. One valve is set for 400 psig and the other for 410 psig. Opening of any of the four will shut down the reactor. An auxiliary circuit indicates the open valves.

11. Reactor steam pressure, high. A pressure-type switch set at 630 psig initiates shutdown of the reactor if the set pressure is exceeded.

12. Containment-shell-radiation monitoring rate, high. Air ejector exhaust activity above normal will indicate a fuel element failure and will initiate shutdown. The normal level is set during operation. As power is raised in the reactor, the activity increases rapidly as transit times decrease from the reactor to the air ejector.

13. High-flux trip No. 1. A shutdown is initiated if magnetic high-flux No. 1 is tripped. The trip level is set for about 140 percent of reactor power. In case of an electronic trip, shutdown will not be initiated unless the trip is in coincidence with an electronic trip of either high-flux trip No. 2 or No. 3. The electronic trip level is set about 125 percent of reactor power.

14. High-flux trip No. 2. Same conditions as high-flux trip No. 1 above.

15. High-flux trip No. 3. Same conditions as high-flux trip No. 1 above.

16. Reactor period trip between 30 and 0.3 seconds. The reactor power can rise and fall on a period until power level is reached at which steam formation takes place. The period circuit monitors operator technique and prevents the reactor power from rising on a period that is too short and possibly reaching excessive peak values. Keyswitch 99 is used to bypass the period interlock after the reactor water has started to boil. Period information does not have any value during the boiling process, and false shutdowns may result from the erratic variations in flux.

17. Reboiler-steam-valve trip (P-11D). Closing of valve P-11D will initiate shutdown unless bypass keyswitch 103 is used. It is necessary to bypass this circuit whenever the reboiler plant is not in operation. When operating the reboilers at low powers, the bypass may be used since the condenser can accept appreciable steam load.

18. Reactor water temperature, low (less than 325°F). The plant must be preheated with an external steam source and heat exchanger to clear this interlock.

3.6.3 Turbine Plant Control

3.6.3.1 Steam Bypass

No changes were made in the steam bypass system for 100-megawatt operation of EBWR. The steam bypass system operates in the same way as in the original plant for startup of the turbine plant and warmup of the new heat-dissipating system. After the reboilers begin accepting steam, the bypass steam is reduced to zero by increasing the bypass system pressure reference point and simultaneously setting the pressure reference point of the new steam pressure controller to take over regulation. Should the pressure of the reactor rise, the bypass valve will assume load as soon as the reactor pressure reaches the pressure reference setting of the bypass system. The exchange feature between turbine and bypass was retained in the 100-megawatt system. In case of generator load loss, the bypass system would assume a comparable steam load. This helps the new steam control valve, P-11D, and its associated controller handle the transient.

3.6.3.2 Turbine Plant Feedwater

The turbine-plant feedwater control system is essentially unchanged from the original plant. Provisions for choosing reactor water level or hotwell water level as the controlling parameter were originally available behind the panel. This choice is now made from panel 12 detail 70, the control level selector in Figure 3-22. This selector simultaneously switches the reboiler-plant feedwater control (refer to Figure 3-34). The three positions of the selector are as follows:

Position No. 1

Turbine-plant feedwater control on hotwell level

Reboiler-plant feedwater control on reactor level

Position No. 2

Turbine-plant feedwater control on hotwell level

Reboiler-plant feedwater control on deaerator level

Position No. 3

Turbine-plant feedwater control on reactor level

Reboiler-plant feedwater control on deaerator level

Since the turbine plant must be in operation whenever the reactor is producing power, the turbine-plant feedwater system must also be in operation. If the reboiler plant is not operating, the reactor level is maintained by the turbine-plant feedwater control system. The operation remains as described in report ANL-5607. When the reboiler plant is in operation and handling power equal to or greater than the turbine plant, the reactor level is maintained by the reboiler-plant feedwater control system and the turbine-plant feedwater control system will maintain the hotwell level.

3.6.4 Reboiler Plant Control

3.6.4.1 Reactor Steam Pressure

Three valves were installed inside the containment vessel in the reactor steam line to the primary reboilers. Valve PS-1 is a motor-operated gate valve that is used as the reboiler-plant steam stop valve.

Control valve P-11D and bypass valve G-339 follow valve PS-1 in series. Valve G-339 bypasses steam to the reboiler for heatup purposes. Figure 3-28 shows the arrangement of the valves.

Valves P-11D and G-339 are pneumatically operated; both have a solenoid valve in the control-air line to the diaphragm motor that must be energized to operate the valve. Power for operating both solenoid valves is taken from the 125-volt d-c bus through auxiliary relay C-10 which is closed only when reactor control power is available. Upon shutdown, reactor control power is lost and both valves, P-11D and G-339, will close. When control power is available, G-339 may be opened and closed manually from the control room for heatup of the reboiler plant. No automatic control is provided for G-339. Both manual and automatic control is provided for valve P-11D. A controller on the control panel positions P-11D, through an electropneumatic transducer, to maintain a preset reactor pressure. See Figure 3-29 for a functional diagram of the reactor-steam pressure control, P-11.

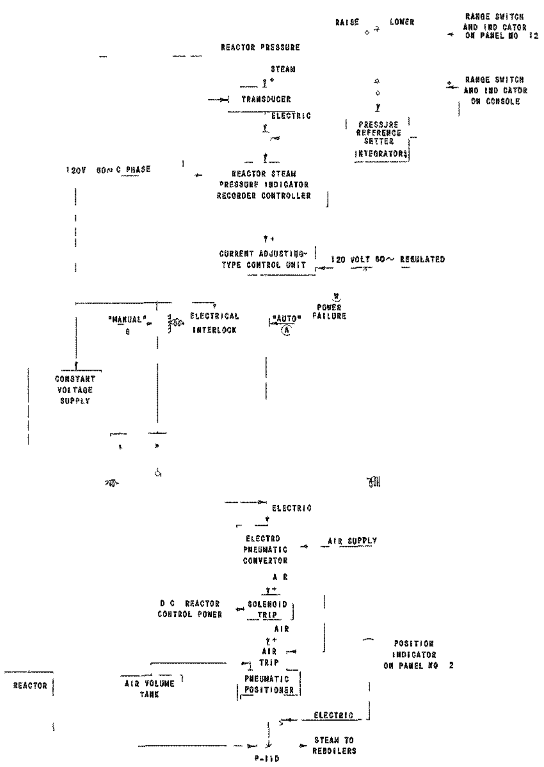

Figure 3-29

Functional diagram of reactor-steam-pressure control, P-11
RE-8-33116-C

A 250 to 650-psi transducer is installed to read the reactor pressure at the top of the water column. The output of this transducer is compared with a pressure reference signal; the difference is displayed on a ±50-psi reactor-steam-pressure minus pressure-reference recorder. The reactor pressure controller operates off a signal from the recorder.

3.6.4.2 Intermediate Steam Pressure

The intermediate-steam-pressure control valve, P-14D, throttles steam to the air-cooled condenser from the intermediate line. A 0 to 500-psig transducer senses the intermediate-steam-line pressure and transmits a signal to the recorder. The recorder develops a signal for the controller which operates the drive unit of the valve. As pressure increases, the valve opens. The controller has reset action so that the pressure will return to the set point. Figure 3-30 shows a functional diagram of the intermediate pressure control, P-14.

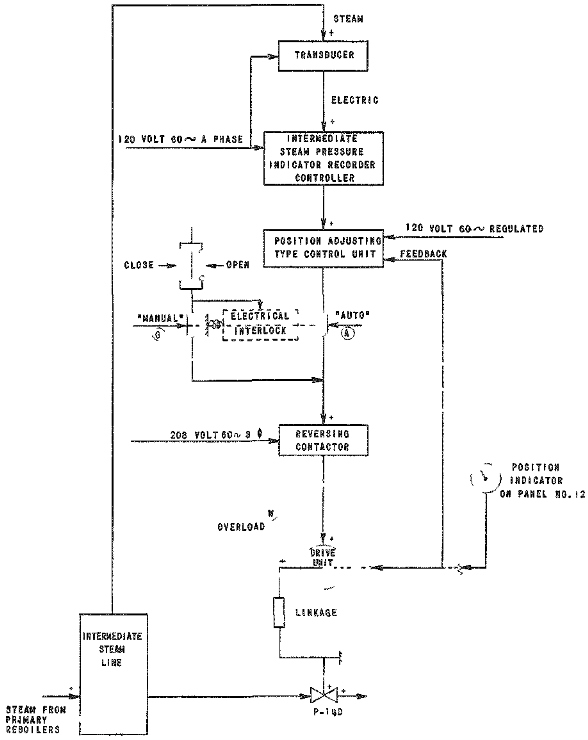

Figure 3-30

Functional diagram of intermediate-steam-pressure control, P-14 RE-8-33118-B

3.6.4.3 Secondary-Reboiler Intermediate Steam Pressure

Laboratory heating steam pressure is controlled at the boiler house at 200 psig. Because the secondary reboiler dumps steam into this line during operation, the secondary side operates at a fixed pressure. By controlling the condensing pressure on the intermediate side, the temperature difference across the tubes is set and, therefore, the heat transferred through the reboiler is fixed.

The condensing pressure is sensed by a 0 to 500-psig transducer. The signal is received by the recorder which in turn produces a signal for the controller. If the pressure tends to rise, the valve moves toward the closed direction. The controller has reset action and will maintain the desired pressure. Figure 3-31 shows a functional diagram of the secondary-reboiler intermediate-steam-pressure control, P-13.

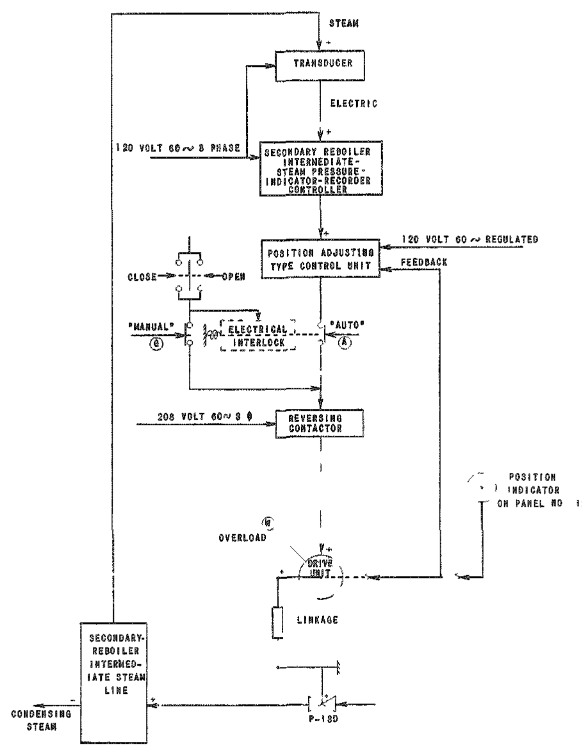

Figure 3-31

Functional diagram of secondary-reboiler intermediate-steam-pressure control, P-13 RE-33119-B

3.6.4.4 Deaerator Pressure

Deaerator pressure is maintained by the pressure regulating valve, P-12D, which discharges into the condenser. Both manual and automatic control is provided at the control panel.

A -15 to 100-psi transducer is installed to read deaerator pressure. The deaerator pressure is presented on a -15 to 100-psi recorder. The recorder has an adjustable set point and controls valve P-12D to maintain the desired pressure. This system is labeled P-12. Figure 3-32 shows a functional diagram of the deaerator pressure control, P-12.

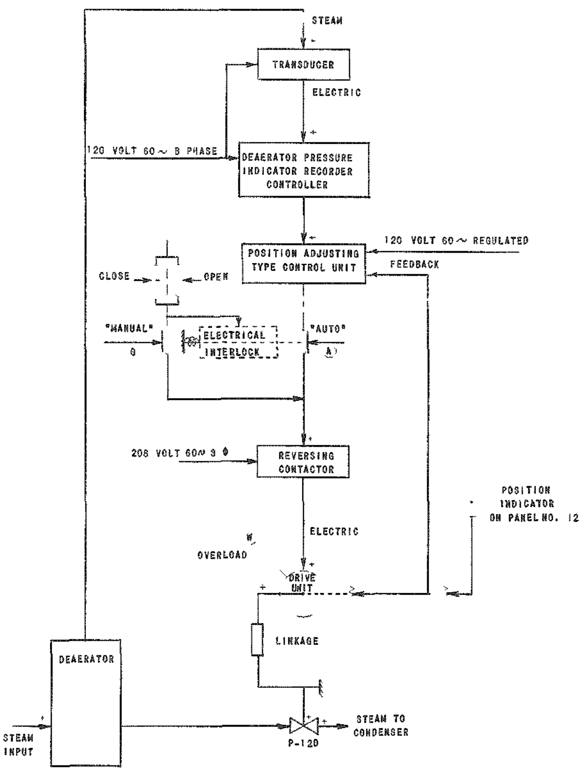

Figure 3-32

Functional diagram of deaerator-pressure control, P-12
RE-8-33115-B

3.6.4.5 Flash Tank Pressure

Flash tank pressure is controlled by controlling the amount of flash steam admitted to the flash condenser. Valve P-15D with the associated equipment comprises the system. A -15 to 100-psig transducer is connected to the flash tank. The signal is read by the recorder in the control room. The recorder develops a signal for the controller, which operates the drive unit. Increase of flash tank pressure moves the valve in the open direction to admit more steam to the air-cooled flash condenser. The controller has reset action and returns the flash tank to the proper pressure. A functional diagram of the flash-tank-pressure control, P-15, is shown in Figure 3-33.

3.6.4.6 Reboiler Plant Feedwater

The flow of reactor feedwater from the deaerator is controlled by valve P-21E. Both manual and automatic control is provided. For automatic control, the choice of proportional level control or three-way control (level, steam flow, and feedwater flow) is optional. The deaerator level or reactor level can be chosen as a control parameter for P-21E. This selector also switches hotwell level or reactor level as the control parameter for the turbine-plant-feedwater control system. The selector is so arranged that reactor level control can be applied to only one of the feedwater regulating valves at a time. Figure 3-34 is a functional diagram of the primary feedwater system.

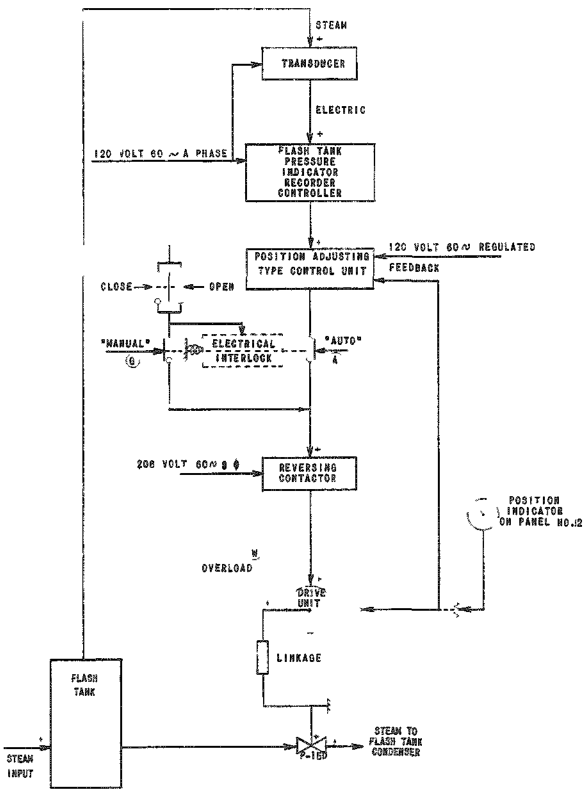

Figure 3-33

Functional diagram of flash-tank-pressure control, P-15
RE-8-33114-B

An 8-inch steam flow nozzle with a capacity of 0 to 400,000 pounds per hour is installed in the steam line before pressure regulating valve P-11D. This system is designated P-22. This measurement is one of the control parameters for the reboiler-plant-feedwater valve.

A 4-inch flow nozzle with a capacity of 0 to 400,000 pounds per hour is installed before the reboiler-plant-feedwater control valve. Feedwater-flow rate is displayed on the feedwater-flow recorder. The flow rate is one of the control parameters for the feedwater-valve, P-21E, controller. This system is designated P-23.

The deaerator operates with the water level below the trays. A 0 to 100-inch transducer is installed across the lower part of the deaerator. This level is displayed on the deaerator-level recorder which has an adjustable set point. The signal of deviation from the set point can be applied to the feedwater-valve, P-21E, controller as one of the controlling parameters. This system is designated P-21.

3.6.4.7 Intermediate Feedwater

Each primary reboiler and drain cooler combination has a level control system. The reboilers operate at a constant level above the tubes. Since the level is constant, no indication of level is brought to the control room except for high and low annunciation. The controls consist of displacer-tube force-to-pneumatic transducers with reset action. The air signal is fed to the pneumatic motors driving the valve. The system is labeled G-333 on Figure 3-28.

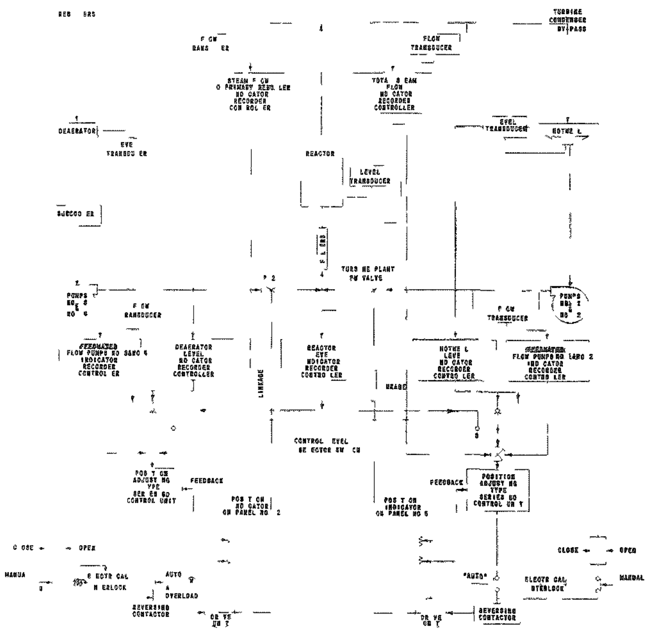

Figure 3-34

Functional diagram of primary feedwater control RE-8-33111-C

3.6.4.8 Secondary Feedwater

The secondary side of the secondary reboiler and drain cooler receives feedwater that is pumped by the Laboratory boiler house. Valve G-335 and controller, shown in Figure 3-28, admit feedwater to the secondary drain cooler and reboiler.

The controller is of the displacer-tube force-to-pneumatic type. The valve is driven by a pneumatic operator. Reset action is used in the controller to maintain small error in the level.

No level indication nor valve position indication is transmitted to the control room. Annunciator points alarm on high and low water levels. Feedwater flow indication is presented in the control room (see Figure 3-22, detail 10).

3.6.4.9 Primary Condensate Flow to Deaerator

Two identical systems control the primary condensate flow to the deaerator from the two primary reboilers. Each system is independent of the other. Since the reboilers are outside of the containment vessel and the flash valves (G-332) are inside the vessel, the control signal is transmitted electrically. The signal is developed by a displacer force-to-electric transducer on the drain tanks and routed through control room panel 12 (details 45, 46, 50, and 51 on Figure 3-22) and into the containment vessel to electropneumatic converters. The converters change the electrical signal into air pressure signals for the pneumatic operators. No level indication is presented in the control room. The valve position is indicated on panel 12, details 44 and 49. Interlocks with the radiation monitor on the tunnel and shield cubicle vent will automatically close the flash valves in case of a high radiation trip. Manual control is permissible through the use of the knobs, details 47 and 52, on panel 12.

Each flash valve is preceded in the flow pipe by a stop valve (PS-15). These valves are controlled by the control stations designated details 48 and 53 on panel 12.

3.6.4.10 Intermediate Condensate Flow to Flash Tank

Condensate from the secondary drain cooler and from the air-cooled drain cooler flows into the flash tank. Condensate from the secondary drain cooler is controlled by valve G-334 and controller. The controller senses the level of the secondary drain tank and opens and closes the valve to maintain a steam-water interface in the drain tank. Since it is important only to maintain an interface, the controller operates as a proportional controller. No level or valve position indication is presented in the control room. Secondary-reboiler intermediate condensate flow and high or low level annunciation is presented (see Figure 3-22, detail 11).

The system for the air-cooled condensate is identical to the above system except that the valve and controller is labeled G-336 and the flow indicator is detail 12 of Figure 3-22.

3.6.4.11 Reboiler-Plant Feedwater Temperature

The temperature of the reactor feedwater is controlled by valve P-42C, which controls the amount of circulating water passing through the subcooler. Both manual and automatic control is available. Figure 3-35 shows a functional diagram of the condensate-temperature control, P-42.

An iron-constantan thermocouple senses the temperature of the feedwater as it leaves the subcooler. The temperature is indicated by the condensate temperature instrument (see Figure 3-22, detail 19) and a signal is developed for the controller. As temperature rises, valve P-42C opens to allow more circulating water to pass through the subcooler.

3.7 CORE HEAT TRANSFER*

3.7.1 Internal Modifications

The performance data of EBWR has been evaluated for operation at 20, 40, and 60-megawatt (thermal) power levels. As a result of this evaluation, the modifications to the reactor vessel and internal components for operation at power levels up to 100 megawatts were made. Modification of EBWR internal components for higher power operation were directed

*M. Petrick and P. A. Lottes

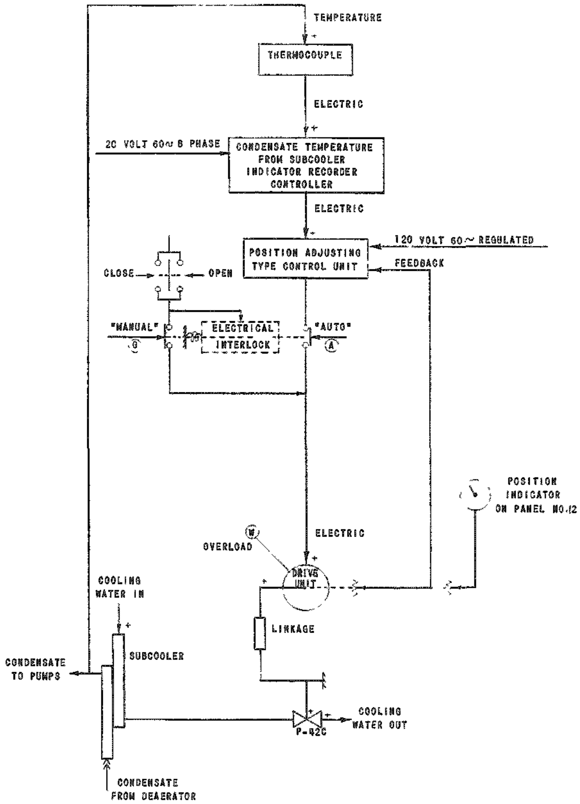

Figure 3-35

Functional diagram of condensate-temperature control, P-42
RE-8-33117-B

primarily toward increasing the recirculation flow rates within the reactor and minimizing the steam volume fraction in the core. Although the performance parameters of the reactor were altered considerably by these modifications, the basic operational characteristics of the reactor and core remain essentially unchanged.

3.7.1.1 Redesign of Riser

In the 20-megawatt-capacity reactor, the control-rod-guide structure also functioned as a core riser to improve the natural circulation of reactor water. However, the effectiveness of this riser could not be accurately defined since the benefit of the increased height in the central region, where it is needed most, tended to be nullified by the flow of steam from the shorter peripheral sections. As a result, the core shroud-riser-riser extension system described in Section 4.1.2 was installed in the reactor. The greater height of the new riser system was necessary to boost the recirculation flow rate to maintain an acceptable void level within the core. The new riser arrangement shown in Figure 2-2 consists basically of an addition to the original system. It retains the desirable features of the original riser geometry of control rod guidance and partial self-regulation over individual sectors of the core. The top necked-down portion of the riser provides additional driving head which is unaffected by local core-power fluctuations. The total height is approximately 7.5 feet. The cross sectional area is equivalent to the coolant passage area through the core proper. This height was chosen as the probable optimum because of operational considerations for maintaining the steam-water interface within a desired level range and minimizing the liquid carryover. The top portion of the riser was necked down in the expectation that the increased downcomer area and resultant reduced water velocity would aid the de-entrainment of steam in the downcomer.

3.7.1.2 New Make-up Water Injection Ring

The effectiveness of the riser system has been enhanced by the installation of a new feedwater injection ring just below the top of the new riser. Any steam entrained in the downcomer is immediately mixed with the cold feedwater, which results in quick quenching of the steam. As a result the maximum amount of the net natural-circulation driving head available is utilized. The feedwater ring was designed so that water emerging from jets would mix with the recirculating water and result in maximum distribution to provide maximum downcomer coverage. The feedwater-ring drilling configuration that was most effective was a pattern of 72 holes, $\frac{3}{8}$ inch in diameter, spaced 5 degrees apart along the circumference of the feedwater ring and placed alternately at 30 and 60-degree angles relative to the horizontal plane. The feedwater analysis was based on a reactor power of 75 megawatts. The diameter of the feedwater ring was also increased to 4-inch pipe-size to allow injection of the desired water-flow rates without excessive pressure drop.

3.7.1.3 Core Shroud Installation

The core was enclosed with a cylindrical shell and a tight closure was provided between the riser and shroud to prevent possible short circuiting of fluid from the downcomer into the peripheral riser sections and core because of static pressure differences that are inherent in such systems. Such short circuiting of fluid may explain some of the discrepancy between measured and calculated coolant-flow rates. The make-up water from the feed ring in the original EBWR geometry was injected at the junction of the riser and core. Short circuiting of colder water at this point would collapse some of the riser voids. Thus, the net driving head would decrease concurrently with the recirculation flow rates.

3.7.1.4 Redesign of Steam Discharge System

The original steam-collecting ring was removed and the 6-inch steam outlet was enlarged to accommodate a 10-inch discharge line. The larger outlet was necessary because of the increased steam flow at the higher powers; at 100 megawatts the steam-flow rate can vary between 300,000 and 360,000 pounds per hour, depending upon final feedwater temperature. The steam is collected from the top of the reactor vessel by means of a duct arrangement as shown in Figure 2-2. The duct arrangement was needed to accommodate the taller riser and higher liquid levels. The cross sectional area of the duct is 112 square inches, or greater, at any plant normal to the steam flow. The cross sectional area of the 10-inch discharge line is 74.7 square inches. The maximum steam velocities in the discharge line are approximately 144 feet per second at a power level of 100 megawatts.

As mentioned previously, the modifications were made to assist in boosting the core recirculation-flow rate. However, by adding the taller riser and raising the operating water level, the steam water separation problem is greatly aggravated. The degree of primary steam separation that will take place within the vessel is not known with any degree of certainty. The over-all gravity-vapor-liquid separation problem can be broken down into three parts:

1. Steam carryunder in the downcomer
2. Liquid carryover in the effluent steam
3. Vapor holdup in the two-phase mixture above the riser discharge.

The seriousness of the steam carryunder problem will be determined by the rapidity with which the remaining entrained steam bubbles are quenched and, of course, by the quantity of steam entrained. Although the feedwater ring was designed to induce rapid mixing of the colder feedwater and saturated recirculating core coolant, it is possible that complete, thorough mixing is not attained, and steam bubbles are not collapsed immediately. Any steam entrained over a substantial downcomer length reduces the net natural-circulation driving head and, hence, reactor performance.

The problem of water carryover by the steam is acute due to the reduction in the volume of the steam dome with the use of the taller riser which causes a sharp increase of the superficial steam velocity (based on reactor vessel diameter) to 1.89 feet per second at 100 megawatts. As the superficial velocity increases, greater dome heights are required to maintain steam purity. Some preliminary measurements on liquid carryover were made in EBWR over a power range of 5 to 30 megawatts. Extrapolation of this data to 100 megawatts shows that excessive carryover would occur with the expected available dome height. The top 1-foot section of the riser was made detachable so that, in the event of excessive liquid carryover, it may be removed if additional steam dome height is desired. The 1-foot increase in steam dome could help the primary vapor separation if the controlling variable is not the superficial steam velocity alone but also the volume of the dome.

The height of the two-phase mixture above the riser depends upon the initial hot saturated operating level and the vapor holdup. It is desirable to keep the interface height at a minimum to ensure the maximum possible steam dome. For the higher powers it is planned to start reactor operation with the saturated water level just above the top of the riser. Under such conditions the estimated height of the two-phase interface at 100 megawatts will be approximately 2.5 feet above the top of the riser. This would leave only 2.5 feet of steam dome height for primary separation.

3.7.2 Performance Characteristics

A performance analysis of EBWR was made for 100-megawatt operation. The analytical procedures used are described in report ANL-6063.[2] The axial power distribution used for the analysis is shown in Figure 3-36. The core recirculation flow rates should more than double, provided the steam carryunder is not excessive. The calculated recirculation flow rate is approximately 10^7 pounds per hour, as compared to the measured value of 4×10^6 pounds per hour of the original core operating at 60 megawatts. The radial core inlet velocity profile is shown in Figure 3-37. The average inlet velocity to the core is approximately 7.4 feet per second. The high recirculation flow rates are primarily due to the action of the upper portion of the riser on the peripheral fuel assemblies. The velocity, instead of dropping off at the core periphery, remains high even though the power density decreases. The reason is that the outer fuel assemblies have lower steam volume fractions and, hence, less frictional resistance, but still share the high driving head of the common riser. The riser will have a higher steam volume fraction than many of the outer fuel assemblies.

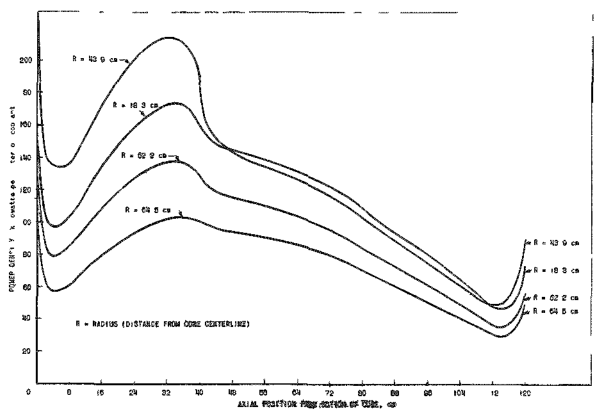

Figure 3-36

Axial power density distribution
RE-7-32963-C

The velocity dip in the spike zone (see Figure 3-37) is primarily due to the equivalent diameter of the spike fuel assembly being 0.45 inch as compared to 0.680 inch, the mean value of the thin and thick fuel assemblies.

With the use of the new riser, the problem of parallel channel operation normally associated with forced circulation systems is introduced, since in a sense the necked-down portion of the riser behaves as a pump. The problem is less acute than might initially be suspected. The reason for this is that the net driving head in the individual fuel assemblies constitutes a substantial percentage of the total driving head available. As a result the driving head for each fuel assembly is automatically varied by any occurrence,

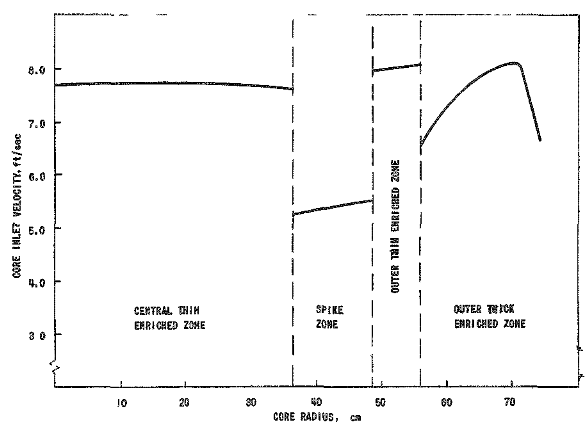

Figure 3-37

Core inlet velocity versus core radius
RE-7-32964-A

such as a local power surge or flow restriction, that may tend to increase the steam volume fraction within the assembly; that is, each fuel assembly possesses some degree of self-regulation. The percentage of the total driving head that exists within the fuel assemblies as a function of radius is shown in Figure 3-38 for 100-megawatt operation.

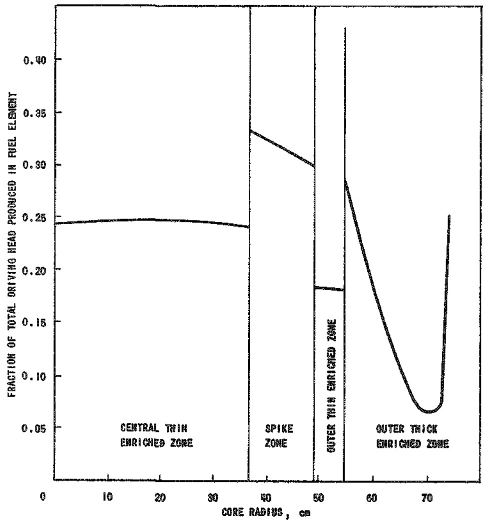

Figure 3-38

Radial driving head distribution
RE-7-32965-A

The axial and radial steam-volume-fraction variations are shown in Figures 3-39 and 3-40 respectively. The radial void distribution in Figure 3-40 results from the unique velocity profile obtained with this type of riser and from the radial power distribution. As expected, the peak voids occur in the spike zone. The axial void profiles are given for points at 7, 40, 48, and 57 centimeters on the core radius in Figure 3-40. Integrating and weighing the radial and axial void profiles yields a value of 0.197 for the mean core volume fraction at 100-megawatt operation.

Table 3-3 summarizes the probable performance parameters relative to the operation of EBWR at 100 megawatts. The tabulated void and flow data are based on calculation procedures which have been developed from Laboratory experiments. The fuel element temperatures listed in the table are based on scale-free elements. The problem of the scale deposits which have occurred on the EBWR elements and their effect on reactor operation is discussed in Section 6.2.

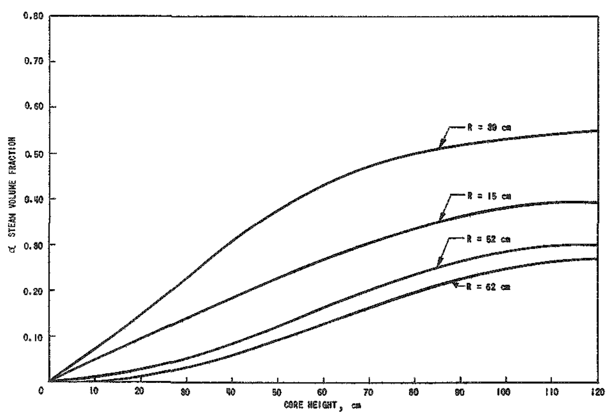

Figure 3-39

Axial void profiles at
various core radii
RE-7-32967-A

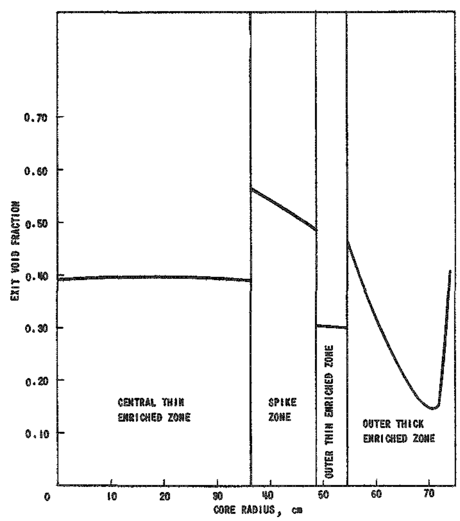

Figure 3-40

Radial exit void distribution
RE-7-32966-A

Table 3-3. Calculated Heat Transfer Data at 100-Megawatt Operation

Design power of reactor	10 mw
Operating pressure	600 psig
Operating temperature	489°F
Steam flow	300,000 lb/hr
Average steam voids in reactor core (based on coolant vol.)	19.7%
Equivalent diameter of coolant channel in spiked element	0.45 in.
Equivalent diameter of coolant channel in thin element	0.781 in.
Equivalent diameter of coolant channel in thick element	0.653 in.
Equivalent diameter of coolant channel in unspiked element	0.68 in.
Total fuel heat transfer area in core	2480 ft^2
Average power density in CTE (Central Thin Enriched) zone	113.3 kw/L
Average heat flux in CTE zone	225,353 Btu/hr ft^2
Maximum heat flux in CTE zone	346,000 Btu/hr ft^2
Percent of power removed in CTE zone	31.25
Average power density in spike zone	130 kw/L
Average heat flux in spike zone	147,160 Btu/hr ft^2
Maximum heat flux in spike zone	246,100 Btu/hr ft^2
Percent of power removed in spike zone	23.27
Average power density in OTE (Outer Thin Enriched) zone	92.8 kw/L
Average heat flux in OTE zone	185,000 Btu/hr ft^2
Maximum heat flux in OTE zone	283,605 Btu/hr ft^2
Percent of power removed in OTE zone	11.37
Average power density in KE (Thick Enriched) zone	77.1 kw/L
Average heat flux in KE zone	118,000 Btu/hr ft^2
Maximum heat flux in KE zone	172,000 Btu/hr ft^2
Percent of power removed in KE zone	34.11
Maximum fuel centerline temperatures (assuming no scale deposit)	
Thin enriched elements	665°F
Thick elements	587°F
Rod element (spike zone)	2000°F
Average fuel centerline temperature	
Thin enriched elements	612°F
Thick elements	564°F
Rod element (spike zone)	1250°F
Average recirculation rate	38.5 lb H_2O/lb steam
Average exit steam quality	2.6%
Inlet subcooling	9.81 Btu/lb or 8.5°F
Average core inlet velocity	7.4 ft/sec

3.8 CHEMICAL CONTROL*

3.8.1 Purpose

Though the concept of soluble poison control of nuclear reactors has been known and used for some time, it has been applied in boiling water power reactor design only as an emergency-shutdown control mechanism. The purpose of the emergency high-pressure and low-pressure boric acid injection systems is only to provide a method of shutting down or holding down the reactor when other methods have failed. An additional application of boric acid in EBWR 100-megawatt design is for operational control of the reactor. Its primary purpose is to hold down the highly reactive fuel loading during shutdown periods, but soluble-poison shim control of the reactor will be fully evaluated during low and medium power operation of the reactor. Design advantages of soluble poison shim control are based on the following(3,4):

1. Excess reactivity required to compensate for fuel burnup is readily controlled with soluble poisons.

2. Heat transfer considerations limit the maximum power level of a reactor. Distortions of the neutron flux near control rods often cause hot spots which limit the power from a given core configuration. A soluble poison shim uniformly depresses the flux and alleviates the hot spot problem.

3. Since the reactor neutron flux is uniformly regulated (see 2 above), fuel is more economically used because uneven burnup is minimized.

4. The presence of a soluble poison in the operating reactor should enhance reactor stability by reducing the magnitude of the negative void coefficient in the boiling water reactor.

With the information and experience gained from EBWR operated with soluble poison control, a more advanced core design may be possible with these additional advantages(3):

5. More compact and uniform fuel lattices can be designed if soluble poison replaces space-consuming control rods.

6. Less complex and probably cheaper controls can be used.

The particular advantages of boric acid as the soluble poison are as follows: (1) ready solubility in the range of interest, (2) sufficiently high thermal neutron cross section (4020 barns for B^{10} which is 18.7 percent abundant in natural boron), (3) ready availability in high purity at a

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moderate cost, (4) stability at the radiation flux and temperatures encountered in the operating reactor core (does not decompose, plate out, or form scale), (5) relative inertness (noncorrosive and has no effect on water decomposition rates), (6) chemically anionic (simplifies chemical removal without interference with usual reactor-water purification methods), and (7) absence of gamma-emitting activation product daughters of consequence in shielding considerations.

The use of boric acid as the soluble poison in EBWR is primarily for control of excess reactivity, particularly in the cold reactor. To attain the proposed high power levels it is necessary to provide a more reactive fuel loading than previously required. Additional reactivity must be available to compensate for that lost through increased voids and xenon concentrations at higher power levels. Thus, to balance the additional reactivity in the cold system and provide a satisfactory safety margin beyond the control possible with all nine control rods fully inserted in the core, a poison is added to the reactor water.

The original design of EBWR specified shutdown of the cold configuration by insertion of seven control rods; i.e., the system would be subcritical with any seven of the nine rods inserted.[5] This will no longer be the case with the modified EBWR. The 100-megawatt core loading will be subcritical in the cold condition without soluble poison only when all nine control rods are fully inserted. Additional control with soluble poison is provided to insure that the system can be held subcritical in case of maloperation of one of the control rods, during rod-drop tests, and during minor maintenance operations.

Not only will the soluble poison be necessary to guarantee an eight-rod safe hold down criterion for the cold reactor, it will also be necessary during preheating of the reactor to 325°F, and even in the hot (489°F) reactor when neither voids nor xenon are present. Table 3-4 shows the effect of various reactor parameters on the effective reactivity and the probable boric acid management for each case. It will be noted that the reactor could become critical, in the absence of boric acid, due to the withdrawal of only one control rod throughout the temperature range. The safety requirement of nine-rod shutdown is never violated, however.

Removal and addition of boric acid in the reactor system are both relatively slow processes. Removal follows an exponential decay curve dictated by the volume of the system and the flow rate through the purification system. About 30 hours are required to reduce the boric acid concentration from 5 grams (0.011 pound) per gallon to less than 0.1 gram (0.0002 pound) per gallon under normal conditions. The addition of boric acid is limited by the total pumping rate of the addition pumps so that a minimum of 39 minutes of pumping will be necessary to bring the boric acid concentration from 0 to 5 grams (0 to 0.011 pound) per gallon. These slow rates are compatible with the reactor start-up, operation, and shutdown practice in all except emergency situations in which the high-pressure boric acid injection system must be activated. Start-ups are, of necessity, slow. The reactor cannot be heated at faster than a prescribed rate. The slow increase in negative reactivity due to rise in temperature makes the gradual removal of boric acid possible during startup. The presence of 2 or 3 grams (0.004 or 0.0066 pounds) of boric acid per gallon of reactor water should not hinder the initial attainment of full power, and this amount can be removed at a faster rate than xenon fission product poison can build up to equilibrium. Use of shim control at less than full reactor power is readily possible, and traces of boric acid (~ 0.1 gram per gallon) will not hamper full power operation even with equilibrium xenon.

Table 3-4. Reactivity Conditions and Boric Acid Management for EBWR Core IA

Power	Temperature	Control rods	Boric acid conc., g/gal	Effective reactivity, ρ	Suggested boric acid management
Zero	Cold	9 inserted	0	0.995	Boric acid required for cold shutdown.
Zero	Cold	8 inserted	0	1.015 - 1.045[a]	
Zero	Cold	9 inserted	5	0.925[b]	Boric acid shutdown concentration,
Zero	Cold	8 inserted	5	0.945 - 0.975	5 g/gal
Zero	Preheat to 325°F	9 inserted	0	0.98[c]	Boric acid still required at preheat
Zero	Preheat to 325°F	8 inserted	0	1.00 - 1.03	conditions
Zero	Preheat to 325°F	8 inserted	5	0.93 - 0.96	At least 2 to 5 g/gal can be safely removed
Zero	Preheat to 325°F	8 inserted	3	0.96 - 0.99	at preheat conditions - more likely re-
Zero	Hot (489°F), no voids	9 inserted	0	0.96	moval will occur while going to operating
Zero	Hot (489°F), no voids	8 inserted	0	0.98 - 1.01	conditions.
Initial full, No xenon	Hot, voids	9 out	0	1.05[d]	The presence of some boric acid will not hinder attainment of initial full power.
Equil. full, Equil. xenon	Hot, voids	9 out	0	1.01	The poison can be removed at a faster rate than xenon builds in.

[a]Based on experimental determinations with fuel follower control rods. Individual rod worths vary from 0.02 to 0.05. These may be up to 0.01 higher with Zircaloy-2 follower rods.

[b]Experimental - 1 g/gal = k_{eff} of -0.014.

[c]Experimental - temperature coefficient from cold to hot, 9 inserted or 8 inserted, ~-0.035.

[d]Temperature coefficient decreases from ~-0.035 at 9 inserted to ~ 0.005 at 9 out.

In the event of a shutdown, the slow addition of boric acid will suffice to maintain a completely safe condition if: (1) the reactor has operated long enough to build up short-lived fission products, or (2) in the absence of fission products, no emergency occurs that would dictate sudden and complete shutdown of the reactor. In the event of a shutdown after power operation, the decay heat of the short-lived fission products will continue to generate steam for some time. The cooling of the reactor by blowing steam to the condenser will also cause steam formation in the core, so that voids in the core will contribute to lowering the effective reactivity of the core for some time, dependent on various reactor operating and shutdown parameters. Xenon will also lower the effective reactivity of the core so that the reactor can be at least temporarily held down with less than nine rods and possibly with only six or seven. As each of these effects - voids and xenon - decrease in their contribution to the control of the reactor, sufficient time will elapse for addition of boric acid to a suitable concentration to maintain an eight-rod safe shutdown condition.

Several situations of an emergency nature, such as loss of two or three control rods during operation, can probably be adequately handled in a controlled manner by boric acid additions with the addition pumps rather than by the rather extreme method of high-pressure injection of a large excess of boric acid poison.

During the evaluation of soluble-poison shim control of the reactor, the boric acid concentration will be maintained at a sufficiently high level so that only one control rod need be used. This rod will provide fine adjustment of the reactor power, since changing the gross inventory of the boric acid in the reactor as a fine-control mechanism is not practical. Changing of the reactor water level and, consequently, the effective boric acid inventory in the reactor core by manipulation of the feedwater control valve will be evaluated as a fine-control technique. Even this method would have a slow response compared to a control rod. During shim control studies all rods except the one used for control will be at some bank position, probably above the core. They will be inserted in the normal manner for reducing the power or scramming the reactor. Depending on reactor operating parameters, shutdown and holddown may be possible with less than eight rods. The presence of boric acid at all times in the reactor would certainly enhance the control of the reactor with Core IA.

3.8.2 Chemical Control Systems

Three systems are required for chemical control of EBWR 100-megawatt system: (1) boric acid preparation and addition system, (2) soluble poison monitoring system with associated alarms, and (3) boric acid removal system which includes provisions for reactor water purification during operation.

3.8.2.1 Boric Acid Addition System

Boric acid additions may be made during normal operation to provide additional shim control, during shutdown to control excess reactivity, during make-up water additions to prevent dilution of the boric acid concentration in the core, and in an emergency in which relatively slow additions of boric acid will suffice to maintain a suitable margin of shutdown control. Figure 3-41 shows a block diagram of the system.

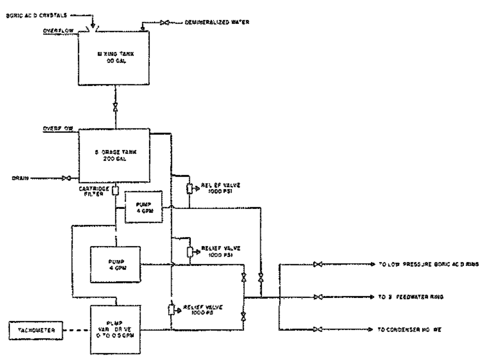

Figure 3-41

Boric acid addition system of 100-mw EBWR
RE-8-32664-B

The addition system includes a 100-gallon mixing tank, a 200-gallon storage tank, and three positive-displacement addition pumps, all located in the reactor building. The mixing tank, located on the steam drier platform, is provided with a 3000-watt Calrod heater and a variable-speed electric stirrer. Water from the plant demineralized-water system is used. Forty pounds of C. P. boric acid are added to about 75 gallons of water in the tank and diluted to 100 gallons. The mixture is heated and stirred until the crystals are dissolved. The solution is stored in the mixing tank and in the 200-gallon storage tank located on the east instrument platform. From 200 to 300 gallons of stock solution are kept on hand at all times. Concentration of the stock solution is 182 grams (0.4 pound) per gallon.

The positive-displacement pumps pressurize the stock solution for injection into the desired addition points. Three such points are available: (1) to the low-pressure boric acid ring, (2) to the 3-inch feedwater ring, and (3) to the condenser hotwell. When not in use, the valve to the feedwater ring is locked in the open position so that remote operation of the addition equipment from the control room is possible, particularly during an emergency. Normally, additions are made through the feedwater ring except when make-up demineralized cold water is being added through the low-pressure boric acid ring. All water added to the reactor must contain a boric acid concentration equal to or greater than that in the reactor water to prevent a hazardous condition from developing. Such a hazardous condition might occur if the boric acid concentration in the reactor core were reduced, by dilution or other means, to a level at which the reactor could become critical without the knowledge of the reactor operator. A continuous monitor instrument, described later, is provided so that this condition can be readily detected and corrected.

The addition pumps are Union Steam Pump Company units capable of pumping pressures in excess of 1000 psi. The pumps are identical except for their motor drives. Each pump has three cylinders and each cylinder has a 0.42-cubic-inch displacement. Two of the pumps are driven at fixed speeds so that each delivers 1.41 gallons of stock solution per minute. The other pump is driven by a variable-speed motor drive capable of delivering from zero to 0.5 gallon per minute. A tachometer, calibrated for actual pump delivery, indicates the pumping rate of the variable-speed unit. Relief valves on bypass lines protect the pumps in the event that none of the valves to the addition points are open when the pumps are started. The relief valves, set for 1200 psi, will automatically reset after relieving excess pressure.

The fixed-rate pumps are operated by switches located in the control room on the chemical control section of panel 7. The variable-speed pump is operated locally.

3.8.2.2 Monitoring Systems

Two monitors are used to determine the soluble poison concentration in the reactor water as natural boric acid.

A manual monitor is located on the pump floor of the reactor building. This instrument is similar to instruments described in the literature[6,7] and incorporates a 16-gram Pu-Be neutron source (about 2×10^6 neutrons per second) and a slow-neutron detector with associated amplifiers, high-voltage supply, and scaler. The sample-cell well is surrounded by a water moderator-reflector-shield. Constant-temperature water is circulated through this shield, where the source is located tangent to the sample well. The detector is located in a tube centered in the sample well. Neutrons from the source are moderated in the water of the shield and the sample. In the absence of a soluble poison in the sample-well water, a reproducible number of neutrons are detected, depending on the fixed physical geometry of the sample cell. The presence of a soluble poison in the water in the sample well will cause the thermal neutron flux to be attenuated. The decrease in neutrons detected has been found to be an exponential function of the concentration of the soluble poison, in the range of 0 to 12 grams (0 to 0.026 pound) per gallon as natural boric acid.

The manual soluble-poison monitor is based on conventional proportional-counting techniques and is provided with an accurate preset timer. When background checks are made with demineralized water immediately before a test run, the general accuracy is ± 0.05 grams up to 10 grams (0.00011 to 0.022 pound) of boric acid per gallon.

The second monitoring device, a continuous monitor, includes a sample cell located in the reactor building basement as a component of the water purification system (see Figure 3-2). The cell is designed for operation at reactor pressure and at water temperatures of 80° to 120°F normally flowing through the purification system. The method is described in the literature.[8] A 110-gram Pu-Be neutron source produces about 1.25×10^7 neutrons per second. The detector is a slow-neutron ion chamber. The associated electronic components measure the current induced by the neutron flux. This current is measured continuously with a highly stable vibrating-reed electrometer. A highly stable, transistorized, low-voltage power supply is used to offset the electrometer zero so that the electrometer output zero voltage corresponds to a soluble-poison concentration of zero in the reactor water flowing through the cell.

The voltage output of the continuous monitor is indicated in several ways. A circular chart recorder located in the basement near the sample cell gives a continuous recording of this information. The recorder reads directly in 0 to 10 grams (0 to 0.022 pound) of soluble poison as natural boric acid per gallon of water. A meter gives continuous indication in the control room. This meter, located on panel 7, is calibrated directly from 0 to 10 grams per gallon to correspond to the recorder.

The continuous monitor is also connected to an alarm which can be preset to a desired concentration. A variation in actual concentration from the preset value of $\frac{1}{6}$ to $\frac{1}{2}$ gram (0.00035 to 0.0011 pound) per gallon will flash the respective high or low warning light, while variations of over $\frac{1}{2}$ gram per gallon will sound the annunciator (Panalarm No. 44). When additions of boric acid are necessary to maintain a desired concentration level, the addition pumps are manually activated for a preset period of time computed on an inventory basis. If the concentration is high, the excess can be removed by the boric acid removal beds of the purification system until the desired concentration is attained.

3.8.2.3 Boric Acid Removal System

The purification system originally designed and installed in EBWR was modified for EBWR 100-megawatt operation to provide for boric acid removal. Figure 3-2 shows the modified system. The pumps, coolers, and filters are connected so that either system No. 1 or No. 2, shown in the figure, may be used to circulate water through the continuous monitor and the ion exchange units that may be in service. The two large units are each filled with 9 cubic feet of super-regenerated strong anion resin. Provision is made for bypassing both or either of these units, or operating them in series with either unit upstream of the other. With this arrangement it is possible (1) to use the full capacity of the resin by expending one unit with a second relatively fresh unit being used to scavenge any leak-through boric acid, and (2) to replace the resin charge of either unit while operating the other unit. Boric acid removal from the reactor system is very slow because the maximum circulation rate through this system amounts to only about 7 gallons per minute. The removal rate of boric acid from the reactor system decreases exponentially with concentration of boric acid remaining in the reactor water. During removal, the small 1.2-cubic-foot unit filled with standard mixed resins is on the effluent side of the anion units and is used to remove any corrosion products from the water. During operation of the reactor with soluble-poison control, the reactor water is circulated only through the 1.2-cubic-foot unit filled with boric acid-saturated mixed resin. This unit will remove corrosion or fission product ions without affecting the boric acid concentration. In the absence of boric acid in the reactor water, the small unit filled only with standard mixed resin is used for purification.

Whether the ion exchange units are in use or not, a flow of reactor water through the purification system is maintained to provide sample water to the continuous monitor sample cell. A low flow through the system is detected by the high-pressure flow detector on the effluent side of the continuous monitor cell. An annunciator alarm (Panalarm No. 42) sounds when flow is less than 3 gallons per minute.

3.8.3 Operation with Chemical Control

Chemical control of EBWR will be initiated with startup of the 100-megawatt reactor. Because of the complexity of the operating parameters involved, and since a preliminary study[9] has shown the feasibility of using boric acid in EBWR, no preliminary mockup experiment was attempted. The operation and performance of the equipment, materials, and processes have been evaluated during preliminary operations of EBWR prior to loading for 100-megawatt operation.

A study of simulated human errors and mechanical failures received attention during the preliminary testing period. The design philosophy assumes that only the inadvertent loss of soluble poison can cause an unsafe reactor condition. This loss can only occur very slowly during reactor operation. Three sources of error or failure conditions that could lead to this loss are:

1. Misvalving of anion exchange units so that poison removal occurs when a fixed concentration level is desired. The 8.5-gallon-per-minute flow through these units makes removal a very slow process that is easily detected by the continuous monitor, the manual monitor on bi-hourly checks, the control rod positions, and by conductivity cells on the effluent side of the anion units and the purification system.

2. Dilution with demineralized water without addition of boric acid during make-up operations. The addition of demineralized water to the operating reactor through the condenser hotwell can proceed at up to 10 to 12 gallons per minute due to the capacity of the water demineralizer units. Dilution would occur very slowly and would be detected by the methods listed above and by increased water inventory in the reactor or the hotwell.

3. Rupture of a condenser tube during reactor power operation would dilute the reactor water with cooling water. This condition would cause the most sudden change in boric acid concentration. However, the rate of dilution would be limited by the size of the rupture (each tube has a flow of about 10 gpm) and, in the case of multiple ruptures, by the capacity of the feedwater pumps. These pumps normally handle 120 gallons per minute from the condenser hotwell but can handle up to 180 gallons per minute on demand. In the case of a rupture, the dilution would be detected by increased water inventory (particularly in the hotwell), increased conductivity both at the hotwell and in the reactor water from the high chemical content of the cooling water, and by the normal removal-dilution indication explained above. At a dilution rate of 60 gallons per minute, the rate of change of the effective negative reactivity of the reactor water containing soluble poison is so small that detection and correction can be readily accomplished.

The probability of inadvertent dilution may be slightly greater when the reactor is not operating, and particularly when the head is off. In addition to the errors listed above, the possibility exists that (1) Laboratory or demineralized water could be run or dumped into the reactor vessel, or (2) water from the overhead storage tank could be run into the reactor vessel rather rapidly. Since the cold EBWR 100-megawatt system will be safe with nine control rods inserted, dilution or removal of boric acid would simply remove the desired safety margin. If a control rod were removed purposely or inadvertently without sufficient soluble poison being present, the reactor could go critical. It is necessary to continue to operate the monitors and to pay particular attention to changes in water inventory and system valving during reactor shutdown periods. Laboratory or demineralized water is never run into the open reactor through hose or other unmonitored piping. Any indication of water flowing to the reactor vessel from the overhead storage tank or other source is sufficient cause to initiate boric acid addition while correcting the condition.

3.9 PLANT SAFETY*

3.9.1 Shutdown Cooling

Immediately after reactor shutdown the core decay heat is approximately 6 percent of the full operating power prior to shutdown. Hence, after 100-megawatt operation the decay heat at shutdown amounts to 6 megawatts. The decay heat from the core decreases with time to about 3 megawatts after 5 minutes and to 1 megawatt after 7 hours. Figure 3-42 is a plot of decay heat versus time for 100-megawatt operation assuming that the reactor was at power for an infinite time.

Normally, decay heat is removed by boiling for the first three hours, and the steam generated is discharged to the condenser until the reactor water reaches about 263°F. This rate of cooling of the reactor amounts to 75°F per hour. Thereafter, the reactor water is circulated through the secondary cooler of the reactor-water purification system and returned to the reactor at 120°F. During normal shutdown the condenser circulating pumps provide cooling water for the condensers. Reactor steam pressure and flow are regulated by the steam bypass valve which is adjusted by the pressure reference (P_0) setting so that the resulting pressure drop is consistent with a 75°F per hour temperature drop in the reactor water.

If sufficient cooling is unavailable in the secondary cooler of the reactor-water purification system, the secondary cooler in the standby reactor-water purification system can also be utilized. Should the steam bypass valve become inoperative, the 4-inch by 8-inch steam-powered

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pressure regulator will automatically admit steam to the condenser and limit reactor pressure to 640 psig until the decay heat decreases sufficiently to allow the reactor pressure to drop. In the event of a power failure, the normal shutdown system cannot be used and the emergency cooler is utilized to dissipate decay heat. Operation of the emergency cooler can be initiated from the control room. Loss of instrument air or 125-volt d-c power automatically turns on the emergency cooler.

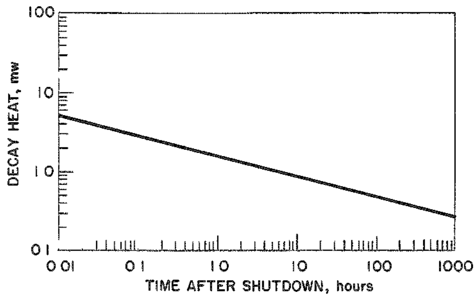

Figure 3-42

Energy generation versus time after shutdown from 100-mw operation

3.9.2 Reactor Vessel Rupture

In the event of a major vessel-nozzle rupture, a limited amount of cooling is available to the core, provided complete loss of water does not occur and that the break is above the core. Introduction of the cooling water from the 15,000-gallon overhead tank through the spray ring is either automatically or manually initiated. When both reactor level and pressure fall below preset limits the system is automatically actuated. In practice, the new riser will interrupt the spray and prevent efficient cooling of the core. Therefore, whatever cooling is obtained from the spray ring will be accomplished by the addition of water to the downcomer area.

Should a major vessel-nozzle rupture occur at the bottom of the vessel, all the water will be lost from the reactor vessel. Under these conditions no cooling is available, and because of the new riser, the spray ring will have limited usefulness.

A major rupture of the reactor vessel resulting in a loss of water and exposure of the core may result in melting of the fuel elements. The high temperatures reached in the fuel plates can cause the zirconium cladding to oxidize, exposing the hot uranium fuel, and fission products can be released to the atmosphere. Therefore, for a major rupture measures can be incorporated in the plant other than confining the fission gases and products to the vapor containment vessel.

A minor rupture of the vessel cannot result in loss of all water from the vessel since provisions have been incorporated to replace the leaking water. If the rupture occurs above the top of the core, flashing caused by pressure drop will provide cooling and prevent overheating of the fuel elements. When the reactor pressure and level drop to a preset point, water from the 15,000-gallon overhead tank will be injected automatically. The core can also be flooded manually by water from the Laboratory water system. If the rupture occurs below the core, the operator can reduce the pressure in the reactor by opening either the steam bypass valve or the reactor-pressure control valve, P-11D, and dump the steam to the condenser or reboilers. As in the previous case, low reactor water level and pressure will automatically cause injection of water from the overhead tank.

3.9.3 Power Failure

Following loss of power to the magnetic clutch, the control rods drop to a position of maximum insertion in less than 1.5 seconds (fail-safe operation). A 25 percent increase in length of the new rods contributes to a longer insertion time than for the original drives. If the control rods do not drop sufficiently to bring the reactor subcritical, the operator may inject the high-pressure boric acid solution. Injection is initiated from the control board or manually through a release level located in the control room on the containment shell wall.

Flow of water to the emergency cooler is automatic on loss of the 125-volt d-c power or instrument air. Water from the overhead storage tank can be admitted to the emergency cooler upon signal from the control room.

In case of loss of power from the utility lines, the reactor is automatically shut down and the turbine-generator is tripped off. The reactor-steam-pressure control valve, P-11D, to the primary reboilers closes automatically upon a loss of reactor control power. The steam bypass valve bypasses steam to the main condenser and prevents reactor pressure from rising above approximately 610 psig. Without electric power, continued operation of the turbine-bypass system is assured through the use of an oil accumulator tank. Since the circulating water pumps are without power, the flow of cooling water to the condenser is obtained by natural convection. The natural circulation is obtained by the automatic opening of a bypass valve which allows water to flow directly to the storage basin of the cooling tower. Unless this bypassing is done, the height of the cooling tower risers would prevent circulation.

The ventilation openings in the containment building are closed automatically on power failure. All primary system lines penetrating the containment shell are isolated automatically on power failure.

3.9.4 Pump Failure

The most probable reason for pump failure is the loss of electric power. Safety measures to prevent failure of the circulating-water pumps, turbine-generator-system feedwater pumps, cooling-water pumps, shield-cooling pumps, and reactor-purification pumps are adequately described in report ANL-5607.[1] None of the systems mentioned above have been changed, and operation still remains as described in ANL-5607.

If the primary feedwater pumps in the reboiler system fail because of power interruption, an electrical interlock will automatically shut down the reactor. Reactor shutdown causes automatic closure of the reactor-pressure control valve, P-11D, and admission of steam to the primary reboiler is interrupted. Failure of the reactor feedwater pumps to supply feedwater even though electric power is available will result in a decrease in reactor water level; low water level will initiate reactor shutdown. Should the deaerator water level drop to a preset low level, a low-level pump-cutoff switch will automatically interrupt the power to the primary feedwater pumps; this action will result in automatic reactor shutdown.

A failure of an intermediate pump or decrease in flow of intermediate feedwater will cause the pressure in the reactor to rise since low level will decrease the heat removal capacity of the reboilers. As a result, the steam bypass valve will begin to open between 602 and 610 psig to maintain pressure in the reactor. The operator will have to start the standby intermediate feedwater pump to resume normal operation. If the standby pump fails to start or flow is not resumed, one of the following actions will occur:

1. Reactor overpressure interlock will initiate reactor shutdown at 630 psig.

2. Primary-reboiler-shell low-level alarm will warn the operator that water is being boiled away with little or no make-up.

3.9.5 Control Valve Failures

The primary steam system is designed without any quick-action closure valves with the exception of the turbine trip-throttle valve and steam-pressure control valve P-11D. For 20-megawatt operation without the reboilers, experience indicates that one of the following three events will occur because of sudden closure of the turbine trip-throttle valve:

1. The reactor will be shut down by the turbine trip interlock.

2. If the turbine trip interlock is bypassed with a key switch, the bypass valve will open fully. The reactor will continue operating without sensing the interchange if the pressure does not increase beyond 600 psig.

3. Should the bypass valve fail to make the interchange in case 2 above, control rods will be inserted by (1) high-flux and overpressure interlocks or (2) overpressure interlock. EBWR has been shut down by the high-flux interlock when the power rate of increase was approximately 1 megawatt second. This corresponded to a rate of increase of pressure of about 3 psi per second.

During reactor operation at less than 100 megawatts in which the reboilers are accepting steam and the turbine trip interlock is bypassed, sudden closure of the turbine trip valve will initiate opening of pressure control valve P-11D to maintain reactor pressure at 600 psig. The additional steam will be directed to the primary reboilers. Since the secondary reboiler is normally base loaded, the added heat load will be dissipated in the air-cooled heat exchanger, this being accomplished by the intermediate-steam-pressure control valve, P-14D. Once the design heat removal capacity of the primary reboilers is reached, reactor pressure will rise and the steam bypass valve will open and dump steam to the condenser.

Should the primary reboilers be operating at peak capacity (100-Mw reactor power) at the time of the incident, reactor pressure will rise immediately. Since the primary reboilers cannot handle the entire load of 100 megawatts at a reactor pressure of 600 psig, the steam bypass valve will open and maintain reactor pressure slightly above 600 psig by dumping steam to the condenser. In practice, the transfer of 100 megawatts of heat to the intermediate system requires a primary condensing pressure of approximately 635 psig. Therefore, the steam bypass will eventually take up the majority of the load, a small part of the load goes to the reboilers because of a slightly higher condensing pressure.

The steam bypass valve is normally closed during operation of the new reboiler plant since it is set to open slightly above reactor pressure. Therefore, the steam bypass valve will not compensate for the failure of the turbine admission valve mentioned above until the reactor pressure rises sufficiently to open the valve.

A failure resulting in full closure of the steam bypass valve or failure to open when required will not affect the operation of the turbine admission valve which will continue to function in accordance with demand. However, any changes in reactor pressure occurring thereafter must be fully compensated by pressure control valve P-11D, otherwise the pressure will continue to rise. If reactor pressure exceeds 630 psig because of a subsequent incident, the control rods will automatically insert and the reactor is shut down.

If the primary reboilers are in use and pressure control valve P-11D closes suddenly, the reactor will be automatically shut down by the valve closure interlock. If the valve closure interlock fails, the reactor will be shut down by the high-flux interlock. If this trip fails, the reactor will be shut down by the overpressure interlock.

Failure of the air supply to the reactor-pressure control valve, P-11D, results in closure of the valve which, in turn, automatically initiates reactor shutdown. As an added precaution, overpressure (630 psig) or high-flux interlocks will automatically insert the control rods.

3.9.6 Control Rod Failure

All nine rods are required to keep a cold, clean reactor subcritical. An eight-rod holddown of a cold, clean reactor requires the addition of boric acid. To assure an eight-rod holddown during rod withdrawal tests, it is necessary to introduce boric acid as described in Section 3.8.

Should a rod fail to insert after a hot shutdown, a normal shutdown procedure can be followed if time permits, this being dependent upon reactor temperature, xenon poison build-up, and available void coefficient. During cooling, boric acid can be introduced from the chemical control system. If conditions do not permit a normal shutdown procedure, boric acid can be injected through the high-pressure boric acid system. The latter method will be used only as an emergency control measure.

In the event a control rod does not insert by force of gravity alone the overriding clutch feature of the drive mechanism may be used to help drive the rod into the fully inserted position.

3.9.7 Excessive Pressure

If the pressure exceeds 630 psig, the reactor is shut down automatically by an overpressure shutdown interlock. If the pressure increases to 640 psig, the 4-inch by 8-inch steam-powered back-pressure regulator opens and dumps steam to the condenser. Further increases in pressure will trip the safety valves, which are set to open at 700, 725, 750, and 775 psig, and also cause steam to be dumped into the condenser. Opening of any of the valves will automatically shut down the reactor since the control rods are electrically interlocked with the valves. If the condenser pressure exceeds 20 psig, which may result if all steam valves open, two rupture disks in the condenser are sized to burst and discharge steam into the vapor-containment building. Consequent radioactivity release will automatically initiate closing of the ventilating valves in the building.

3.9.8 Reactor-Water Level and Temperature

Control rods cannot be withdrawn if the water level in the reactor is too low or too high. The rods will be inserted automatically if the water level exceeds preset limits during operation. Also, the control rods cannot be withdrawn if the reactor water temperature is below 325°F; hence, excess reactivity cannot be added to the cold reactor.

3.9.9 Steam and Water Leaks from the External System

The dominant radioactivity in the external steam and condensate circuits is caused by the fast neutron irradiation of oxygen-16 to yield nitrogen-16. Although the reactor water contains several radioactive corrosion products, the major one being sodium-24, no problems or hazards are created by their presence because of their low carryover in the steam. Nitrogen-16 is not a problem because of its short half-life (7.4 seconds). This short period of time prevents accumulation of this isotope in the vicinity of either a steam leak or a water leak. Further, like sodium-24, there is very little carryover of nitrogen-16 in the reactor. Even if all of the water in the external system should flash to atmospheric pressure, the integrated exposure from nitrogen-16 would not pose a serious hazard to operating personnel.

Development of leaks in condenser tubes will permit leakage of circulating water into the primary fluid in the condenser. An increase in electrical conductivity of condensate or of reactor water will be the first indication of major leakage in the condenser. Also, water-level indicators would eventually show an increase in primary water inventory. With evidence of a leak, the operator can isolate the half of the condenser involved by closing butterfly valves in the inlet and outlet circulating-water lines. The water boxes and tubes on the leaking side can then be drained to a tank in the basement of the power plant area. The remaining half of the condenser can still condense the turbine exhaust and/or bypassed steam, although, at a slightly higher back pressure. The operator can open the desuperheater spray valve to admit spray water into the bypass desuperheater to prevent damage to the condenser because of high thermal gradients.

Condenser tubes have failed by erosion in the original condenser because of steam deflecting off condenser bracing. The above sequence was followed for locating and plugging the faulty tubes; thereafter, circulating water was restored to both halves of the condenser. The entire operation took place without reducing reactor power from the 20-megawatt level.

If a leak or tube break occurs in the primary reboiler or drain cooler, radioactive fluid will enter the 350-psig intermediate system. This system will be monitored constantly for radiation, and if radioactivity is detected, a leak or tube failure would be suspected. Further verification of a leak can be made by observing whether there are changes in liquid inventory in both the primary and intermediate systems. By isolating the affected unit, operation can be continued in the parallel unit under reduced power.

The same analysis applies to the primary subcooler, except that the cooling tower water leaving the unit is monitored.

3.9.10 Startup Accident

The existing antimony-beryllium neutron source, previously irradiated to saturation and maintained at full strength by operation of the EBWR, will be used in the new core.

A period meter, set to release control rods at periods of less than 5 seconds, will be used in all reactor startups. The use of a period trip meter at full power is not desirable because of the short period variations of the neutron flux.

Control rods cannot be withdrawn unless the flux-indicating galvanometer is on its most sensitive scale and the water temperature is 325°F or above. The rate of addition of reactivity by rod withdrawal is limited to approximately 0.01 percent per second. With this rate of reactivity increase, the water will begin to boil as soon as the fuel plates reach a temperature just slightly higher than the boiling point of the water. The negative steam and temperature coefficient will automatically compensate for the reactivity added by rod withdrawal. Since the burnout heat flux is at least twice the maximum operating heat flux, this does not appear to be a dangerous situation. Burnout occurs at a steam volume fraction greater than 80 percent.

Experience with EBWR has shown that the safety procedures employed during startup are adequate.

3.9.11 Failure of Steam System

A failure of the main steam system may occur by: (1) rupture of the steam piping, steam drier, or a valve upstream of any control mechanisms, (2) rupture of the steam piping or valves downstream of the control mechanism, the turbine or condenser casing, a reboiler, or tubes in the reboiler, or (3) internal mechanical failure of turbine valves or control mechanisms.

Rupture of the main steam piping or steam drier upstream of the shutoff valves on the primary reboiler will permit the steam to escape into the gas-tight building. In case of a catastrophic failure, the rate of steam flow will be sufficient to reduce the reactor power instantaneously because of flashing. If the operator does not insert the control rods, they will automatically insert when any one of the following conditions occurs: (1) when the general level of radioactivity in the building rises above a predetermined level, (2) when the water level in the reactor drops, or (3) when the water temperature drops to 325°F.

The effect on the reactor of a failure of the condenser or turbine casing or other elements downstream of the control mechanisms would be similar to that discussed in the preceding paragraph.

In the event the EBWR turbine shaft or casing should fail, the reinforced concrete lining of the steel shell will prevent puncture of the shell by flying fragments. The failure of the turbine will either trip the throttle valve or completely open the governor valves, depending on the sequence and nature of the failure. Tripping the throttle valve will shut off steam to the turbine, and reactor pressure will be maintained by valve P-11D or the steam bypass valve. Interlocks will insert the control rods to shut down the reactor when any one of the following conditions occurs: (1) high radioactivity in the building, (2) low water level in the reactor, (3) loss of vacuum in the condenser, or (4) safety action by the operators. Opening of the governor valves will increase the flow of steam from the reactor and, in addition to the shutdown signals just mentioned, will reduce power in the reactor temporarily because of flashing in the core. The reactor, after being shut down, will be cooled by one of the cooling systems previously described since the condenser will be inoperative.

In the event a tube rupture occurs in either the primary reboilers or drain coolers, control rod insertion may be automatically initiated by the low water level in the deaerator which will cut off the reactor feedwater pump and result in reactor shutdown. If a tube rupture is detected before any of the interlocks operate, the operator can initiate reactor shutdown. Shutdown cooling is accomplished by any one of the cooling systems described in Section 3.2.4.

A $\frac{5}{8}$-inch tube rupture in one of the primary drain coolers will result in a flow of approximately 250 gallons per minute of primary fluid (at 482°F and a differential pressure of 210 psi) into the intermediate system. The amount of primary fluid entering the intermediate system is based on flow from two nozzles since this would be the case for a complete rupture. The calculated flow is assumed to be 82 percent of the theoretical discharge from a long nozzle running full and without flashing prior to discharge.

A change in inventory will occur in the flash tank since liquid level is controlled in all the remaining vessels of the intermediate system. The change in inventory will be equal to the primary system leakage. For a single tube rupture the leakage is less than one-half of the total intermediate feedwater rate. A rise in reboiler level will cause valve G-333 in the line to the affected cooler to partially close and decrease the intermediate feedwater rate by the amount of leakage. Hence, the reboiler associated with the leaking drain cooler is furnished feedwater equivalent to that prior to the rupture. Discounting the small amount of flashing, the feed to the affected reboiler is saturated and the steam generated still remains essentially as before. Therefore, for a single tube rupture the inventory increase in the flash tank will be approximately 209 gallons per minute based on a temperature of 278°F. With the 1000-gallon flash tank normally operated at one-half of its volume capacity, the elapsed time between rupture and complete filling is approximately 2.5 minutes.

If more than one drain-cooler tube ruptures and the leakage is greater than one-half of the total intermediate feedwater rate, valve G-333 in the line to the affected cooler will close completely; valve G-333 to the other cooler will operate normally. Under these conditions the flash tank will be filled in about 1.5 minutes at a rate equal to one-half of the original pumping rate. However, in this case a rise in the affected reboiler level will occur at a rate equal to the difference between leakage and one-half of the original intermediate feedwater pumping rate. The other reboiler will operate normally since it is isolated from the affected circuit. Steam-flow to each reboiler will be unaffected by the rupture in one of the drain coolers because condensing rate in each of the reboilers will remain essentially the same. Also, drain tank level on the affected circuit is maintained, which limits the amount of primary leakage to one-half of the primary steam flow to the reboilers.

Simultaneously, the primary system is being depleted by an amount equal to the leakage. An inventory change will occur in the deaerator since it is the only vessel in the primary circuit where water level is not normally controlled. During operation, 1000 gallons is stored in the deaerator; thus, the stored water will be depleted in about five minutes for a single tube rupture. Low deaerator water level will result in pump cut-off which in turn will initiate reactor shutdown. In the event the pump interlock fails, reactor level will begin to drop approximately five minutes after drain cooler tube rupture. In approximately 7.25 minutes after a single drain cooler tube rupture, the water level drops 2 feet (575 gallons) which uncovers the top shroud and causes the reactor low-water level to initiate reactor shutdown.

Rate of inventory change in the deaerator is directly proportional to leakage. However, rate of change of inventory in the flash tank is directly proportional to leakage up to, but not exceeding, the intermediate feedwater rate to the leaking drain cooler.

A $\frac{3}{4}$-inch tube rupture in one of the primary reboilers will result in a maximum possible steam flow of approximately 210 pounds per minute into the intermediate system. Although the pressure drop across the rupture is 210 psig, the calculated flow is based on critical flow; the actual flow will be slightly less. Since negligible leakage will occur from the direction of the water box, the amount of primary steam escaping to the intermediate system is based on flow from a single nozzle. Steam leakage in the reboiler will cause the associated valve, G-333, to close partially; as a result, the flash tank will fill up at a rate equal to the leakage and the deaerator level will decrease by a like amount. Changes in inventory are not as great for ruptures involving steam as for ruptures involving water. It is not expected that the intermediate steam pressure will rise because of one reboiler tube rupture. Other than radioactive carryover, the main difficulty is the change in flash tank inventory; however, safety valves have been provided on the flash tank to relieve pressure.

4. NUCLEAR COMPONENT MODIFICATIONS AND ADDITIONS

4.1 REACTOR

4.1.1 Vessel Alterations*

4.1.1.1 Purpose

Operating experience with EBWR indicated that experiments and tests conducted at powers above 20 megawatts nominal would require certain modifications and additions to the plant. Components that have been redesigned and altered to accommodate the new requirements include the reactor pressure vessel. Reactor vessel alterations are more fully described in report ANL-6162.(10)

Experiments already completed and others anticipated have indicated the necessity for the installation of five $\frac{3}{4}$-inch nozzles in the reactor-vessel-closure cover plate and three 2-inch nozzles in the reactor-vessel-closure ring forging. Operation at 100 megawatts required that the reactor vessel shell be revised to: (1) increase the size of the steam outlet from 6 to 10 inches, (2) add a 6-inch feedwater supply, and (3) add two $1\frac{1}{2}$-inch water-level-measurement connections. Operational experience dictated an increase from $\frac{1}{2}$ to 1 inch in the size of the drain connections located at the bottom of three forced-circulation nozzles in the reactor vessel lower head. It was recognized at the time when addition of new nozzles in the pressure vessel wall appeared necessary that installation problems would be unique. The radioactive background inside the shell would make it necessary to perform all operations in minimum time and to provide an initially unknown amount of personnel shielding. Consequently all work procedures and tools were developed and tested before starting work in the vessel. Provisions were also made to install the required amount of shielding on the shield and work platforms.

The revisions are most conveniently reported according to their locations in the vessel, as summarized in Table 4-1. The east and west instrument nozzles, as well as the nozzles in the closure cover plate, were made prior to the final shutdown for the 100-megawatt plant alterations.

Table 4-1. Summary of EBWR Reactor Vessel Revisions

Location in vessel	Component	Number and size
Closure cover plate	Instrument nozzles	Five, 3/4-in. pipe-size
Closure ring forging	East instrument nozzle	One, 2-in. pipe-size with thermal sleeve. Accommodates three 1/2-in. pipes and one 3/4-in. pipe
	West instrument nozzle	One, 2-in. pipe-size with thermal sleeve
	Northeast instrument nozzle	One, 2-in. pipe-size with thermal sleeve
Shell	Steam outlet	One, 10-in. pipe-size with thermal sleeve
	Feedwater inlet	One, 6-in. pipe-size with thermal sleeve
	Water level measurement connections	Two, 1-1/2-in. pipe-size
Nozzles in lower head	Drain connections	Three, 1-in. pipe-size

*T. L. Kettles

4.1.1.2 Nozzle Installation

4.1.1.2.1 Closure Cover-Plate Nozzles - Five $\frac{3}{4}$-inch pipe-size nozzle flanges were installed in the top cover. A slight amount of radioactivity was detected on the interior surface of the top cover flange during installation, and as a safety measure, welders wore air masks and other personnel were excluded from the area during welding operations.

The 80 $\frac{7}{8}$-inch-diameter closure cover plate is made of SA-105, grade II carbon steel and is clad with submerged-arc welded 308 stainless steel on the bottom surface which is exposed to the reactor atmosphere. It is 9 $\frac{1}{16}$ inches thick including the $\frac{1}{4}$-inch cladding in the thickest section where the five $\frac{3}{4}$-inch pipe-size flanged nozzles are installed.

The nozzle flanges are located at the top surface of the cover plate. Each mating flange accommodates three ferule-type fittings for $\frac{1}{8}$-inch-diameter tubing penetrations into the reactor vessel.

4.1.1.2.2 Closure Ring-Forging Nozzles - The closure ring forging is made of SA-105, grade II material and provides forty-four 2$\frac{1}{2}$-inch studs on a 75$\frac{1}{2}$-inch-diameter bolt-circle for bolting down the closure cover plate. The minimum inside diameter of the forging is 66 inches and tapers downward and outward at an angle of 27 degrees 18 minutes to a maximum inside diameter of 84 inches, where it is welded to the vessel shell. Three nozzles were installed in the forging perpendicular to the inner face of the transition section.

The east instrument nozzle consists of a 3-inch, schedule-80 carbon steel pipe attached by dissimilar-material welding techniques to a stainless steel extension pipe and a loose stainless steel liner. This nozzle accommodates three $\frac{1}{2}$-inch and one $\frac{3}{4}$-inch pipe connections into the reactor through pipe couplings welded into an elliptical-head pipe cap by full penetration welds. The pipes are continued through the reactor shielding in a 6-inch pipe sleeve and are intended for thermocouples and for water attemperation of superheated steam.

The west instrument nozzle is similar in design to the east nozzle but provides one 2-inch pipe penetration rather than a group of smaller pipes. The west instrument nozzle is designed for a pressure of 800 psig and temperature of 800°F, being intended primarily for use as a superheated-steam test outlet.

The northeast instrument nozzle is almost a duplicate of the west nozzle but is intended to carry a large group of tubes and thermocouples into the reactor for various tests associated with 100-megawatt operation.

Each nozzle required a 3 $\frac{9}{16}$-inch-diameter hole through $\frac{1}{2}$ inch of stainless steel cladding and 5 inches of carbon steel forging with a machined J-groove weld profile measuring 5 $\frac{11}{16}$ inches maximum diameter.

The east and west instrument nozzles were installed during the period from February 23 to March 24, 1959. At that time the core was fully loaded and had undergone 195,331 megawatt-hours of operation. The water level was maintained at 10 feet above the top of the fuel elements during the nozzle installation.

Two platforms were used during the nozzle installation. A lower platform was suspended just over the water level by rods from the top of the vessel, and an upper platform was supported on top of the steam dry-pipe ring. The lower platform was sealed against the vessel wall and was intended primarily to catch any debris that might pass the upper platform. The upper platform was also sealed and was used to support the drilling rig.

The nozzle holes were drilled to $2\frac{1}{2}$ inches in diameter and machined with piloted boring bars to $3\frac{9}{16}$ inches in diameter. The J-groove was also machined with the boring bar. The nozzle assemblies were fabricated, hydrostatically tested, and helium-leak tested in the shops. They were welded into the forging with 200°F minimum preheat.

The northeast instrument nozzle was installed during the shutdown period for the 100-megawatt revisions, at which time, the fuel elements and control rods had been removed from the reactor vessel. The steam dry-pipe ring, having been removed, was not available to support the air-powered drill as before; therefore, a $\frac{3}{4}$-inch-thick steel working platform was used instead. This platform was supported with rods and was held rigid by a generous pattern of adjusting screws bearing against the vessel wall.

Because of the difficulties previously experienced on breakthrough, the northeast hole was drilled to a 3-inch diameter and finished to a $3\frac{9}{16}$-inch diameter with a piloted counterbore. The J-groove weld profile was machined with a piloted boring bar as before.

4.1.1.2.3 Shell Nozzles - The procedures and tools required for installation of nozzles in the shell were developed by welding the three nozzles (one of each size) into a 4-foot-high by 8-foot-long curved test section of clad plate which was fabricated to duplicate the vessel shell. Procedures and tools for installation of the steam and feedwater nozzles were similarly developed.

After the three nozzles had been installed in the test plate, they were cut to provide specimens for Charpy V-notch, subsize-weld tensile, root bend, face bend, side bend, weld-profile hardness, and weld-profile macroetch tests. The specimens were tested and gave results that met applicable ASME power boiler code requirements.

The vessel shell is made of $2\frac{7}{16}$-inch thick SA-212, grade B carbon steel clad with 0.109-inch thick type-304 stainless steel sheet spot welded to the base material. The inside diameter of the vessel shell is 7 feet 0 inches and extends from the stud-ring forging 18 feet 5 inches down to the lower head. One 10-inch steam, one 6-inch feedwater, and two $1\frac{1}{2}$-inch water-level-measurement nozzles made for vessel design conditions of 800 psig and 650°F were installed in the vessel shell (see Figure 2-2). Holes for these four new nozzles in the vessel shell were made from the working platform. The procedures developed on the test plate as previously described were used.

The steam nozzle is designed with a thermal sleeve because the upper-vessel temperature lags that of saturated steam during startup, and because production of superheated steam through this nozzle may become practical at a later date. The hole was reinforced by providing extra thickness in the carbon steel nozzle. The nozzle was welded in the shop at its outer end to a $\frac{1}{8}$-inch-thick loose stainless steel liner and to a 10-inch, schedule-80S stainless steel pipe penetration. A $\frac{1}{8}$-inch-thick circular stainless steel plate, welded to the inner end of the loose liner and to the existing shell cladding, allows differential expansion between the stainless and carbon steels with minimum restraint. A $\frac{1}{4}$-inch pipeline connected to a pressure gage outside the reactor shielding continuously monitors the integrity of this construction during operation. The nozzle was fabricated in the shop and then welded into the shell.

The 6-inch feedwater nozzle is designed with a thermal sleeve because the feedwater is about 300°F colder than the vessel wall at operating conditions. The design is the same as for the steam nozzle with the exception that the T-segment of the new 4-inch feedwater distribution ring must be rotated inside the vessel to permit welding to the arc segment. The distribution ring was assembled and welded in two sections.

In the upper and lower water-level-measurement nozzles, differential expansion is provided by a diaphragm plate arrangement but, as there is no necessity for a thermal sleeve, the diaphragm is welded directly to the $1\frac{1}{2}$-inch pipe penetration.

The completed installation of the nozzles and the distribution ring is shown in Figures 4-1 and 4-2.

The steam, feedwater, and water level measurement piping was continued from the nozzles to terminations outside the reactor shielding and plugged for a 1200 psig hydrostatic test. After the test the shielding materials were installed in the pipe tunnel.

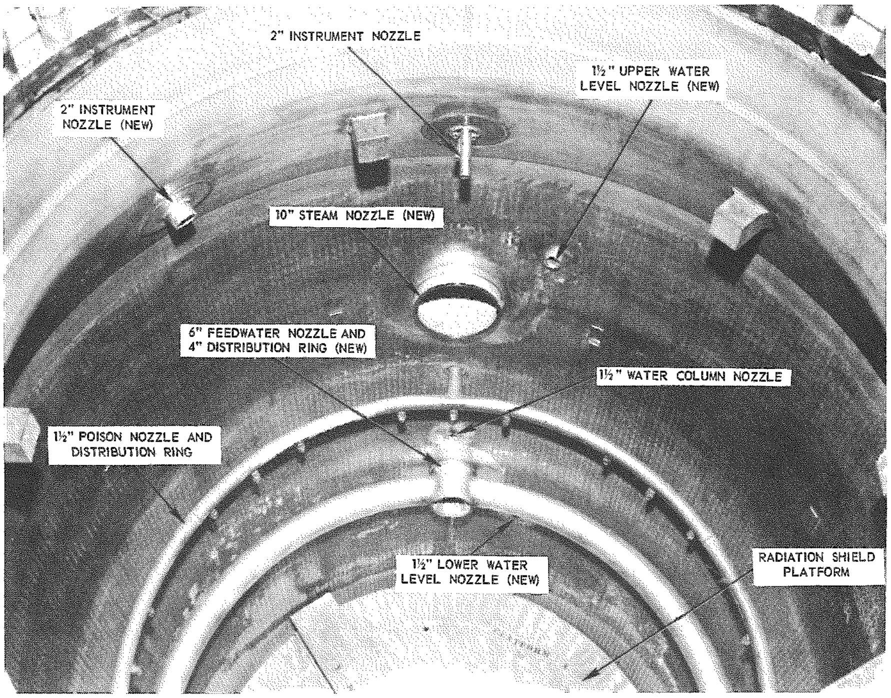

Figure 4-1

Inside view of completed installation
Neg. 111-8784B

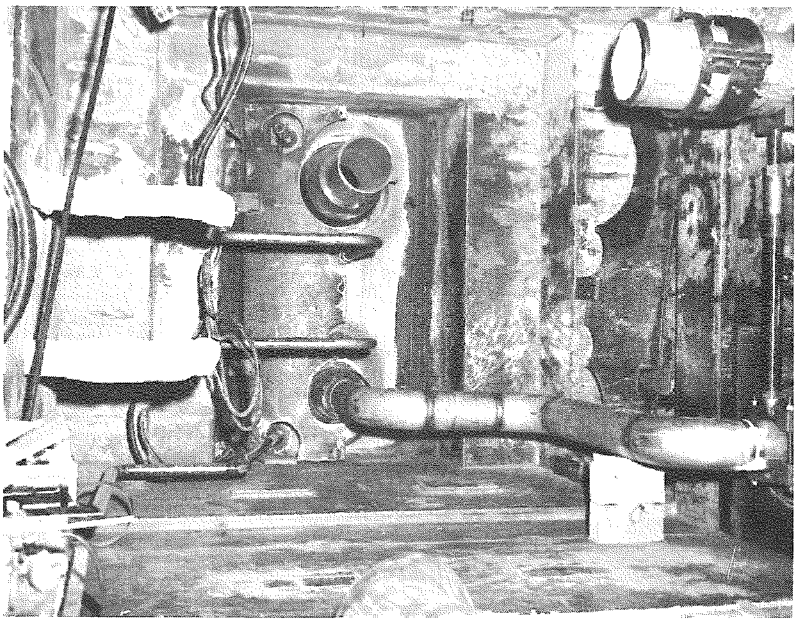

Figure 4-2

Outside view of installation
Neg. 111-8621

4.1.1.3 Drain Connections in Lower-Head Nozzles

The EBWR reactor vessel includes four 6-inch pipe-size nozzles which extend vertically downward 6 feet from the lower head. The four nozzles are intended to supply water to the reactor for forced-circulation-type operation. For natural-circulation-type operation one of these nozzles is used for high-pressure boric acid supply and the other three are capped. To prevent collection of radioactive waste in these dead ends, a $\frac{1}{2}$-inch valved flushing connection was originally provided on each cap. However, the $\frac{1}{2}$-inch-size connection proved inadequate to prevent waste buildup so they were replaced with 1-inch valve connections during the shutdown for 100-megawatt revisions. The radioactive waste buildup in one cap is shown in Figure 4-3.

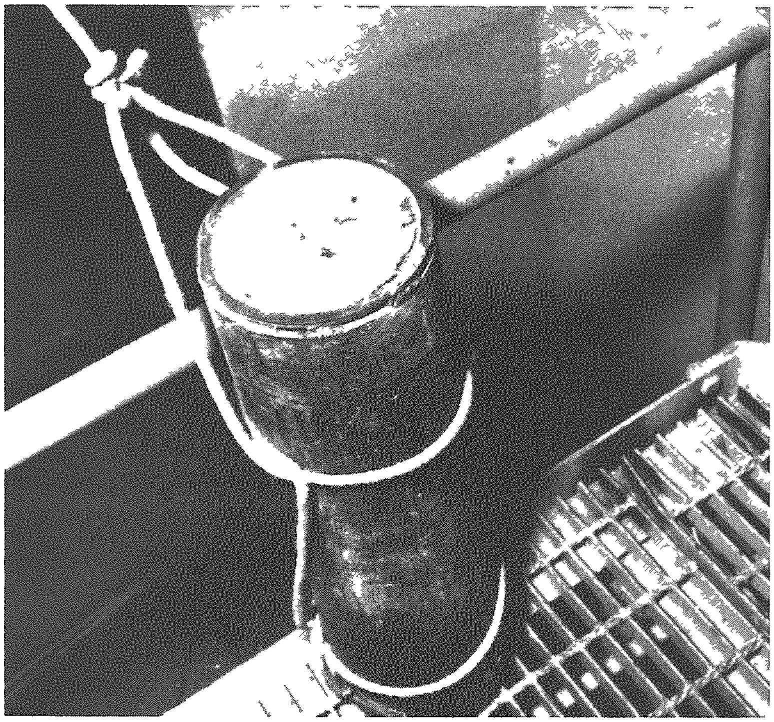

Figure 4-3

Waste collection in forced-circulation thimble cap
Neg. 111-8670

4.1.1.4 Lower Shock Shield

The shock shield serves to prevent sudden chilling of the hot wall of the pressure vessel in the event of injection of cold water through the boric acid distribution ring.

The original shock shield was removed to provide an area for the new feedwater nozzle. Since the shock shield that was removed required alterations and could not be decontaminated below a radioactivity of 2.5 roentgens, a new two-part shock shield without louvers was installed, as shown in Figure 2-2. The lower section is bolted to the thermal-shield flange and rises to within 1 inch of the new 6-inch feedwater return line. The lower section is composed of four longitudinal plates of $\frac{1}{8}$-inch-thick type-304 stainless steel bolted together to form a $6\frac{1}{2}$-foot-diameter by $3\frac{1}{2}$-foot-high cylinder spaced 3 inches from the vessel wall.

The top section of the new shock shield is described in Section 4.1.2.

4.1.2 Internal Alterations*

4.1.2.1 Core Shroud and Riser

Operation at 100 megawatts requires an increased rate of natural circulation of coolant through the core to permit higher rates of heat extraction while maintaining steam void fractions below the values that cause instability. A core shroud, riser, and riser extension were incorporated in the 100-megawatt-capacity design to produce the chimney effect necessary to obtain the higher flow rates. All three components are fabricated of type-304 stainless steel.

The core shroud, shown in Figure 4-4, is a $\frac{1}{4}$-inch-thick, $62\frac{5}{8}$-inch-OD by $96\frac{1}{4}$-inch-long cylinder bolted to the core support plate in a fashion identical to the previous forced-circulation shroud. A ring with a 1-inch by $3\frac{1}{2}$-inch cross section and $63\frac{5}{8}$ inches in outside diameter is welded to the top of the cylinder to provide reinforcement and a machined metal-to-metal seat for the riser. To insure parallel alignment of the machined seat and bolting surface, the cylinder was machined while in one piece. The cylinder was then cut in half to permit entrance through the containment building freight door. A circumferential band, bolted to both halves, fastens the two sections together.

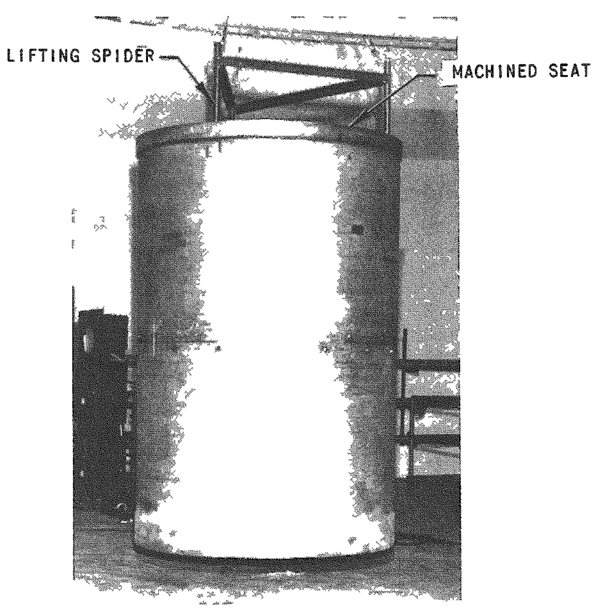

Figure 4-4

Core shroud
Neg. 111-7725

The riser, shown in Figure 4-5, is fabricated of $\frac{1}{4}$-inch-thick plate rolled into a cylindrical section 46 inches in inside diameter by 36 inches long and welded to a frustrum of a cone $18\frac{7}{8}$ inches high.

*L. E. Genens

The seat, which rests on the core shroud, is machined on an L-shaped section welded to the bottom of the cone. This results in a riser height of $57\frac{3}{8}$ inches and a total core shroud-riser height above the core support plate of $153\frac{5}{8}$ inches. The conical section was incorporated to provide a larger downcomer area at the restriction of the large flange fixed to the top of the thermal shield. Use of the conical section also provides diametral reduction to the 46-inch inside diameter of the cylindrical portion, which is sized to equal the core cross sectional flow area.

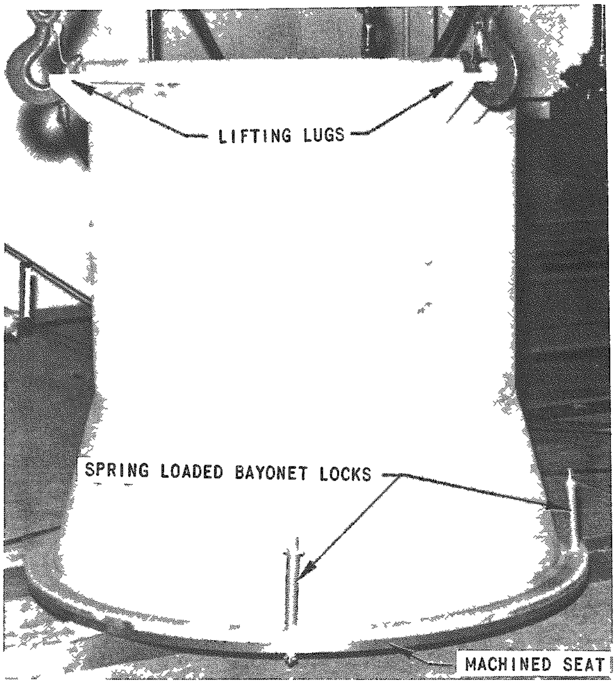

Figure 4-5

Core riser
Neg. 111-8772

In the event of a temperature excursion, a pressure buildup in the core would exert a force on the conical portion of the riser of sufficient magnitude to lift it off the seat. To prevent this, three spring-loaded, bayonet-type, holddown locks are welded to the riser and lock into lugs located on the inside surface of the core shroud thus, transmitting all the load to the core support plate. The locks are sized to withstand a 15-psi buildup, chosen as a possible value from BORAX experiment data. In normal operation, holddown is not required because hydrodynamic and weight forces act downward.

To determine experimentally the effect of a 1-foot increase in overall shroud height, the top of the riser is designed to receive a 1-foot cylindrical extension. The $\frac{1}{4}$-inch-thick, 46-inch-ID cylinder, shown in Figure 4-6, can be removed or added independently of the riser. Since hydrodynamic forces do not act on the extension, a locking device is not needed. Three bars welded to the inside surface, extending one foot below the bottom of the extension and bent in a U-shape at the top, facilitate installation and guide the cylinder back into place in the event it is forced out of position.

Three guide rails are used to guide the riser into and out of the vessel. The primary purpose of the rails is to permit remote handling should contamination from radioactivity occur. Each rail is clamped and positioned on a head bolt and hangs into the vessel. The top 2-foot section of each rail extends above the vessel and is bent outward at 15 degrees. The bend provides a lead-in and makes the rails more visible from the operating floor during remote operation.

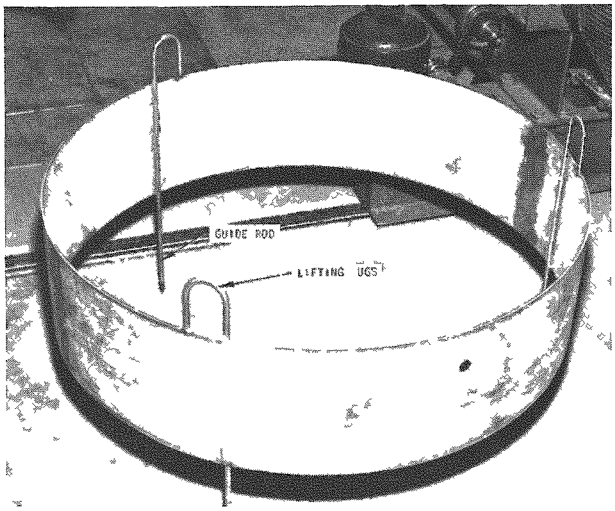

Figure 4-6

Core riser extension
Neg. 111-8775

4.1.2.2 Upper Shock Shield

The upper shock shield, which is located between the 6-inch condensate-return line and the poison ring, services two purposes: (1) to prevent shocking the vessel wall if injection of demineralized water were to occur when the vessel is hot and (2) to reduce the waste buildup and radiation level on the vessel wall. The upper shock shield is illustrated in Figure 2-2. The shield is made up of four 90-degree segments fabricated from $\frac{1}{8}$-inch type-304 stainless steel and bolted together to form a cylinder $23\frac{1}{2}$ inches high by 78 inches in outside diameter. Support is achieved by bolting the shield to a three segment mounting ring welded to the 6-inch condensate-return line. Eight Z-shaped brackets are equally spaced around the outside diameter of the shield to maintain concentricity with the inner vessel wall.

4.1.2.3 Steam Duct

The high-water level and large steam quantity production resulting from 100-megawatt operation dictates the following: (1) steam take off high in the vessel to obtain the driest possible steam and (2) an inlet area of sufficient size to prevent excessive velocities. These conditions are achieved through the use of a steam duct.

The steam duct, fabricated of $\frac{1}{8}$-inch type-304 stainless steel, is a shaped duct designed to conform with the internal vessel configuration on the backside without restricting the minimum vessel opening of 64 inches on the inside. The steam duct is shown in Figure 4-7. To allow access to the west instrument nozzle and upper gage line, the center of the 112-square-inch, rectangular steam-inlet area is displaced 90 degrees from the steam outlet and sustains an arc of 82 degrees. The bottom of the inlet is located at the 726-foot $7\frac{7}{16}$-inch elevation, which is 3 feet $\frac{1}{16}$ inch above the centerline of the 10-inch steam line. The duct is supported by two angle brackets bolted to existing shield ring lugs and to one end of the duct which is welded directly to the steam line. An L-shaped bracket, welded to the bottom of the duct and to the vessel wall, is used to provide additional support and to minimize vibration caused by the pulsating coolant entering the space between the duct and vessel wall. To assure a water-tight assembly, the duct is welded directly to the steam line. This required fabrication in two parts, illustrated in Figures 4-7 and 4-8, to permit sealing the duct after the weld on the steam line was secured.

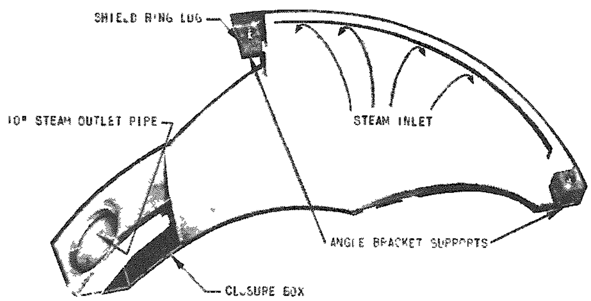

Figure 4-7

Steam duct
Neg. 111-8828

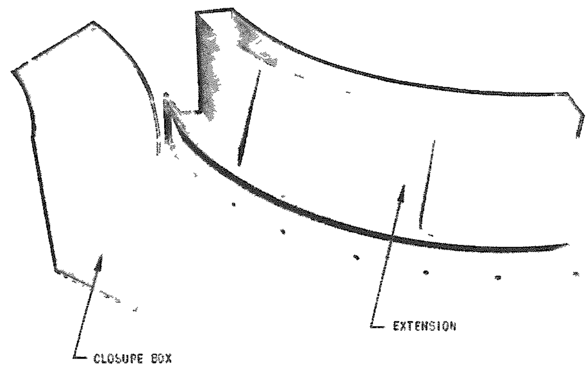

Figure 4-8

Steam duct extension and closure box
Neg. 111-8821

A removable steam duct extension, fabricated of $\frac{1}{8}$-inch type-304 stainless steel and shown in Figure 4-8, is provided to raise the inlet area an additional 13 inches. The extension will allow installation of a wire mesh demister, at a later date, without modification of the duct assembly. Since the extension protrudes into the vessel at the point of minimum inside diameter, a bolted connection secures the extension to the steam duct and permits easy removal. The inlet area is sized to produce the same flow velocity as that at the entrance of the duct. This results in a steam duct assembly with a minimum cross sectional area of $1\frac{1}{2}$ times that of the 10-inch steam line.

4.1.3 Instrument Location Additions

4.1.3.1 Water Column

To obtain signals for adequate water-level control during operation using the riser, the following modifications were made to the existing water column (see Figure 3-27).

The existing 6-inch column was extended $39\frac{3}{8}$ inches to an elevation of 724 feet $9\frac{3}{8}$ inches. The $39\frac{3}{8}$-inch extension was selected so that new piping connections could be incorporated and permit continued support of the column from the underside of the operating floor. Two new $1\frac{1}{2}$-inch reactor gage lines, which penetrate the biological shield at elevations of 724 feet $5\frac{3}{8}$ inches and 719 feet $3\frac{3}{8}$ inches, are connected into the column and are used in conjunction with the two existing gage lines. A vent is installed from the reactor vent line, outside of the biological shield, to the top of the column extension. Since the vent is located at the extreme top of the column, a tap is installed to provide a blowdown connection for radiation decontamination of the column. Five $\frac{3}{4}$-inch instrumentation taps are provided, one in each of the three upper reactor gage lines and the remaining two in the vent and poison line. The column connections of the 96-inch differential pressure cells (instruments H-1A-D3 and H-1A-D11) have been moved up 3 feet. This permits the existing instrumentation to cover the operating-water-level range for high and low power and moves the constant-level pot to the 723-foot elevation. A 4-foot sight gage, similar to the original, has been added for visual inspection of water level when the riser is installed. The 4-foot range covered by the sight gage is from 719 feet $1\frac{1}{4}$ inches to 723 feet $1\frac{1}{4}$ inches. The unit is positioned so that the distance between its bottom port and the top port of the original gage is equivalent to any two adjacent ports on either unit. The new gage is independent of the existing gage, therefore, either unit can be operated as a 4-foot range or combined to give a total visual range of 8 feet $2\frac{1}{4}$ inches. Three Magnetrol level switches have been added to provide alarm signals when the riser is used. A high-level alarm set at the 723-foot elevation will allow the water level to rise to $\frac{7}{8}$ inch below the steam duct. Two low-level alarms set at elevations 721 feet 3 inches and 720 feet 3 inches will prevent the level from dropping below the riser, with or without the riser extension. The following instrument connections were raised to the top of the column (elevation 724 feet $7\frac{3}{8}$ inches): existing pressure transducer (pressure reference), Norwood precision pressure transducer, and pressure gage G-101. Two new instruments have been connected at elevation 724 feet $7\frac{3}{8}$ inches: pressure transducer P-11A and pressure gage G-101A. Gage G-101A is located at elevation 720 feet 9 inches and provides a visual pressure indication (viewed by remote television camera) when the upper sight gage is used. Shutoff valves located in all pipe lines allow isolation of the reactor water column from the reactor or independent isolation of any alarm, transducer, pressure cell, or gage from the water column.

4.1.3.2 Instrument Nozzles

Two instrument nozzles (discussed in Section 4.1.1) have been installed to permit direct entrance to the vessel from the operating floor The west nozzle is located on reactor axis W and is a temporary nozzle. The northeast nozzle, located midway between axes X and Y, is a permanent nozzle. At each location a $4\frac{1}{2}$-inch-diameter hole was core-drilled through the concrete to allow nozzle installation with minimum structural weakening of the floor.

Installation of the temporary nozzle, shown in Figure 4-9, was facilitated by cutting the previously installed superheat-outlet line and welding to the existing vessel nozzle. The superheat line was cut so that sufficient material remained for reinstallation upon completion of the instrumentation tests and removal of the instrument nozzle. Because the nozzle is installed for temporary usage, the 2-inch pipe and the cored hole are sealed to maintain the negative pressure envelope around the reactor with a temporary sealing material.

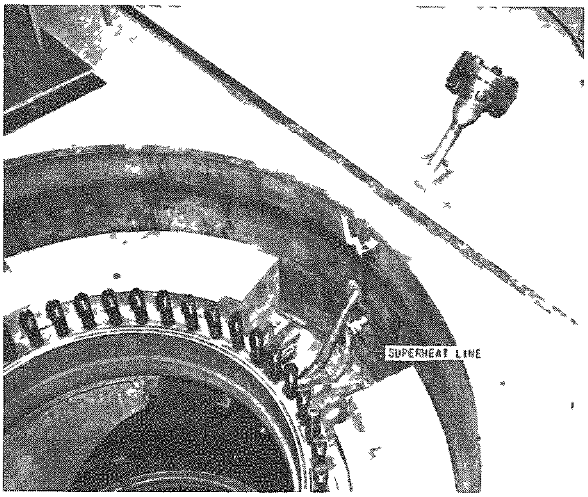

Figure 4-9

West instrumentation nozzle
Neg. 111-9022

The northeast instrument nozzle, shown in Figure 4-10, provides the only "straight-in" opening to the vessel. This greatly simplifies installation of instruments and instrument lines and also results in a port to obtain measurements of radiation emitting directly from the vessel. The upper end of the nozzle terminates in a flange located in a diamond-shaped box that is sunk into the operating floor. Two $\frac{1}{2}$-inch fittings are provided on the upper end of the nozzle, one on the flange to vent the system to the fuel storage pit and the other to receive cooling water for the instrumentation. A $3\frac{1}{2}$-inch carbon steel liner, welded at the bottom to the reactor-pit wall and at the top, through a bellows, to the 2-inch pipe, allows expansion of the instrument nozzle and forms the seal for the negative pressure envelope.

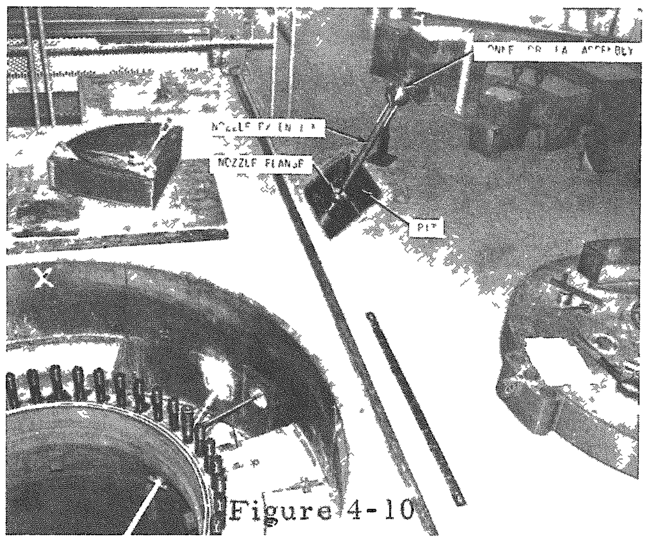

Figure 4-10

Northeast instrumentation nozzle
Neg. 111-9021

The northeast nozzle extension, fabricated entirely of type-304 stainless steel and shown in Figure 4-11, simplifies the installation and external termination of 26 instrument tubes. Fixed to the northeast nozzle, through the flange provided, the extension creates a new high point in the system and is therefore supplied with a vent connection located at the top of the instrument-connector-head assembly. Since the instrument tubes originate in the connector head, a $\frac{1}{2}$-inch cooling water tap is located at the upper end of a 2-inch pipe to furnish cooling over the entire tube length. The bulbous connector-head assembly is incorporated because it provides more convenient access to the Swagelok termination of the 26 instrument tubes than that permitted by the more commonly used tube-sheet arrangement. A support, in the form of a saddle located on a pedestal, eliminates the bending load on the flange connection by guiding the nozzle extension just below the connector-head assembly.

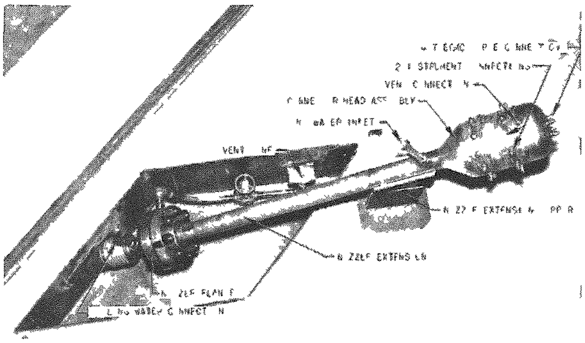

Figure 4-11

Northeast instrument nozzle extension
Neg. 111-9028

An instrument connector box is located in the vessel at an elevation (to bottom of box) of 724 feet $3\frac{5}{8}$ inches and is in line with the northeast nozzle. The connector box acts as a water reservoir for instrument-line cooling and a junction point for disconnecting the removable instrumentation in the vessel from the permanent instrumentation in the nozzle. The angle-shaped box, 15 inches deep by $8\frac{5}{8}$ inches wide by approximately 22 inches long, is fabricated of $\frac{3}{16}$-inch type-304 stainless steel plate and fits in the vessel without restricting the minimum 64-inch-diameter opening. The box is supported by four $\frac{5}{8}$-inch-diameter bolts that screw into two rails welded to the vessel wall. A 150-pound flange is welded to the portion of the nozzle that protrudes into the vessel and provides a connection for the water jacket that extends from the nozzle inlet into the connector box.

4.2 CONTROL RODS AND DRIVES

4.2.1 Operating Control Rod*

4.2.1.1 Description

The control rods for 100-megawatt EBWR incorporate several features which are improvements over the previous rods. The length and stroke of the absorber section were increased to 152 centimeters (60 inches) to accommodate a future 152-centimeter core. The control rod is shown in Figure 4-12. The improvements consist of (1) new handling fixtures that enhance postoperation handling and (2) substitution of boron-stainless steel for hafnium to reduce material costs. Postoperation handling of the new rod has been facilitated by design improvements. This provides for separation of the 25-centimeter (10-inch) poison and follower cross from the balance of the rod. The poison and follower cross can then fit in the

*V. M. Kolba

transfer coffin for removal to the storage pit. The remaining portion of the rod is then handled by the handling tool, crane, and a small clamp-on shield can. Cost reduction was accomplished by use of 2 percent boron-stainless steel in place of hafnium.

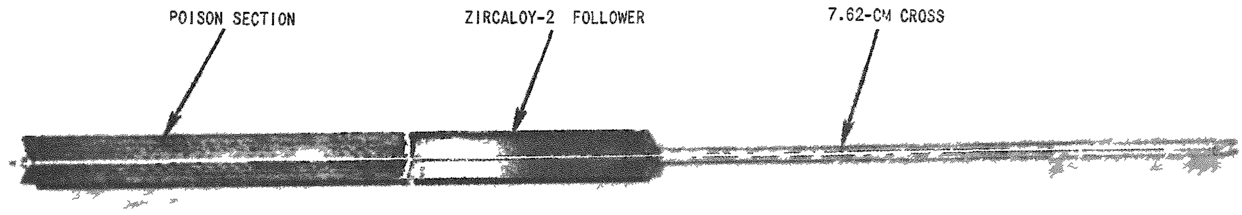

Figure 4-12

Control rod assembly
Neg. 112-1320

As shown in Figure 4-13 the control rod is composed of the poison section, the Zircaloy-2 follower section, and the 7.62-centimeter (3-inch)

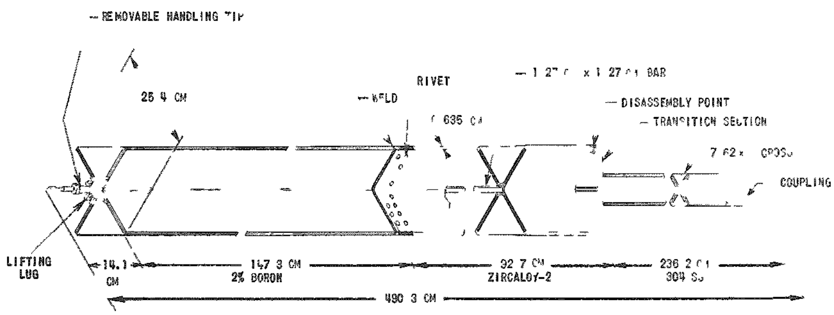

Figure 4-13

Operating control rod with Zircaloy-2 follower
RE-6-35284A

cross. The poison section is composed of nominally 2 percent natural boron-304 stainless steel. Analysis of the material is shown in Table 4-2. The boron-stainless steel labeled old heats and new heats were the materials of construction of Core I and Core IA rods respectively.

Table 4-2. Analysis of Boron Stainless Steel[a]

	Old heats[b]		New heats[c]		
	KA-272	KA-274	KH-761	KH-762	Desired
C	0.046	0.046	0.014	0.016	
Mn	1.81	1.66	2.03	1.96	1.55 min
Si	0.47	0.56	0.64	0.70	0.5 min
Cr	18.33	19.06	19.03	18.91	18 - 20
Ni	9.60	9.92	9.70	9.59	18 - 10
S	0.009	0.011	0.007	0.006	-
P	0.014	0.011	0.011	0.010	-
B	2.02	1.91	2.20	2.19	2.0

[a]All values in percents
[b]Material of construction of original control rods
[c]Material of construction of improved control rods

4.2.1.2 Fabrication and Assembly

The sheet material is received from the vendor in 0.318-centimeter-thick ($\frac{1}{8}$-inch) stock, 25.4 centimeters (10 inches) wide and approximately 178 centimeters (70 inches) long. The sheets are hot formed to 90-degree angles with a 1.27-centimeter ($\frac{1}{2}$-inch) internal radius. (Previous boron sheet had been cold formed; however, the new material exhibited a gross tendency to crack on cold forming. Hot forming was investigated and found to be adequate. All remaining sheets were successfully hot formed at 915°C.) Four of these angles are spot welded together to form a cross 25.4 centimeters by 25.4 centimeters (10 inches by 10 inches) with blades approximately 0.635 centimeter ($\frac{1}{4}$ inch) thick.

Examination of irradiated boron-stainless rods from Core I revealed circumferential cracks approximately 0.635 centimeter out from the center of the spot weld in approximately 180-degree arcs. This phenomenon was also noted in a sample Core I control rod which was autoclaved at reactor conditions except for flux. However, a control blade that was fabricated at the same time as the other Core I rods but never autoclaved or operated in a reactor did not show any cracks upon examination. Several methods for eliminating these cracks were investigated. The method selected was to heat treat the spot welded crosses. This consists of a stress relief for 6 hours at approximately 700°C followed by a furnace cool. All crosses are heat treated in this manner prior to final assembly with the follower unit.

The follower section is composed of Zircaloy-2 material. Sheet material 0.159 centimeter ($\frac{1}{16}$ inch) thick, 25.4 centimeters (10 inches) wide, and approximately 96 centimeters (~38 inches) long is cold formed to a 90-degree angle having a 1.27-centimeter internal radius. Sheets of 0.318 centimeter thickness are sheared to 11.4-centimeter by 89-centimeter (~$4\frac{1}{2}$-inch by 35-inch) pieces to serve as spacer plates for the followers. Four of the 0.159-centimeter-thick ($\frac{1}{16}$-inch) angles and four spacer plates are spot welded together to form a cross 25.4 centimeters by 25.4 centimeters (10 inches by 10 inches) with blades approximately 0.635 centimeter ($\frac{1}{4}$ inch) thick. A 304 stainless steel transition piece was riveted between the thin Zircaloy-2 angles at one end. Inconel rivets of 0.318-centimeter ($\frac{1}{8}$-inch) diameter are used to join the Zircaloy-2 followers and the stainless steel transition pieces.

After the transition section is riveted to the followers, the sections are corrosion tested for two weeks at saturated conditions of pressure and temperature (252°C).

The poison cross is welded to the follower cross, at the 304 stainless steel transition, to form one integral unit and then machined to length. Holes are machined in the blades of the Zircaloy-2 follower to receive 0.317-centimeter-diameter ($\frac{1}{8}$-inch) locating dowel pins which are in the lower transition piece. A lifting lug is also welded to the poison section to permit handling in the transfer coffin.

The connector, 7.62-centimeter (3-inch) cross, lower transition section, and 1.27-centimeter by 1.27-centimeter ($\frac{1}{2}$-inch by $\frac{1}{2}$-inch) bar are made of 304 stainless steel and welded together to form an assembly approximately 482.6 centimeters (190 inches) long. The connector is machined from a piece of 304 stainless steel. The 7.62-centimeter (3-inch) cross is made of four 0.317-centimeter-thick ($\frac{1}{8}$-inch) sheets formed to a 90-degree angle and spot welded together to form the 7.62-centimeter (3-inch) cross with 0.635-centimeter-thick ($\frac{1}{4}$-inch) blades. The removable handling tip is machined from 17-4 PH stainless steel to provide a gall-resistant joint.

The control rod is assembled by sliding the poison section and Zircaloy-2 follower over the 1.27-centimeter ($\frac{1}{2}$-inch) bar and locating on the transition section by means of the 0.317-centimeter ($\frac{1}{8}$-inch) dowel pins at each arm extremity. The removable handling tip is threaded on the 1.27-centimeter ($\frac{1}{2}$-inch) bar and when tightened is tack welded to the lifting lug. The rod is then loaded into the reactor for locking to the drive mechanism for reactor operation.

A sample control rod was constructed and tested in the control-rod-drive test rig. The sample rod was an exact dimensional mockup of the design, except for the materials of the Zircaloy-2 follower and the poison section. Stainless steel was used to simulate the poison and follower sections. This rod was cycled in increments of 5000 scrams and examined after each increment. The test unit was operated at EBWR pressure but the water in the test rig was at room temperature. A pressure of 600 psi was maintained by nitrogen overpressure.

Post-test examination revealed no indication of failure of any joints at the end of 15,000 scrams. A few burnished areas were noted in several of the blade radii where the blade had rubbed the shroud. The amount of wear was negligible and not considered detrimental to the operation of the unit

Using the same sample rod, disassembly and handling techniques were also tested. The Core I control-rod handling tool was modified by an adapter. In the disassembly and handling tests, the tack weld was broken between the handling tip and the lifting lug and the handling tip unscrewed and removed. The simulated poison and follower section was then removed from the balance of the control rod, the handling tip was replaced on the bar, and the balance of the control rod removed.

4.2.2 Oscillator Control Rod*

4.2.2.1 Description

Prior to operating EBWR at 100 megawatts, transfer functions of the reactor core must be measured to confirm empirically the stability of

*J. F. Pohlman

the core. The reactivity input signal for these measurements is provided by oscillating a specially designed control rod through a wide range of frequencies at various strokes and core elevations. In addition to good neutron absorption, a control rod used for power transfer-function measurements must possess structural rigidity, physical strength, and light weight. The most severe loading of the rod occurs when the oscillating frequency is 8 cycles per second, amplitude ± 1 inch, and the rod withdrawn 40 inches from the core. The maximum load occurs at the top and bottom of the stroke because of the inertia of the rod.

The over-all dimensions of any control rod used in EBWR are essentially fixed by reactor operating requirements. Control considerations dictate approximately 4 feet of absorber of 10-inch by 10-inch cross section with the follower section full size to maintain a minimum water hold.

In all respects, the oscillator control rod closely resembles the original 20-megawatt EBWR hafnium control rod. The absorber section is made of four 5-inch by 5-inch by $\frac{1}{16}$-inch angles of hafnium spot welded together to form a 10-inch by 10-inch cruciform $\frac{1}{8}$ inch thick and 46 inches long (see Figure 4-14). This produces an absorber section of 97 pounds which is the lightest weight currently available.

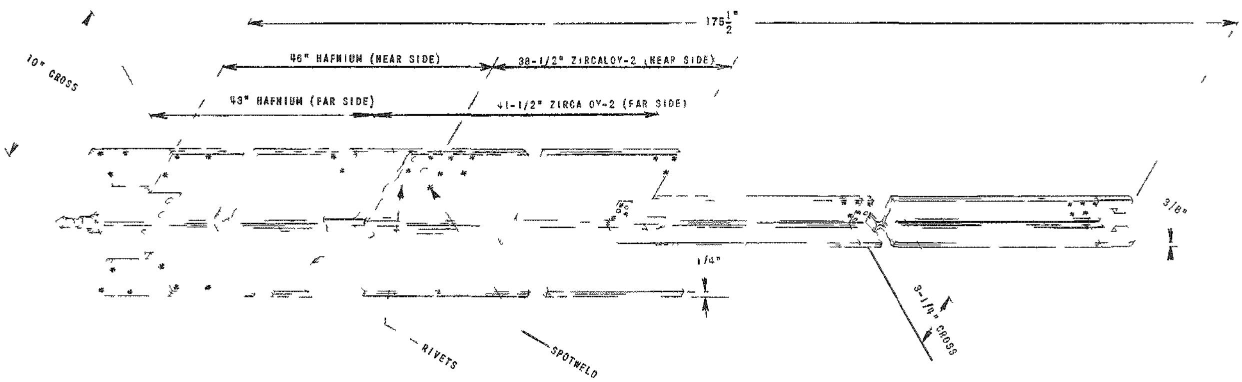

Figure 4-14

Oscillator control rod
RE-6-35172-C

The follower section is made from four 5-inch by 5-inch by $\frac{1}{16}$-inch Zircaloy-2 angles spot welded together to form a 10-inch by 10-inch by $\frac{1}{8}$-inch cruciform section $38\frac{1}{2}$ inches long. The absorber and follower sections are joined by a combined spot weld and riveted lap joint to insure the integrity of the transition.

The lower end of the follower section is made of 304 stainless steel since this section requires both column strength and resistance to flexure. To provide sufficient stiffness of the follower during rapid oscillation, the largest possible cruciform section, $3\frac{1}{4}$ inches by $3\frac{1}{4}$ inches, is used. The connection between follower and extension, as well as the Zircaloy-2-stainless steel transition in the follower, are riveted lap joints.

The lower end fitting is similar to those of the 100-megawatt operating control rods. However, it was necessary to increase the thickness of the end fitting from $\frac{1}{4}$ inch to $\frac{3}{8}$ inch to achieve the strength required while oscillating. A special cross-guide connector is used for the oscillator control rod; this connector can also be used on any of the 100-megawatt control rods or in conjunction with the 100-megawatt rack-and-pinion drives. This design is discussed in Section 4.2.4.

4.2.2.2 Testing

The oscillator control rod is similar to the hafnium control rod used for the first series of transfer-function measurements, except for the change to riveted joints. This change provides a large margin of safety against failure from tensile loading during oscillation.

Prior to actual transfer-function-measurement experiments, the rod is to be installed in the reactor and oscillated throughout its entire operating range. The rod will remain in place until the conclusion of the transfer-function experiments, since it must be destroyed upon removal from the reactor.

4.2.3 Operating-Control-Rod Drive Mechanism*

4.2.3.1 Description

The control-rod drive mechanisms originally installed in EBWR were replaced with units of a different design. Replacement was made for the following reasons: (1) increase in stroke length, (2) performance improvement, and (3) elimination of wast-collecting pockets. The change of core length from four to five feet requires increased control rod travel. The original 4-foot-stroke unit could not conveniently be converted because of increased column loading requirements and need for other major alterations. The original units were subject to sticking, which was attributed to settlement of waste into the seal rings and shaft guide bushing. Due to design limitations, it was not possible to remove the waste by flushing. The new rack-and-pinion type units can be flushed, and the worth of this feature has been demonstrated. In addition, the original connector-dashpot part used

*D. J. Roy

for connecting the control rod to the drive shaft was a natural reservoir for collecting radioactive waste that could not be readily recovered.

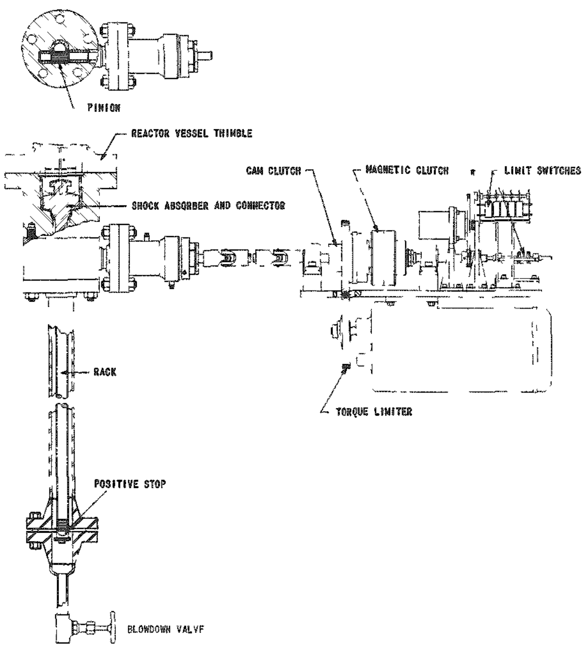

Figure 4-15

Operating-control-rod drive mechanism
RE-6-23979-B

The new control-rod drive mechanism incorporates a conventional rack-and-pinion design which drives through a rotating shaft fitted with a pressure breakdown seal (see Figure 4-15). The original unit, with a 4-foot stroke, was considered as a backup design for EBWR reference drives. It was adapted from the original Army Package Power Reactor (APPR) model to meet EBWR operating requirements. Development work was continued over the interim and resulted in elimination of original objectionable features and reduction of cost from about \$12,000 to less than \$4000 per unit. The principal improvement was the substitution of graphite bushings and sleeves for Stellite antifriction bearings in the pinion shaft support area. The housing changed from a weldment to a more compact and less expensive casting. Waste-collecting pockets were eliminated and provisions for flushing incorporated.

Previously established reactor operation requirements and evaluation of test information have resulted in the design criteria listed below:

1.	Operating pressure	600 psig
2.	Operating temperature of seal	110°F
3.	Number of control rods	9
4.	Rod travel	60 in.
5.	Control rod spacing	$12\frac{3}{4}$ by $12\frac{3}{4}$ in.
6.	Rod speed in reactivity	
	Minimum	1×10^{-5} k/sec
	Maximum	4×10^{-5} k/sec
7.	Maximum rate of linear travel	28 in./min
8.	Position ability	$\pm$ 0.025 in.
9.	Scram speed at 200 psi	56 in. in 1.35 sec

10. Deceleration distance 4 in.
11. Maximum leakage rate per rod 200 cc/min
12. Complete fail-safe characteristics
13. Flushable during operation
14. Removal of each control rod and drive unit without draining the reactor vessel.

The control-rod drive mechanism consists of a drive unit and a power unit connected to each other by a shaft with two universal joints for alignment. The main components of the drive unit are:

1. Dashpot housing assembly
2. Pinion housing assembly
3. Seal housing assembly
4. Lower housing assembly

A cross section of the drive unit assemblies is shown in Figure 4-16.

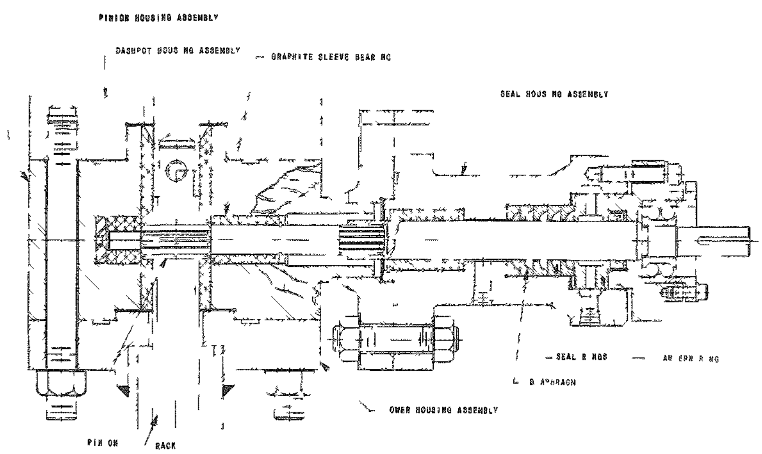

Figure 4-16

Cross section of control rod drive unit
RE-1-24670-F

The dashpot housing is made of type-304 stainless steel and contains a removable flash-chromium-plated 304 stainless steel dashpot sleeve that is tapered for proper dashpot action. The combination dashpot piston and cross connector is made of type-304 stainless steel with a Stellite inlay which is ground to the contour of the seat in the bottom of the dashpot housing. This feature enables removal of the portions of the drive below the seat while the reactor vessel is full of water.

Although the valve seats do not normally leak, a tool (Figure 4-16) is used to prevent loss of water from the thimble while the mechanism is removed from the operating position. This tool also prevents inadvertent lifting of the connector when the control rod is removed from the core.

The control rod is connected to the rack by the combination connector-dashpot piston with a bayonet-lock joint. The rack cannot be disconnected from the control rod during operation. The rack is made from 17-4 PH stainless steel heat treated to condition H-1100, and the end containing the connector-pin hole is annealed to improve impact resistance.

The pinion housing is made from type-304 stainless steel casting and the combination pinion and pinion shaft is made from 17-4 PH stainless steel heat treated to condition H-1100 and flash plated with chromium. The splined end of the shaft is annealed to increase impact resistance. The Graphitar sleeve bearings and guide bushings are held in place by a shrink fit. Bearingbores are machined for size-to-size shaft fits. Rotating shafts are allowed to seek natural centers during preliminary run-in.

The seal housing, which is a type-304 stainless steel casting, contains a 5-stage, pressure-breakdown seal unit. The drive shaft, which extends through the seal, is made from 17-4 PH stainless steel with the same physical properties as the rack and pinion parts. It is suspended between a Graphitar bushing on the pressure end and a combination radial-thrust ball bearing on the atmospheric end. A Sirvene synthetic rubber lip seal is inserted between the ball bearing and the leak-off connection.

The rack housing tube is made of type-304 stainless steel pipe and flanges and provides space for rack travel to the fully inserted position. The flanged joint in the lower portion of the tube is necessary for installation of the mechanical stop on the rack end which is required to limit upward travel. The main components of the power unit are:

1. Gear motor
2. Torque limiter
3. Magnetic clutch
4. Synchrotransmitter
5. Limit switch
6. Indicating light limit switches
7. "One-way" clutch

The units are mounted as a package on wall brackets adjacent to the control-rod-drive-mechanism access platform (Figure 4-17). A plan view of the orientation of the nine drives is shown in Figure 4-18.

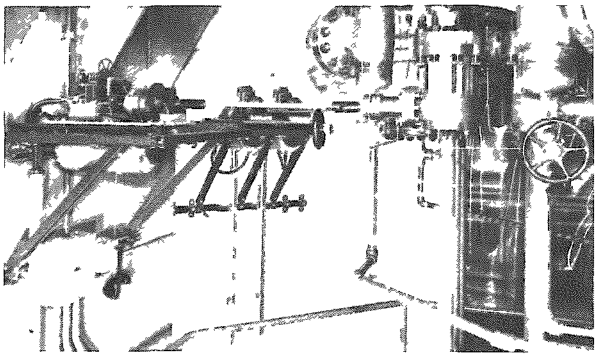

Figure 4-17

Power unit mounting arrangement
Neg. 111-9279

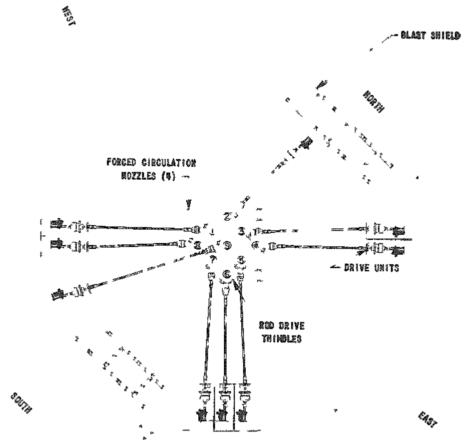

Figure 4-18

Orientation of control-rod
drive mechanisms
RE-6-34892-B

Normal operation of the power unit consists of energizing the magnetic clutch and driving the output shaft with the gear motor through the torque limiter (set at $2\frac{1}{2}$ times the torque required to raise the rod) and magnetic clutch. Position indication is achieved through the synchrotransmitter and receiver, the transmitter being driven by a gear on the output shaft.

The 5-unit, rotary-adjustable, cam-operated microswitch assembly is also driven off the output shaft. A switch actuates the "Rod Out" light on the position indicator dial which should read in the 59 to 60-inch range at this time. Another switch actuates the "Rod In" light on the indicator dial when the rod is in the 0 to 4-inch range. Two other limit switches are provided for "Rod In" and "Rod Out" limits. These switches are mounted on an Arch Instrument Co. limit device connected to the output shaft. The limit device may be adjusted to a limit of $\pm \frac{1}{20}$ inch.

Rapid insertion of the control rods is effected by de-energizing the magnetic clutch and allowing the rod and rack to fall by gravity to the fully inserted position. Safety requirements prevent re-energizing the magnetic clutches until all nine rods are inserted and insertion is indicated by signal lights connected to the "Rod In" limit switches. Since the motor circuit is active and causes the motor to run for downward travel, a one-way clutch is provided to apply downward force only. This feature provides additional thrust in the event a rod does not fall in by gravitational force. An adjustable torque-limiting clutch is provided in this train to prevent excessive thrust that might result in damage to the mechanism.

4.2.3.2 Testing

The testing of developmental, prototype, and the nine operating control-rod drive mechanisms was performed in the facility shown in Figure 4-19. This rig simulated EBWR operating conditions of pressure, temperature, and stroke length. During the last stages of development and testing, the rig was operated at 600 psi and room temperature. Gas pressure was used instead of steam because temperature gradients along the vessel length caused the mechanism to operate at low temperature under all circumstances.

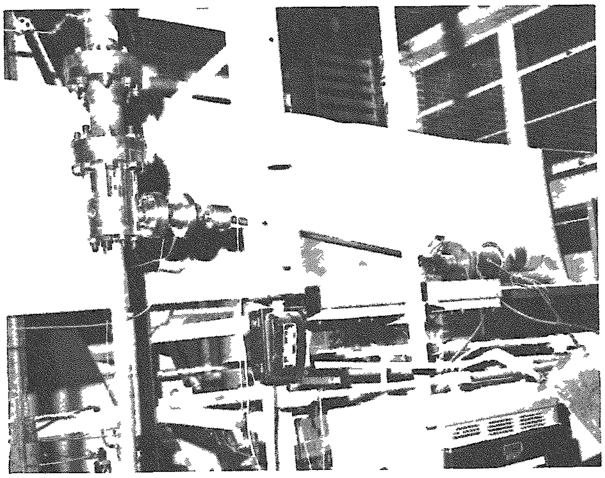

Figure 4-19

Facility for testing control-rod drive mechanisms
Neg. 111-7344

Development of the rack-and-pinion drive mechanism was started in late 1955 as a backup effort to the reference design. The rack-and-pinion concept was selected as an alternative because of its simplicity, performance characteristics, and state of development. The APPR drive design, then under development, was adapted to meet EBWR design requirements. Testing was continued after the commitment was made to develop the reference design. Possibilities for design improvement were observed and a series of changes began. Tests were made for both 4- and 5-foot-stroke lengths. The design was thus available when the decision was made to increase rod stroke to five feet.

The prototype model successfully demonstrated the feasibility of flushing out corrosion products in the test facility. Shortly after installation on EBWR it was discovered that radioactivity was increasing around the blowdown valve. When the level reached 1 roentgen at the pipe surface, the radioactivity level was reduced by blowdown to 20 milliroentgens. The permanent effectiveness of the flushing system can be determined only after prolonged operation.

In addition to checking the flushability of the drive, extensive tests were conducted to determine the minimum rapid-insertion time and life expectancy of the complete mechanism attached to a control rod. The minimum insertion time for a full 56-inch stroke (the 4-inch dashpot was not included) was found to be 1.25 seconds. This rapid-insertion time was recorded after the mechanism had been through 9000 cycles at 28 inches per minute rack speed and 3000 rapid insertions. Inspection of all the wear surfaces after the tests indicated negligible wear except that the bore of the Graphitar pinion bearings increased by 0.013 inch. However, this wear did not affect the performance of the mechanism. The pinion bearings were replaced. The replacement bearings were originally sized to create an interference fit with mating parts and run-in in water until free (approximately 40 cycles run-in). Inspection of the bearings after run-in revealed a glazelike contact surface which improved wear resistance. The mechanism was reassembled and the rapid-insertion time for a 56-inch stroke, as determined with a Brush Company recorder, was found to be 1.35 seconds (see Figure 4-20).

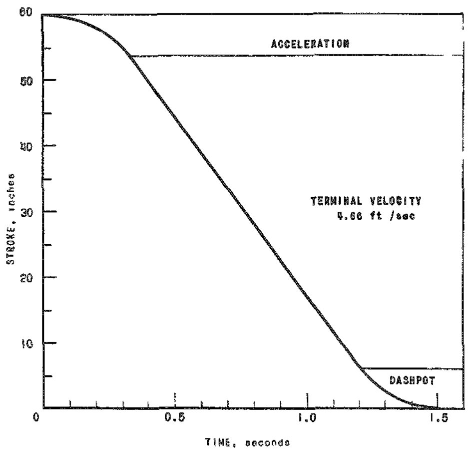

Figure 4-20

Control rod rapid-insertion curve
RE-7-34893-A

Design drawings and specifications were submitted to vendors for the purchase of nine complete drive packages, excluding seal rings, diaphragms, lanterns, synchrotransmitter and gear motors. The cost of each complete mechanism including the parts furnished by ANL was less than $4400. As the drive mechanisms arrived at ANL each unit was assembled and checked in the test facility. The checkout consisted of assembling the drive mechanism and cycling until a 56-inch insertion in less than 1.35 seconds was obtained. A tabulation of run-in data is shown below:

Drive mechanism number	Cycles required for run-in	Rapid insertion time, sec.	Seal leakage rate at 600 psi, cc/min
1	20	1.24	75
2	245	1.26	175
3	256	1.32	80
4	195	1.24	123
5	35	1.32	55
6	281	1.28	48
7	13	1.16	75
8	20	1.32	48
9	24	1.28	40

4.2.4 Control-Rod-Oscillator Drive Mechanism*

4.2.4.1 Design Objectives

The control-rod-oscillator drive mechanism is used in experiments to measure power transfer functions of EBWR. When linked to a control rod in place of the conventional rack-and-pinion drive, the rod-oscillator drive mechanism, shown in Figure 4-21, acts as a reactivity input function generator. The center control rod, a specially designed lightweight (97 pounds) hafnium rod, is oscillated by this mechanism with approximately harmonic motion over a wide range of frequencies and amplitudes; these oscillations are possible at various conditions of reactor pressure and power and at nearly any desired core position as shown in Table 4-3. Mechanical backlash, play in the linkages, and vibration are minimized to yield a very clean reactivity input signal.

Figure 4-21

Oscillator-control-rod drive mechanism
RE-6-34894-D

Basically, the oscillator drive mechanism consists of a hydraulic drive unit, a crankshaft with an adjustable crank throw, a vertical connecting rod, a piston rod, piston, and cylinder as shown in Figure 4-22. The control rod is attached to the piston by an adjustable control-rod cross-guide connector which is pinned to a lead screw that engages an internal thread within the piston. The connecting rod is jointed so that the piston can be rotated about its axis while under vertical restraint from the connecting rod. Thus, the lead screw must move up or down. In this manner, the control rod can be positioned at any setting. The rod position is read from a scale on the oscillator housing.

*J. F. Pohlman

Table 4-3. Control-Rod-Oscillator Operating Range

Frequency, cps	Amplitude, by $\frac{1}{8}$ths	Core position above seat, inches	Reactor steam pressure, psig	Power level, mw(t)
0.001 to 2.0	0 to $\pm$ 5	10 to 40	0 to 600	0 to 100
2.0 to 8.0	0 to $\pm$ 1	10 to 40	0 to 600	0 to 100

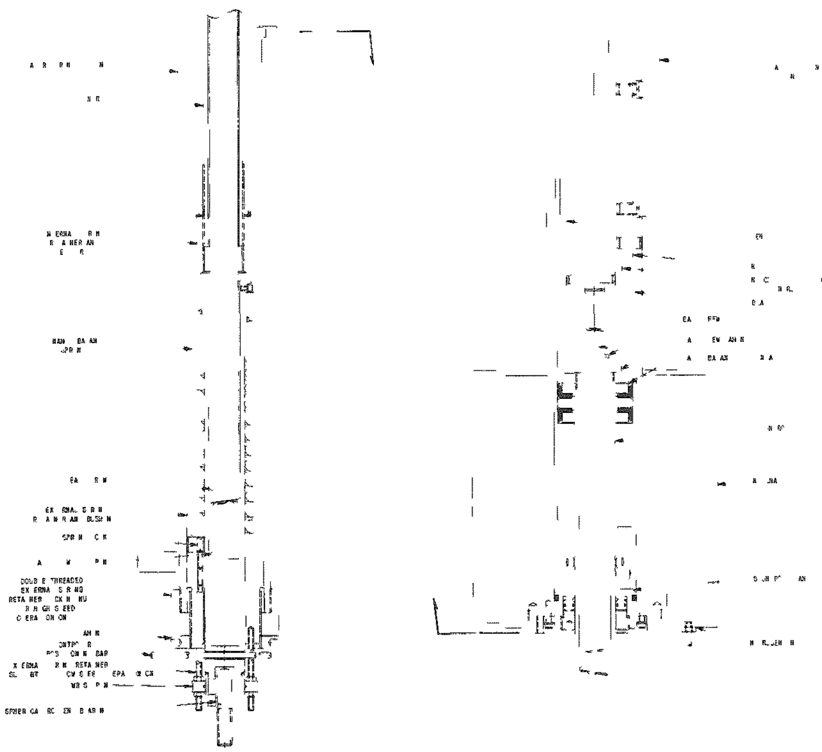

Figure 4-22

Oscillator housing assembly of oscillator control rod
RE-34895-D

The control-rod extension rod passes through an atmospheric seal; therefore, the piston must counteract both the reactor pressure and the weight of the control rod and connecting members. The latter force is constant, but the reactor pressure is variable; hence, a variable static balancing force is required. The balancing force is provided by pressurizing the underside of the piston with nitrogen. The nitrogen pressure is regulated to a value proportional to the reactor pressure. For example, at 600 psig reactor pressure, 140 psig nitrogen pressure is required for static balancing.

As the rod is oscillated, the cylinder volume below the piston is continually changing, thereby varying the nitrogen pressure within the cylinder. To keep the pressure fluctuations to a tolerable level, the cylinder volume is augmented by a reservoir. The reservoir is self-contained to eliminate any surging or pumping which might occur with an external reservoir

At low rod oscillations the inertia force is negligible and only static balancing is required. However, at oscillations above 2 cycles per second the inertia becomes appreciable and must be reduced by a system of dynamic balancing. Reduction is partially accomplished by linking the piston and the cylinder with a coil spring. The spring acts both in compression and tension, absorbing all or part of the inertia force on both the upward stroke and the downward stroke.

The spring is permanently attached to the piston but is detached from the cylinder housing at rod oscillations below 2 cycles per second. With the dynamic balancing spring in operation, amplitude is restricted to $\pm$ 1 inch

In practice, the spring balances the inertia at only one particular frequency (approximately 6 cps). At oscillations below this resonant frequency, work is done on the spring; at higher speeds, the spring absorbs only a portion of the inertia, which thus limits the top operating speed of the oscillator to 8 cycles per second.

4.2.4.2 Description

The drive unit consists of a servo-controlled, variable-speed, hydraulic transmission and a four-speed gear reducer. In the hydraulic transmission, a variable-displacement aircraft-type piston pump drives a fixed-displacement aircraft-type piston motor. The desired speed is obtained with a servo motor which positions the pump yoke by means of a worm gear and a rack-and-pinion drive, the position of the pump yoke determines the direction and rate of output flow from the pump. The usable range of this transmission is from 200 to 2000 rpm.

The gear reducer receives the output of the hydraulic pump and reduces the speed in a ratio of 1 to 4. This shaft is joined to the gear reducer output by means of a pinned, solid-flange coupling. This provides a direct drive for oscillating frequencies in the range of 2 to 8 cycles per second. A series of worms and worm gears provide three additional drive shafts with speed reductions of 10 to 1, 100 to 1, and 1000 to 1. These shafts are joined to the gear reducer output by individual electric clutches which can only be engaged separately, provided the pins from the direct-drive flange coupling are removed.

Control of the hydraulic transmission and selection and energizing of the clutches is accomplished in the reactor control room. Thus, all rod oscillating frequency changes, except the transition to the high range, are controlled remotely.

All four drive shafts contain shear pins to protect the reactor and the oscillator from possible damage due to binding of the control rod.

The output flange of the drive unit is slotted and serrated to accept a threaded and serrated T-stud which functions as a crank pin. Amplitude regulation is accomplished by matching the 8-pitch serrations at any crank radius from 0 to 5 inches. A locknut secures the connecting-rod bottom end bearing to the T-stud and locks the mating serrations in place.

An amplitude mean-reference slot is provided in the drive flange and a mean-reference hole in the gear reducer housing. When the slot and hole are aligned by insertion of a locking pin, the control rod is at the mean-reference position, or zero position, in the amplitude selector. Whenever the transmission is not in operation, the control rod is returned to the mean-reference position, the stroke is adjusted to zero amplitude, and then locked in place with the locking pin.

The connecting rod design is an economical optimization of lightness and resistance to flexure from column, bending, and inertia loadings. The rod is tapered lengthwise in an approximation of the bending stress curve, it has a large rectangular cross section to yield a high radius of gyration, and is made of aluminum to achieve light weight.

A universal-rod end bearing is used at the upper end of the connecting rod, but a caged needle bearing is required at the bottom end because of the high speed of the crank pin.

The spring housing contains the dynamic-balance coil spring which is internally threaded onto the clevis rod. The clevis rod attaches to the connecting rod and is an extension of the piston rod. The lower end of the coil spring is externally threaded into an aluminum bronze bushing. Both the inside and outside surfaces of the bushing are used for bearing surfaces. At low oscillating frequencies, the outside surface of the bushing slides within the spring housing. At oscillations above 2 cycles per second, the bushing is locked to the spring housing by means of a double-threaded nut (which locks the lower end of the spring), and the clevis rod slides within the inside surface of the bushing.

The outer support ring on the outside of the housing is used to steady the oscillator mechanism by preventing horizontal movement during oscillation.

The lower end of the piston housing contains the static-balancing cylinder, piston, piston rod, and the self-contained reservoir. The upper end of the cylinder is open to the atmosphere; the piston is single acting.

The lead screw has been machined for a $1\frac{3}{8}$ - 4-stub Acme, Class 4C thread. A precision thread is needed to keep the pitch clearance between screw and nut to a minimum. Although a lead-screw lockout is provided to hold the screw and threaded piston solid, it cannot be relied on because of the coarse thread. This is the basis for the requirement of the close fit between screw and threaded piston.

The control-rod extension rod is made from AISI 4130 aircraft quality steel hardened to a value of 48 on the Rockwell C scale and chromium plated to a minimum thickness of 0.002 inch.

The lead screw is pinned to the control-rod extension rod. This juncture provides a means of restraining the lead screw from axial rotation. A notched aluminum-bronze sleeve is fitted over the joint of the lead screw and extension rod, and all three pieces are pinned together. This notched sleeve then rides on a vertical guide key which is fixed to the inside surface of the piston housing.

A series of vertical slots in the housing wall permits access to a pointer located on the notched sleeve. Rod position is indicated by the pointer on a scale attached to the piston housing.

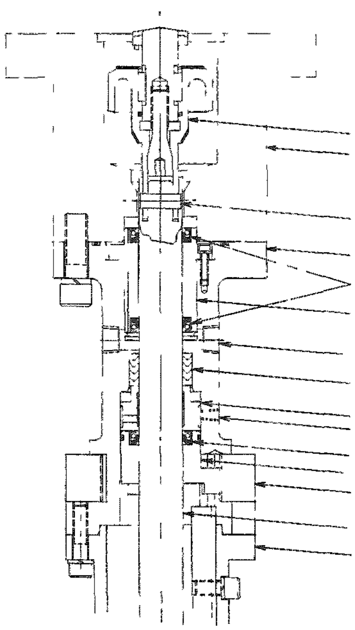

Figure 4-23

Oscillator-seal-housing assembly of oscillator control rod RE-6-34896-C

The seal housing shown in Figure 4-23 provides the guide bushing and atmospheric seal for the control-rod extension rod. The seal is provided by a stack of three Teflon V-rings $1\frac{1}{2}$ inches in inside diameter held in place by an externally threaded packing clamp and a split nut. The split nut, restrained from vertical movement by the seal housing, forces the packing clamp upward against the V-rings and is locked at the proper position. In an emergency, this design allows the seal housing to become a stuffing box.

Teflon V-rings are used because of their low friction, although a mica-coated homogeneous V-ring could be substituted. Wipers are provided both above and below the seal to keep the seal reasonably free of particulate matter. The V-rings can be easily replaced, if necessary, when the oscillator drive mechanism is removed from the reactor at the completion of a test run.

The aluminum-bronze guide bushing is also provided with wipers above and below to keep out foreign material. The bushing is longitudinally slotted along its periphery to provide a pressure bypass and to allow cooling of the bushing. Two tapped holes through the housing immediately below the bushing can be used for thimble cooling or back flushing.

Installation of the solid connector used to join the boron-stainless steel control rod to the rack-and-pinion drive depends on clearance between the connector and the control-rod end fitting. Since any clearance is objectionable in an oscillator application, the redesigned connector clamps tightly to the control-rod end fitting to eliminate end play. The larger bearing surface of the redesigned connector greatly reduces the stress reversals occurring in the rod end fitting during oscillation.

The adjustable cross-guide connector is fully interchangeable with the rack-and-pinion drive or with the boron-stainless steel control rods.

4.2.4.3 Testing and Installation

The oscillator drive mechanism received limited testing on the control rod mockup test facility. Space limitations prevented use of the full length connecting rod or the use of amplitudes greater than $\pm$ 2 inches. Lack of rigidity of the test stand prevented operating at high speed for any length of time. The testing, although limited, was entirely satisfactory and informative.

The mechanism was also tested in the reactor during a shutdown period. The drive functioned satisfactorily throughout the entire range of frequencies, amplitudes, and elevations listed in Table 4-3. Observation of the oscillating control rod through the open top of the reactor vessel revealed that, although the rod rubs against the control rod channel, no appreciable vibration in the reactor vessel structure resulted.

The gear reducer is bolted to an I-beam pad sunk into the subreactor room floor to achieve sufficient rigidity; the entire drive unit is a semipermanent fixture which will be removed only during extensive layoffs between transfer-function measurements or when the space is needed for other purposes. The housing assembly will be installed and removed as needed, since this involves only the steps required to change a conventional rod drive.

4.3 FUEL ELEMENTS*

4.3.1 Fuel Elements For 100-Megawatt Core

The spike fuel elements are composed of two units: (1) a fuel element frame and (2) the fuel rod assembly which is removable from the frame. The fuel assembly, shown in Figure 4-24, is a rectangular boxlike structure 226.7 centimeters ($89\frac{1}{4}$ inches) long and 9.84 centimeters ($3\frac{7}{8}$ inches) square.

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*V. M. Kolba

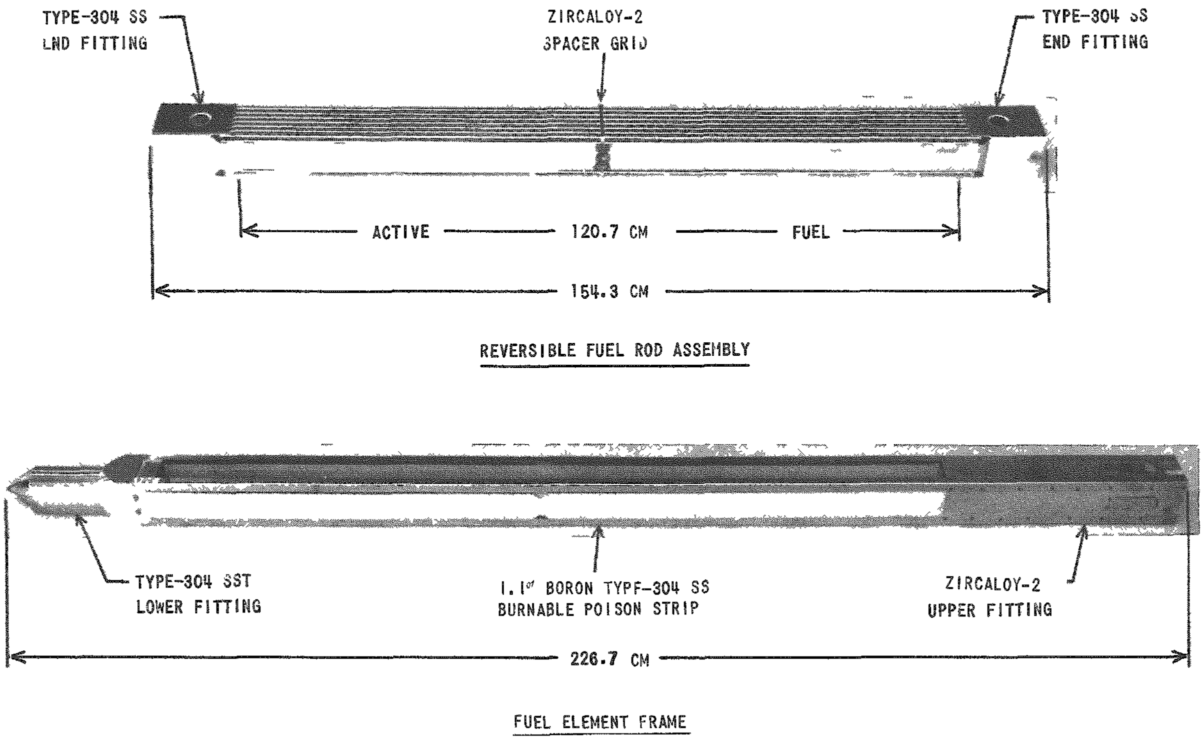

Figure 4-24

EBWR spike-fuel-rod assembly and fuel element frame

The components of the fuel element frame, shown in Figure 4-25, are the lower end fitting, the angles, and the top lifting plates.

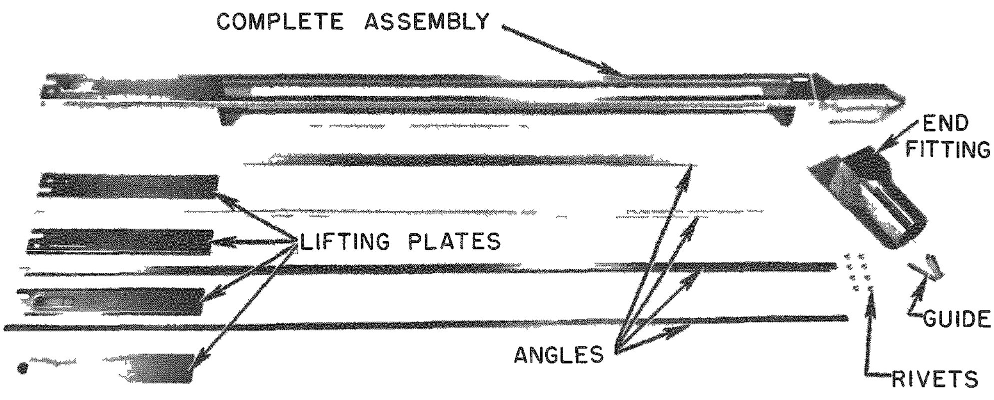

Figure 4-25

EBWR spike-fuel-element frame and components
Neg. 106-5283

The end fitting is finish machined from a type-304 stainless steel casting. A tapered guide tip is provided on the bottom end to guide the assembly into the locating holes of the grid plate. A single chamfered seat is machined on the periphery of the casting to provide a seating area for the assembly when in position in the support plate. A recessed area is machined in the upper surface of the fitting to provide a seating area for the angles and to eliminate all projections that might cause interference during fueling operations.

The angles are formed from 0.206-centimeter (0.080-inch) Zircaloy-2 sheet material and provide the connecting link between the top lifting plates and the lower end fitting. In addition these angles form the basket which accepts the fuel rod assembly. A 0.103-centimeter ($\frac{1}{24}$-inch) deep recess, 0.953 centimeter ($\frac{3}{8}$-inch) wide by 46.04 centimeters ($18\frac{1}{4}$ inches) long, is milled in the inside of each leg of the angles at the end connected to the lifting plates.

The lifting plates are made of 0.206-centimeter-thick (0.080-inch) Zircaloy-2 plate. A 0.953-centimeter-wide ($\frac{3}{8}$-inch) 0.103-centimeter-deep ($\sim\frac{1}{24}$-inch) lip is milled on each plate edge for the entire length of the lifting plate. An elongated hole and a slot are punched in the top end of each plate. A larger hole is punched somewhat lower in the plate to provide for a leaf spring in the compressed condition.

Leaf springs for spacing the fuel spikes are sheared and formed from Zircaloy-2 sheet material. The leaf spring is spot welded to the lifting plate. Four of the lifting plates are lap joined, by spot welding, to the angles to form the top lifting plate. These four angles are riveted to the end fitting to complete the fuel element frame. Four elongated holes in the top lifting plates provide lifting points for the pins of the fuel-handling tool. At a given time only two holes, those on opposite sides, are used in lifting. Slots are provided adjacent to the holes. These slots permit the pins to extend sufficiently through the holes for proper lifting without engaging adjacent fuel elements since the pins will slide in the slots of adjacent elements.

The burnable poison strips for the spike fuel elements are fabricated from nominally 1.1 percent natural boron-304 stainless steel. Each strip is 155.9 centimeters ($61\frac{3}{8}$ inches) long, 5.8 centimeters ($2\frac{9}{32}$ inches) wide, and 0.159 centimeter ($\frac{1}{16}$ inch) thick. A slight step is formed on one end of the strip to permit positioning of the upper end of the poison strip. A reduced section is provided at the center of the strip to permit future remote removal. The poison strips are shown in the view of the fuel element frame in Figure 4-24.

The strips may be attached in any of four positions on the fuel element frame. The strips are supported on a 0.206-centimeter (0.080-inch) ledge and tack welded to the end fitting casting. On assembly the strip is bowed out and the upper end is tucked into the space between the end fitting of the fuel rod bundle and the lifting plate of the fuel element frame. The 1.1 percent boron-stainless steel is corrosion resistant and has less notch sensitivity than the 2 percent boron-stainless steel material of the control rods. The reactivity worth of a boron-stainless steel strip has been determined by cold critical experiments and is reported in ANL-6305.(11) One strip is worth approximately 0.09 ± 0.035 percent reactivity.

Since the strip rests on a ledge, no stress is exerted on the tack weld by the strip when in the reactor. With two strips 180 degrees apart, there is also a snug fit at the top of the fuel element between the strips, fuel bundle end-fitting box, and the lifting plate of the fuel element frame.

This form of assembly makes possible the remote removal of the irradiated poison strip in the storage pit should this be desired.

The components of the fuel rod assembly are the two end-fitting boxes, the fuel rods, the two side plates, and the center spacer grid. These components are shown in Figure 4-26.

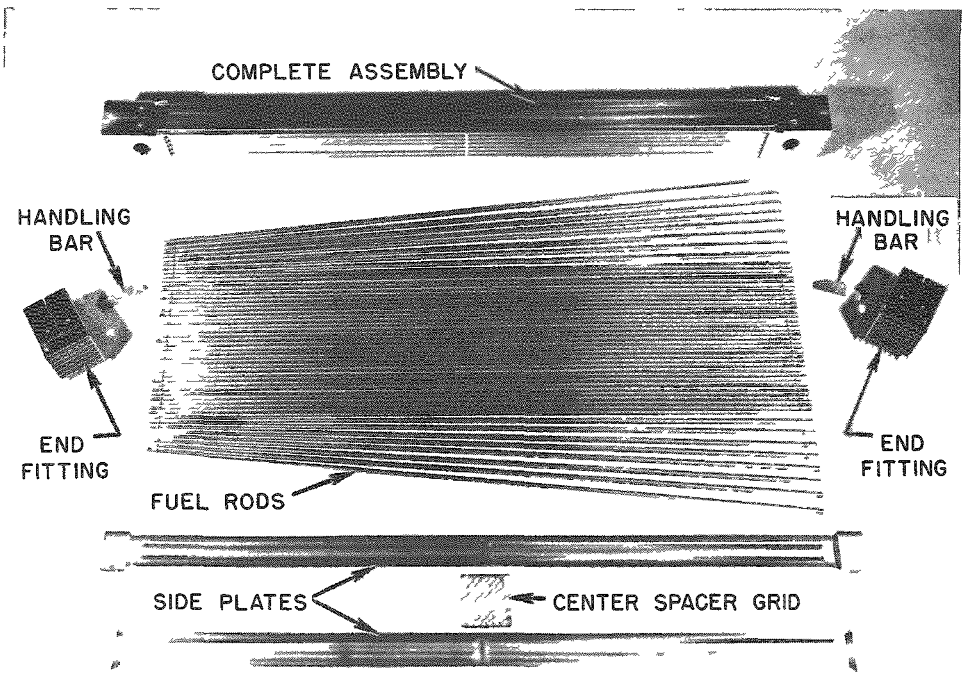

Figure 4-26

EBWR spike-fuel-rod assembly and components
Neg. 106-5282

The end-fitting box is made of 0.159-centimeter-thick ($\frac{1}{16}$-inch) stainless steel. The sheet steel is formed into a channel, and two channel sections are heliarc welded together to form a box section. The box section is cut to length and one end is slotted to accept the fuel-rod support grids and the other end is slotted to accept the handling bar. Two holes are formed in the box sides to permit handling for inversion of the fuel unit. The fuel-rod support grid is a comb which spaces and guides the fuel rods.

The comb is machined from 0.159-centimeter-thick ($\frac{1}{16}$-inch) stainless steel. Seven combs are required for each end fitting and are welded into the machined slots to form the completed box. The handling bar is a flat 0.317-centimeter-thick ($\frac{1}{8}$-inch) stainless steel sheet with the head slightly tapered to allow for seating on the fuel-element-frame end fitting. One handling bar is required for each end box.

The side plates are made from Zircaloy-2 material 0.159 centimeter ($\frac{1}{16}$ inch) thick with the ends stepped to permit fastening to the inside of the end-fitting box. The longitudinal ribs are formed in the side plates to supply additional rigidity. These plates are riveted to the stainless steel end-fitting boxes and serve to tie the end boxes together. The fuel rods slide freely and are merely guided on the support grids.

The center spacer grid, which is made of Zircaloy-2 material, maintains proper spacing of the fuel rods. The grid is machined from a solid sheet of Zircaloy-2 material by drilling, milling, and filing. This eliminates the corrosion testing required for a welded assembly and permits more positive placement of the fuel rod.

Fuel rods for the spike elements are shown in Figure 4-27. The reference-design fuel rod consists of ceramic pellets, cladding, and two end plugs.

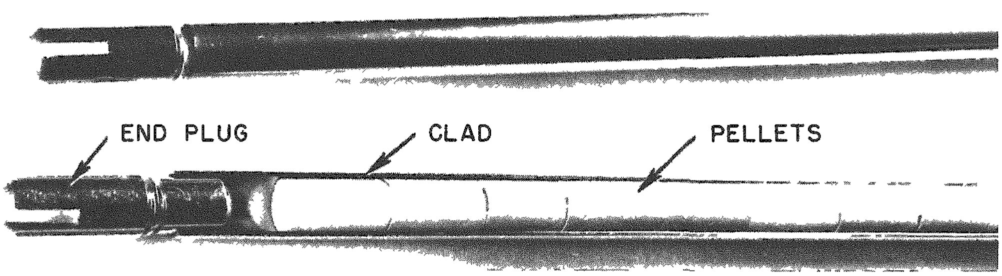

Figure 4-27

Typical spike-fuel rod
Neg. 106-5281

The ceramic pellets are prepared from urania, zirconia with low hafnium content, and calcia powders. The UO_2, ZrO_2, and CaO are blended, cold pressed, and sintered to form a solid solution. The pellets, 0.815 centimeter OD by 1.27 centimeters long (0.321 inch by 0.5 inch) are used in the as sintered condition. The end plug is machined from Zircaloy-2 bar stock. Zircaloy-2 cladding material is purchased as drawn seamless tubing of 0.978 centimeter (0.383 inch) outside diameter with a

0.064-centimeter-thick (0.025-inch) wall, and finish drawn to a 0.95-centimeter ($\frac{3}{8}$-inch) outside diameter. An end plug is welded to the Zircaloy-2 clad tube and the pellets placed in the Zircaloy-2 tube. After insertion of the pellets, the tubes are evacuated and backfilled with helium and the second end plug is welded in place. A helium atmosphere is maintained in the core-clad gap. Slots are milled in the Zircaloy-2 end plugs for positioning in the fuel support grid. Before final assembly into the fuel boxes, a percentage of the fuel rods are corrosion tested for two weeks at 550°F saturated conditions.

Prior to final fuel rod assembly, the end fittings for the fuel rod bundle are completed except for the handling bar. The two side plates are riveted to one of the end fittings and plug welded to the positioned center guide-spacer grid. Forty-nine fuel rods are slipped through the center guide-spacer grid and into the grid of the end fitting. The second end fitting is then positioned at the opposite end, slid into place and riveted to the two side plates. The final assembly operation consists of welding the handling bars to the end fittings. Thereafter, the reversible fuel rod bundle is slipped into the fuel element frame for loading into the reactor. A cross section of the spike element is shown in Figure 4-28. All subsequent handling is done with the fuel rods in the fuel frame. The method of assembly of the spike fuel permits:

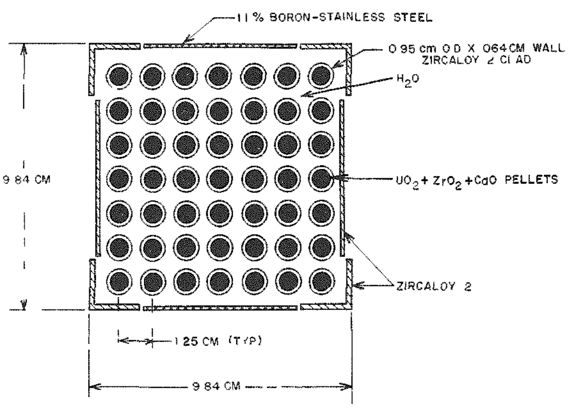

Figure 4-28

Cross section of spike-fuel element
Neg. 111-9496

1. Individual axial expansion of the fuel rods.
2. No riveting or fastening of the Zircaloy-2 clad fuel rods to the stainless steel end-support grids.
3. A removable and invertible fuel assembly. This also permits the reuse of the frame for sample element irradiations.

4.3.2 Fuel Element Performance in 20-Megawatt Core

Since going critical in December 1956, the EBWR core has operated a total of 11,164 hours, which represents 211,509 megawatt-hours. The average power level was approximately 18.9 megawatts. A great portion of the initial operation was at relatively low power; however, both a sustained operation of 40 megawatts was made and a short time peak output of approximately 62 megawatts (thermal) was reached. The core contained nominally 5,508 kilograms (12,140 pounds) of uranium during operation from 1956 to 1959. The calculated average burnup of the U-Zr-Nb alloy fuel was approximately 1600 megawatt-days per metric ton.

The fuel elements were visually examined several times. In January 1958 a rather intensive investigation of the core was made by gamma probe and ferrous sulfate dosimetry.(12) This work revealed that the center 36 fuel elements were producing approximately 50 percent of the power and that the fuel element in the center of the cell of nine had the highest gamma flux and possibly was in the highest neutron flux.

At this time it was also noted that there was considerable scale deposit on several fuel elements. In one element, ET-51, this scale flaked off, and was measured to be about 0.0076 centimeter (0.003 inch) thick.

Two fuel elements have been removed from the core and examined metallurgically.(13) Table 4-4 presents data of fuel elements T-23 and ET-51. Figure 4-29 shows the core location of the two elements.

Table 4-4. Fuel Elements Data

Fuel element	T-23	ET-51	
Position	129-53	101-266	
Date removed	1/2/58	4/20/59	
Megawatt hours	72,963	196,091	
Average core burnup, mwd/tonne	551.9	1,483.3	
Fuel plate analyzed	No. 606	No. 828	
Measured burnup			
Average atomic percent	~0.06	~0.21	
Average mwd/ton	~600	~2,100	
Maximum atomic percent	0.11	0.33	0.39[a]
Maximum mwd/ton	1,100	3,300	3,900[a]

[a]Edge of plate values

On January 2, 1958 T-23, a natural-uranium-bearing thin fuel element, was removed from the core and sent to the hot cells for examination. There was no evidence of plate warpage, swelling, or excessive scale deposit. The end fittings were firmly attached and ultrasonic tests indicated complete bonding of the clad and core.

On April 20, 1959 ET-51, an enriched-uranium-bearing thin fuel element, was removed from the core and sent to the hot cells for examination. A foil probe had stuck in the channel of the element. This element occupied the center position of one of the inner cells of nine fuel elements, and gamma probe and dosimetry measurements had indicated maximum burnup. Hot cell examination of the element showed a high burnup compared to the calculated average value of the core. Fuel element ET-51 was found to be in good condition with no ruptured cladding, good clad-to-core bonding, and no excessive plate warpage or swelling. The clad was found to be ductile, but the core was extremely brittle and hard (a hardness of 50 on the Rockwell C scale). Some evidence of hydrogen pickup was found in portions of the clad. Scale formation was observed in varying thickness; scale that spalled off was approximately 0.013 centimeter (0.5 mil) thick. Composition of the scale is nominally $Al_2O_3 \cdot H_2O$, boehmite, with some nickel and iron also present.

At core operating conditions of 400°C and approximately 0.4 atomic percent burnup a volume increase of 6 to 7 percent per atomic percent burnup was found. Calculated integrated exposure of this fuel clad and core was approximately 1.6×10^{21} nvt (fast) at a temperature of 260-325°C.

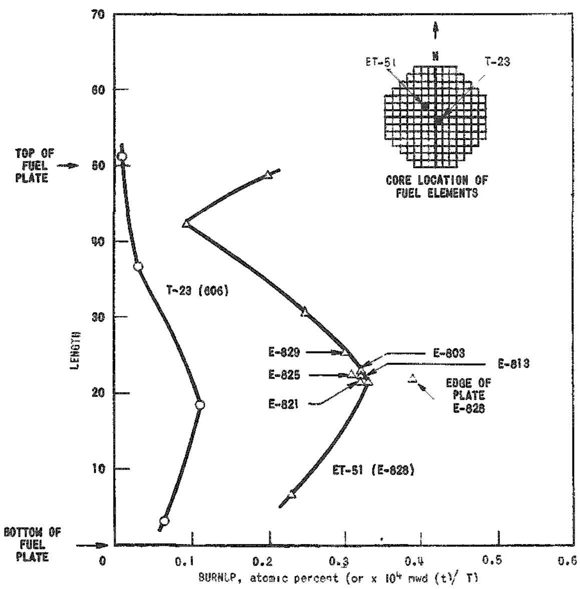

Figure 4-29

Axial burnup of EBWR fuel
RE-7-34897-A

The plot in Figure 4-29 of the axial burnup of the fuel in EBWR shows a distorted flux pattern with the bulk of the burnup being in the lower 60 percent of the core. On comparison it appears that the maximum burnup point between T-23 and ET-51 has shifted slightly. This may be due to either (1) operating on reducing reactivity with the rods progressively further out or (2) to a nonuniform axial flux pattern in the radial direction.

The subject of scale formation on the fuel plates is discussed in the Appendix (Section 6.2).

4.4 FUEL ELEMENT AND CONTROL ROD HANDLING*

4.4.1 Coffin Redesign

The coffin shown in Figure 4-30 is a hollow, thick-walled, lead cylinder 32 inches in outside diameter. The diameter of the center cavity is stepped from $11\frac{1}{4}$ inches at the lower half to $13\frac{1}{4}$ inches at the upper half. This large center opening permits the coffin to be used for the removal of the control rod. A removable, lead-filled, shield plug with a $6\frac{3}{8}$-inch internal diameter is inserted into the coffin for handling fuel elements. A rotating plug which carries the fuel retrieving tool assembly is mounted at the upper end of the coffin opening. The lower opening is closed by a motorized door which is provided with a watertight seal; this permits the center cavity to be flooded should cooling be necessary during fuel transfer.

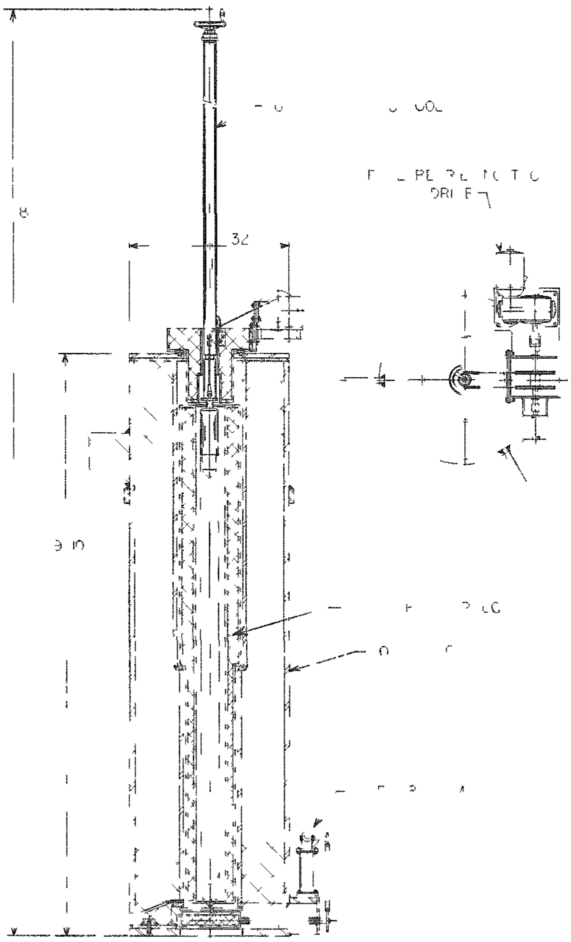

Figure 4-30

Fuel transfer coffin
RE-6-32077-E

Although only 6 inches taller than the original coffin assembly, the new coffin can handle irradiated fuel or a 94-inch-long section of a control rod.

The only major change to the carriage was to motorize the bridge.

4.4.2 Fuel Removal

The fuel storage rack is monitored by instrumentation during the entire fuel transfer operation. The neutron-count rate is recorded after each fuel element is placed in the storage rack and the reciprocal count rate is plotted to detect any change in multiplication. During the storage of fuel from the 20-megawatt reactor no net increase in the count rate throughout the fuel additions was observed; this indicates that the boron-stainless steel in the fuel storage rack [see description in ANL-5607,(1) p. 67] keeps the stored assembly of fuel subcritical.

*W. J. Kann

4.4.3 Control Rod Removal

Prior to making the extensive pressure vessel modifications delineated in Section 4.1.1, it was necessary to remove the control rods from the reactor. To facilitate the removal of the irradiated control rods and make use of the new coffin cylinder, the rods were first cut into three pieces approximately 60 inches long.

The saw shown in Figure 4-31 was used to cut the control rods. The commercial, air-driven guillotine saw, modified slightly to operate under water, was mounted on a platform which contained four knife-edge grippers. The grippers kept the control rod from returning to the shroud during the cutting operation. Two hydraulic power cylinders equipped with special locking jaws were used to position and hold the control rod during the cutting operation.

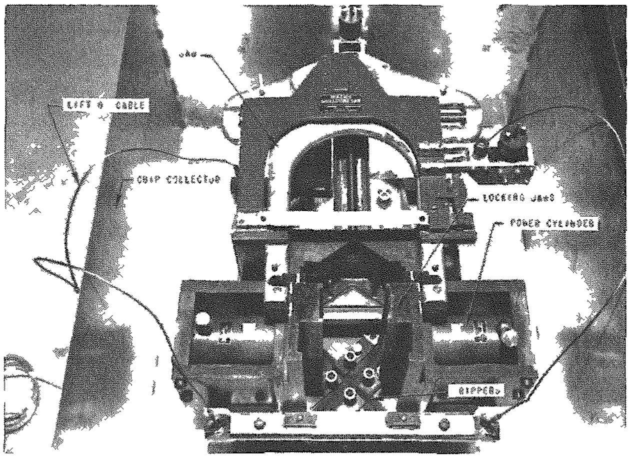

Figure 4-31

Control rod cutoff tool
Neg. 111-8202

The saw, grippers, and power cylinders were put in a sheet metal box (chip collector) to minimize contamination within the reactor. The cruciform slot in the box was covered by a flexible diaphragm, cut to fit tightly against the control rod, to deflect the cutting chips into the box.

Each rod was handled essentially in the same manner. The saw was located over a control rod opening and set on the shroud structure. The control rod was then pulled up into the first cutting position using the control-rod handling tool (see ANL-5607,[1] p. 42). The power-cylinder locking jaws were then clamped on the control rod. The handling tool was removed and a hydraulic collet was attached. The hydraulic collet was used to pull the cut pieces into the coffin. During the control rod removal, the telescoping fuel-retrieving tool was removed from the coffin assembly. For the second (last) cut, the hydraulic collet was used to pull the piece held by the knife-edge grippers into the cutting position. After the cutting, the pieces were brought into the coffin and transferred to the fuel storage pit.

The removal of the nine control rods was accomplished with only the normal difficulty that had been anticipated. The number 9 rod was found to have separated at the Hf-Zr joint (see ANL-5607, Figure 21, p. 30). The separation was probably caused by the power transfer-function experiments in which the rod was cycled repeatedly at high frequencies.

All of the rods were cut under 12 feet of water. The radiation background at the surface of the water was negligible. The maximum radiation fields around the reactor top and storage pit during the transfer of a cut control rod section were: 6 to 25 roentgens per hour for transfer from reactor vessel to coffin, and 30 to 750 milliroentgens per hour for transfer from coffin to storage pit. The major radiation hazard occurred during the cleaning of the saw after each control rod had been cut and stored. The radiation level of the chips in the chip collector was from 15 to 50 roentgens per hour, but the accumulation in the vacuum cleaner trap used for removing the chips gave readings as high as 100 roentgens per hour (hard) at 1 inch. The maximum dosage obtained by any operator during saw cleaning was 70 milliroentgens.

The entire removal operation required approximately 90 hours from the first cut until the last control rod piece was stored.

4.4.4 Tool Redesign

The 100-megawatt-reactor fuel-handling tool is operated in the same manner as the original fuel-handling tool (see description in ANL-5607, p. 61). The lifting cables for vertical travel have been motorized and the design of the telescoping-fuel-retrieving-tool head assembly has been modified slightly. The latch pins are not diametrically opposite, but are offset to match the holes in opposite sides of the top lifting plates (see Figure 4-24). The design of four holes and four slots in the top lifting plates eliminates the need of aligning the tool for rotation since two holes for lifting are available at all times.

The lower telescoping actuator tube has been keyed to permit locking or unlocking of the latch pins during any part of its extension. This permits use of the tool at various levels within the core or fuel storage well.

Since the original fuel-handling tool and 100-megawatt-reactor fuel-handling tool operate in the same manner, they can be interchanged easily by removing a pin at the bottom end of the lower telescoping actuator tube and exchanging tool heads. This easy exchange feature will be necessary during the period when both 20-megawatt and 100-megawatt-reactor fuel are to be in use.

5 REBOILER PLANT COMPONENT DESCRIPTION*

5 1 PRIMARY SYSTEM COMPONENTS

5 1.1 Reboiler

The two primary-system reboilers are located within the shielded cell inside the reboiler building. The reboilers are floor-mounted, horizontal, single-effect type with a two-pass tube side steam condensing circuit (see Figure 5-1) The primary steam from the reactor on the tube side of the reboiler provides the necessary heat to produce steam from the intermediate-system water on the shell side of the reboiler.

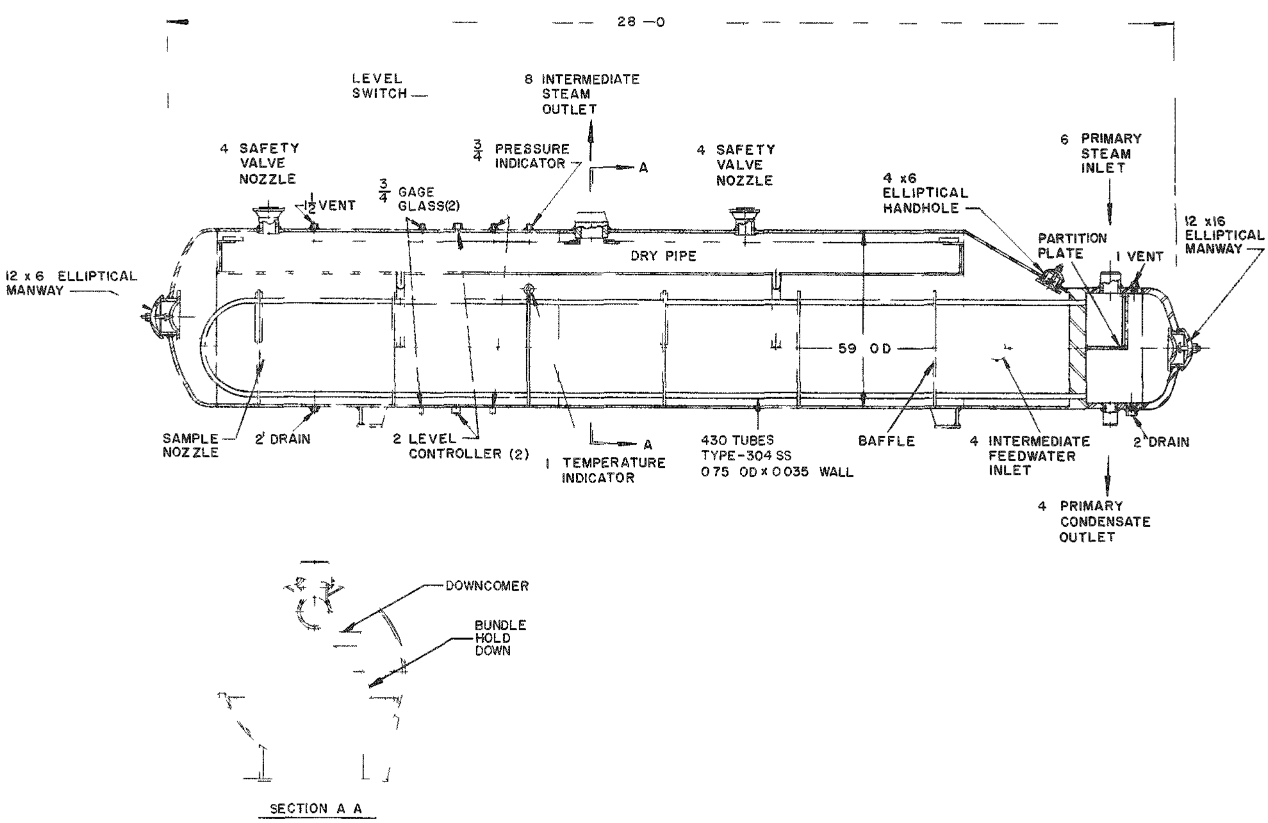

Figure 5-1

Primary reboiler
RE-6-32683-D

The reboiler is a conventional kettle type with U-tube bundle construction The shell is typical of heat exchangers designed for vapor generation since it provides a large volume over the boiling liquid This permits the vapor to travel at a sufficiently low velocity toward the outlet so that entrained liquid settles out before the vapor leaves the shell The shell outlet connection is located at approximately the center of the tube

*E L Martinec

bundle length. This location allows a maximum volume of vapor to leave the shell for a given vapor velocity since no vapor has to traverse more than half the dome area to exit. Entrained liquid is further disengaged by a dry pipe which is attached to the outlet nozzle and extends the length of the vapor space. The operating conditions and capacities of the reboiler units are given in Table 5-1.

The reboiler design conforms to the design requirements of Section VIII of the ASME Code and also, where applicable, with TEMA standards for Class A construction. The shell, tube sheet, and channel are welded together to form an integral unit without flanged joints. As indicated in Table 5-2, all construction materials are carbon steel except where the primary-reactor fluid is in contact with the surface material. All surfaces in contact with the primary fluid are stainless steel. In the channel end all nozzles and channel partition plates are type-304 stainless steel, and the inside surfaces of the channel shell and bonnet and the channel side of the tube sheet are clad with $\frac{3}{16}$-inch-thick arc-deposited type-304 stainless steel. Access to the shell and channel end are made through two manways, one at each end of the vessel. This permits inspection of the tube bundle for leakage at the tube-seal welds at the tube sheet, corrosion or erosion of the exterior of the tubes, and stress cracking at the tube bends.

The tube bundle is of two-pass U-tube design. The tubes are rolled and seal welded to the tube sheet to eliminate any leakage of the primary fluid into the intermediate system. Any leakage past the tubes will cause contamination of the intermediate system by radioactive corrosion products and other radioactive materials.

Table 5-1. Operating Conditions and Capacities of the Primary Reboiler

Reboiler tube surface	4000 sq ft
Heat exchanged - maximum	111×10^6 Btu/hr (32.5 mw)
Quantity of primary steam to tube side	150,000 lb/hr (max)
Tube side pressure - operating	560 psig
- design	800 psig
- test	1500 psig
Tube side temperature - operating (both steam inlet and condensate outlet)	482°F
- design	520°F
Quantity of intermediate steam leaving shell side	140,500 lb/hr (max)
Shell side pressure - operating	350 psig
- design	400 psig
- test	600 psig
Shell side temperature - operating (both water inlet and steam outlet)	435.7°F
- design	448°F
Steam quality leaving shell dry pipe	98% (approx)
Fouling factor	0.0005
Shell side water capacity at normal operating conditions	1925 gal
Weights - dry	31,500 lb
- operating	47,500 lb
- flooded	59,500 lb

Table 5-2. Primary Reboiler Materials of Construction

Shell	ASTM-A-212 Gr B Firebox quality steel
Shell weld stubs	ASTM-A-105 Gr 1 Forged steel
Channel	ASTM-A-212 Gr B Firebox quality steel with minimum $\frac{3}{16}$-in.-thick arc-deposited AISI-304 cladding on inside surface
Channel weld stubs	ASTM-A-240, Type AISI 304
Tube sheet	ASTM-A-212 Gr B Firebox quality steel with minimum $\frac{3}{16}$-in.-thick arc-deposited AISI-304 cladding on tube side
Channel partition rib and cover	ASTM-A-240, Type AISI 304
Tubes	ASTM-A-249, Type AISI 304L
Shell handholes and weld coupling	Steel

5.1.2 Reboiler Drain Tank

Two primary-reboiler drain tanks serve as accumulators for the condensate from the primary reboilers. The design of the vessel conforms to Section VIII of the ASME Code The 30-inch-ID by 6-foot-long tanks are horizontally mounted between each reboiler and drain cooler within the shield cell of the reboiler house. The vessels are fabricated entirely of stainless steel (ASTM-A-240 type 304). Two 6-inch-diameter inspection openings have been provided: one on the top of the shell and one in the south head. The 4-inch pipe-size inlet and outlet nozzles are located tangentially to the top and bottom of the shell respectively. Pipe couplings are provided for instrumentation and operational purposes, such as level control, level alarm, vents, and drains.

5.1.3 Drain Cooler

Two primary-system drain coolers are located on opposite walls of the shielded cell in the reboiler building. Drain cooler No. 1 is against the west wall of the cell and drain cooler No. 2 is against the east wall. Each drain cooler is comprised of two shells connected in series, mounted one above the other (see Figure 5-2). The primary-steam condensate from the primary-reboiler drain tank flows through the tube side of the drain cooler and provides the heat necessary to raise the temperature of the intermediate-system water, on the shell side of the drain cooler, to the saturation temperature. Drain cooler operating conditions and capacities are given in Table 5-3.

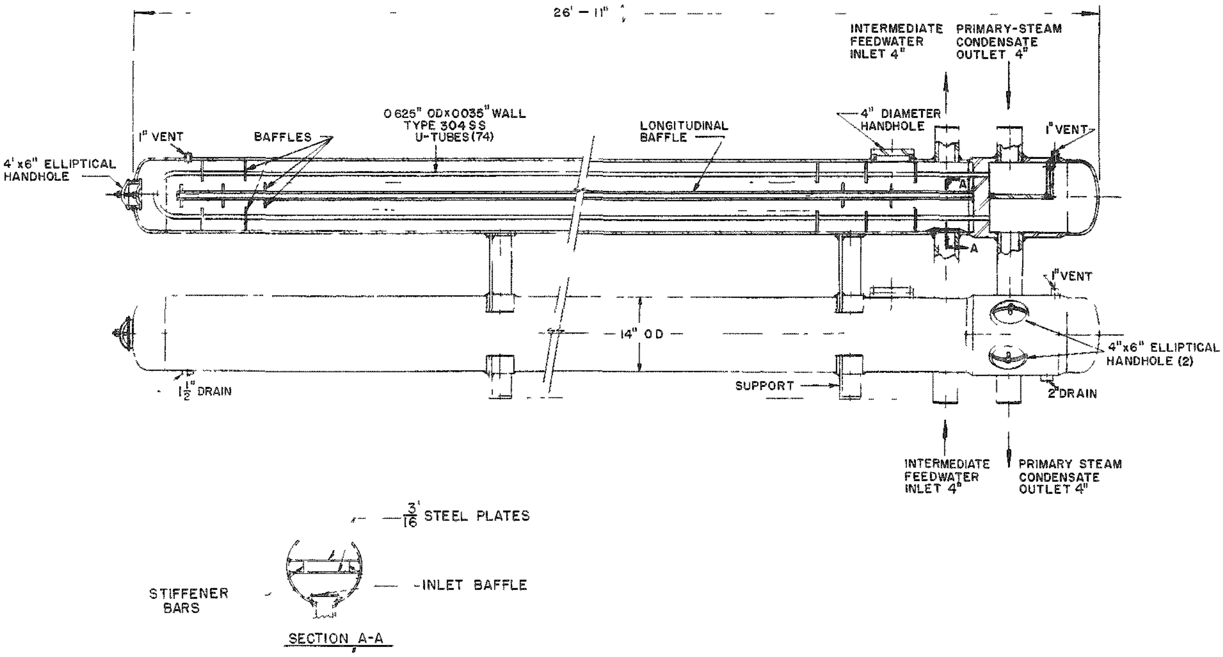

Figure 5-2

Primary drain cooler
RE-6-32686-D

The primary drain cooler is similar to a conventional horizontal liquid-to-liquid heat exchanger of two-pass shell side and two-pass tube side design. The tube bundle is of U-tube construction. The exchanger is of all welded construction, i.e., integral shell, tube sheet, and channel end. The drain cooler design conforms to the design requirements of Section VIII of the ASME Code and also, where applicable, with TEMA standards for Class A construction. The materials of construction are given in Table 5-4.

Each drain cooler unit consists of two separate shells and tube bundles with each being a two-pass design. The shells are mounted horizontally, one on top of the other. A longitudinal baffle separates each shell into two passes. The baffle is constructed of two sheets of carbon steel, $\frac{3}{16}$ inch thick, reinforced by two 1-inch stiffener bars. Each sheet is welded along its periphery to the shell and tube sheet to minimize leakage of fluid across the baffle.

The tubes are rolled and seal welded to the tube sheet to eliminate any leakage of primary fluid on the tube side into the intermediate fluid on the shell side. An impingement plate at the shell-side inlet prevents erosion of the tubes in this region from the influent water. A handhole at the head end of the shell enables the inspection of the tubes at the tube bends for stress cracking. Handholes on the channel side of the tube sheet allows inspection of the tube ends for leakage at the seal welds.

Table 5-3. Operating Conditions and Capacities of the Primary Drain Cooler

Drain cooler tube surface	1080 sq ft
Heat exchanged - maximum	35.4×10^6 Btu/hr (10.4 mw)
Quantity of primary steam condensate to tube side	132,200 lb/hr
Tube side pressure - operating	560 psig
- design	800 psig
- test	1500 psig
Tube side temperature - operating (inlet)	482°F
(outlet)	231°F
- design	520°F
Quantity of intermediate feedwater to shell side	123,500 lb/hr
Shell side pressure - operating	350 psig
- design	400 psig
- test	600 psig
Shell side temperature - operating (inlet)	160°F
(outlet)	435.7°F
- design	448°F
Fouling factor	0.0007
Weights - dry	10,000 lb
- flooded	14,200 lb

Table 5-4. Primary Drain Cooler Materials of Construction

Shell	ASTM-A-106 Gr B
Shell nozzles	ASTM-A-106 Gr B or ASTM-A-53 Gr B Seamless steel
Shell head	ASTM-A-234 Gr WPB Seamless steel
Channel	ASTM-A-240, Type AISI 304
Channel nozzles	ASTM-A-312, Type AISI 304
Tube sheet	ASTM-A-182, Type AISI 304 - Forged
Channel partition rib	Type AISI 304
Tubes	ASTM-A-213, Type AISI 304
Shell handholes and weld couplings	Carbon steel
Shell longitudinal baffle	Carbon steel plate

5.1.4 Subcooler

The primary-system subcooler is located inside the EBWR containment vessel at the 696-foot elevation. The subcooler, as shown in Figure 5-3, is a conventional shell and U-tube heat exchanger with condensate from the primary system on the shell side and cooling-tower water on the tube side. The condensate leaving the deaerator can be cooled in the subcooler to any desired temperature before being pumped to the reactor. Also, cooling is available, if necessary, to increase the NPSH available to the pump, thus eliminating any possible cavitation. The operating conditions and capacities are given in Table 5-5.

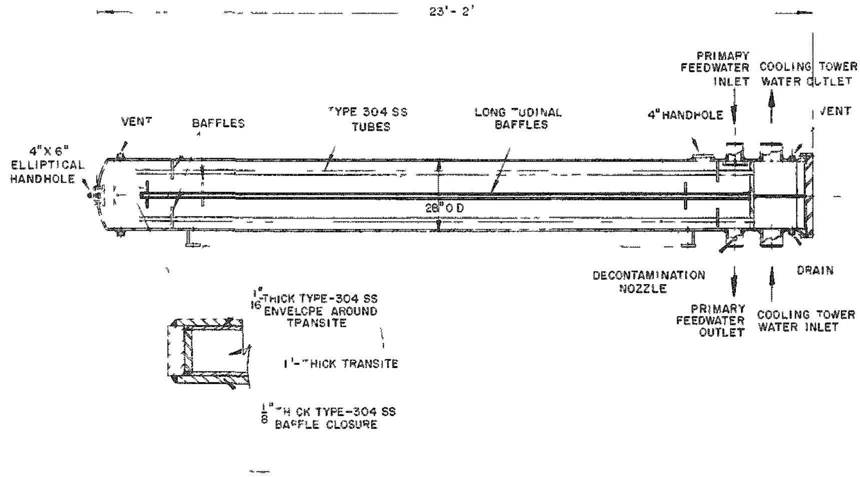

Figure 5-3

Primary subcooler
RE-6-32681-D

The subcooler is a conventional, horizontal, liquid-to-liquid heat exchanger with a two-pass shell side and two-pass tube side design. The tube bundle is of U-tube design. The exchanger is an integrally welded shell, tube sheet, and channel, with a flanged channel closure. The subcooler design conforms to the design requirements of Section VIII of the ASME Code and also, where applicable, with TEMA standards for Class A construction.

The shell, being of two-pass design, is separated by a longitudinal baffle. The baffle design is similar to the drain cooler baffles except that a 1-inch-thick Transite sheet is included as a thermal barrier. The Transite is completely enclosed in a $\frac{1}{16}$-inch-thick stainless steel sheath which is inserted between two $\frac{1}{8}$-inch-thick stainless steel sheets welded to the shell. This provides two separate barriers to any leakage path. Leakage might leach out the Transite used as a thermal barrier between the two passes and result in contamination of the primary fluid. Each envelope is helium tested for leakage to insure a leak-tight weld joint. This type of baffle construction increases the exchanger efficiency by eliminating any short circuiting between shell passes and by decreasing heat conduction through the baffle.

The tubes are rolled and seal welded to the tube sheet to eliminate any leakage of primary fluid from the shell side into the cooling-tower water on the tube side. An impingement plate, installed at the shell-side inlet, prevents erosion of the tubes in this area. Handholes at the shell side of the tube sheet and at the shell head allow inspection of the exterior of the tubes. The channel cover is removable for inspection of the tube-seal welds and to give access to the tubes to plug any leaks which may occur.

Table 5-5. Primary Subcooler Operating Conditions and Capacities

Subcooler tube surface	1943 sq ft
Heat exchanged - maximum	17.25×10^6 Btu/hr (5.05 mw)
Quantity of primary condensate to shell side	235,570 lb/hr
Shell side pressure - operating	10 psia
- design	100 psig
- test	195 psig
Shell side temperature - operating (inlet)	193.2°F
(outlet)	120°F
- design	338°F
Quantity of cooling water to tube side	495,000 lb/hr
Tube side pressure - operating	50 psig
- design	100 psig
- test	150 psig
Tube side temperature - operating (inlet)	95°F
(outlet)	129.8°F
- design	338°F
Fouling factor	0.0022
Weights - dry	8500 lbs
- flooded	15,500 lbs

5.1.5 Deaerator

The main function of the flash deaerator is to separate and expel any noncondensable gases that may be present in the primary reboiler condensate returning as reactor feedwater. The deaerator also serves as a water-holdup reservoir for the feedwater pump. In addition, since the operating water level in the deaerator is normally 30 feet above the feedwater pumps, the NPSH available for the feed pumps is enhanced.

The deaerator pressure vessel is built to conform to all requirements of Section VIII of the ASME Code as well as ASME Code Case Rulings 1234 and 1224-1. The unit has a 10-inch-pipe-size tangential inlet nozzle, a 12-inch pipe-size steam-outlet nozzle at the top, a 6-inch pipe-size water-outlet nozzle at the bottom, a 24-inch-diameter manhole at the side, and numerous 1-inch and $\frac{3}{4}$-inch pipe-size nozzles for instrumentation. Figure 5-4 shows the vessel layout and nozzle locations.

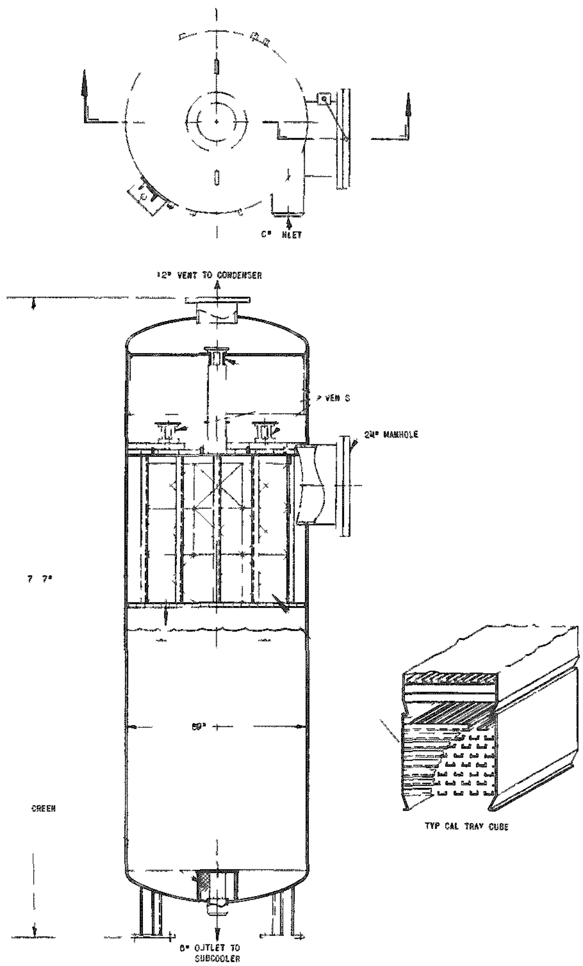

Figure 5-4
Deaerator
RE-6-32673-D

The primary reboiler effluent passes through two valves, G-332, where flashing occurs, and then flows into the deaerator where it passes over a multiple layer of separating trays. The flow labyrinth formed by the trays provides sufficient dwell time for the noncondensable gases present in the effluent stream to separate from the water at a low partial pressure in accordance with the provisions stated by Henry's law.

Upon separation the noncondensable gases pass up through the deaerator vent via a 12-inch line to the main condenser where they are monitored and discharged through the stack to the atmosphere. See Table 5-6 for the deaerator design specifications.

The deaerator is fitted with a Model 751 X Magnetrol high-level switch, a similar Model 751X low-level switch, and a Model 751X low-level pump-cutoff switch. In addition, two 20-inch gage glasses are attached to the vessel for visual level inspection.

Table 5-6. Deaerator Design Specifications

Capacity	300,000 lb/hr
Design pressure	100 psig and full vacuum at 100°F
Design temperature	338°F
Empty weight	10,000 lb
Operating weight	21,000 lb
Flooded weight	30,000 lb
Water storage volume	1,000 gal
Shell	
Material	304 SS ASTM-A-240S
Thickness	$\frac{3}{8}$ inch
Outside diameter	59 inches
Plate length	13 ft 0 inch
Over-all length (including heads)	14 ft 9 inches
Heads	
Material	304 SS ASTM-A-240S
Thickness	$\frac{3}{8}$ inch
Trays	
Number	18 tray cubes
Size	Each cube 15 in. x 15 in. x 15 in.
Total area	405 sq ft
Total effective surface	5400 sq ft
Thickness	17 gage
Material	304 SS ASTM-A-240S

5.1.6 Feedwater Pumps

The two new reactor feedwater pumps are six-stage, double-volute, split-case, centrifugal units as shown in Figure 5-5. The pumps are horizontally mounted and driven by conventional squirrel-cage induction motors. During normal 100-megawatt operation, one new pump is used in parallel with one of the original 180-gpm feed pumps; the other new pump is available as a standby. The head-capacity curve of the new pumps is shown in Figure 5-6 and the operating characteristics are given in Table 5-7.

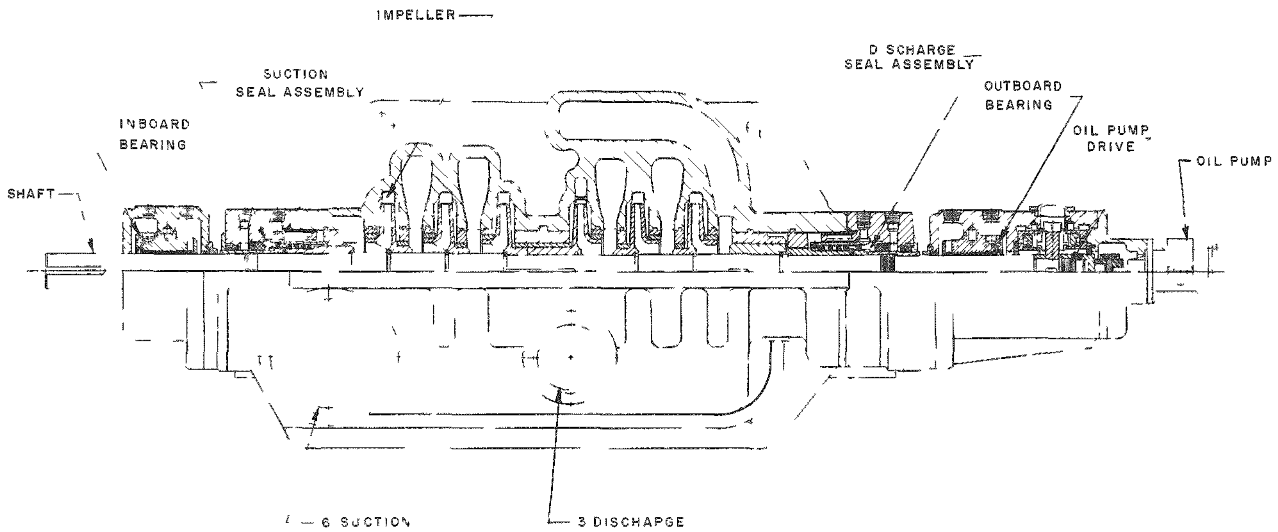

Figure 5-5

Cross section of primary feedwater pump
RE-6-32689-D

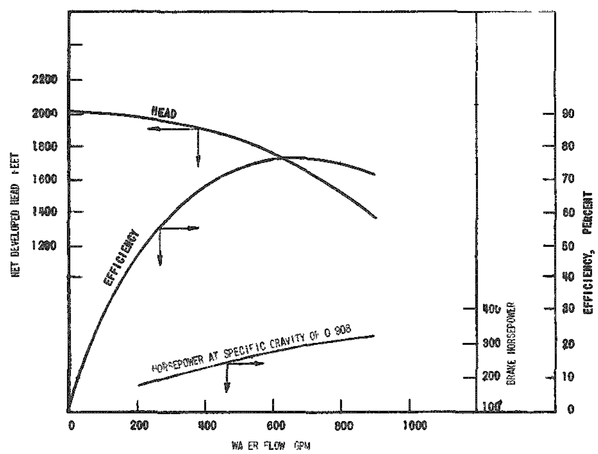

Figure 5-6

Performance curves of primary feedwater pumps
RE-7-32670-A

The pump casings were manufactured in accordance with Section VIII of the ASME Code, and full radiography was carried out in accordance with Case Ruling 1234 of the Code. The materials of construction for the pumps are given in Table 5-8.

Table 5-7. Primary Feedwater Pump Operating Characteristics

Capacity	645 gpm at 320°F
Shut-off head	2200 ft
Head (total dynamic)	2021 ft (not including friction losses)
Suction head	221 ft
Developed head	1800 ft
NPSH required	30 ft
Design pressure	900 psig
Design temperature	320°F
Motor horsepower, (rated)	350 hp
Motor voltage (3 phase, 60 cycle)	4000 volts
Motor speed	3550 rpm
Weights	
Total (pump and motor)	7170 lb
Pump and baseplate	3350 lb
Motor	3820 lb

Table 5-8. Primary Feedwater Pump Materials of Construction

Casing	11-13% Chromium-steel alloy
Shaft	11-13% Chromium-steel alloy (heat treated)
Shaft sleeves	11-13% Chromium-steel alloy
Impeller	11-13% Chromium-steel alloy
Casing wear ring	11-13% Chromium-steel alloy (Nitrided)
Impeller wear ring	11-13% Chromium-steel alloy (Stellited)
Packing gland	11-13% Chromium-steel alloy
Suction nozzle size	6-inch, schedule 40[a]
Discharge nozzle size	3-inch, schedule 80[a]

[a] The weld ends of these nozzles are "safe ended" with type-304 stainless steel.

The pump journal bearings are of self-aligning sleeve design, solid bronze, and ring oiled. The thrust bearing is the pivot shoe type. An oil pump, integral with each feedpump and mounted on the end of the shaft, provides continuous pressure lubrication to the bearings. Filters for the oil are located in a reservoir mounted on the end of the baseplate. An auxiliary oil pump is provided for standby service and for use prior to and during startup. For startup operation, the auxiliary oil pump is interlocked with the integral oil pump and remains in service until the main oil pump increases the oil line pressure to greater than 7 psig. Should the integral oil pump fail, the auxiliary pump takes over when the oil-line pressure drops to 3 psig and should the oil-line pressure drop below 2 psig, the feedwater pumps are automatically shut down. The lubrication system and cooling system are connected to suitable individual heat exchangers mounted adjacent to the pumps in order to assure proper lubrication and cooling for safe operation.

To minimize shaft seal leakage, the pumps are equipped with a modified Borg-Warner type-D mechanical seal utilizing a special leak-off and disaster bushing. The general arrangement of the seal is shown in Figure 5-7. A water jacket is provided in the casing casting for proper cooling of the mechanical seal. In normal operation, any leakage through the seal is collected and drained from a chamber on the outboard end which is backed up by a floating-type throttle bushing. In the event of a seal failure, pump leakage is controlled by a throttle bushing on the inboard side of the mechanical seal which restricts the flow sufficiently to enable it to be collected and drained off from the collecting chamber.

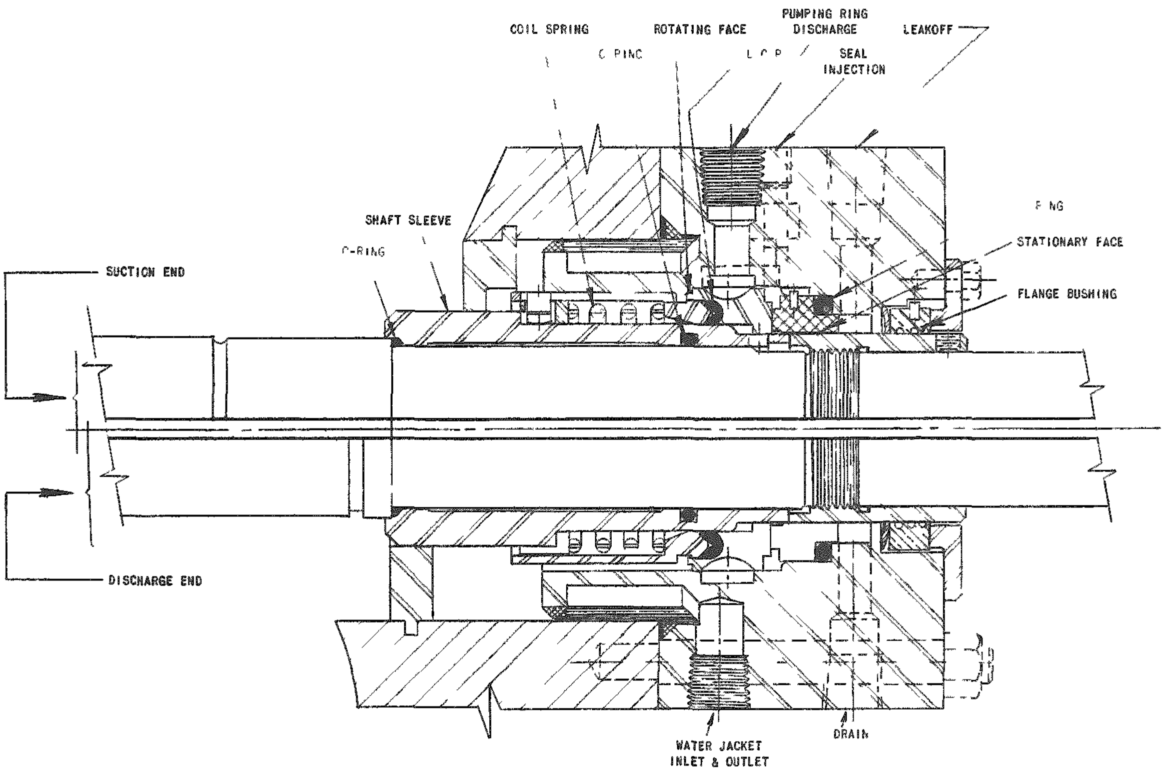

Figure 5-7

Primary-feedwater pump shaft seal
RE-6-32672-C

Each pump is provided with a warming line and a suitable breakdown orifice to maintain the standby unit at system temperature for ready service. The warming line breakdown orifice is sized to pass 8 gpm at an operating head of 1800 feet. Each pump is also equipped with a minimum-flow bypass line and breakdown orifice to assure a minimum flow of 30 gpm at the 2200-foot pump-shutoff head.

Tandem basket strainers, located in the suction line of each pump, remove foreign material from the operating fluid to prevent possible damage to the pump. Iron-constantan thermocouples continuously monitor pump bearing temperatures.

5.1.7 Filters

For 100-megawatt operation, additional filter capacity is necessary to accommodate the higher flow requirements; therefore, two new filters are connected in parallel with the two existing filters. The new filter, shown in Figure 5-8, has the same flow capacity as the original filters, 180 gpm at 120°F.

The new filter shells are fabricated entirely of type-304 stainless steel; the original filter shells are carbon steel clad with stainless steel on the internal surface. The change in shell material necessitated a redesign of the head closure flanges, shown in Figure 5-8, to attain proper bolting stresses. The redesign changed the diameter of the Viton O-ring seals and thereby precluded the possibility of basket interchangeability of the two sets of filters. The internal design and arrangement of the new filter baskets are of the same design described in ANL-5607.(1)

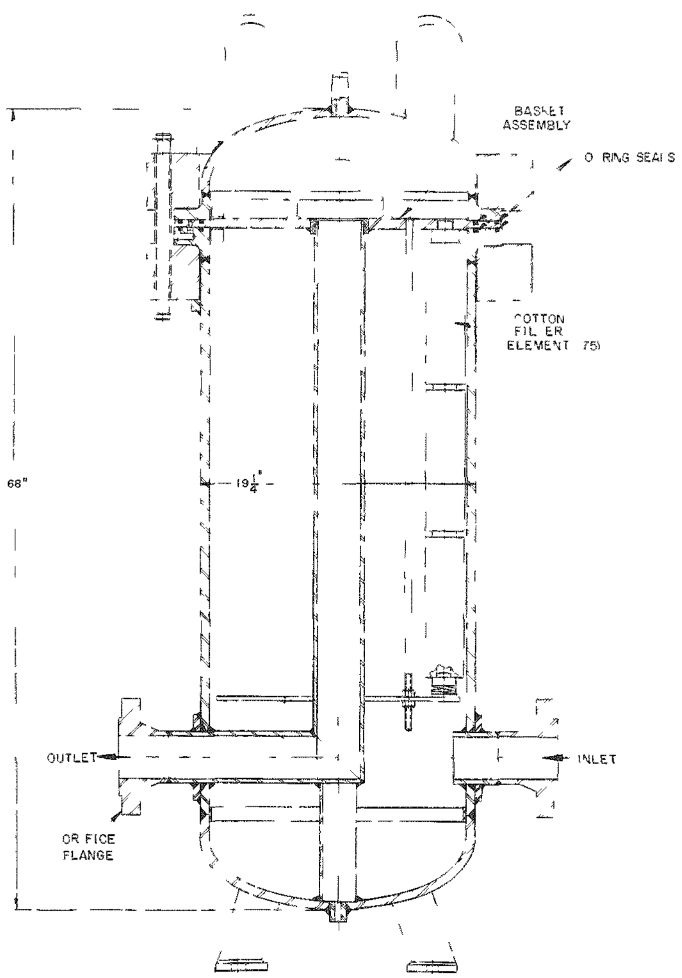

Figure 5-8

Primary feedwater filter
RE-6-32682-D

The four feedwater filters are equipped with orifice plates in the discharge line from each filter to measure the flow through each filter. The orifice plate taps are manifolded to a central differential pressure cell which enables selective monitoring of the flow or pressure drop associated with each filter to ascertain the degree of fouling. To facilitate easy removal of the filter cages, a monorail is mounted above the filter installations.

The baskets are removed by disassembling the closure rings, removing the upper head, and vertically lifting the basket from the filter shell by means of a chain-fall or suitable hoist. The general design specifications of the new feedwater filters are given in Table 5-9.

Table 5-9. Feedwater Filter Design Specifications

Capacity	180 gpm at 120°F
Dry weight	1900 lb
Shell	
Material	ASTM-A-312, Type-304 SS
Thickness	$\frac{5}{8}$ inch
Outside diameter	$19\frac{1}{4}$ inches
Over-all length (including heads)	68 inches
Heads	
Material	ASTM-A-312, Type-304 SS
Thickness	$\frac{5}{8}$ inch
Filter elements	
Number	75
Size	$2\frac{3}{4}$ in. diameter by $9\frac{3}{4}$ in. long
Material	Cotton thread
Pressure drop	5 psi
Minimum particle size	2 microns

5.1.8 Air Compressor

A second compressor was added to the EBWR control-air supply system to service the requirements of the new controls of the 100-megawatt-capacity plant. The new compressor, shown in Figure 5-9, supplies all of the control air for the air-operated controls within the containment shell, those within the reboiler building, and the controls for the variable-pitch fans on the air-cooled heat exchangers. The original compressor is maintained on standby service and will automatically pick up the load upon failure of the new compressor. The air supplied by the new compressor is entirely free of oil or oily vapors because of the nonlubricated construction of the compressor.

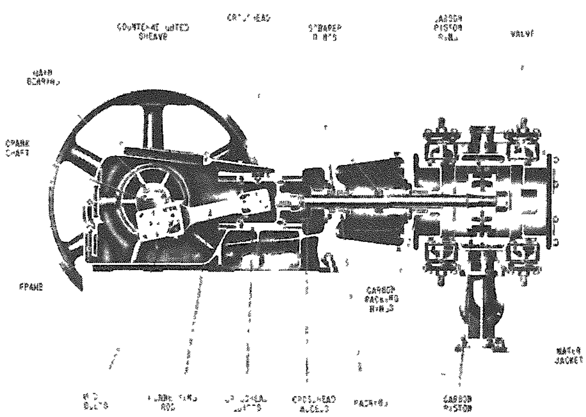

Figure 5-9

Instrument air compressor
RE-6-34898-A

The compressor is a single-stage, double-acting, horizontal, straight-line, water-cooled, roller-bearing, nonlubricated type. The cylinder bore is 5 inches with a piston stroke of 5 inches and piston displacement of 46.75 scfm. The compressor operates at 425 rpm with an atmospheric intake pressure and discharge pressure of 100 psig. The actual air delivery as referred to intake conditions is 25.2 scfm.

The compressor components include a nonlubricated cylinder arranged with top intake and bottom discharge. The valve seats and guards are Ni-resist cast iron. The piston, piston rings, and piston rod packing are carbon. A special extra-long open-type distance piece separates the cylinder and frame. The distance piece is of sufficient length to prevent any part of the piston rod that enters the frame from alternately entering the cylinder stuffing box and thereby injecting oil into the cylinder. The crank case is the enclosed oil-tight type with a breather and oil gage.

The compressor is equipped with dual regulation consisting of: (1) regulation for continuous operation with lifting inlet-valve unloaders and a pneumatically operated pilot valve and (2) regulation for start-and-stop control comprised of lifting inlet-valve unloaders, a three-way solenoid-operated pilot valve and diaphragm-type pressure regulator, a three-way selector switch and automatic water valve. The compressor can be operated under either mode of regulation.

The compressor drive consists of grooved compressor and motor sheaves and standard V-belts. The drive is protected by an expanded metal mesh guard mounted on the floor and supported by angle irons. The motor is a $7\frac{1}{2}$-horsepower, 1800-rpm, open, drip-proof, sleeve-bearing induction type with power station insulation, 40°C rise, arranged for 3-phase, 60-cycle, 540-volt current.

The air intake filter-silencer is of the dry type. The air receiver is an 18-inch-diameter by 6-foot-high vertical type constructed to comply with ASME Code requirements. The air receiver is suitable for a maximum working air pressure of 125 psig (maximum safety valve setting) and is equipped with a safety valve, pressure gage, and drain cock.

The aftercooler is a horizontal-pipeline type complete with moisture separator, automatic moisture trap, gage glass and is built to ASME Code requirements. The shell and air headers are of welded steel construction. The tubes and tube sheets are copper alloy. The aftercooler is capable of handling 100 scfm of free air compressed from one atmosphere pressure to 100 psig (single stage compression). The compressed air leaves the aftercooler at a temperature within 15°F of the cooling water inlet temperature when supplied with not less than $1\frac{1}{2}$ gpm of 80°F cooling water.

The control-air compressor is located at the southwest side of the condenser floor within the containment shell. The pressure is reduced to 30 psig for distribution within the containment shell and reboiler building and is reduced to 90 psig for the variable-pitch fan controllers. The containment vessel is equipped with a solenoid shut-off valve to cut off the air supply and to isolate the shell during a nuclear incident or period of air-borne radioactive contamination. During an isolation period or low air pressure (40 psig), the service-air compressor will carry the load of the reboiler house and the air-cooled heat-exchanger controls. The service air is connected to the control air distribution line upstream of the pressure reducing station.

5.1.9 Primary Piping

To minimize the additional load placed on the existing water treatment system by the new steam plant, the added primary piping is fabricated of ASTM-A-312 type-304 seamless stainless steel. All joints in the stainless lines are of welded construction and utilize type-308 stainless steel, solid, consumable, insert rings with V-groove pipe-end preparations. The ring is consumed by an inert gas-tungsten arc root-pass weld using helium for the purge atmosphere and argon for the cover atmosphere. Every root weld was examined for integrity with dye penetrant, and defects detected by the test were ground out, repaired, and then re-examined. This procedure was repeated until the weld was satisfactory. In extreme cases, the weld was cut out, the mating surfaces refaced, a new consumable insert ring inserted, and the welding process repeated. In addition to the dye penetrant test, all internal joint surfaces were individually checked by visual means using mirrors and periscopes, where necessary, in order to assure complete fusion of the ring. Thus, a smooth transition between mating parts was assured and resulted in a joint of high integrity.

Where joints were made in existing lines, such as the placement of the two motor-operated globe valves in the original feedwater-pump discharge lines, radiographs were used to determine weld quality since visual methods were not feasible. After the root-pass weld was approved by the inspecting engineer, a second inert gas-tungsten arc weld using type-304 stainless steel filler rod was made, and the remaining welds for completion of the joint were accomplished with a metallic arc using type-308-15 lime-ferritic coated stainless steel welding rod.

The general specifications for the primary system piping are as follows:

1. For all lines 10 inches and larger, schedule-80S butt welded
2. For all lines 8 inches and smaller but not less than $2\frac{1}{2}$ inches, schedule-40S butt welded

3. For lines 2 inches and smaller, schedule-80S socket joint, socket welded with at least one weld being of the inert gas-tungsten arc type

The 6-inch line to the new reactor relief valves and the 12-inch deaerator vent line to the turbine condenser are fabricated of ASTM-A-106 grade B carbon steel. An E6015 weld rod was used for the metallic arc method of welding the butt joint. It was considered feasible to fabricate these lines of carbon steel since the relief line is essentially dead ended and will not place any additional burden on the water treatment system. In the case of the deaerator vent, the flow through the 12-inch line is small and only a negligible amount of contamination will be introduced into the system. The joints in these two carbon steel lines were spot checked by radiography to assure integrity and quality.

5.1.10 Primary System Valves

Twenty-eight new shutoff valves, designated PS-1 through PS-18, were added to the primary system. Table 5-10 is a tabulation of the quantity requirements, service, maximum pressure and temperature, size, type, pattern design, and material for each valve.

Table 5-10. Primary System Shutoff Valves

Valve	Number req'd	Service	Maximum pressure, psig	Maximum temp. °F	Size, inches	Type	Design standard	Mat'l
PS-1	1	Steam to primary reboilers	800	520	8	Gate	600 lb	316 SS
PS-2	2	Steam to primary reboilers	800	520	6	Gate	600 lb	316 SS
PS-5	1	Condensate to subcooler	100	320	6	Gate	150 lb	316 SS
PS-6	1	Condensate from subcooler	100	320	6	Gate	150 lb	316 SS
PS-7	2	Reactor feed pump suction	100	320	6	Gate	150 lb	316 SS
PS-8	2	Reactor feed pump	900	320	6	Check	600 lb	316 SS
PS-9	2	Reactor feed pump discharge	900	320	6	Globe	600 lb	316 SS
PS-10	1	Feedwater regulator valve bypass	900	320	2	Globe	600 lb	316 SS
PS-11	4	Feedwater filter inlet and outlet	900	285	3	Gate	600 lb	316 SS
PS-12	1	Feedwater to reactor	900	285	6	Stopcheck	600 lb	316 SS
PS-13	2	Deaerator vent steam	100	320	10	Gate	150 lb	CS
PS-14	1	Deaerator steam press regulator valve bypass	100	320	6	Globe	150 lb	CS
PS-15	2	Condensate to deaerator	800	520	4	Gate	600 lb	316 SS
PS-16	1	Subcooler bypass	100	320	4	Gate	150 lb	316 SS
PS-17	2	Drain cooler bypass	800	520	3	Globe	600 lb	316 SS
PS-18	2	Condensate from drain cooler	800	520	4	Globe	600 lb	316 SS
FW-3R	1	Feedwater to reactor (replacement of original manually operated gate valve).	900	285	3	Globe	600 lb	316 SS

All valves are manually operated with the exception of valves PS-1, PS-9, PS-10, PS-12, and FW-3R which are motor operated. The motor operators utilize 60-cycle, 208-volt, 3-phase circuits with 115-volt a-c control circuits. Valves PS-1, PS-9, PS-10, PS-12, and FW-3R are equipped with limit switches connected to indicating lights at the control panel to indicate the full-open and full-closed positions. In addition, valves PS-10, PS-12, and FW-3R incorporate slide wires and remote position indicators for continuous intermediate-position indication in the control room.

All gate valves have removable seats and disks. Globe valve seats are faced with Stellite.

Nonreturn (stop-check) valves are installed on the steam-outlet nozzles of the primary and secondary reboilers. The purpose of a nonreturn valve is to prevent flow between two steam sources in a parallel system when one steam source is temporarily below the system pressure.

The nonreturn valves on the primary reboilers are flanged 10-inch globe type and on the secondary reboiler are flanged 12-inch angle type. The material of construction for these valves are as follows:

1. Body and bonnet - cast carbon steel
2. Stem and piston rings - 13 percent chromium-stainless steel
3. Disk - carbon steel with stainless steel seating face

The nonreturn valves are equipped with a handwheel for manual operation: (1) to permit them to be closed during operation, or (2) if they have already been closed automatically, to hold the disk in the closed position. There is no mechanical connection between the disk and the stem. When the stem is raised to the open position by the handwheel, only the reboiler pressure can lift the disk.

The control valves listed in Table 5-11 for the primary, intermediate, and secondary systems include air-operated and motor-operated types.

Table 5-11. Control Valves for Primary, Intermediate and Secondary Systems

Valve	Service	Size, inches	Mat'l	Design standard	Direction of failure	Operated by	Fluid
G-332	Primary-reboiler drain tank level	3	316 SS	600 lb W.E.[a]	Close	Air	Water
G-333	Primary-reboiler shell level	3	5 Cr-0.5 Mo[b]	300 lb flanged	Close	Air	Water
G-334	Secondary-reboiler drain tank level	4	5 Cr-0.5 Mo	300 lb flanged	Close	Air	Water
G-335	Secondary-reboiler shell level	4	5 Cr-0.5 Mo	300 lb flanged	Close	Air	Water
G-336	Air-cooled condenser drain tank level	4	5 Cr-0.5 Mo	300 lb flanged	Close	Air	Water
G-339	Primary-steam-control valve bypass	1	316 SS	600 lb S.W.[c]	Close	Air	Steam
P-11D	Reactor steam pressure controller	8	316 SS	600 lb W.E.	Close	Air	Steam
P-12D	Deaerator pressure controller	10	5 Cr-0.5 Mo	150 lb W.E.	In position	Elec. motor	Steam
P-13D	Steam to secondary reboiler pressure controller	12	CS	300 lb W.E.	In position	Elec. motor	Steam
P-14D	Intermediate steam pressure controller	10	CS	300 lb W.E.	In position	Elec. motor	Steam
P-15D	Intermediate flash tank pressure controller	8	CS	150 lb W.E.	Close	Elec. motor	Steam
P-21E	Reactor feedwater regulator	4	316 SS	600 lb W.E.	In position	Elec. motor	Water
P-42C	Condensate from subcooler temperature controller	6	CS	150 lb W.E.	In position	Elec. motor	Water

[a]W.E. - Weld end

[b]This material is a steel alloy containing 5 percent Cr and 0.5 percent Mo.

[c]S.W. - Socket weld

All of the valves listed are single-seated design except valves G-339, P-11D, P-13D, P-14D, and P-42C which are double seated. The valves are designed with flow-to-open construction. In addition to being automatically controlled, all valves are completely equipped for local manual control.

The valves with the prefix P and valve G-332 are also equipped for remote manual operation from the gage board panel. The air-operated valves are actuated by air pilots with integral air-pressure-regulator, filter, and pressure gages except valve G-332 which uses an electrical pilot and electropneumatic transducer for actuation of the valve operator. The valves that regulate the liquid level in a tank or reboiler are controlled by liquid-level controllers of the displacement type that use a cylindrical float to sense level.

The safety valves added to the system are all standard valves of conventional design for protection against overpressure in the various systems or vessels. The two primary-system and four primary-reboiler-steam safety valves are constructed in accordance with the requirements of Section I of the ASME Code. The deaerator safety valve is constructed in accordance with Section VIII of the ASME Code. All the valves are of similar construction and a typical cross section of a primary-system safety valve is shown in Figure 5-10.

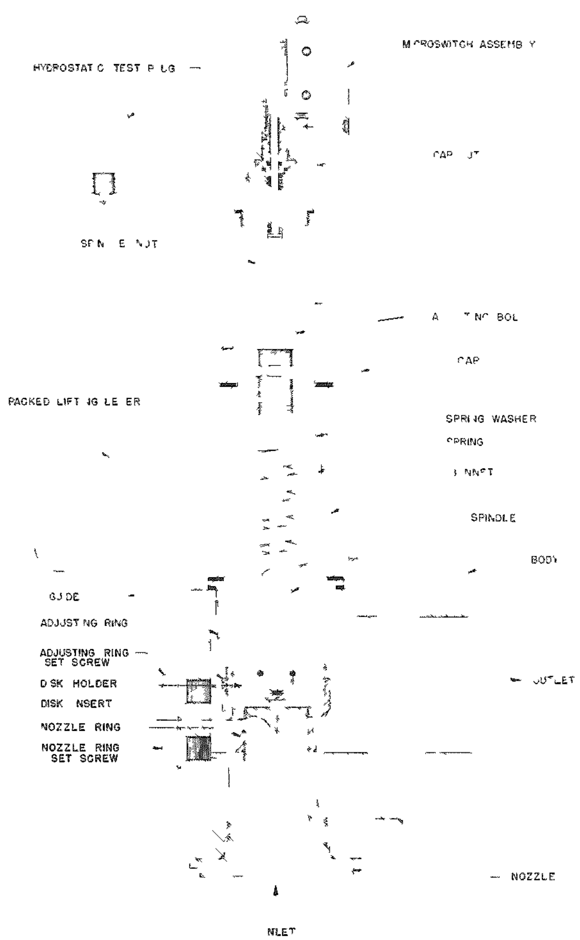

Figure 5-10

Cross section of typical safety valve
RE-6-32687-D

In the primary steam system, one safety valve is set to open at 750 psig and the second valve is set to open at 775 psig should the reactor pressure exceed these set points. Each of the new primary-system safety valves have a rated capacity of 160,000 pounds per hour of steam. The valve relieving capacity is rated, according to Section I of the ASME Code, at 90 percent of full flow at 800 psig inlet pressure with 3 percent pressure accumulation.

Each of the two original primary-steam-system safety valves have a relieving capacity of 65,000 pounds per hour of steam and a set pressure of 700 and 725 psig. The total relieving capacity of the four primary-steam safety valves is approximatly 390,000 pounds per hour of steam and therefore is capable of relieving the total steam generating capacity of the reactor at an operating level of 100 megawatts and producing 361,600 pounds per hour of steam.

Each primary reboiler has two safety valves; one has a set pressure of 400 psig and a relieving capacity of 122,500 pounds per hour of steam, and the other has a set pressure of 410 psig and a relieving capacity of 85,350 pounds per hour of steam. The relieving capacity is rated at 90 percent of full flow at 3 percent accumulation above set pressure as required by Section I of the ASME Code.

The deaerator safety valve has a set pressure of 100 psig with a relieving capacity of 84,700 pounds per hour. The relieving capacity is rated at 90 percent of full flow at 10 percent accumulation above set pressure as required by Section VIII of the ASME Code. All of the new safety valves have a 4 percent blowdown before the valve reseats itself.

The valves are constructed with a cast-steel body, bonnet, and cap, type-304 stainless steel nozzle and disk insert, S-Monel guide and disk holder, high-speed steel spring and Stellite-faced seating surfaces on nozzle and disk inserts. Each safety valve is equipped with a positive-action microswitch device (see Figure 5-10) which senses the opening of the valve and actuates certain safety control devices which shut down the reactor. A packed lifting lever is provided on each valve to permit periodic testing of the mechanical action of the valve. All valves are capable of being easily "gagged" for hydrostatic testing of the various systems. The safety-valve inlet flanges are American Standards Association raised face except the primary-system safety valve which has a ring-type joint.

The primary-system safety valves discharge directly into the condenser. The deaerator safety valves discharge into the desuperheating pipe connection to the condenser. The primary-reboiler safety valves discharge to the atmosphere through vent pipes extending through the roof of the reboiler house.

5.1.11 Changes to Condenser

To insure containment of primary system steam should a reactor over-pressure occur during 100-megawatt operation, two relief valves were added to exhaust to the main condenser. In the event that all safety valves (the two new valves and the two original valves) were to simultaneously dump steam into the condenser while a failure or malfunction occurs in the condenser coolant system associated with the outside cooling tower, the condenser pressure would rise above the 20-psig design pressure and cause the rupture disk originally provided in the unit to rupture. Calculations have shown that under these conditions the original rupture disk does not provide sufficient area to allow a rapid release of pressure from the condenser; therefore, an additional rupture disk 12 inches in diameter was installed in the condenser wall to make the total flow capacity of the two rupture disks in excess of 400,000 pounds per hour, enough to prevent the condenser pressure from rising above its design pressure.

The 12-inch deaerator vent line is connected near the bottom of the condenser, at the far end of the 14-inch desuperheater line, to allow the noncondensable gases in the deaerator effluent to be combined with the noncondensable gases emanating in the condenser hotwell. After suitable radiation monitoring, the combined gases are dispersed to the atmosphere through the air ejectors and stack.

Two 6-inch flanged nozzles are attached to the side of the condenser for accommodation of the two new 4-inch by 6-inch reactor relief valves. In addition, the 1-inch vent line from the primary reboilers and drain tanks as well as the 1-inch effluent line from the main-steam-line traps and the reactor-relief-line trap are connected to the condenser.

5.2 INTERMEDIATE SYSTEM COMPONENTS

5.2.1 Feedwater Pumps

The two feedwater pumps for the intermediate system are located in the southwest corner of the reboiler building. The pumps are 2-stage, horizontally split-case, centrifugal units, as shown in Figure 5-11, driven by conventional squirrel-cage induction motors. The pump suction comes from the flash tank above the primary-system shield cell and the pump discharges to the shell side of the primary drain coolers and reboilers. Only one pump operates at a time; the second pump is on standby service. The specifications for the pump are given in Table 5-12, the materials of construction are given in Table 5-13, and the head capacity curve is plotted in Figure 5-12.

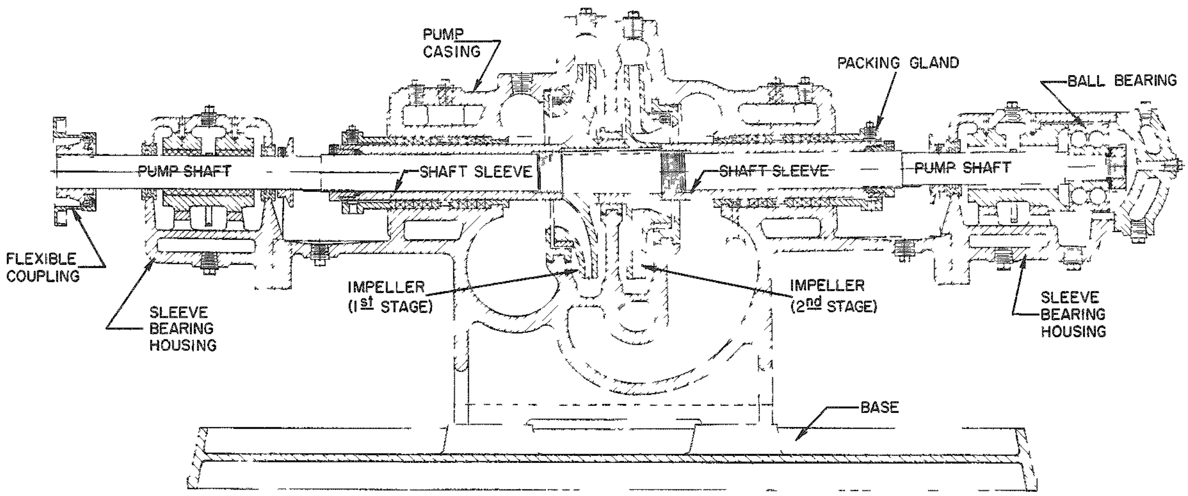

Figure 5-11

Cross section of intermediate feedwater pump
RE-6-34899-C

Table 5-12. Operating Conditions and Capacities of Intermediate Feedwater Pumps

Performance	
Capacity	600 gpm
Shutoff head	1152 ft
Total dynamic head	1050 ft
Differential head	1038 ft
NPSH required	12 ft
Efficiency at rated capacity	66%
Design temperature	260°F
Brake horsepower	220
Motor	
Rated horsepower	250
Speed, rpm	3570
Weights	
Pump	2000 lb
Base plate	800 lb
Motor	2500 lb
Total	5300 lb

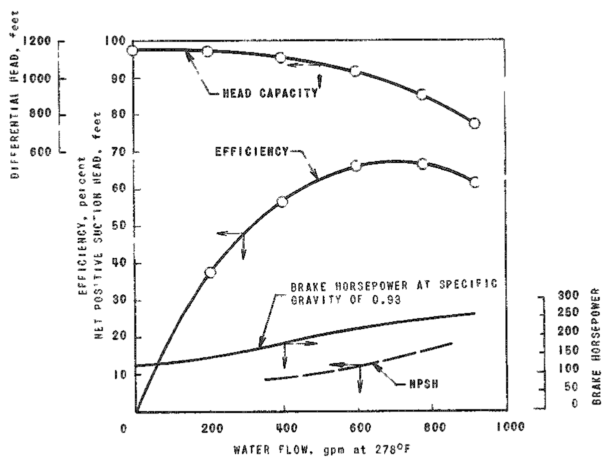

Figure 5-12

Performance curve of intermediate feedwater pump RE-7-34900-A

Table 5-13. Materials of Construction of Intermediate Feedwater Pumps

Casing	Carbon steel
Shaft, shaft sleeve, impeller, casing and impeller wearing ring	11-13% chromium-steel alloy
Packing gland	11-13% chromium-steel alloy with bronze bushing
Suction nozzle	6-inch raised face
Discharge nozzle	4-inch raised face

The pump casings were manufactured in accordance with Section VIII of the ASME Code with an additional requirement of 100 percent radiography. The pump has rigid sleeve-type journal bearings and ball-type thrust bearings. An oil sump with an oil-splash ring lubricates the bearings. The cooling water enters the pump-shaft-seal water jacket and then flows to the bearing-housing jacket for cooling the lubricating oil. This method of piping eliminates the possibility of water condensation in the bearing housing from the cold cooling water because of heat absorbed in the shaft-seal water jacket.

Since this pump is a standard commercial unit, the shaft seals are conventional solid packing. Any leakage past the shaft seal that flashes into steam will be quenched by a flow of cooling water at the packing gland. The pumps are provided with warmup and minimum-flow orifices that maintain flow through the pump at 5 and 45 gpm respectively.

Strainers located in the suction line of each pump prevent large particles from entering and possibly damaging the pumps.

5.2.2 Air-Cooled Heat Exchangers

Heat supplied by the intermediate system in excess of the secondary system requirements is dissipated to the atmosphere by four air-cooled heat-exchanger units. The heat exchangers are physically located in two areas. The air-cooled steam condenser is located to the east of the reboiler building; the air-cooled drain cooler, the air-cooled flash condenser, and the air-cooled subcooler are grouped to the south of the building.

The four heat exchangers are similar except for variations in the number, size, and design ratings of the components indicated in Table 5-14. Vertical-shaft fans are mounted on a common fan deck and force ambient air upward through horizontal banks of finned tubes containing the intermediate system coolant. Below the fan deck each group of heat exchanger units is enclosed on all sides by adjustable shutters to provide airflow regulation. Doors installed in the shutters provide an entrance to the areas beneath the fan decks. Each unit is separated from the adjacent unit by vertical steel-ribbed enclosure panels placed below and above the fan deck. Each unit is enclosed with steel-ribbed enclosure panels which extend 4 feet above the heat exchanger sections to enhance protection against freezing. Further protection from freezing during periods of nonoperation is provided by a manually operated drain system capable of draining all water from the tubes, headers, and external piping. A typical air-cooled heat exchanger arrangement is shown in Figure 5-13.

Each fan assembly consists of a constant-speed vertical-shaft fan, a speed-reduction gear, and an electric drive motor. These three major components are all mounted on a drive pedestal which is anchored to a single footing.

Table 5-14 Design and Equipment Data of the Air-Cooled Heat Exchangers

Parameters	Heat exchanger			
	Steam condenser	Drain cooler	Flash condenser	Subcooler
Heat exchanger				
Number of sections	8	2	1	2
Surface per section sq ft	16 941	16 941 8 102	5 410	25 780
Total surface sq ft	135 528	25 043	5 410	51 560
Design pressure psig	400	400	100	100
Design temperature °F	448	448	338	338
Intermediate coolant				
Flow lb/hr	245 400	245 400	4 930	246 600
Operating pressure psia	310	310	19 8	19 8
Temperature in °F	424 7	424 7	258 5	258 5
Temperature out °F	424 7	291 7	258 5	160
Pressure drop psi	Negligible	16 7	Negligible	4 2
Heat exchanged Btu/hr	196 5 x 10^6	34 6 x 10^6	4 64 x 10^6	24 5 x 10^6
Passes	1	4	1	4
Air				
Flow lb/hr	3 4 x 10^6	0 72 x 10^6	0 6 x 10^6	1 2 x 10^6
Temperature in °F	95	95	95	95
Temperature out °F	336	295	127	180
Components				
Tubes				
Total number	736	136	46	280
Material	Admiralty SB-11 1 Type C	Admiralty SB-11 1 Type C	Admiralty SB 11 1 Type C	Admiralty SB-11 1 Type C
Size and gage	1 in OD x 16 BWG	1 in OD x 16 BWG	1 in OD x 18 BWG	1 in OD x 18 BWG
Pitch in	2-5/8 by 2 11/32	2-5/8 by 2 11/32	2-5/8 by 2-11/32	2 5/8 by 2 11/32
Layers	4	4	2	4
Length in	359-1 2	359 1/2	360	360
Fins				
Material	Aluminum Type 1100-0	Aluminum Type 1100 0	Aluminum Type 1100 0	Aluminum Type 1100-0
Number per in	11	11	11	11
Diameter in	2 1/4	2 1/4	2-1/4	2 1/4
Thickness in	0 016	0 016	0 016	0 016
Type bonding	Mechanical	Mechanical	Mechanical	Mechanical
Design	Grooved	Grooved	Marley	Marley
Headers	Steel	Steel	Steel	Steel
Fan assemblies				
Fans (Hartzell Propeller Fan Co)				
Number of fans				
Variable pitch	1	1	1	1
Fixed pitch	3	2	2	1
Number of blades	6	6	4	4
Blade diameters ft	12	7	7	12
Blade material	Plastic	Aluminum	Aluminum	Plastic
Speed rpm	292	500	500	292
Horsepower per fan	28 4	9 45	4 5	13 3
Gear reducers (Philadelphia Gear Corp)				
Model	3415-CT	3405-CT	3405-CT	3415-CT
AGMA rating	40	20	20	31
Motor drives				
Horsepower	30	10	5	15
Speed rpm	1750	1750	1750	1750
Electric rating	480 volt 60 cycle 3 phase	480 volt 60 cycle 3 phase	480 volt 60 cycle 3 phase	480 volt 60 cycle 3 phase

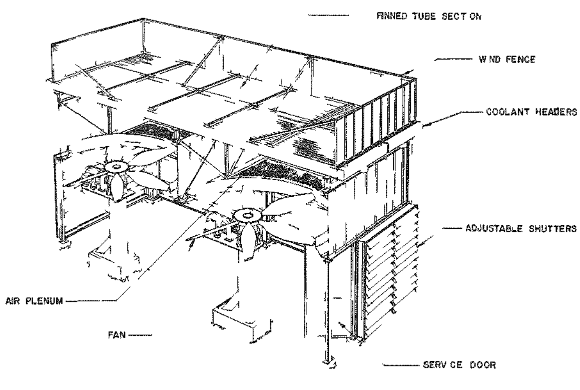

Figure 5-13

Typical air-cooled heat exchanger
RE-6-32684-C

One fan with each heat exchanger unit is a pneumatically operated variable-pitch design and provides stepless airflow regulation necessary to maintain desired operating conditions in summer and winter. Blade position is continuously monitored by an electromechanical position detector which is directly connected to the pneumatic operater and which relays information to a position indicator located at the control panel. The remaining fans with each heat exchanger unit are of fixed-pitch design. A vibration cutoff switch mounted on the drive-unit support pedestal protects all fan assemblies from excessive vibration arising from imbalance.

The tubes in all heat exchanger sections are made of Admiralty metal (ASTM-SB-111 type-C) rolled into steel headers at each end. Grooved or Marley-type aluminum fins, depending upon the requirements of the particular unit, are attached to the outside or air side of the tubes by mechanical bonding for maximum metal-to-air heat transfer surface.

The air-cooled steam condenser operates in parallel with the tube side of the secondary reboiler as shown in Figures 2-1 and 3-3. The chief function of this heat exchanger is to condense intermediate steam that is not utilized by the secondary reboiler. The unused steam must be dissipated if desired reactor operating conditions are to be maintained independent of secondary system requirements. The exchanger dissipates to the atmosphere the heat of vaporization of the intermediate fluid flowing in the tubes. Flow to the air-cooled condenser is regulated by control valve P-14D, the back pressure regulator for the intermediate circuit. Hence, the air-cooled steam condenser, in conjunction with valve P-14D, maintains the steam pressure in the intermediate loop at 350 psig by presenting a constant heat sink that is unaffected by fluctuations in secondary-system demands.

The air-cooled steam condenser consists of eight equal-size, horizontal, single-pass, tubular, heat exchanger sections installed in parallel. The sections are designed to condense 245,000 pounds per hour of steam at 310 psig when the ambient-air temperature is 95°F. Under these conditions the heat removal is accomplished by passing up to 3.4×10^6 pounds per hour of air across the finned tubes. Four motor-driven fans, one with variable pitch, circulate the air, which is heated to 336°F as it passes through the steam condenser. The variable-pitch fan, through regulation of airflow, and consequently by temperature differential, controls the pressure in the condenser. Condenser pressure must be sufficient to cause adequate steam-flow to the steam condenser from the intermediate system and prohibit the development of back pressure which would affect the intermediate system pressure. Remote adjustments to the angles of the variable-pitch blades are initiated manually by the plant operator. The intermediate coolant, in flowing through the tubes of the steam condenser, experiences only a negligible pressure drop.

In the air-cooled portion of the intermediate system, the steam-condenser drain tank furnishes the location for the interface between the high-pressure steam and condensate. Maintaining a water level in the tank permits operation of the intermediate system without complete or partial flooding of the air-cooled steam condenser and, at the same time, assures that no intermediate steam will be condensed in the drain cooler or flow directly to the flash tank. The drain tank is equipped with a sight glass, high and low water level alarms, and a water level controller. The desired water level in the tank is maintained by control valve G-336, which regulates the flow from the air-cooled drain cooler to the flash tank.

The 300-gallon-capacity, horizontally mounted, cylindrical tank is located against the east wall of the reboiler house. The 7.5-foot-long by 2.75-foot-inside-diameter tank is made of carbon steel and designed for a pressure of 400 psig and temperature of 448°F in accordance with Section VIII of the ASME Code. The operating conditions are 310 psig and 425°F.

The air-cooled drain cooler accepts all of the intermediate coolant flow from the air-cooled steam condenser drain tank and further cools the condensate before it flows into the flash tank. The temperature of the condensate leaving the drain cooler is established as the temperature necessary to provide sufficient flashing of the condensate in the flash tank to furnish adequate steam flow to the flash condenser for maintenance of flash tank pressure control.

The drain cooler unit consists of two horizontal, four-pass, tubular, heat exchanger sections installed in parallel. The sections are of unequal size; one is designed for 67.6 percent of the flow.

The air-cooled flash condenser functions to maintain the desired operating pressure in the intermediate flash tank. To maintain operating pressure, the necessary quantity of flashed steam is permitted to flow from the flash tank through a motor-operated control valve, P-15D, to the flash condenser. In the flash condenser the steam is condensed by transferring heat of vaporization to the atmosphere. The resulting condensate is returned to the flash tank by means of the flash-condenser condensate-return unit.

The flash condenser consists of one single-pass heat exchanger section and three fan assemblies. The unit is designed to condense 4,930 pounds per hour of steam at 19.8 psig.

The air-cooled subcooler, installed between the flash tank and intermediate feedwater pumps, transfers heat energy from the flash tank drains to the atmosphere. In operation, the temperature of the drains is lowered below that of saturation conditions. The result is an increase in the available NPSH of the intermediate feedwater pumps and constitutes a method of controlling feedwater temperature on the shell side of the primary drain coolers.

Two equal-size, four-pass, heat exchanger sections installed in parallel and two fan assemblies constitute the subcooler. The unit is designed to subcool the intermediate coolant as much as 98°F. The intermediate system can be operated with full bypass of the subcooler and all flash tank drains passed directly to the pumps.

5.2.3 Flash Tank

In the intermediate system, the flash tank serves as a hold tank for the intermediate water and provides space for the flashing of condensate from the air-cooled drain cooler and secondary drain cooler. Water level is maintained by the manual addition of make-up water. The pressure in the flash tank is maintained by pressure control valve P-15D which regulates the steam-flow to the air-cooled flash condenser. As shown in Figure 5-14, the flash tank is equipped with three safety valves, a sight glass, high and low water level alarms, and a pump cutoff switch for shutdown of the intermediate feedwater pumps at extremely low water level. Any change in water inventory in the intermediate system will be noticed by a change in water level in the flash tank.

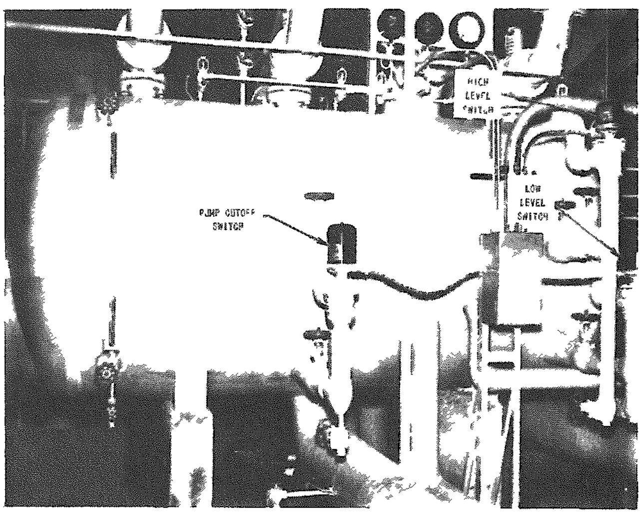

Figure 5-14

Flash tank (south side)
Neg. 112-599

The 1000-gallon-capacity, horizontally mounted, cylindrical tank is located on top of the shielded cell within the reboiler building. The 102-inch-long by 54-inch-inside-diameter vessel is fabricated of carbon steel and has design conditions of 100 psig at 338°F. The tank is built in accordance with Section VIII of the ASME Code.

5.3 SECONDARY SYSTEM COMPONENTS

5.3.1 Heat Exchangers

The secondary system receives intermediate steam from the primary reboilers on the tube side and generates steam on the shell side of the secondary reboilers for the Laboratory heating system. The reboilers generate only the amount of steam required by the heating system. This limits the amount of intermediate steam which is absorbed by the secondary system. The remainder of the intermediate steam not absorbed by the secondary system is diverted by pressure regulating valve P-14D to the air-cooled heat exchangers. The amount of steam generated is established by fixing the temperature differential across the tubes of the reboiler. This is accomplished by selecting and maintaining the condensing pressure.

The intermediate steam condensed in the secondary reboiler is accumulated in the secondary-reboiler drain tank and then passed through the secondary drain cooler to the flash tank. The condensate accumulated in the drain tank is maintained at a constant level by regulating the flow to the flash tank through control valve G-334 which receives its control signal from float-cage level controller LC-334 on the drain tank.

The feedwater to the shell side of the secondary drain cooler comes from the feedwater supply of the Laboratory heating system boiler. The feedwater flow is governed by the steam generated in the secondary system reboiler and is regulated by the water level of the reboiler The reboiler level controller, LC-335, is a float-cage type that controls the feedwater flow to the drain cooler shell through air-operated control valve G-335.

5.3.2 Reboiler

The secondary system reboiler is located in the east end of the reboiler house. The reboilers are floor-mounted, horizontal, single-effect type with a two-pass, tube side, steam condensing circuit (see Figure 5-15). Two line-type steam separators remove moisture from the steam to 99.75 percent quality before it enters the Laboratory steam system. The intermediate system steam coming from the primary reboilers to the tube side of the secondary reboiler provides the necessary heat to produce the secondary steam on the shell side of the unit. The operating conditions and capacities are shown in Table 5-15.

The secondary reboiler is a conventional kettle-type steam generator with U-tube removable-bundle construction. The design of the reboiler conforms to the design requirements of Section VIII of the ASME Code and also, where applicable, with TEMA standards for Class A construction. See Table 5-16 for the materials of construction of the secondary reboilers.

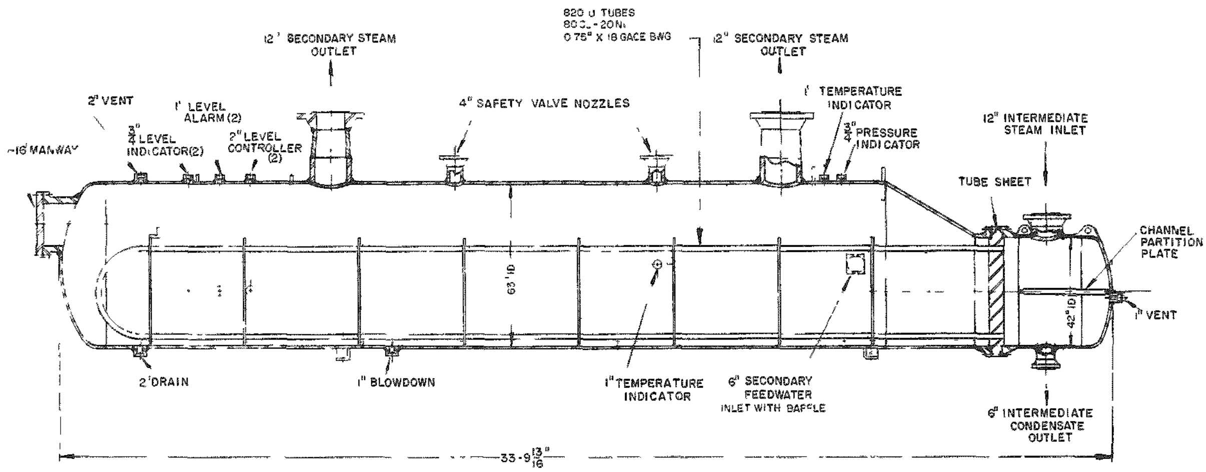

Figure 5-15

Secondary system reboiler
RE-6-32680-D

Table 5-15. Operating Conditions and Capacities of the Secondary Reboiler

Reboiler tube surface	9340 sq ft
Heat exchanged	193 x 10^6 Btu/hr (56.6 mw)
Quantity of intermediate steam to tube side	241,000 lb/hr
Tube-side pressure - operating	310 psig
- design	400 psig
- test	600 psig
Tube-side temperature - operating (both steam inlet and condensate outlet)	424.7°F
- design	448°F
Quantity of secondary steam leaving shell side	227,000 lb/hr
Shell-side pressure - operating (inlet)	376.5°F
(outlet)	388°F
- design	406°F
Steam quality leaving reboiler after steam separators	99.75%
Fouling factor	0.0005
Shell-side water capacity at normal operating conditions	2,525 gal
Weights - dry	45,900 lb
- operating	54,000 lb
- flooded	64,500 lb
Heat-up rate - normal	20°F/min
- maximum	30°F/min

Table 5-16. Materials of Construction of the Secondary Reboiler

Shell	ASTM-A-212 Grade B, Firebox quality
Channel	ASTM-A-212 Grade B, Firebox quality
Shell and channel flanges	ASTM-A-105 Grade 2, (30% max carbon)
Shell and channel nozzles	ASTM-A-106 Grade B
Shell steam outlet nozzles	ASTM-A-105 Grade 2, (30% max carbon)
Relief valve flanges	ASTM-A-105 Grade 1, (35% max carbon)
Shell and channel flanges	ASTM-A-111 Grade 1, (35% max carbon)
Tubes	ASTM-B-111 80 Cu-20 Ni, Type A annealed
Tube sheet	ASTM-A-212 Grade B, Firebox quality

In reboiler construction, the large shell diameter (compared to the tube bundle diameter) provides a large open volume, or disengaging area, over the boiling liquid. This permits the vapor to travel at a sufficiently low velocity toward the outlet so that a majority of the entrained liquid droplets settles out before leaving the shell. Two steam outlets are located at the quarter and three-quarter points along the length of the shell. This allows a maximum amount of steam to leave the shell without traversing any great distance and also maintain a low velocity. An impingement plate is placed at the feedwater inlet of the shell to protect the nearby tubes from erosion by the influent water.

The tube bundle is a two-pass, U-tube design. The stationary tube sheet is clamped between the gasketed surfaces of the shell and channel flanges and is drilled to the same bolt-circle diameter as the flanges. This enables the tube sheet to act as a shell closure when the channel end has been removed for hydrostatic testing of the reboiler shell and tube bundle. The tubes are mechanically joined to the tube sheet by rolling the tube ends into two machined grooves in the tube sheet holes to provide a joint seal and mechanical integrity.

The selection of copper-nickel alloy tubes instead of Admiralty metal alloy tubes was governed by the presence of 0.32 ppm of ammonia in the feedwater influent to the shell side of the secondary heat exchangers. Vendors of copper and copper alloy tubing have expressed the opinion that water containing slight amounts of ammonia may cause stress corrosion cracking

Copper-zinc alloys containing more than 15 percent zinc should not be used with alkaline solutions such as sodium hydroxide because of the possibility of dezincification corrosion. The Laboratory condensate on the shell side of the secondary reboiler may have a pH as high as 11 depending on the amount of blowdown and the concentration of solids due to the continual evaporation of liquid. Alkaline salts, such as sodium phosphate used in water treatment, also act like hydroxides but are less corrosive and therefore less likely to cause dezincification. The dezincification most apt to occur in alkaline or alkaline salt solutions is the plug type, in which brass dissolves as an alloy and the copper constituent redeposits from solution onto the surface of the brass as a metal, but in porous form. The zinc constituent is deposited in place as an insoluble compound. The plug-type dezincification proceeds in localized areas. Arsenic added to copper-zinc alloys, as in Admiralty brass, is an excellent inhibitor of dezincification.

Of the copper alloys, copper-nickel containing 1 percent zinc is the most resistant to dezincification and corrosion by ammonia or ammonium hydroxide. Another factor considered in the selection of the tubes was that copper-nickel is less affected by thermal cycling than Admiralty metal and therefore would reduce the tendency of the tubes to loosen in the tube sheet. The reason for considering Admiralty metal tubes for this application was the appreciable cost savings involved.

Type-A copper-nickel alloy, 80 Cu - 20 Ni with 1 percent zinc, was chosen instead of type B (3-4 percent zinc) because of its lower zinc content. Therefore, type A is less vulnerable to dezincification in the event that adverse water conditions occur.

The channel or water box end of the exchanger is a two-pass, bonnet-type design with a partition plate welded in place and two flanged inlet and outlet connections extending radially from the channel shell.

5.3.3 Reboiler Drain Tank

The secondary-reboiler drain tank accumulates the intermediate-steam condensate from the tube side of the secondary reboiler. The drain tank is located on the northeast side of the reboiler house between the secondary reboiler and the secondary drain cooler. The vessel is 36 inches in inside diameter by 7 feet in over-all length, horizontally mounted, and fabricated of carbon steel (ASTM-A-212 firebox quality). Two 6-inch-diameter inspection nozzles have been provided, one on the top of the shell and the other in the south head. The 6-inch pipe-size inlet and outlet nozzles are located tangentially to the top and bottom of the shell. Several pipe coupling nozzles are provided for operation and for instrumentation purposes such as liquid-level controller, high and low level switches, vents and drains. The float cage for liquid level controller G-334 senses the level of the condensate in the tank and controls the position of control valve G-334 which in turn regulates the condensate flow to the flash tank to maintain a constant level in the secondary-reboiler drain tank. The design of the vessel conforms to Section VIII of the ASME Code.

5.3.4 Drain Cooler

The secondary-system drain cooler is located outside the east wall of the shield cell within the reboiler building. The drain cooler, as shown in Figure 5-16, is a conventional two-pass shell, two-pass tube side, horizontal, U-tube, water-to-water heat exchanger. Intermediate condensate on the tube side raises the temperature of the incoming Laboratory condensate to the desired or saturation temperature before it enters the secondary reboiler. The operating conditions and capacities of the drain cooler are given in Table 5-17. The design of the drain cooler conforms to the design requirements of Section VIII of the ASME Code where applicable, otherwise to TEMA standards for Class A construction. The materials of construction are given in Table 5-18.

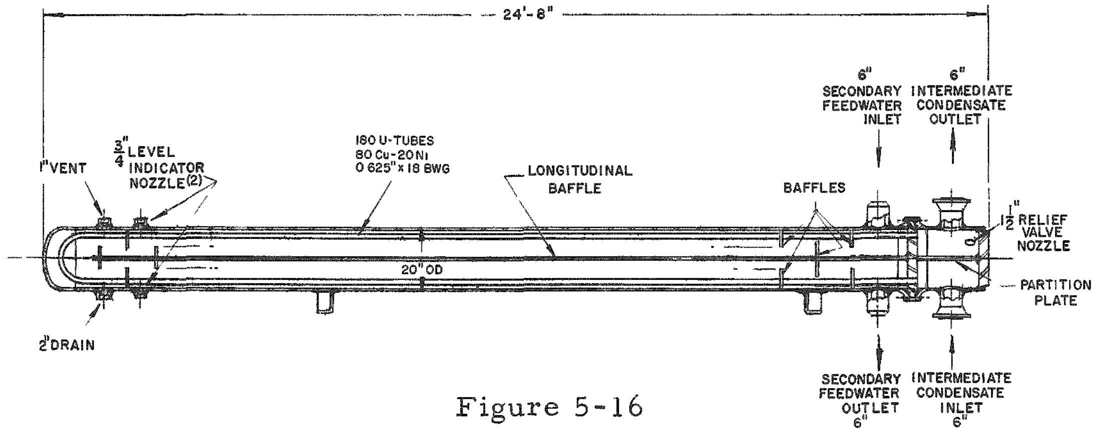

Figure 5-16

Secondary system drain cooler
RE-6-32685-D

Table 5-17. Operating Conditions and Capacities of Secondary Drain Cooler

Drain cooler tube surface	1349 sq ft
Heat exchanged - maximum	34.4×10^6 Btu/hr (10 mw)
Quantity of intermediate steam condensate to tube side	241,000 lb/hr
Tube-side pressure - operating	310 psig
- design	400 psig
- test	600 psig
Tube-side temperature - operating (inlet)	424.7°F
(outlet)	289.8°F
- design	448°F
Quantity of secondary system condensate to shell side	227,000 lb/hr
Shell-side pressure - operating	200 psig
- design	250 psig
- test	375 psig
Shell-side temperature - operating (inlet)	230°F
(outlet)	376.5°F
- design	406°F
Fouling factor	0.0005
Weight - dry	7200 lb
- flooded	10,000 lb

Table 5-18. Secondary Drain Cooler Materials of Construction

Shell	ASTM-A-106 Grade B Seamless
Shell head	ASTM-A-234 Grade WPB
Channel shell and cover	ASTM-A-212 Grade B Firebox quality
Flanges	ASTM-A-182 Grade 2 (30% max carbon)
Nozzle necks	ASTM-A-106 Grade B
Nozzle flanges	ASTM-A-181 Grade 1 (35% max carbon)
Tube sheet	ASTM-A-212 Grade B Firebox quality
Baffles and impingement plate	ASTM-A-283 Grade D

The shell is fabricated from seamless carbon steel pipe with a flange on the channel end for removal of the tube bundle. A longitudinal baffle, which is welded to the tube sheet as part of the tube bundle, divides the two-pass shell into two sections. Leakage around the edges of the longitudinal baffle between the upper and lower shell sections is presented by a Lamiflex seal. The seal is composed of eight light-gage strips of stainless steel sheet which leaves one side of the seal held flat against the baffle by a bolting bar and other side flexed upward to conform to the contour of the shell inside surface when the tube bundle is inserted into the shell.

The channel end is a flanged, two-pass, bonnet design. The channel head is a flat billet welded to the channel shell. The partition plate is welded to the channel shell and head, and the flanged inlet and outlet connections extend radially from the channel shell. A pipe coupling is welded to the channel for the installation of a small relief valve to guard against over-pressure from thermal expansion of the liquid during isolation of the exchanger from the system. The tube bundle is a two-pass U-tube configuration.

5.4 INTERMEDIATE AND SECONDARY SYSTEM WATER TREATMENT AND BLOWDOWN

In the intermediate and secondary systems, the solids entering each of the systems with the feedwater or addition of chemical water-treating agents accumulate in the shells of the primary and secondary reboilers since they normally do not leave with the steam. The total concentration of solids in the reboiler water is a function of the total solids in the feedwater, amount of chemical water-treating agents, corrosion products, and amount of blowdown. If a limited concentration is to be maintained, the solids introduced into the system must be removed by either continuous or periodic blowdown. Blowdown is the process of removing some of the concentrated-solids-bearing water and replacing it with feedwater to effect a general lowering of solids concentration in the systems. Excessive solids in the water caused deposition of sludge in the reboilers and may alter the surface tension of the water which could cause carryover of water with the steam due to excessive foaming of the water.

The intermediate system is a closed loop, and the concentration of solids in the primary reboiler shell does not change appreciably with time. The makeup for the intermediate system comes from the demineralized water system in the reactor service building, which is used to fill the primary system. The accumulation of solids in the primary reboiler shell is mainly due to corrosion products created in the intermediate system.

The amount of dissolved gases in the water is negligible. However, vents are provided on the system to bleed off any accumulated gases evolved after the initial fill or after long periods of operation.

Chloride concentration is maintained as close to zero as possible by using demineralized water for initial fill and by providing demineralized water for makeup. The maximum chloride concentration in the primary reboiler shell is attained after several weeks of operation and is equal to the chloride concentration of the makeup water multiplied by the ratio of feedwater flow to blowdown. A concentration below 0.3 ppm can be maintained. Periodic analysis of the water is made to check the chloride content.

The total dissolved solids are kept at a level of 100 to 300 ppm to prevent sealing of the heat transfer surfaces. Dissolved and suspended solids are controlled by periodic blowdown. Prior to blowdown of the primary reboilers, the water is checked for radioactivity.

The secondary system receives a constant supply of Laboratory feedwater and the steam produced is introduced to the Laboratory heating system. The constant supply of fresh feedwater results in an accumulation of solids in the reboiler shell. Shown below is a typical analysis of the Laboratory feedwater in which components are given in parts per million.

pH	9.4 to 9.7
P alkalinity	1.8
M alkalinity	7.2
Cl as NaCl	<1
PO_4 as Na_3PO_4	<1
SO_4 as Na_2SO_4	29.5
NH_3	9.32
Total dissolved solids	45.6

The secondary-reboiler blowdown is continuous and is maintained at approximately 5 percent of the feedwater flow rate. At maximum feedwater flow, the blowdown is approximately 11,500 pounds per hour. The dissolved solids, as determined by conductivity measurements, are maintained between 1000 and 2000 ppm. Although the typical Laboratory feedwater analysis shows 45.6 ppm dissolved solids, the dissolved solids content occasionally reaches 105 ppm and accounts for the higher range of dissolved solids concentration in the secondary reboiler.

The chloride content (reported as NaCl) in the secondary reboiler is maintained at a level of 10 to 20 ppm. Chlorides tend to accumulate in the reboiler shell with the concentration being equal to the feedwater content multiplied by the ratio of feedwater to blowdown.

Ammonia introduced with the feedwater does not accumulate in the secondary reboiler, because it is evolved with the steam and passes out to the Laboratory heating system. The concentration of ammonia in the secondary reboiler is 0.32 ppm, essentially the same as the feedwater.

The water in the secondary reboiler is analyzed daily, and the water in the primary reboilers is analyzed twice a week. Conventional methods are employed for the chemical treatment of the primary and secondary reboiler feedwater. The following treatment is performed by the addition of chemicals to the reboilers.

1. Sodium sulfite, Na_2SO_3, is added to scavenge dissolved oxygen, thereby lessening oxygen attack. Oxygen is removed by converting the sulfite to a soluble sulfate salt. The sodium sulfite concentration is maintained at 25 to 100 ppm.

2. Phosphates are added to the feedwater to maintain the pH in a range of 10.6 to 11.0. The mono-, di-, and trisodium orthophosphates are used and are maintained at a concentration of 100 to 300 ppm.

Figure 5-17 is a schematic diagram of the chemical feed system for the shell side of the primary reboilers. The intermediate chemical-feed system is of the batch type which utilizes the pressure drop across either of the intermediate-feedwater control valves (G-333) to inject the chemicals into the primary reboilers. The feed tank is initially loaded with solid chemicals, filled with water to dissolve the solids, and then closed and placed on-stream. The intermediate system requires only an occasional addition of chemicals.

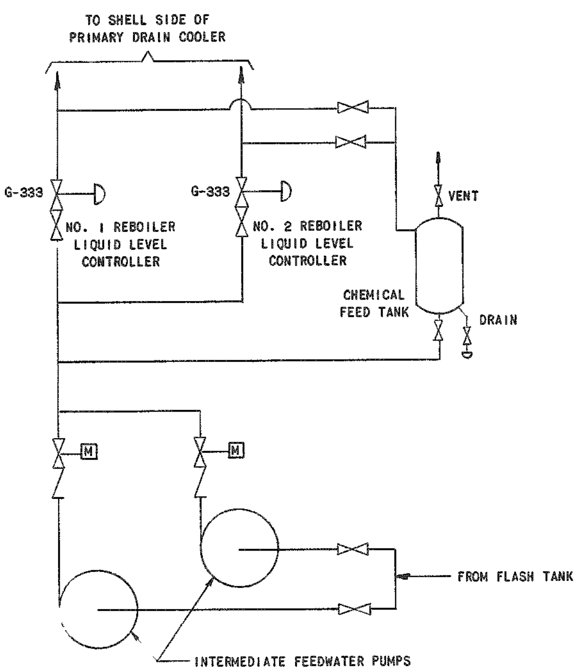

Figure 5-17

Flow diagram of intermediate system water treatment
RE-8-32725-A

The secondary chemical-feed system is a continuous type consisting of a chemical solution tank and a small feed pump for pumping the solution into the shell of the secondary reboiler. The pump has an adjustable stroke for varying the rate of feed to correspond with the blowdown rate. A schematic flow diagram of the system is shown in Figure 5-18.

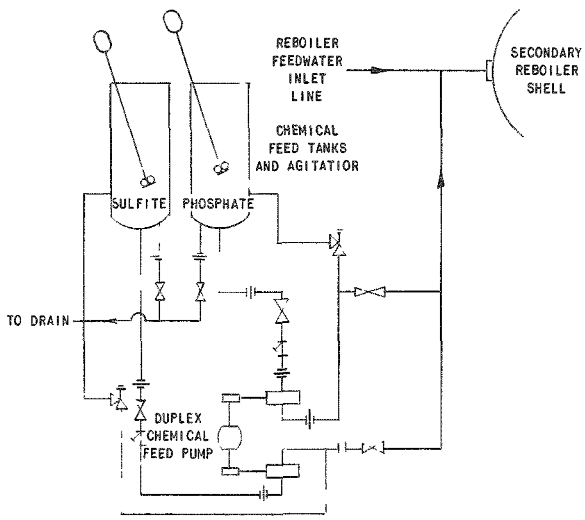

Figure 5-18

Flow diagram of secondary system water treatment
RE-8-32668-A

A blowdown tank is provided in the reboiler house for flashing the secondary-reboiler blowdown water to atmospheric pressure before disposal. The vessel is fabricated of carbon steel and is located in the pit along the south wall of the reboiler building. The capacity of the vessel is 300 gallons, and it is equipped with an 8-inch vent that extends through the roof for venting any flashed steam to the atmosphere and a 2-inch drain to the Laboratory sewer for the remainder of the blowdown water. A manhole is provided to allow for periodic inspection.

5.5 REBOILER BUILDING VENT AND EXHAUST SYSTEM

Five thermostatically controlled unit heaters are located in the reboiler building. Four heaters are in the building proper and the remaining heater is located at the end of the piping tunnel adjacent to the containment shell. The four heaters maintain the temperature within the reboiler building. The heater for the piping tunnel preheats the influent tunnel air to eliminate the possibility of freezing the stagnant condensate water in the pipe lines during winter operation when the plant is shut down.

The piping tunnel and the concrete shield cell within the reboiler building form a relatively gas-tight enclosure which surrounds the primary system. A blower in the exhaust duct above the cell maintains a negative pressure within the shield cell. The heater fan at the piping tunnel inlet draws 1400 scfm of fresh air into the enclosure. This air plus any inward leakage to the shield cell and tunnel is removed by the exhaust duct blower, which has a capacity of 2470 scfm at $\frac{3}{4}$-inch static pressure. A schematic diagram of the reboiler building ventilation system is shown in Figure 5-19.

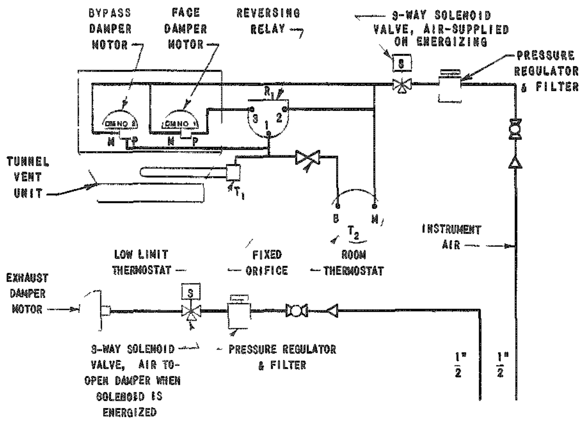

Figure 5-19

Schematic diagram of reboiler
building ventilation system
RE-8-33016-A

6. APPENDIX

6.1 PHYSICS CALCULATIONS OF EBWR*

The original physics calculations for EBWR(1,14) were made by two-group diffusion-theory methods, with appropriate choice of the group constants based on experiments on H_2O-U lattices. For multiregion problems, UNIVAC was used.

The original calculations resulted in an overestimation of the reactivity by 3.39 percent. The theory was then modified by describing the fast neutron leakage by exp. (τB^2) instead of $(1 + \tau B^2)$ and by taking the value of the resonance integral to be 1.209 times larger than that used originally.(14) With these modifications, there was agreement between experimental and calculated values.

6.1.1 Three-Group Theory Calculations

The original calculations were followed by the use of three-group structure for the cross sections, appropriately normalized in some instances, to conform with experimental results. The three-group equations are:

$$-D_1\nabla^2\Phi_1 + \Sigma_1\Phi_1 = \epsilon\nu(\Sigma_{f_2}\Phi_2 + \Sigma_{f_3}\Phi_3)$$

$$-D_2\nabla^2\Phi_2 + \Sigma_2\Phi_2 = \Sigma_{sl\,1}\Phi_1$$

$$-D_3\nabla^2\Phi_3 + \Sigma_3\Phi_3 = \Sigma_{sl\,2}\Phi_2$$

where

$$\Sigma_1 = \Sigma_{sl\,1}$$

$$\Sigma_2 = \Sigma_{sl\,2} + \Sigma_{(1/v)a2} + \Sigma_{res2}$$

$$\Sigma_3 = \Sigma_{a3}$$

$$\left.\begin{array}{l} \Sigma_{sl\,1} = D_1/\tau_1 \\ \Sigma_{sl\,2} = D_2/\tau_2 \end{array}\right\}$$ as defined by Deutsch,(15)

$\Sigma_{(1/v)a2}$ includes $(1/v)$ absorption by U^{238}

$$\Sigma_{res2} = \frac{1-p}{p}\Sigma_{sl\,2}$$

*H. P. Iskenderian and J. A. Thie

where p = resonance escape probability obtained in the usual way, using values of resonance integral $(RI)_{eff}$ of U^{238}, given by Sher,[16] with appropriate self-shielding corrections and with allowance for resonance in niobium.

The energy groups used in the three-group equations are:

Group 1 (fast) ∞ to 0.180 Mev

Group 2 (epithermal) 0.180 Mev to 0.625 ev

Group 3 (thermal) 0.625 ev to 0

Fast and epithermal cross sections were obtained by use of the equivalence-factors method given by Deutsch.[3] Epithermal cross section data for uranium was obtained from the OCUSOL 30-group cross section data (see ANL-5800[17] p. 157). Thermal cross sections were obtained using a Wigner-Wilkins spectrum according to Amster's compilation,[18] except that $\alpha^{U^{235}} = (\sigma_c/\sigma_f) = 0.184$ was used as in BNL-325. In obtaining cell cross section values, P_3 spherical harmonic approximations were used to determine the disadvantage factors.

6.1.2 Volume Fractions

Three types of fuel elements comprise the revised EBWR core and include the two plate-type elements of the original EBWR core. These two elements differ in the thickness of the fuel plate only and are referred to in this section as the thick (abbreviated to K) and the thin (abbreviated to T) elements. The third fuel element is the new, fully enriched, rod-type element and is referred to as the spike element (abbreviated to S). Physical constants of S given in this report were derived from design specifications and not necessarily from the constructed spike elements.

The basic data from which the volume fractions of the different elements were calculated are given in Tables 6-1 and 6-2. Volume fractions and isotopic densities for a fuel cell 12.75 inches, bound by the centerlines of the control-rod channels and including all reactor materials (guides, spacers, control rods or followers, and total water), are listed in Table 6-3. The data in this table were used for making RZ-geometry reactivity calculations.

Similar data for a cell with dimensions of 12.25 inches by 12.25 inches, including nine fuel boxes and bound by the control-rod guides, are given in Table 6-4. In this cell the control-rod channels are considered as distinct regions and data in the table are suitable for calculations of control-rod worth in XY-geometry.

Table 6-1. Areas Occupied by Materials at Room Temperature in a 12.75-Inch by 12.75-Inch Cell Consisting of Nine Similar Subassemblies

	Area, sq. in.	
	Thin elements	Thick elements
U	25.06	35.54
Nb	0.88	1.24
Zr meat	3.86	5.48
Zr clad	10.32	11.38
Zr sides	2.93	2.70
Zr guides	3.33	3.33
Zr spacers	1.25	1.25
Zr follower	2.50	2.50
H_2O between plates	104.12	90.83
H_2O in control channels	8.31	8.31
	162.56	162.56

Table 6-2. Characteristics of Fuel Elements

	Plate type fuel elements			
	Thin natural	Thick natural	Thin enriched	Thick enriched
Meat thickness, inches	0.172	0.240	0.172	0.240
Total plate thickness, inches	0.212	0.280	0.212	0.280
$\frac{H_2O\ \text{volume}}{U^{235} + U^{238}\ \text{volume}}$	4.116	2.597	4.116	2.597
Weight U^{235} in 6-plate element, grams	291.1	413.0	582.2	826.0

	Spike fuel elements
Composition, wt percent	9 UO_2 - 82.4 ZrO_2 - 8.1 CaO
Fuel dimensions	
Diameter of pellet	0.322 inch
Diameter of rod	0.375 inch
Length of rod	48 inches
Zircaloy thickness	0.025 inch
Clad-to-pellet gap thickness	0.0015 inch
Weight U^{235} in spike element	864 grams

Table 6-3. Volume Fractions and Atomic Densities in a 12.75-Inch by 12.75-Inch Cell at Room Temperature

Material / Fuel element type	Thin elements				Thick elements				Spike elements	
	Enriched		Natural		Enriched		Natural			
	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})
U^{235}	0.00244	0.0001156	0.001276	0.0000604	0.00336	0.0001592	0.001739	0.0000823	0.0036	0.0001702
U^{238}	0.1598	0.00756	0.1610	0.007615	0.2230	0.01055	0.22474	0.01063	0.000262	0.0000124
U	0.1623	0.007675	0.1623	0.007675	0.2264	0.01071	0.2264	0.01071	0.003862	0.0001826
H_2O	0.668	0.02239	0.668	0.02239	0.588	0.01970	0.588	0.01970	0.582	0.01950
Zr	0.134	0.005650	0.134	0.005650	0.148	0.006245	0.148	0.006245	0.250	0.01057
Nb	0.00578	0.0003151	0.00578	0.0003151	0.00813	0.0004431	0.00813	0.0004431		
Ca									0.0365	0.0003511
O in spike meat										0.00757
Fe	0.0302	0.00256	0.0302	0.00256	0.0302	0.00256	0.0302	0.00256	0.0302	0.00256

Notes:

1. The rod follower has a volume fraction of 0.0302. In the stainless steel fueled follower, N(U^{235}) = 0.0001737 x 10^{24} and N(Fe) = 0.0848 x 10^{24}.
2. The control rod has the same volume fraction as the follower. In the rod N(Fe) = 0.0765 x 10^{24} and N(B) = 0.00875 x 10^{24}.
3. U^{235} in the fueled follower is included with U^{235} in the cell. When Zircaloy-2 follower is used the required correction is then made.

Pertinent volume fraction data on control-rod followers are listed in Table 6-3. Included in the table are data for control-rod followers composed of Zircaloy-2 material and fuel-containing stainless steel plates. The stainless steel fueled followers were tested in EBWR as a possible replacement for the Zircaloy-2 followers in the 100-megawatt-capacity system. When test objectives were not realized, the fueled followers were abandoned in favor of the Zircaloy-2 followers.

The physical dimensions of the followers are the same as the control rods except that the followers are 14.37 inches long.

Table 6-4. Volume Fractions and Atomic Densities in a 12.25-Inch by 12.25-Inch Cell at Room Temperature

Material / Fuel element type	Thin elements				Thick elements				Spike elements	
	Enriched		Natural		Enriched		Natural			
	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})	Volume fraction	Atomic density (x 10^{24})
U^{235}	0.00253	0.0001196	0.001265	0.0000598	0.00353	0.000167	0.001765	0.00008454	0.00378	0.0001789
U^{238}	0.1735	0.00819	0.1747	0.00825	0.24164	0.01143	0.24341	0.011513	0.000285	0.00001345
U	0.176	0.0083096	0.176	0.0083096	0.245	0.0011597	0.245	0.0011597	0.00407	0.00019235
H_2O	0.683	0.02287	0.683	0.02287	0.596	0.01995	0.596	0.01995	0.589	0.01973
Zr	0.135	0.005742	0.135	0.005742	0.151	0.006386	0.151	0.006386	0.262	0.01107
Nb	0.00626	0.0003414	0.00626	0.0003414	0.00881	0.00480	0.00881	0.00480	0	0
Ca									0.040	0.000922
O in spike meat*										0.00802

Notes:

In the spikes, the meat is in the form of ZrO_2-UO_2 and CaO (with a slight excess of O). The number densities in the meat are (x 10^{24}): U - 0.000818, Zr - 0.016233, Ca - 0.003924, and O - 0.038048. The volume fraction of the meat in the cell is 0.2352.

6.1.3 Three-Group Cross Sections

Hot (485.6°F), clean cross sections with voids as a parameter are given in Table 6-5. These cross sections correspond to their diffusion-theory definitions as given by the three-group-theory equations given above.

The cross sections referred to above were normalized to yield the correct hot, clean reactivity of the original EBWR core. In view of the large uncertainty in the value of the effective resonance integral (RI_{eff}), this parameter was chosen to be adjusted. The value of RI_{eff}, calculated by Sher's formula[16] and corrected for spatial self-shielding and for the contribution due to niobium, was increased by a factor of 1.096 in order to match the experimental reactivity. This correction is within the limits of uncertainty in the parameter.

Some loadings of the core, in which the outer region of the core was composed primarily of KE elements, contained several KN elements as well. The KN elements were regarded as perturbations and their effect on the region was compensated for by multiplying Σ_{a2} and Σ_{a3} in Table 6-5 by the factor 0.9603, and $\nu\Sigma_{f2}$ and $\nu\Sigma_{f3}$ by the factor 0.9385.

Table 6-5. Hot Macroscopic Cross Sections

Material ↓ / Reactor constants →	D_1, cm	$\Sigma_{1\to2}$, cm^{-1}	D_2, cm	Σ_{a2}, cm^{-1}	$\Sigma_{2\to3}$, cm^{-1}	$\nu\Sigma_{f2}$, cm^{-1}	D_3, cm	Σ_{a3}, cm^{-1}	$\nu\Sigma_{f3}$, cm^{-1}
TE, 0% voids	1.492	0.0356	0.567	0.012015	0.0382	0.004887	0.2778	0.0603	0.08586
TE, 30% voids	1.749	0.0247	0.669	0.011850	0.0257	0.004887	0.3912	0.05766	0.08647
TE, 60% voids	2.107	0.0144	0.813	0.012075	0.0144	0.004887	0.5848	0.05470	0.08783
KE, 0% voids	1.366	0.0319	0.509	0.01428	0.0327	0.006372	0.2947	0.07382	0.11199
KE, 30% voids	1.550	0.0224	0.580	0.01493	0.0222	0.006372	0.4167	0.07122	0.11130
Spike, 0% voids	1.840	0.0344	0.9862	0.005506	0.0384	0.00855	0.321	0.07213	0.1313
Spike, 30% voids	2.171	0.0285	1.292	0.005411	0.0263	0.00855	0.434	0.06730	0.1255
Spike, 60% voids	2.469	0.0285	1.831	0.005316	0.0143	0.00855	0.760	0.05965	0.1139
TN, 0% voids	1.492	0.0356	0.567	0.01550	0.0382	0.002443	0.2839	0.04100	0.043870
TN, 60% voids	2.107	0.0144	0.813	0.01068	0.0144	0.002443	0.6135	0.03506	0.044820
Reflector H_2O	2.067	0.05117	0.7951	0.000553	0.0620	0	0.2934	0.01138	0
SS-fueled follower	0	0	0	0.341	0	0.230	0	4.02	3.807
Burnup - TE	0	0	0	1.17	0	0	0	6.70	7.34
Burnup - KE	0	0	0	0.560	0	0	0	5.40	6.94

Notes:

1. The cross sections for the TE and KE elements have been computed for the 12.75-Inch by 12.75-Inch sublattice containing 9 elements. Zircaloy-2 control rod followers are assumed to be in the core, and all structural materials (guides and spacers) are included.
2. The cross sections for the spikes have been computed for a 3.875-Inch by 4.25-Inch elemental box, and include, besides the fuel rods, only the structural materials associated with the spike element itself (sides and corner angles). Since the spike elements are never clumped together and are always adjacent to other types of elements it would have been incorrect to consider a 9-element sublattice composed of spike elements alone.
3. The slowing down cross sections are slightly dependent on the leakage from the reactor (that is, on the product τB^2). In order to avoid an unnecessary multiplicity of materials, an average B^2 of 0.00155 cm^{-2} was used for the quoted slowing down cross sections.
4. The parameter exhibiting the largest nonlinearity with voids is p (the resonance escape probability, here included in the Σ_{a2} of the TE, TN, KE, and KN elements). It was approximated with two lines of differing slopes (from 0% voids to 30% voids to 60%) with a maximum deviation at 50 percent void of 1.2 percent. At all of the other void concentrations the approximation is more accurate.

6.1.4 Control-Rod and Control-Rod Follower Cross Sections

In the calculation of control-rod and control-rod-follower cross sections for the operating reactor, the effect of control-rod-bank insertion depth was simulated. The problem of partially inserted rods was resolved by including an equivalent control-rod poison in the cross sections of the control-rod-containing regions. Equivalent poison is defined as a uniformly distributed poison that has the same experimental reactivity effect as the control-rod bank.

The revised core, containing control rods with fueled followers, also had the problems arising from the fueled followers which added to the absorption and fission cross sections.

The cross sections affected by the equivalent poison technique of calculation were Σ_{a_2} and Σ_{a_3} only. The required values were obtained by an iterative procedure that compared calculated reactivities with the experimentally obtained control rod calibration curve. The converged values of cross sections for the original grouping of five hafnium and four boron control rods were:

$$\Delta\Sigma_{a_2} = 0.000702 \text{ and } \Delta\Sigma_{a_3} = 0.0114 \quad .$$

These macroscopic cross sections are simply added to the epithermal and thermal cross sections of the regions containing control rods. They were found to be fairly insensitive to control rod position.

A separate experiment to compare boron and hafnium control crosses for the same location showed that $\frac{1}{4}$-inch-thick boron-stainless steel was more effective by a factor of 1.03 than $\frac{1}{8}$-inch-thick hafnium.

The cross sections given in Table 6-5 are computed for a core with Zircaloy-2 control-rod followers. When this is not the case, as in a core with fueled followers, the other materials (control rods or fueled followers) are assumed to have the same effect on neutron diffusion properties as the Zircaloy-2 followers, and only the absorption and fission cross sections need to be altered. Therefore, these materials were assumed to occupy negligible volume and are specified correctly by giving them a volume fraction of 0.001. For problems in which control rods are only partially inserted, the cross sections contributed by the followers are added to the regions containing control-rod followers. Cross section data on followers is given in Table 6-5.

6.1.5 Reactivity and Flux

All experimental reactivity measurements reported here are based on the original control rod calibration.(14) In view of the use of β_{eff} = 0.00825 in the original calibration work, the experimental measurements were converted to dollar values at 0 megawatt-day exposure ($°) to make the experimental measurements independent of β_{eff}. Subsequent results for the effective delayed-neutron fraction based on recent Keepin data (ANL-5800[17] p. 23) and perturbation theory have shown:

$$\beta^{\circ}_{eff} = 0.00717$$

$$\frac{1}{\beta^{\circ}_{eff}} \frac{d\beta_{eff}}{d(\Phi t)} = 1.42 \times 10^{-5} \text{mwd}^{-1} \text{ (Calculated)}$$

$$= 2 \pm 0.4 \times 10^{-5} \text{mwd}^{-1} \text{ (Measured)}$$

These values are used to relate calculated reactivity to dollars.

The measurement of the changing delayed-neutron fraction due to burnup was accomplished by comparing periods late in core life with those obtained at 0 megawatt-day with the same control rod settings. If this measurement is corrected for a calculated decrease in control rod sensitivity (due to an increasing core Σ_a with burnup),

$$\frac{1}{\left(\frac{d\rho}{d\Sigma_a}\right)_0} \frac{d\left(\frac{d\rho}{d\Sigma_a}\right)}{d(\Phi t)} = -0.5 \times 10^{-5} \text{ mwd}^{-1} \quad ,$$

the measured burnup coefficient of β_{eff} would be:

$$\frac{1}{\beta_{eff}} \cdot \frac{d\,\beta_{eff}}{d(\Phi t)} = -2.5 \times 10^{-5} \text{mwd}^{-1} \quad .$$

For the proposed loading X of Figure 6-3 (spiked core with 8813 megawatt-days operation), the value of β_{eff} was calculated to be 0.00655.

Figures 6-1 and 6-2 show two almost identical loadings in which most of EBWR operation took place. From periods and rod positions when loading changes were made, the relative reactivities shown in Table 6-6 were determined. These reactivities do not include burnup effects.

Table 6-6 Measured Reactivities of EBWR Loadings

Loading number	Relative reactivity, $°[a]
46	0 00
47	0 92
48	0 74
49	0 76
50	0 92
51	0 58
52	(not measured)
53	0 85

[a]Values given in terms of dollars at 0 megawatt-day.

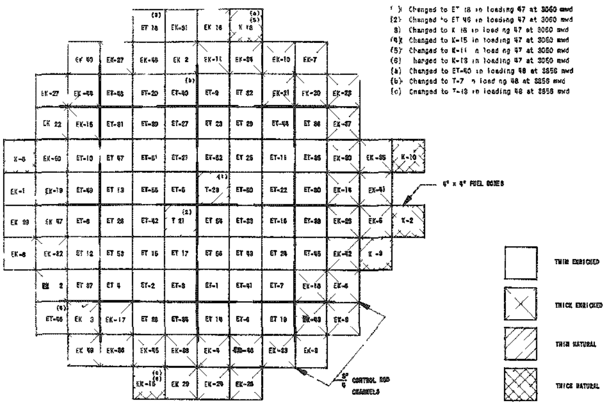

Figure 6-1

Loading 46 operated from
0 to 3060 megawatt-days
Neg. 111-9124

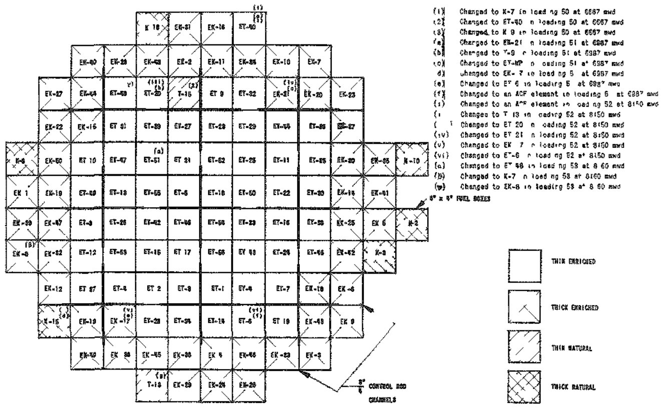

Figure 6-2

Loading 46 operated from
4629 to 6667 megawatt-days
Neg. 111-9125

Theoretical reactivity calculations of these loadings were performed using zones of standardized composition, rather than attempting exact simulations of all of these minor changes. Figure 6-4 shows the basic zones. A reduced fuel enrichment is used in the calculations in lieu of the few natural uranium elements in the outer zone. Void and burnup nonuniformities are adequately approximated by averages in these regions.

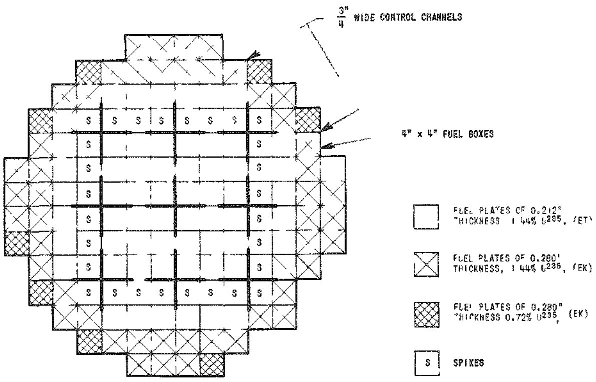

Figure 6-3

Loading X, a possible spike-fuel loading
RE-6-35111-A

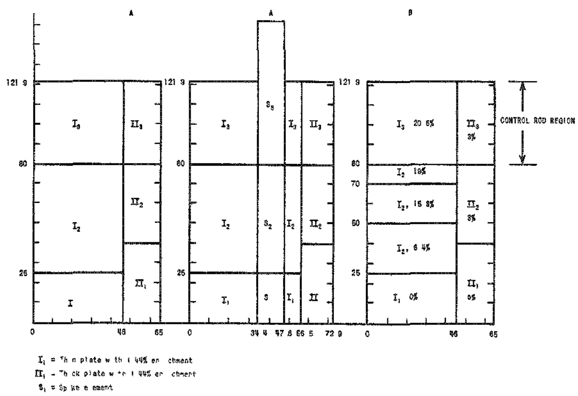

Figure 6-4

Zone patterns used in
EBWR calculations
RE-7-35112-B

Measurements of reactivity in voids were made at various powers and times during core life. These measured values were obtained from differences in critical experiment rod positions at power and zero power, and corrected for possible difference in temperature and xenon. The precision of the zero-power critical experiment measurements in the presence of transient temperature and xenon effects was ensured by actually measuring the transients (by determining their effect on a series of reactor periods).

The results in Table 6-7 show the reactivity in voids to be insensitive to minor burnup and loading changes. However, for a given power, the reactivity in voids was found to depend on the position of the control rods. Reactivity is reduced by about 0.035 \$° for every inch upward in the location of the tips of the nine rods for the 20-megawatt, 600-psig conditions encountered. This stems from a reduction in power density and void coefficient magnitude.

Table 6-7. Measured Values of the Reactivity in Voids at 600 psig

Loading number	Burnup, mwd	Distance of rod tips from top of fuel, inches	Power, mw	Reactivity in voids, \$°
46	0	15.2	19.6	2.67
51	6987	19.0	13.5	2.00
51	6987	16.5	20.4	2.67
52	8150	0	42.7	4.04
53	8450	14.4	20.4	2.68
53	8450	4.9	38.4	3.99

Measurements of power and flux distributions shown in Figures 6-5 and 6-6 were obtained from gamma-ray and cobalt-activation data respectively.

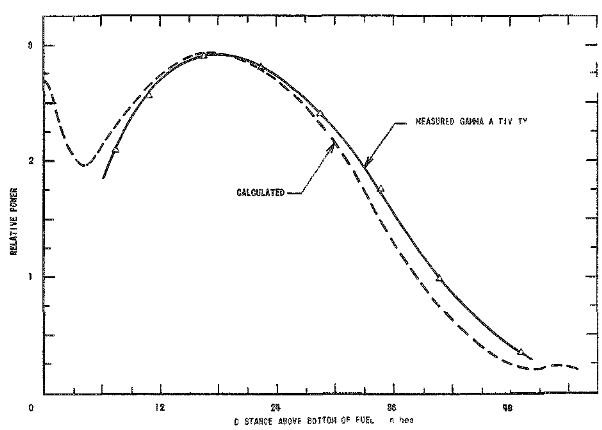

Figure 6-5

Axial power distribution at 20 megawatts early in core life RE-7-35113-B

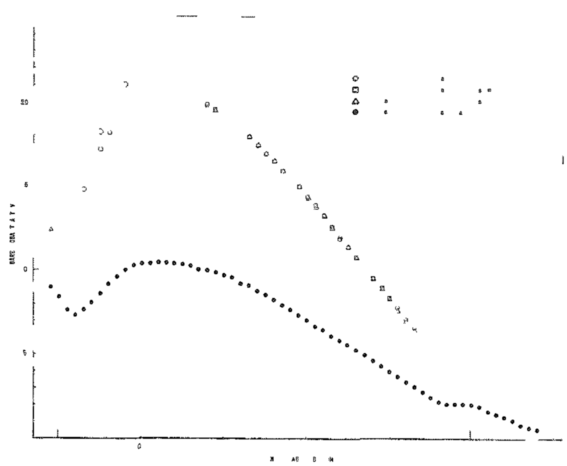

Figure 6-6

Axial flux distributions obtained from bare cobalt wires Neg. 111-9127

It can be seen that agreement with the three-group calculations described is satisfactory. Figure 6-7 shows a determination of the coolant void fraction from its theoretical effect on the measured cadmium ratio.[19]

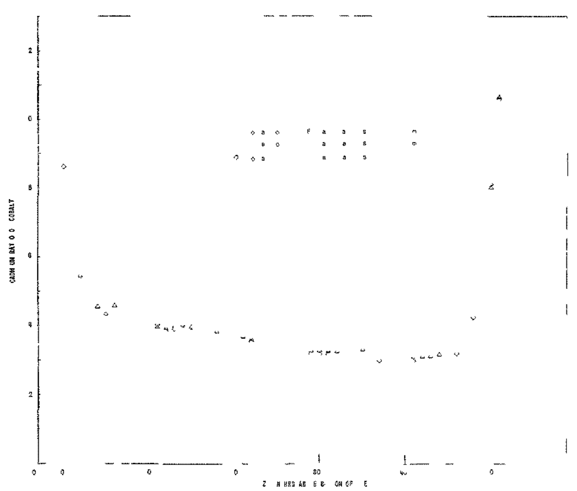

Figure 6-7

Cadmium ratio of cobalt wires as a function of height
Neg. 111-9129

A number of void-dependent calculations were made to check the cross sections and methods with experiment. In particular, the experimental run on loading 51 given in Table 6-7 was well suited for theoretical analysis. It was performed after a long shutdown and was xenon free.

The core was divided into eight regions and the void content of each region was represented by an appropriate average. The calculation was performed by the usual physics and heat transfer iterations. The final void distribution converged upon is given in Figure 6-4. Two approximations present in this distribution are:

1. Hydraulic calculations assume voids to be produced only in coolant adjacent to fuel, whereas this void fraction is used in the water everywhere.
2. Local boiling in the subcooled region is ignored.

To some extent the errors in these approximations tend to cancel. The reactivity in voids calculated by this procedure was 2.67 dollars for a power of 20 megawatts. The agreement with experiment is very close, for Table 6-7 indicates 2.67 dollars for 20.4 megawatts.

The fission-density distribution along the centerline of the converged problem is given in Figure 6-5. An experimentally obtained gamma-flux distribution is plotted alongside the calculated curve so that the two curves are normalized at their maxima. As seen in the figure, the agreement is satisfactory. The absence of end effects for the experimental curve is due to the measuring technique. The curve was obtained by a small gamma chamber moving between the plates of an element. As the chamber approached the fuel element end, its volume of integration extended beyond the element where the flux dropped abruptly. Furthermore, the measurement is for a central element of a nine-element sublattice. That is, the element is as far away from a control rod as possible, accordingly its flux is higher in the control rod region than for the theoretical curve which incorporates a homogenized control rod effect.

6.1.6 Isotope Buildup and Burnup

A boiling water reactor encounters uneven burnup for two reasons. First, the unavoidable power gradient is intensified by the presence of partly inserted control rods, and second, the uneven void distribution strongly influences the resonance escape probability. To obtain adequately averaged values for these parameters, the 20-megawatt EBWR core was divided into six regions as shown in Figure 6-4. Average power densities and void content were calculated for each region, average concentrations of the self-shielding isotopes (Pu^{239} and Pu^{240}) were estimated and their concentration-dependent cross sections were obtained. Effective one-group cross sections were calculated by using the resonance and thermal flux ratios available from three-group calculations. Then the various isotope concentrations at a burnup of 10,000 reactor megawatt-days were computed independently for each region. The isotope concentrations are given in Table 6-8.

Table 6-8. Isotope Concentrations[a] at 10,000 Reactor Megawatts-Days Exposure

Region[b]	ΔN^{25} [c]	N^{26}	N^{49}	N^{40}	N^{41}	FP[d]
I_1	-2.309	0.47	1.44	0.192	0.14	2.87
I_2	-2.54	0.49	1.66	0.240	0.148	3.21
I_3	-0.824	0.16	0.640	0.093	-	0.927
II_1	-1.25	0.22	0.91	0.089	-	1.48
II_2	-1.38	0.242	1.12	0.11	-	1.66
II_3	-0.403	-	0.28	-	-	0.48

[a]All units are times 10^{19} atoms/cm^3.

[b]See Figure 6-4 for location of regions.

[c]ΔN^{25} is the change in 25 concentrations.

[d]FP refers to general fission products.

The concentrations in Table 6-8 were translated into three-group operating temperature macroscopic cross sections. The changes in cross sections are given for each region in Table 6-9. The values as shown are fine-structure-flux averaged values which can be added to the void-dependent cross sections of Table 6-5.

Table 6-9. Changes in Cross Sections with Burnup at 10,000 Reactor Megawatt-Days Operation

Region	$\Delta \Sigma_{a_2}$	$\Delta \Sigma_{a_3}$	$\Delta \nu \Sigma_{f_3}$
I_1	0.001030	0.00604	0.00637
I_2	0.00132	0.00737	0.00831
I_3	0.000550	0.00346	0.00444
II_1	0.00482	0.00462	0.00568
II_2	0.000638	0.00618	0.00820
II_3	0.000128	0.00136	0.00160

Notes:

All $\Delta\Sigma$ values are positive. $\Delta\nu\Sigma_{f_2}$ is very close to zero as the opposite sign contributions of $-\Delta N_{U^{235}}$ and $+\Delta N_{Pu^{239}}$ cancel each other. $\Delta\nu\Sigma_{f_3}$ is positive because of the higher reactivity of Pu^{239}.

The reactivity change due to the cross sections given in Table 6-9 was calculated by the two-dimensional PDQ-2 code and was found to be -5.55 \$ or -0.55 \$ per 1000 reactor megawatt-days.

Figure 6-8 shows the reactivity loss due to samarium and long-term burnup effects as obtained from control rod positions at zero power during core life. The lower rate of loss at 489°F is associated with the temperature coefficient becoming less negative as burnup progresses because of plutonium buildup. This effect can be observed by measuring the temperature coefficient as illustrated in Table 6-10.

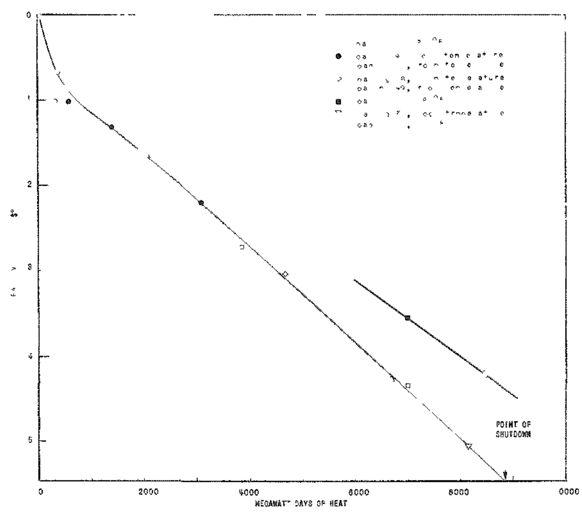

Figure 6-8

Reactivity loss due to samarium and burnup in loadings 46 through 53

Neg. 111-9244

Table 6-10. Reactivities Due to Temperature Changes

Burnup at time of measurement, mwd	$\Delta\rho$ in \$° T_1 = 68°F T_2 = 489°F	$\Delta\rho$ in \$° T_1 = 328°F T_2 = 469°F
-	3.55	1.82
6987	1.76	-
8150	-	1.06

As measured, the reactivity loss due to a temperature rise is:

$$\Delta\rho = \int_{Z(T_1)}^{Z(T_2)} \frac{d\rho}{dZ}(T)\, dZ$$

where $d\rho/dZ$ (T) is the measured reactivity per unit height at various temperatures and positions of the control rods (see Figure 6-9).

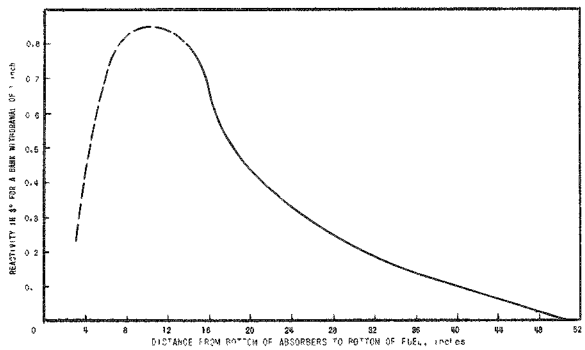

Figure 6-9

Differential worth of a nine-control-rod bank at room temperature in loading 46
Neg. 111-9230

6.2. RADIOACTIVITY AND CORROSION PRODUCTS IN EBWR*

Radioactivity and corrosion products are formed in a water-cooled, water-moderated reactor. These contaminants cause problems of hazards to personnel, interference of moving parts, and fouling of heat transfer surfaces. All three of these problems have been encountered in EBWR.(20)

6.2.1 Sources of Radioactivity

The primary radioactivity originating from water in EBWR is N^{16} (half-life 7.35 seconds) produced from O^{16} by a fast (n, p) reaction. The N^{16} nuclide decays with emission of β^- and gamma rays and is responsible for most of the radiation background around the external parts of the system during operation. Radiation levels at various components of the EBWR system, attributed to N^{16} at different reactor powers, are listed in Table 6-11. Because of the short half-life of N^{16}, this radiation disappears very quickly after plant shutdown.

*C. R. Breden

Table 6-11. Equipment Surface Activities Attributed Primarily to N^{16}

Location	10 mw(t)	20 mw(t)	40 mw(t)	61.7 mw(t)
	Milliroentgens per hour			
Air ejector aftercooler	90	400	4000	7000
Condenser hotwell	10	40	240	580
Steam drier	80	150	600	720
Turbine exhaust casing	2	4	14	20
Feedwater filters	8	10	50	100
Plant air exhaust	6	12	40	120

Small amounts of fission product activities are usually present in EBWR reactor water and steam, but their source is often difficult to establish. In the early days of EBWR operation, fission product activities were believed to have been caused by about 50 milligrams of fuel that was assumed to be surface contamination on fuel plates. This contamination resulted when two or three fuel elements ruptured during preliminary autoclave tests. Fuel elements subsequently tested in these autoclaves picked up the contamination. Other sources of small amounts of fission products that have been suggested are: (1) the reported presence of about 1 ppm of uranium as an impurity in Zircaloy-2 and (2) diffusion of fission products through fuel element cladding.

Large amounts of fission products result from defects or ruptures in fuel elements. Experiments with deliberately defected fuel element samples in EBWR showed that a high percentage of fission products that entered the water and steam were decay products of fission gases that diffused out of the fuel through the defect and into the water.(21)

Corrosion products from EBWR structural materials become radioactive because of exposure to the neutron flux and are transported by the water to external parts of the system. Structural materials in or near the core may become activated "in situ" and, through either the corrosion process or recoil, are released to the water to be carried and deposited elsewhere in the system. Another source of nuclides is the corrosion products released in other parts of the system and carried by the water into the high-neutron-flux area of the system. A further developement of this condition occurs when the corrosion products originating in out-of-flux areas are carried into the core, become deposited on fuel elements, reside for a time, and are released again to the water to be carried and deposited elsewhere. This condition was encountered in EBWR when radioactive corrosion products were deposited in the recirculation nozzles and control-rod-guide bushings below the reactor pressure vessel. As a result the nozzles and bushings gradually reached activity levels as high as 90 roentgens at contact with accompanying dose levels at head height above the control-rod-drive-motor platform of 500 milliroentgens per hour. The hazard to personnel resulting from redeposited radioactive corrosion products necessitated clean-out of the nozzles and bushings in May 1958 to reduce the activity.

The deposit in the recirculation nozzles, shown in Figure 4-3, appears to be largely composed of flakes spalled off of fuel elements. Chemical analysis showed results similar to the analysis of deposits removed from fuel elements. The high activity level in the deposits indicates that the material has been subjected to very high radiation fluxes for significant periods of time. The chemical analysis shows that the corrosion products originated, not on the fuel elements, but elsewhere in the system. They then deposited on the fuel elements, remained for an undetermined period of activation and spalled off to settle in low velocity areas.

It is of interest to note that the purification system was the only other part of the water system where significant amounts of radioactive corrosion products were found to accumulate. This accumulation had been predicted during the design phase and was provided for by shielding.

Because of the high decontamination factor, about 3×10^2, between reactor water and steam, only very small amounts of radioactive corrosion products have been found deposited in the turbine, condenser, or other components of the primary steam system. Radiation levels on internal surfaces of components of the primary steam system, measured after shutdown following one year of operation [22] are listed in Table 6-12.

Table 6-12. Activity Levels in Components of Steam System after Shutdown[a]

Component	Activity, mr/hr[b]
Steam drier	20
Turbine	
Steam chest	5.0
Blades	0.2
Condenser	
Top of tube bank	0.2
Hotwell	2.0

[a]Following one year of operation
[b]Measured at 2 inches on internal surfaces

To determine the long-lived activity levels reached after about $2\frac{1}{2}$ years of operation, a radiation survey was made on March 25 1960, about nine months after reactor shutdown for conversion to 100-megawatt-capacity system. The location and survey meter readings of the highest levels are given in Table 6-13.

Table 6-13. Activity Levels Nine Months after Shutdown

Location	Activity level, mr/hr (hard)
Steam drier and emergency cooler	1.3 at 1 in.
Feedwater filter No. 1	1.5 at 1 in.
Feedwater filter No. 2	9 at 1 in.
Startup heater	16 at 1 in.
Feedwater pump filter No. 1	7 at 2 in.
Feedwater pump filter No. 2	7 at 1 in.
Retention tank No. 1	22 at 2 in.
Retention tank No. 2	20 at 2 in.
Prefilter No. 1	18 at 2 in.
Prefilter No. 2	5 at 1 in.
Ion exchanger No. 1	21 at 2 in.
Ion exchanger No. 2	17 at 2 in.
After filter No. 1	32 at 2 in.
After filter No. 2	100 at 2 in.
Regenerative cooler No. 1	100 at 2 in.
Secondary cooler No. 1	310 at 2 in.
Regenerative cooler No. 2	130 at 2 in.
Secondary cooler No. 2	240 at 2 in.

6.2.2 Deposition of Corrosion Products

The deposition of particulate matter in EBWR moving parts has been a serious problem at times. In one instance, when this material settled in the control rod bushings the resulting mechanical interference prevented the control rod from moving closer than two inches to the "In" position. As a result of the corrosion product deposition difficulty, it was necessary to change the design of the bushings (see Section 4.2.3.1).

During the June-July 1957 shutdown period, fuel element ET-5 in position 129-370 was transferred to the storage pit for examination of the scale deposit. The fuel element had a nonadherent film which could be removed by wiping with a cellulose sponge. In December 1957, after another six months of operation, element ET-5 was re-examined. A hard adherent scale had formed on all the plates except in the high flux region of the outside plates where it had spalled off. This condition is shown in Figure 6-10. A piece of scale that flaked off was recovered and measured to be 0.003 inch thick. At this time none of the other elements showed flaking or spalling of the scale.

Another element, T-23, was removed at this time and subjected to destructive examination. Descaling of sample coupons removed from areas of different burnup along the plate showed good correlation between scale thickness and burnup as shown in Figure 6-11. The maximum scale thickness found was about 0.003 inch.

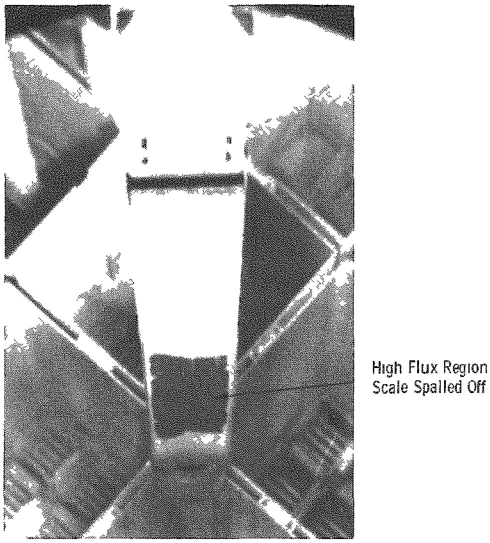

Figure 6-10

Fuel element showing spalled off scale in high flux region
Neg. 111-6082

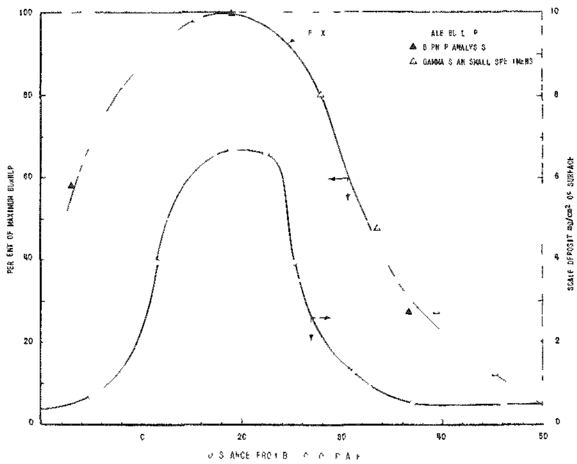

Figure 6-11

Relation between scale buildup and fuel burnup after operation for about one year
Neg. 112-1180

A third element, ET-51, examined for scale thickness was removed April 20, 1959, a few weeks before the final shutdown for 100-megawatt conversion. A piece of scale removed from the element was found to be about 0.005 inch in thickness. After shutdown of the reactor, pieces of scale were found which ranged from 0.004 inch to 0.008 inch in thickness. A visual examination of all the fuel elements during reloading in March 1960 showed that a high percentage exhibited areas of spalling or flaking.

Preliminary analyses showed the deposits to be largely oxides of iron, nickel, and aluminum with the relative proportions decreasing in the order given. Subsequent analyses showed the deposits to be largely aluminum oxide with less amounts of iron and nickel oxides. Relative amounts of iron and nickel have been reversed in some of the analytical results. This apparent reversal may be because the spectrographic method of analysis was accurate only to within a factor of two. A recent wet chemical analysis, with a higher degree of accuracy, was conducted on a sample of fuel element scale deposit; the result is given in Table 6-14.

Two possible sources exist for the aluminum oxide found in the scale: (1) the aluminum-nickel dummy fuel elements and (2) the aluminum condenser tubes. Several sources of nickel oxide are possible, and, in order of probability, are: (1) the aluminum-nickel dummy fuel elements, (2) the Kanigen nickel from the turbine, and (3) minor amounts from the stainless steel piping and components. The iron oxide originates from the plain steel and alloy piping.

Table 6-14. Composition of EBWR Fuel Element Scale

Element	Analysis, wt %	Assumed compound	Calculated, wt %
Al	36.3	$Al_2O_3 \cdot H_2O$	80.6
Ni	10.6	NiO	12.6
Fe	3.7	Fe_3O_4	5.1
Si	0.8	SiO_2	1.6

The existence of silicon in the fuel element scale can be accounted for by its presence in the steel piping and aluminum-nickel dummy fuel elements. Another source of silicon is condenser in-leakage.

The aluminum, nickel, and iron in the scale deposits were assumed to be present as oxides. X-ray diffraction studies have resulted in positive identification of boehmite, $\alpha Al_2O_3 \cdot H_2O$ in amounts estimated at greater than 50 percent, and usually, an unidentified second phase is reported. Assuming that all of the aluminum is present as boehmite and that the other elements are as given, the composition of the scale as determined by calculation is given in Table 6-14. The long-lived radioactive nuclides found on a sample of scale removed from an EBWR fuel plate in January 1958 are given in Table 6-15.

Table 6-15. Radioactive Particles in EBWR Fuel Element Scale Deposit

Nuclide	Half-life	Disintegrations per min per mg of scale	Probable nuclear reaction
Co^{58}	71 d	1.43×10^7	$Ni^{58}(n,p)Co^{58}$
Co^{60}	5.3 y	1.66×10^6	$Co^{59}(n,\gamma)\ Co^{60}$ $Ni^{60}(n,p)\ Co^{60}$
Fe^{59}	45 d	2.43×10^5	$Fe^{58}(n,\gamma)\ Fe^{59}$
Mn^{54}	300 d	2.08×10^5	$Fe^{54}(n,p)\ Mn^{54}$

The possible effect of the scale deposits on the operating temperature of the fuel elements has been of concern, particularly with respect to 100-megawatt operation. The thermal conductivity of the scale, measured on as removed specimens in the dry condition, was 0.44 ± 0.09 Btu/hr-ft-°F. Calculations based on a maximum heat flux of 485,000 Btu/ft^2-hr and scale deposits ranging from 0.002 to 0.008 inch in thickness predict that the maximum center-line temperatures of the fuel at 100-megawatt operation will be as shown in Table 6-16.

Table 6-16. Calculated Fuel Element Central Zone Temperatures

Case	Thickness of deposit, in.	Assumed number of heat transfer surfaces covered	Central temperature, °F		
			Ave	Min	Max
1	0.002	1	848	834	864
2	0.004	1	909	892	935
3	0.002	2	949	919	998
4	0.004	2	1133	1070	1229
5	0.008	2	1499	1375	1692

The values in Table 6-16 are believed to be conservatively high because the calculations are based on thick plates, whereas thin plates occupy the central region of the reactor core. It has been estimated that, with the alloy used, growth problems may occur at temperatures of 900°F or above. Assuming this to be correct, the temperatures in Table 6-16 indicate a high probability that swelling problems may be encountered at 100-megawatt operation. Experiments on fuel plate growth showed a 17 percent increase in the volume of EBWR fuel plate (6-inch-long sections with meat exposed at the ends) when heated for 45 hours at 1112°F.

The possibility of maintaining low temperatures in the central zone of the fuel element by descaling the elements before 100-megawatt operation was investigated. Chemical removal of deposits on coupons with oxalic acid, nitric acid, or hydrochloric acid failed, probably because of the high content of Al_2O_3. More drastic chemical treatments were avoided because of possible adverse effects on the Zircaloy-2 cladding.

The tendency of the scale to spall off the fuel elements in areas of high heat flux suggested that heating in an oven might be an effective method of descaling. An experiment, in which scale-covered specimens from fuel element ET-51 were heated at 850°F for eight hours in an inert atmosphere, resulted in flaking off of the scale so that it could be removed by light brushing (see Figure 6-12). An attempt was then made to descale a full-size fuel subassembly by this method.

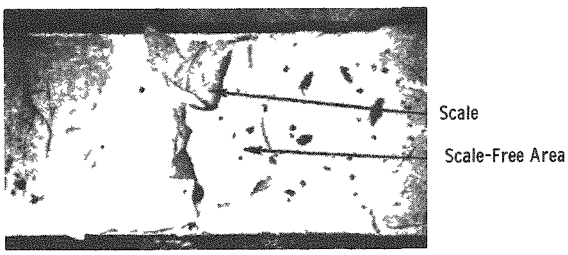

Figure 6-12

Fuel element ET-51 section showing scale deposit flaking off after heating for 8 hours at 850°F in argon
Neg. 112-600

The heat treatment not only failed to remove the scale, which on this element was very thin due to low burnup, but damaged the assembly to such an extent, by buckling the Zircaloy-2 side plates, that reinsertion into the core was impossible. The results of the unsuccessful descaling operation are shown in Figure 6-13.

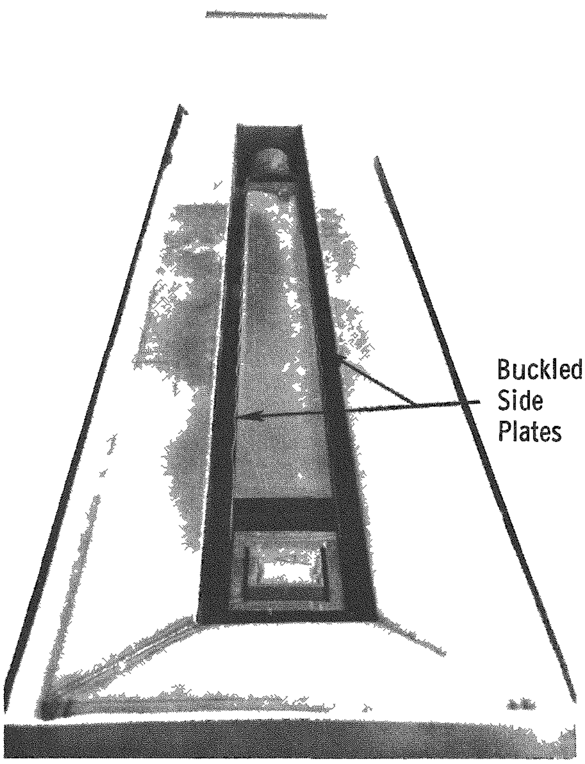

Figure 6-13

Unsuccessful attempt to descale full-size element

The results of additional heating tests revealed the growth to be expected at elevated temperatures. One plate on which the cladding had failed by oxidation grew about 2.3 percent in length when heated to 1292°F; this represents a calculated increase in volume of about 7 percent. Another plate on which the cladding had not failed grew about 0.3 percent in length. Assuming isotropic expansion (because the plate had been quenched), this increase in length corresponds to an increase in volume of about 0.8 percent.

The low volume change observed in the heating of a plate with cladding intact, as compared with the growth of plates on which cladding had failed, suggests that intact cladding on EBWR fuel elements offers considerable restraint to growth tendencies of the fuel alloy. The growth experiments mentioned above give added support to this view. The restraining effect of cladding on growth tendencies of the fuel, it is believed, will tend to minimize or reduce deformation in EBWR fuel elements that may otherwise occur if the buildup of corrosion deposits produce increased central fuel temperatures.

Another factor that may tend to alleviate the possible damage from corrosion product and buildup is the apparent tendency of the deposits to spall off in regions of high heat flux. This may be a self-limiting characteristic.

A surveillance program will be conducted to find any excessive scale deposition or swelling, as indicated by decrease in channel clearances, before dangerous conditions are reached.

6.3. MATERIALS SURVEILLANCE AND POSTOPERATIVE EVALUATION*

6.3.1 Austenitic Stainless Steel

In the preparations for the conversion of EBWR to the 100-megawatt system, the removal of the complete core, control rods, shock shield, and 6-inch steam ring was required. During the conversion period, a visual inspection of the upper half of the pressure-vessel cladding was conducted. The over-all appearance of the vessel-shell cladding is shown in Figure 6-14. The peripheral fernlike structure, located at approximately the normal water line, appeared to be bare metal. The rest of the vessel-shell cladding was coated with the normal, tightly adhering, brownish-gray oxide. Although visually different, mild scraping did not reveal any thickness variations (1) between the bare and oxide-coated areas and (2) across the boundaries of the areas in both the cladding sheets and welds joining the cladding sheets.

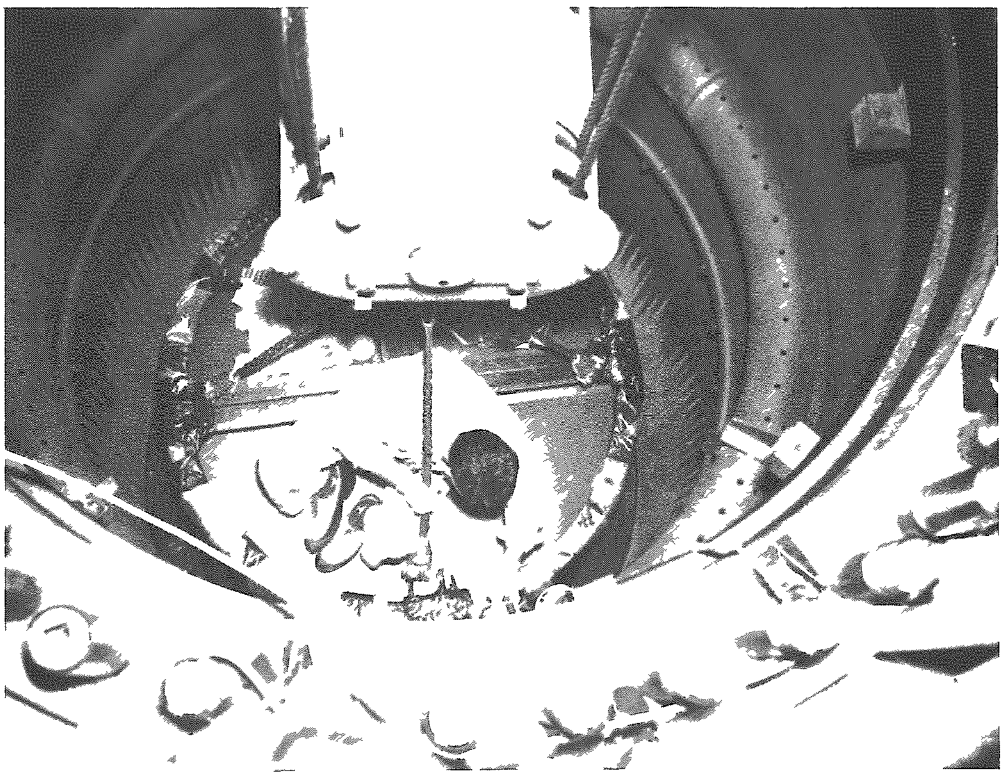

Figure 6-14

View of pressure vessel cladding
Neg. 111-8237

When a number of support blocks of the segmental shield ring was removed from the ring forging to permit drilling for additional instrument

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*N. Balai

nozzles, an opportunity was provided for comparing the unoxidized original weld-metal cladding with the surface exposed in service. General corrosion, as revealed by the contours of the machining marks in the protected areas, was negligible. A few minor pits were found in the ground fillet welds that attach the shield-ring support blocks to the cladding. From their shape, these pits apparently were caused by minor slag inclusions that escaped detection by the dye-check process during shop fabrication.

Inspection during this period revealed that the original resistance-welded 304 stainless steel cladding sheets and the submerged-arc type-308L cladding of the ring forging were intact and free of damage.

6.3.2 Boron-Stainless Steel

The four original corner control rods were fabricated from type-304 stainless steel containing 2 weight percent natural boron. After 9200-megawatt-days exposure, all the original rods were cut into three sections within the reactor vessel and transferred to the EBWR fuel storage pit. At the time of their removal, the 2 weight percent boron-stainless steel control rods showed no visible signs of gross cracking, bowing, or sticking in the top shroud and control rod channels. Postirradiation examination of the 2 weight percent boron-stainless steel control rods, the hafnium rods, and the Zircaloy-2 followers was in progress at the time of writing of this report.

The testing and evaluation of borated stainless steel has included irradiation of samples in MTR as well as operation in EBWR. Preliminary data from a cold irradiation in the MTR at 20×10^{20} nvt (thermal) of miniature specimens sealed in NaK filled containers show the following:

1. The helium gas generated from the (n,α) reaction is contained within the lattice of the alloy as evidenced by the lack of swelling of the capsules and of the samples.

2. The initially notch-sensitive material (impact energy of 1 foot-pound) is further embrittled (impact energy of 0.1 foot-pound).

3. The nil-ductility transition temperature of the irradiated material is as high as that of the unirradiated alloy, i.e., about + 450°F.

4. Irradiated material is stable dimensionally at reactor operating temperatures. No diametral changes were found in samples heated for one hour at temperatures below 1300°F.

Two 1000-pound ingots of 2 percent boron-stainless steel reserve stock from the production of the first EBWR control rods failed to convert to $\frac{1}{8}$ inch-thick sheet at a later date. Partially converted 1-inch and 2-inch-thick slabs of the material were found to be hard (values of 32 to 35 on the

Rockwell C scale), magnetic, and resistant to softening by solution annealing at temperatures as high as 2100°F. The chemical composition statistics from five previous heats of soft material indicated that the two unworkable heats were deficient in both manganese and silicon. Additional confirmation of the ANL statistics was seen from the analyses of 18 heats of material produced for the General Electric Company.[23] From the ANL and G-E data, it was apparent that the 2 weight percent modified 304 stainless steel required a minimum content of 1.5 percent Mn and 0.5 percent Si.

The sensitivity of the alloy to chemical composition was also demonstrated by remelting a portion of the unworkable heat, KA840, and adding manganese and silicon. Ferromanganese and ferrosilicon were added to the heat to raise Mn and Si to 1.40 and 0.68 percent respectively. The material was converted to $\frac{1}{8}$-inch sheet by hot rolling; the resulting alloy was nonmagnetic and soft (a value of 95 on the Rockwell B scale).

Tests of the 2 weight percent boron-modified 304 stainless steel purchased for the fabrication of the new 60-inch-stroke control rods revealed that cross-rolling has a marked effect on bending properties of the material. The boron-stainless steel control-rod material of the 20-megawatt-capacity system was sheared from cross-rolled material which permitted fabrication by cold bending to a $\frac{1}{2}$-inch corner radius. The material for the 5-foot-long boron-stainless steel rods was straight-away rolled and required hot bending at 1600°F to the required $\frac{1}{2}$-inch corner radius.

The 1-inch-thick, 1 weight percent natural boron-stainless steel thermal shield of EBWR is satisfactory as far as can be ascertained. Visual inspection showed the alloy to be bright and free of the dark oxide coating that developed on the boron-free alloy in adjacent areas of the reactor.

Similar to the 2 weight percent alloy, NaK-filled samples irradiated to 20×10^{20} nvt (thermal) in the MTR showed no evidence of helium-gas release from the lattice of the alloy. As in the case of the 2 weight percent boron-stainless steel, the impact resistance of the alloy was severely impaired by irradiation. The irradiated 1 percent boron alloy is as stable dimensionally as the 2 percent boron composition.

6.3.3 Radiation Damage to SA-212B Vessel Steel

At the time of construction of the EBWR pressure vessel, only a limited amount of neutron irradiation-effects data on constructional carbon steels was available. Since the accumulated dosage during the lifetime of the vessel for all of the desired operating modes would be substantially greater than for the known dose-effects data, it was deemed prudent to execute a radiation damage surveillance of the EBWR pressure-vessel steel (SA-212B). The dose-effects region of interest was above 1×10^{20} nvt (>1 Mev). Furthermore, these data were desired well in advance of dosages that would be accumulated by the pressure vessel during normal operation. Therefore, irradiation of steel specimens in the MTR was deemed necessary.

For this irradiation, two of the four cusp-shaped corner remnants of the $3\text{-}\frac{15}{16}$-inch-thick, square lower-head plate were welded to each other by the submerged-arc process by the Babcock and Wilcox Co., builders of the vessel. The welded test plate was then shop (B&W) furnace stress relieved for 4 hours at 1125-1150°F and shipped to ANL where the plate was sawed into: (1) subminiature impact and tensile specimens, (2) Charpy V-notch impact bars, and (3) standard 0.505-inch-diameter by 2-inch-gage length tensile bars.

The subminiature impact and tensile specimens shown in Figure 6-15 were divided into four groups of two capsules per group for irradiation in the MTR at four levels of exposure. Type-304 stainless steel capsules, containing approximately 18 samples each, were then evacuated and filled with NaK to transfer heat from the steel back into the MTR process water. By this method of encapsulation, it was estimated that the maximum temperature attained during the irradiation would not exceed 250°F.

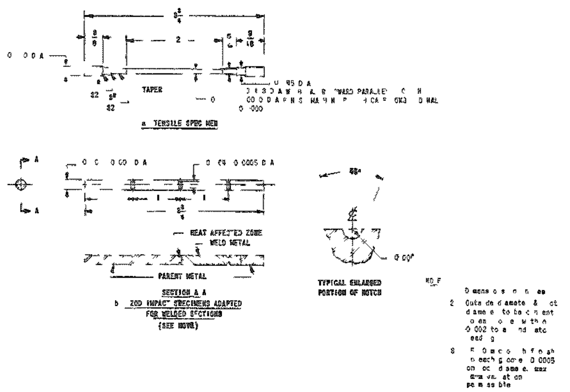

Figure 6-15

Subminiature impact and tensile specimens
RE-6-34901-B

The base plate material is identified from the mill test report data for the EBWR lower head plate as Lukens Steel Co. heat (melt) number 22789. The chemical and physical properties are given in Table 6-17. The radiation history of the three groups examined is summarized in Table 6-18.

Table 6-17. Chemical and Physical Properties of EBWR Lower Head Plate

Grade	SA-212B, silicon killed, aluminum treated, firebox quality
Grain size	7 (McQuaid-Ehn)
Chemical analysis	(From mill test report)
Carbon	0.26%
Manganese	0.75%
Phosphorus	0.018%
Sulfur	0.029%
Mechanical properties	
Ultimate tensile strength	76,000 to 77,000 psi
Yield strength (0.2%)	45,500 psi
Elongation	35%

Table 6-18. Summary of Capsule Irradiation Dosages Irradiated in MTR at 40-Mw Power Level

Group No.	Capsule No.	Position	Exposure, mwd	Neutron flux, n/cm² sec Thermal	Neutron flux, n/cm² sec Fast (>1 Mev)	Integrated dose, nvt Thermal	Integrated dose, nvt Fast (>1 Mev)
I		L-47(9 in.)	590	2.0×10^{14}	0.7×10^{13}		
	1	L-47(11 in.)	1148	2.7×10^{14}	1.2×10^{13}		
		L-47(12 in.)	1217	3.3×10^{14}	1.4×10^{13}		
		L-47(17 in.)	1246	5.3×10^{14}	2.1×10^{13}	32.2×10^{20}	1.3×10^{20}
	2	L-47(33 in.)	4201	3.0×10^{14}	1.0×10^{13}	27.2×10^{20}	0.9×10^{20}
II		L-49SW(11 in.)	2955	0.9×10^{14}	1.1×10^{13}		
	1	L-49SW(16 in.)	686	1.4×10^{14}	2.5×10^{13}		
		L-49SW(25 in.)	560	2.0×10^{14}	2.6×10^{13}		
		L-49SW(29 in.)	581	1.6×10^{14}	2.5×10^{13}		
		L-49SW(17 in.)	1295	1.5×10^{14}	2.7×10^{13}		
		L-49SW(19 in.)	2512	1.7×10^{14}	2.9×10^{13}	25.8×10^{20}	4.0×10^{20}
		L-49SW(5 in.)	2955	0.15×10^{14}	0.13×10^{13}		
	2	L-49SW(10 in.)	686	0.75×10^{14}	1.1×10^{13}		
		L-49SW(19 in.)	560	1.7×10^{14}	2.9×10^{13}		
		L-49SW(23 in.)	581	2.1×10^{14}	2.9×10^{13}		
		L-49SW(28 in.)	1295	1.7×10^{14}	2.5×10^{13}		
		L-49SW(30 in.)	4708	1.4×10^{14}	2.4×10^{14}	25.4×10^{20}	4.1×10^{20}
III		L-47(22 in.)	2386	7.0×10^{14}	7.0×10^{13}		
		L-49SW(12 in.)	581	1.0×10^{14}	2.0×10^{13}		
	1	L-49SW($5\frac{1}{2}$in.)	669	0.15×10^{14}	0.25×10^{13}		
		L-49SW(6 in.)	626	0.2×10^{14}	0.35×10^{13}		
		L-49SW($7\frac{1}{2}$in.)	1288	0.38×10^{14}	0.72×10^{13}		
		L-49SW($8\frac{1}{2}$in.)	1204	0.50×10^{14}	0.95×10^{13}		
		L-49SW($14\frac{1}{2}$in.)	2641	1.3×10^{14}	2.7×10^{13}		
		L-49SW($23\frac{1}{2}$in.)	2319	2.1×10^{14}	2.5×10^{13}	57.7×10^{20}	7.2×10^{20}
		L-47(28 in.)	2386	6.0×10^{14}	6.0×10^{13}		
	2	L-49SW($17\frac{1}{2}$in.)	581	1.6×10^{14}	3.3×10^{13}		
		L-49SW($11\frac{1}{2}$in.)	1295	0.95×10^{14}	1.8×10^{13}		
		L-49SW(13 in.)	1288	1.1×10^{14}	2.2×10^{13}		
		L-49SW(14 in.)	1204	1.2×10^{14}	2.5×10^{13}		
		L-49SW($19\frac{1}{2}$in.)	3226	1.8×10^{14}	3.3×10^{13}	54×10^{20}	7.6×10^{20}

NOTES:

1. Neutron fluxes reported by MTR are:
   a. Peak thermal flux
   b. Estimated unperturbed fast flux
2. Fast flux calculated (by MTR) from disintegration rate of Co^{58} formed in Ni^{58} (n,p) Co^{58} reaction.
3. SA-212B irradiation not monitored.

Impact-transition temperature data from the irradiated and unirradiated parent plate miniature impact specimens are summarized in Figure 6-16 for the three irradiation exposures. These data show that the parent plate material was completely embrittled at the lowest exposure level. Data for the performance of the irradiated weld metal and heat affected zone, although not included on the figure, also showed a like embrittlement. Also plotted on the figure are four points that show how the irradiated material responded to a conventional 1125-1150°F stress-relief heat treatment in air, i.e., the restoration of impact properties to unirradiated levels.

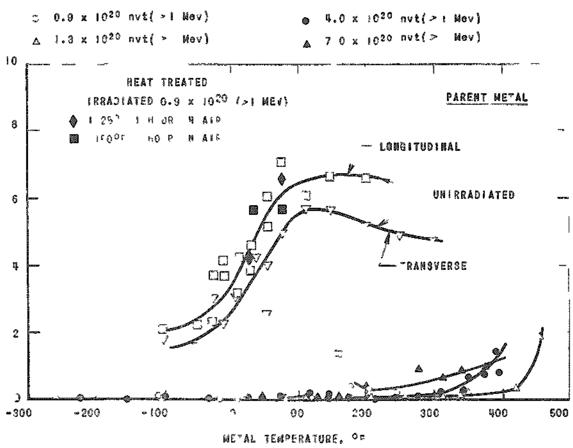

Figure 6-16

SA-212B impact resistance data summary RE-7-34902-A

The effects of lower temperature heat treatments were studied on specimens from capsule 2 of group II. Single-notch-impact data taken at room temperature showed that irradiated SA-212B started to regain impact strength at temperatures as low as 700°F. Figure 6-17 summarizes the response of SA-212B to heat treatment over the range of 700°-1000°F for periods up to 3 hours at temperature in argon. At the higher temperatures, 900° and 1000°F, the steel appears to re-embrittle with a long holding time at temperature. The data of Figure 6-17 shows almost complete recovery of the steel to the unirradiated impact values at 800° or 900°F with a holding time of one hour at temperature. The limited multinotch-impact transition curves for heat-treated irradiated SA-212B indicate that the lower temperature, 800°F, is too low to restore the steel to the lower (transverse) impact strength of the unirradiated steel, as shown in Figure 6-18. Figure 6-19 data, from many bars, confirms that the 900°F heat treatment for one hour is effective in restoring the impact strength of the irradiated material to that of the unirradiated parent plate.

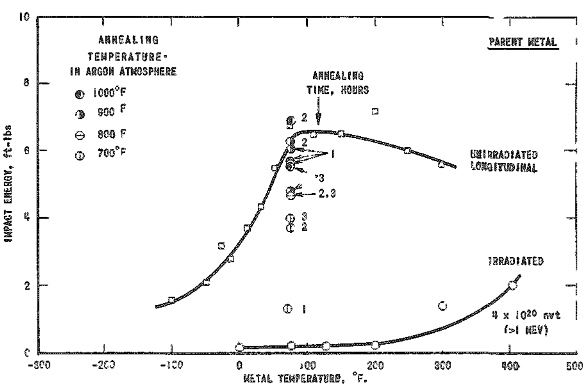

Figure 6-17

Effect of heat treatment on irradiated and unirradiated SA-212B specimens RE-7-34903-A

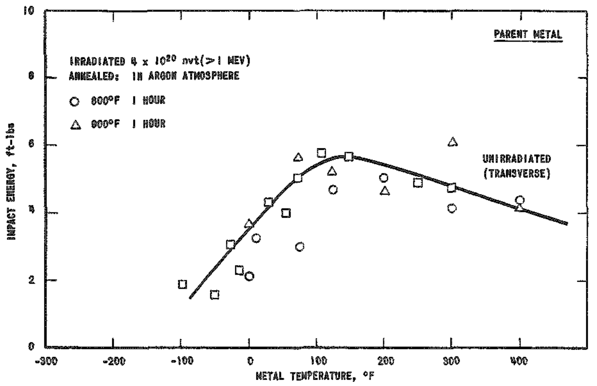

Figure 6-18

Effect of one hour holding time on restoration of impact resistance properties of SA-212B metal
RE-7-34904-A

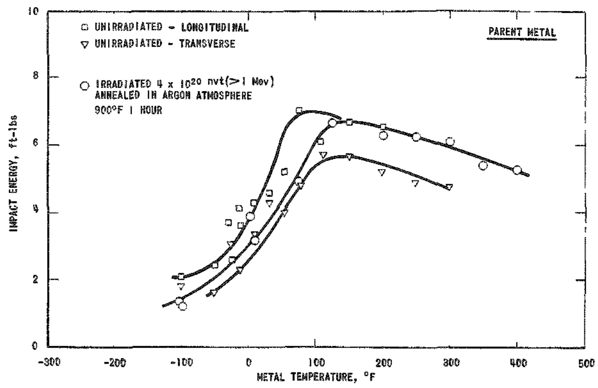

Figure 6-19

Restoration of impact transition temperature of irradiated SA-212B base metal
RE-7-34905-A

Figures 6-20 and 6-21 show that the irradiated, heat-affected zone and weld metal also regain impact resistance, virtually equal to that of unirradiated parent plate, at the end of the 1-hour 900°F heat treating cycle. The impact transition data points for the heat-treated, irradiated weld metal and heat-affected zone show less scatter than their unirradiated counterparts.

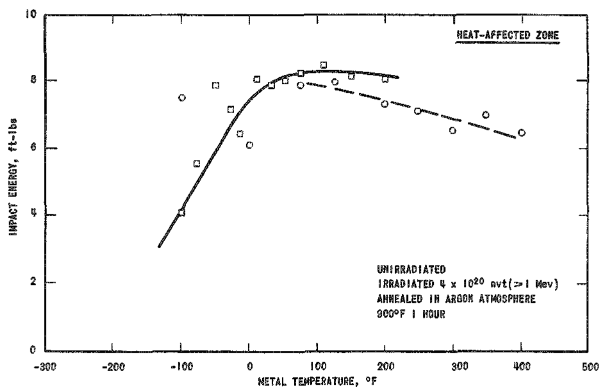

Figure 6-20

Restoration of impact transition temperature of irradiated SA-212B heat-affected-zone material
RE-7-34906-A

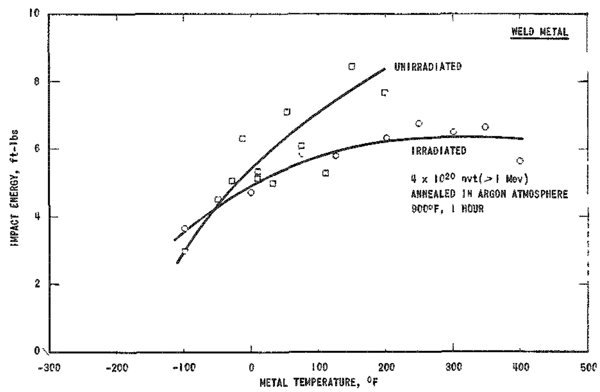

Figure 6-21

Restoration of impact transition temperature of irradiated SA-212B weld material to unirradiated SA-212B base metal properties
RE-7-34907-A

Most of the miniature as-irradiated tensile specimens were lost as a result of a faulty behavior of the remotely controlled tensile test machine. All specimens, however, showed at least a 10 percent elongation and an undetermined reduction of area, estimated from photographs at 30 percent. All samples failed by necking.

The hardness of the base and weld metals was determined on the cylindrical portion of the specimens. The Rockwell 15-N scale was selected because it conveniently measures this property in the soft as well as the hardened condition of the steel - the Rockwell C scale readings are meaningless below the C-20 value.

Figure 6-22 shows that the pressure-vessel steel reached a saturation hardness below the lowest dosage, i.e., 1 x 10^{20} nvt (>1 Mev) and then hardened to a slightly higher value above the 4 x 10^{20} nvt (>1 Mev) dosage. The initially harder weld metal resisted hardening at the 1 x 10^{20} nvt (>1 Mev) exposure but later hardened to about the same level as did the base metal at the higher dosages.

The annealing studies summarized in Figure 6-23 revealed that post irradiation heat treatments were effective in reducing the peak radiation hardness of the base metal and weld metal to that of unirradiated base plate metal at temperatures as low as 700°F. At higher temperatures, 900° and 1000°F, the hardness of the irradiated steel was reduced to that of the base plate material after one hour at anneal temperature.

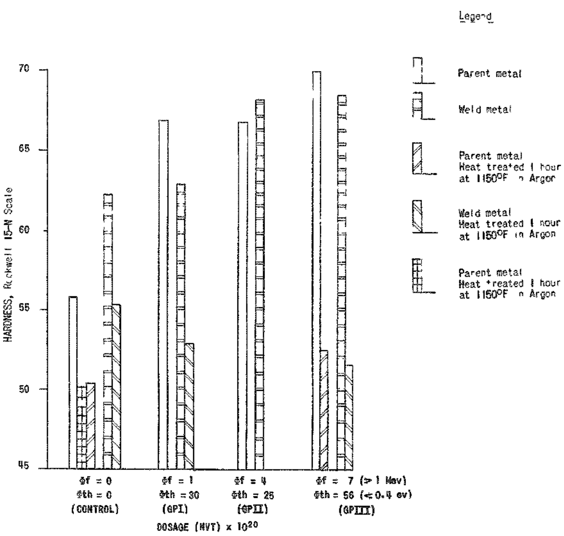

Figure 6-22

Pre- and postirradiation hardness history of SA-212B parent metal and weld metal RE-7-34908-B

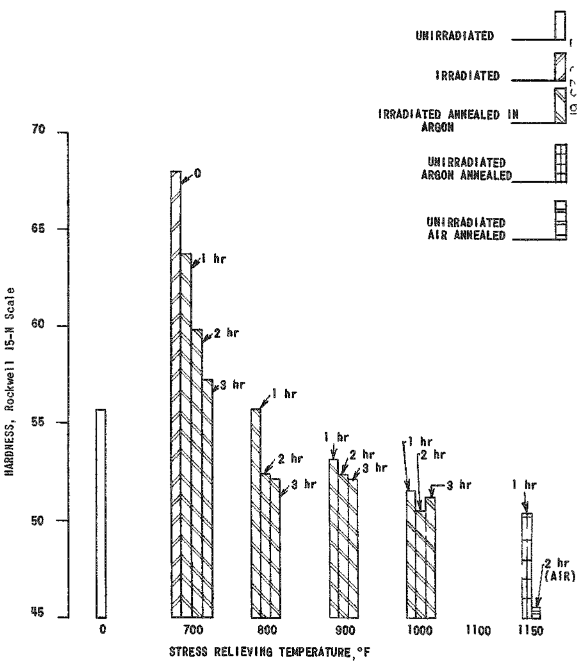

Figure 6-23

Pre- and postirradiation hardness history of group II SA-212B parent metal RE-7-34909-A

The annealing studies also revealed that the unirradiated control samples had been slightly work-hardened during their preparation (from 51 to 56 on the Rockwell 15-N scale as shown in Figure 6-23).

The results from the study of hardness changes closely parallel the results of the changes in the transition temperatures. The changes in hardness, however, do not reveal the re-embrittling effects of protracted anneals at the higher temperatures, 900° and 1000°F, which are shown in Figure 6-17.

A fourth group (of two capsules) was irradiated to about 1.5×10^{21} nvt (>1 Mev) and examination of results shows no significant differences from the data shown on Figure 6-16. It was also shown that this group of specimens responded to heat treatment similar to group 2 in Figure 6-19.

As a result of this examination, it is apparent that the original objective of the irradiation - a surveillance of damage to the EBWR pressure vessel - was not accomplished. Furthermore the results of this irradiation are diametrically opposed to those reported in the early literature. Results reported concurrently by other investigators are in agreement with the results reported herein.

New irradiation facilities are installed within EBWR for the surveillance program. Capsules of "bare" specimens (exposed to the reactor neutron spectrum) and "shielded" specimens (surrounded by $\frac{1}{2}$-inch-thick 2 percent boron-stainless steel to harden the neutron spectrum) will be withdrawn at regular intervals of exposure, starting at the 5×10^{18} nvt (thermal) dosage level.

6.4 FUEL DESCRIPTION AND CALCULATIONS*

The EBWR spike fuel elements were designed to approximate the over-all core in terms of heat transfer area and water-to-metal ratio. Design requirements were: (1) an active fuel length of 121.9 centimeters (48 inches) and (2) a power level of 100 megawatts (thermal) for the entire core.

The selection of a fuel rod geometry was based on ease of fabrication, proper structural capabilities for the fuel and cladding, and ready adaptability to a variety of fuel and cladding material combinations. Some of the combinations of core-clad materials investigated were:

1. Zircaloy-2 clad + U_3O_8 in X-8001 matrix (98.5 Al - 1.0 Ni - 0.5 Fe).
2. Zircaloy-2 clad + ZrO_2 + UO_2 + CaO pellets.
3. X-8001 clad + U_3O_8 in X-8001 matrix.

---

*V. M. Kolba

The combination of core-clad material shown in number 3 above was eliminated as EBWR fuel because of:

1. Corrosion problems at high heat flux.
2. The ratio of aluminum content to the nonaluminum surface area is disproportionate, and this would, theoretically, increase the corrosion rate of the aluminum.
3. Deposition rate of scale on Zircaloy-2 clad plates might be increased.

The combination in number 1 was investigated as the first reference fuel; however, because of the problems outlined in this section, the combination in number 2 was ultimately selected as the fuel composition.

In the combination in number 1, Zircaloy-2 clad was investigated because it is compatible with the over-all system cladding and was purported to be readily available in tubing acceptable for reactor operation. The meat X-8001 + U_3O_8 was selected for investigation because this mixture had been used in Argonaut fuel plates, was extrudable, and corrosion resistant in the event of clad failure.

The combination of Zircaloy-2 clad and X-8001 + U_3O_8 fuel offers the advantage that any small gap existing between the clad and core at room temperature will close at elevated temperature and, thereby, allow intimate contact and a lower thermal resistance during operation.

The total diametral expansion of the fuel and clad materials for the temperature rise postulated are: (1) 0.0058 centimeter (~2.3 mils) for aluminum and (2) 0.0015 centimeter (0.6 mil) for Zircaloy-2. The calculated difference in expansion is 0.0043 centimeter (1.7 mils). However, testing at elevated temperature has indicated the need for a 0.0051-centimeter (2 mils) radial (0.010 cm or 3.9 mils diametral) gap between the meat and clad to provide for differential expansion and also for clad variations which may cause grabbing of the U_3O_8 + Al meat by the Zircaloy-2 clad.

The fuel mixture UO_2-ZrO_2-CaO was investigated as the fuel media to be used in combination with Zircaloy-2 clad because this core-clad combination is corrosion resistant and the thermal expansion of the UO_2-ZrO_2-CaO and the Zircaloy-2 clad systems are compatible. The second reason is an advantage over the U_3O_8 + Al and Zircaloy-2 clad system, which tends to grab during thermal cycling.

Assuming that the cell size utilized is 10.16 centimeters (4 inches) square and that the water-to-metal ratio (w/m) of the original EBWR fuel elements (fuel plates and side plates) is 1.449 for the thick plates and 2.123 for the thin plates will yield an average of 1.786 for a one-half core loading of each type. On this basis the new fuel elements are designed to be near the average value of the w/m ratio. The total area of a unit fuel cell may be expressed by:

$$A_T = A_f + A_m + A_w \quad . \tag{1}$$

Since the fuel element pitch in EBWR is 10.16 centimeters from center-to-center, A_T = 103.2 square centimeters (16 square inches) per cell. The water-to-metal ratio may be expressed in terms of cross sectional area as:

$$w/m = \frac{A_w}{A_m + A_f} = \frac{A_T - (A_m + A_f)}{A_m + A_f} \quad . \tag{2}$$

Using the average of EBWR Core I fuel:

$$\frac{A_T - (A_m + A_f)}{A_m + A_f} = 1.786 \quad . \tag{3}$$

By substitution the nonwater cross sectional area is:

$$A_m + A_f = 37.04 \text{ cm}^2 \quad . \tag{4}$$

The rod diameter required for adequate surface area is next evaluated. For this calculation A_r applies to only fuel cell components (rods or plates) and includes only the fuel and clad but not the fuel element structural material since, at this point in the calculation, no design features are known. For the sake of simplicity the following is used:

$$A_T = A_r + A_w \quad . \tag{5}$$

Total rod cross sectional area in a fuel cell is expressed by:

$$A_r = \frac{N\pi d^2}{4} \quad . \tag{6}$$

The average heat flux may be expressed by:

$$Q_A = \frac{P_r}{C} \text{mw} \tag{7}$$

$$Q_A = 0.68 \text{ mw/cell} \quad . \tag{7a}$$

The surface area required per cell to stay within an assumed average surface heat flux of 37.85 watts per cm^2 is given by:

$$S_A = \frac{Q_A}{F_A} \quad . \tag{8}$$

The surface area in the cell may also be written as:

$$S_A = N\pi Ld \quad . \tag{9}$$

Substituting (9) in (8) yields:

$$\frac{Q_A}{F_A} = N\pi Ld \quad . \tag{10}$$

Solving (10) for N the number of rods per unit cell required is:

$$N = \frac{Q_A}{F_A \pi Ld} \quad . \tag{11}$$

Rearranging and equating (6) and (11) yields:

$$\frac{4A_r}{\pi d^2} = N = \frac{Q_A}{F_A \pi Ld} \quad . \tag{12}$$

Solving (12) for d yields:

$$d = \frac{4L\, F_A A_r}{Q_A} \quad . \tag{13}$$

Substituting F_A = 37.85 watts per cm^2, A_r = 37.04 cm^2 for fuel rods only, and L = 121.9 cm in equation (13) a value of 0.94 cm is obtained for the rod diameter.

A commercially available die of 0.953 centimeter ($\frac{3}{8}$inch) diameter was determined to be sufficiently close to the above value. Using this number in equation (11), a value of 48.6 rods is obtained for N. The closest integral number of rods for a square geometry is a 7 by 7 arrangement of 49 rods. Using 49 rods and a diameter of 0.953 centimeter, the average surface heat flux becomes 38.06 watts per cm^2.

Mechanical design requirements to be considered in the design of the fuel rod assembly include: material strength, material compatability, and fabrication procedures and processes. To lower material costs, the end boxes and grids of the fuel rod assembly are made of 304 stainless steel while the fuel rods are all Zircaloy-2. Two problems encountered in fixing the ends of the fuel rods in the boxes are: (1) an adequate method of joining stainless steel and Zircaloy-2 and (2) the difference in thermal expansion of individual fuel rods. To circumvent these problems, the fuel rod ends are slotted to ride over a 0.159-centimeter-thick ($\frac{1}{16}$-inch) stainless steel grid spacer. The thickness and strength of the grid spacer far exceed the support loads postulated for the member. The weld area also exceeds the thickness and strength required to support these members.

Two side plates hold the upper and lower end-fitting boxes together and, thus, contain the fuel rods between the end-fitting boxes and grids. Since these plates are in the active core, Zircaloy-2 is used. Calculations show the plates to be adequate to sustain the loads encountered. A force of 2,932 kilograms (6463.8 pounds) is required on each side plate before yielding occurs.

The rivets used to hold the side plates to the end-fitting boxes are 0.635-centimeter-diameter ($\frac{1}{4}$-inch) 304 stainless steel. The load that each rivet is capable of sustaining in shear is calculated to be 268 kilograms or 590 pounds (based on Section VIII of the ASME Code), which is more than adequate to sustain the load of the fuel.

Tear out of the rivet is also of concern because of the thinness of the materials. The thickness of both the Zircaloy-2 sheet and the stainless steel end box is 0.159 centimeter ($\frac{1}{16}$ inch). Calculations indicate that a pull of 114 kilograms (251 pounds) is required to pull a rivet through the sheet stainless steel or Zircaloy-2. A similar condition exists in the fuel element frame, where Zircaloy-2 is also riveted to 304 stainless steel with sufficient strength of the rivets to assure adequate performance.

The top lifting plates of the fuel element frame are lap spot welded to the angles. Pull tests conducted on sample spot welds required a 591-kilogram (1303-pound) pull on each weld to cause failure.

Fuel rods have several additional problems associated with materials and operating conditions. One of these encountered in combination 1 above is the determination of the diametral gap requirements between the fuel slug and the Zircaloy-2 clad.

It can be assumed that the optimum condition would prevail during operation of the aluminum slug just touching the clad; in which case, the temperature drop between the slug and the clad would not be appreciable. In this case, if all dimensions are nominal, a 0.0043-centimeter (1.7-mil) diametral gap is required to just cause the clad and fuel slug to touch at operating conditions. However, tests showed that when rods having 0.0051-centimeter (2-mil) diametral gaps were thermal cycled, grabbing of the fuel slug by the clad occurred. Test results showed that because of the tolerances involved a diametral gap of 0.010 centimeter (3.9 mils) or a radial gap of 0.0051 centimeter (2 mils) was required. Since exact center temperatures cannot be calculated until gaps are known, the above calculations must now be reiterated when center temperatures are indicated. The gap requirement necessitates a fuel slug diameter of 0.815 centimeter or 0.321 inch (cold) in order that the aluminum slug just touch portions of the Zircaloy-2 clad at operating temperatures.

Centerline temperatures of the fuel are calculated by equation:

$$T = T_s + \frac{\dfrac{1.9(12Q)^{1/4}}{dL}}{e^{P/900}} + \frac{Q}{L}\left(\frac{X_{sc}}{K_{sc}d} + \frac{X_c}{K_c d} + \frac{X_g}{K_g d} + \frac{1}{4K_p}\right) \quad . \qquad (14)$$

Based on the above equation, the average centerline temperature for a new fuel rod with a nominal 0.0051-centimeter (2-mil) helium-filled gap is (1) 410°C for the U_3O_8 + X-8001 slug and (2) 928°C for the UO_2-CaO pellets. End gaps were also provided in the fuel rod design to compensate for unequal thermal expansion between the aluminum fuel slug and the Zircaloy-2 clad. These gaps are also required in the case of ceramic pellets and Zircaloy-2 clad. However, in this instance a portion of the end gap is required to accommodate gaseous fission product buildup.

The gaseous fission product buildup is based on assumptions for burnup and loss to the tube from the pellets. If 5 percent burnup and 25 percent release is assumed, the pressure in the tube increases to approximately 1.9 kilograms per square centimeter. For a burnup of 50 percent, the entire gaseous fission product released is raised by a factor of about 10, and the internal pressure would reach approximately 18.8 kilograms per square centimeter (267.45 pounds per square inch). These calculations make no allowance for the initial diametral gap or the decrease in volume caused by differential thermal expansion since these effects approximately cancel each other.

A list of definitions of the terms used in the calculations in this section is given below.

A_f - Total cross sectional area of fuel rods in cell, cm^2

A_m - Total cross sectional area of metal as clad, supports, etc. in cell, cm^2

A_r - Total cross sectional area of fuel units in cell, cm^2

A_T - Total cell cross sectional area, cm^2

A_w - Total cross sectional area of water in cell, cm^2

C - Number of cells in core, 148 positions minus 1 source = 147

d - Log mean diameter, cm

F_A - Average heat flux, watts/cm^2

K - Thermal conductivity, cal/cm-sec-°C
  p - pellet
  sc - scale
  c - cladding
  g - gap (pellet-to-clad)

L - Length, cm

m - Metal (all nonwater)

N - Number of rods per unit cell

P - Pressure, kg/cm

P_r - Reactor power

Q - Heat flux per rod, watts

Q_A - Average heat flux per unit cell, watts

S_A - Surface area, cm^2

T - Temperature

T_s - Saturation temperature at reactor pressure

w - Water

w/m - Water-to-metal ratio

X - Thickness, cm
  sc - scale
  c cladding
  g gap (pellet-to-clad)

6.5. REACTOR-VESSEL BLOWDOWN SYSTEM*

In past operation, corrosion products generated on the EBWR reactor vessel and fuel element surfaces tended to settle into the forced-circulation nozzles and control-rod-drive thimbles attached to the bottom of the shell. This caused a high degree of radiation at the bottom of the reactor vessel.

The reactor-vessel blowdown system provides a means of removing corrosion products that settle and accumulate at these points. To permit periodic blowdown during operation, the design includes an accumulator and valve arrangement to limit the amount of water withdrawn during any individual blowdown operation. A flow diagram of the system is shown in Figure 6-24.

Each thimble or nozzle is drained individually to the accumulator tank. The accumulator restricts the amount of drainage at any one time to 5 gallons of liquid. By an interlock valve arrangement, the contents of the accumulator can be drained to a settling tank. Sludge which settles from the water in this tank can be emptied into a shielded container. The remaining water can be monitored for radioactivity and returned to the primary system through the vapor recovery system. This water may also be routed through a portable ion exchange column to remove any radioactive waste not eliminated by previous operations.

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*R. J. Gariboldi

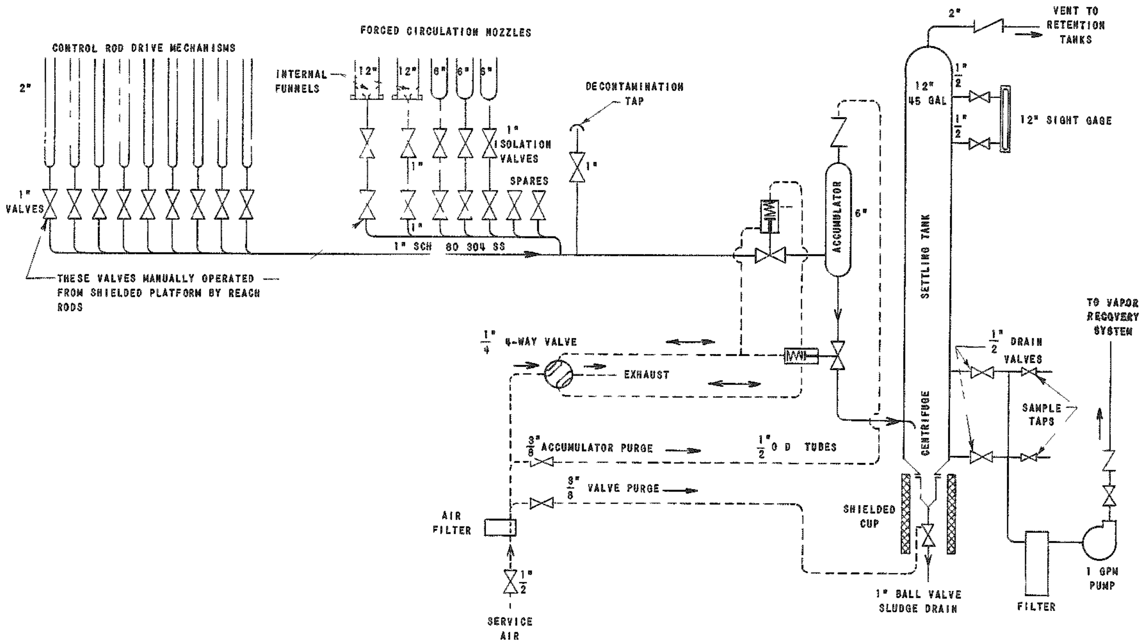

Figure 6-24

Flow diagram of reactor-vessel blowdown system
RE-8-23942-C

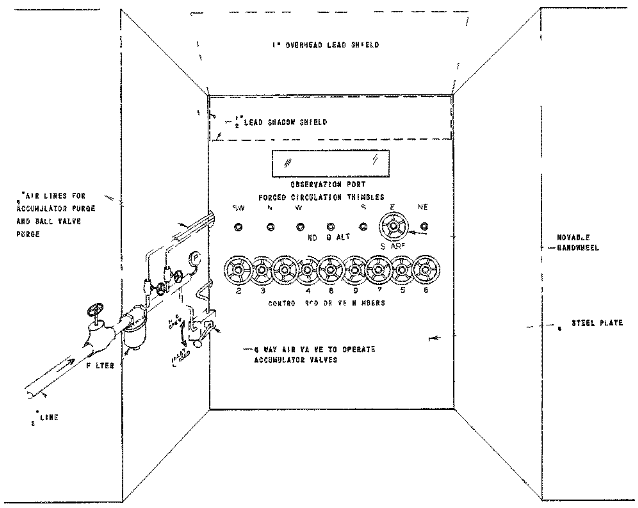

Figure 6-25

Blowdown system valve operating platform
RE-8-23938-C

The blowdown system is operated from a semiremote platform to shield the operator in the event of high-level gamma radiation in the subreactor room. The operating platform is illustrated in Figure 6-25.

The blowdown valves of the control-rod drives, located on the bottom of each of the rack thimbles, are at the lowest points in the reactor vessel. The blowdown valves of the control-rod drives are shown in Figure 6-26. Only eight valves were originally installed because the central rod is an oscillator rod which prohibits this type of installation.

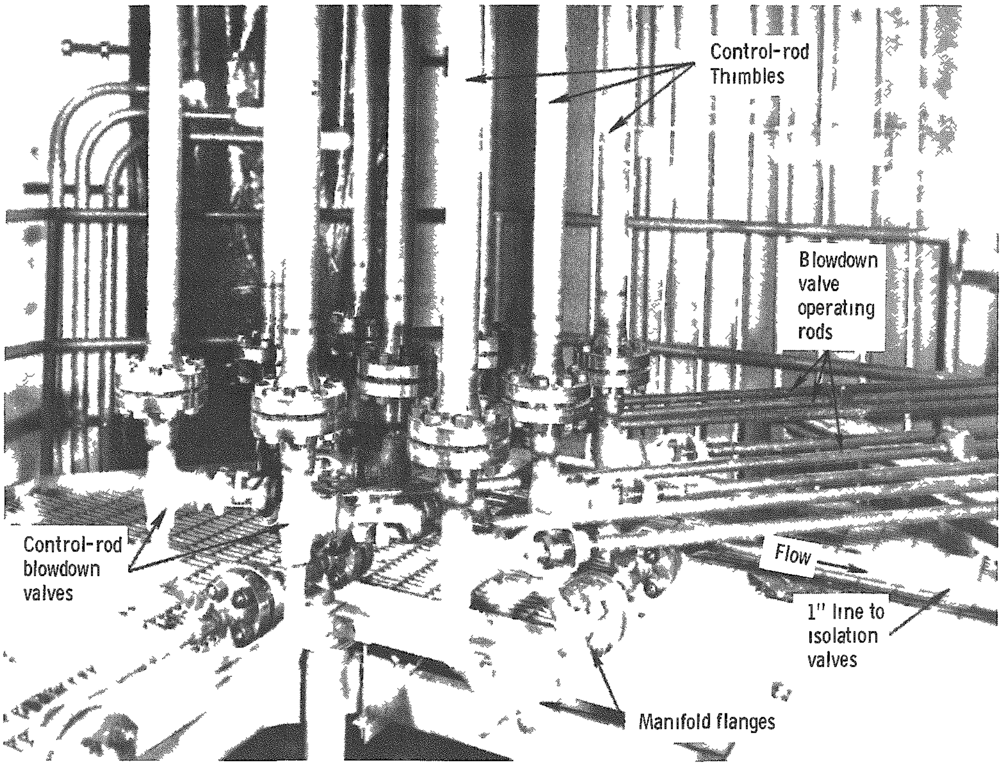

Figure 6-26

Blowdown valves on control-drive rack thimbles

The blowdown valves are furnished with Stellite-faced seats because considerable wire drawing is expected to occur during blowdown. The rod drive design permits removal of these valves to reface the seats. Because of high radiation levels expected near the valves, the valves are to be operated by valve stem extension rods from the shielded platform.

Three of the four forced-circulation inlet nozzles and the two forced-circulation outlet nozzles have blowdown valves mounted on a manifold behind the shielded wall of the operating platform (see Figure 6-27). These valves are also operated by extension rods. The fourth inlet nozzle is not connected to this system because it is in continuous use for the reactor purification system.

Isolation valves are used between the forced-circulation-system nozzles and the blowdown valves to permit removal and repair to the blowdown valves.

Manifolds pipe the discharge from the blowdown valves to the accumulator. The manifold under the control-rod drives is a flanged design that permits removal of any individual rod drive without the need of removing the blowdown connections to other drives.

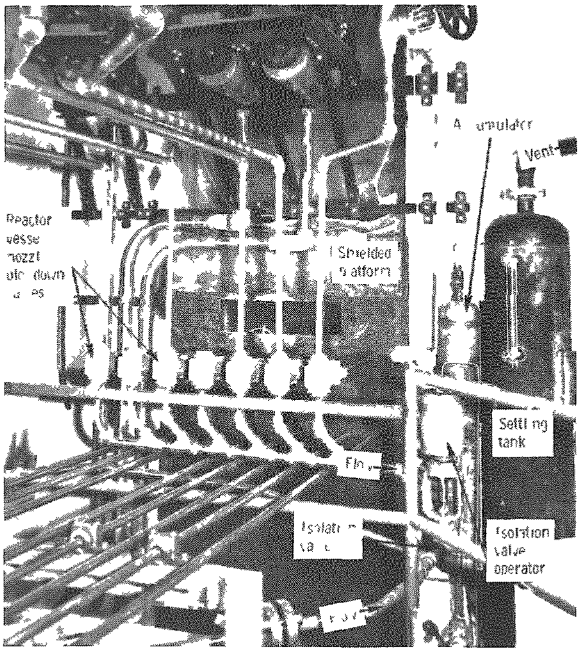

Figure 6-27
Forced-circulation-nozzle
blowdown valves
Neg.111-9283

A second manifold connects the blowdown valves of the forced-circulation nozzles to the accumulator tank. A 1-inch, 1500-pound, spring-loaded, piston-type check valve prevents back flow into the control-rod-drive manifold when flushing the forced-circulation nozzles.

An accumulator with two interconnected, pneumatically operated valves, prevents the possibility of inadvertent release of large volumes of water from the reactor vessel. This is accomplished by a method analogous to a set of canal locks.

The blowdown manifolds are piped to the inlet valve of the accumulator. The outlet valve drains the contents of the accumulator to the settling tank. Both of these valves are the normally closed, spring-loaded type. A four way air valve permits only one accumulator valve to be open at any time. The accumulator has a $5\frac{1}{2}$-gallon capacity which limits the discharge for each cycle to less than 5 gallons. An airline enters the top of the accumulator to: (1) supply 100-psi cushion air and (2) purge the accumulator. A $\frac{1}{4}$-inch, 600-pound, spring-loaded, piston-type check valve is used on the top of the accumulator tank to prevent back flow into the purge airline.

The accumulator is fabricated of 6-inch, schedule-40, type-304 stainless steel pipe.

The settling tank, fabricated of 12-inch, schedule-20, SA105 steel pipe, has a 45-gallon capacity. The bottom end of the tank reduces to a 3-inch-diameter shielded pocket to collect corrosion products. A 1-inch ball valve discharges the settled material into a shielded can for disposal. The upper end of the tank is vented to the main retention tank by a 2-inch pipe. This vent also acts as an overflow line. The operator is warned of excess water in the settling tank by means of a 12-inch sight glass at the top of the tank, which is visible from the observation port.

A Fulflo filter, Model BR x $10\frac{1}{2}$ with W13R10C insert tube, filters the water before it is pumped to the turbine condenser. The settled water is monitored before being pumped to the condenser to avoid transferring radioactivity.

A stainless steel pump moves settling tank water to the vapor recovery system. The pump has a capacity of 1-gpm at 15-psi discharge head, is fabricated of type-304 stainless steel, and is driven by a 115-volt, 60-cycle, single-phase motor.

6.6 PRIMARY SYSTEM DECONTAMINATION*

In the event of a fuel element failure it is possible for the fission products and fissile material thus liberated to carry throughout the reactor primary system. The degree to which the system would become contaminated is dependent upon many factors. Some of these are:

1. The rate of fuel failure and the amount of fuel exposed in the failure.
2. The rate of fission product dispersion into water and steam phases.
3. The time required for detection and the reactor operating power at the time of failure.
4. The time required to shut down and isolate the system in order to limit the extent of contamination.
5. The degree that fission products carry over and plate out on steam-side components.
6. Fission product lifetime.

The limited experience of BORAX IV, in which ceramic fuel was clad with aluminum, indicated that major contamination with fission products is limited to the reactor vessel proper and equipment that directly handles reactor vessel water. In the event of a fuel element failure in EBWR, the defective fuel element will be placed in a can while still below water in the reactor so that fission products will not contaminate the inside of the coffin during the transfer from the reactor vessel to the storage pit.

The first operation in decontamination will therefore be to clean up the reactor water to permit removal of failed fuel elements. The reactor-cleanup-loop ion exchange equipment will be used until the ion exchange resins are depleted. Further ion exchange will be conducted as accessibility permits.

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*R. J. Gariboldi

It is assumed that limited access to the main floor of the reactor containment building will be possible by virtue of the shielding afforded by the concrete floor. It is also assumed that primary-system equipment located outside the containment shell will be isolated in time to prevent serious contamination with long-lived fission products. The steam flowing to the primary system is not dried and may contain up to 10 percent moisture during high power runs. All components within the containment building that handle only dry steam are expected to be approachable within a reasonable time after shutdown. This is feasible since, in general, short-lived gaseous products are carried with the dry steam and long-lived products remain with the moisture.

The decontamination provisions incorporated in the EBWR system do not provide the ultimate coverage and accessibility under any and all circumstances. The system does provide fill, drain, and vent taps on all components for the introduction and circulation of decontaminant solutions. It also provides reasonable accessibility. The components most likely to be inaccessible and yet essential for operation are equipped for remote decontamination. Components least likely to be seriously contaminated or accessible by other routes have their decontamination tap valves located adjacent to the component. The general provisions afforded by this system are:

1. Cleanup of the reactor water by auxiliary ion exchange beds.
2. Removal of remaining residue from equipment surfaces by dissolving with decontaminant solutions.

Figure 6-28 indicates the location of isolation and decontamination valves.

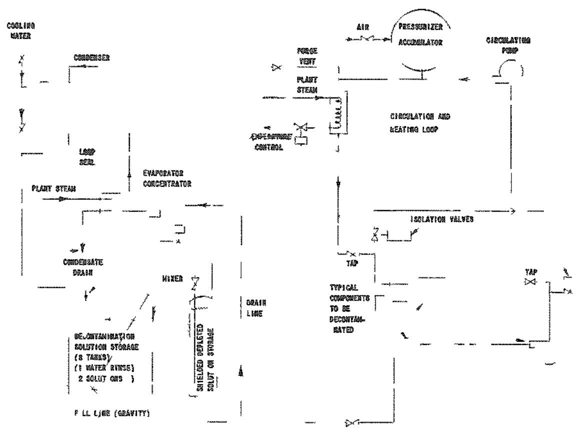

Figure 6-28

Flow chart of decontamination system
RE-8-32674-B

The equipment and piping to be decontaminated are isolated by valves or by gravity. Water is drained from the component through the drain valves or, in the case where components may be located as much as 30 feet below the decontamination equipment on the main floor, the water may be purged from the system by applying air pressure to the vent lines. The drained water is transferred to another part of the primary system.

The first decontamination solution consists of 18 percent sodium hydroxide plus 3 percent potassium permanganate. This solution is added, through a drain valve, to the component to be decontaminated while venting a high point or by metering a sufficient amount of solution to completely fill the system.

Circulating pumps on the main floor will draw the solution from the drain valves and return it to the component through the vent line or alternate line. This will assure circulation through the entire component. The solution is heated by a small heat exchanger utilizing plant steam as the heat source. The first solution is to be circulated for 30 minutes at a temperature of 210°F. The component is then drained of the solution. If radioactive, the solution will be temporarily stored in underground stainless steel tanks. A rinse of fresh water is required to completely remove the first solution.

The same procedure is used with a second solution; 10 percent ammonium citrate is circulated for 15 minutes at 180°F. This solution is then drained, and the component is filled with demineralized water, if required.

In decontaminating the system, the first area of concern is the primary ion-exchange system since it is used for the initial cleanup of water before removing the reactor vessel head. In the event that the ion-exchange beds become exhausted and are inaccessible for recharging, facilities are available for bypassing the beds and installing auxiliary beds on the main floor. These auxiliary beds are contained in shielded casks to minimize radiation. The tube side of the regenerative exchangers and aftercoolers are most likely to be highly radioactive because of sludge settling in the tubes.

Many factors affect the decontamination procedure for the reactor vessel and no attempt is made to establish a fixed procedure in this case. Since more than 6,000 gallons of solution would be required to follow the general procedure outlined above, it appears to be expedient to first use other means to apply spot decontamination that may be required for fuel handling. Immediate and complete decontamination of the reactor vessel is not likely to be required. Considerations that govern the procedure are:

1. The vessel is well shielded and need only be approached for fuel handling operations.

2. The vessel may be filled with clean water to shield the contaminated vessel walls during head removal and fuel handling operations.

3. The thickness of the head provides shielding from the contamination on the lower surface.

4. It may be possible to use spray techniques to apply decontamination solution to the vessel interior.

The steam drier is expected to collect considerable long-lived contamination. The scrubbers within the drier present 820 square feet of surface area for deposition of radioactive material carried in the moisture. Most of the moisture is extracted from the steam at this point. The drier, being also the emergency cooler, will be in operation even after the turbine and condenser have been shut down following a fuel failure incident. Operating experience indicates that this component collects contamination during normal operation.

The cleanup of the startup heater and liquid-level column will be carried out concurrently with the pressure vessel decontamination in the event that radiation levels do not permit isolation of this equipment. A rough cleaning can be accomplished when the reactor is drained. Solutions pumped into the steam drier will drain through the startup heater and then to the reactor vessel. In the case of the liquid-level column, the solutions can be injected into the vent line and drained back to the vessel. This will reduce the radiation levels sufficiently to allow isolation of the equipment for a more thorough cleaning procedure.

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Plant Engineers

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R. King	Foundations and Reboiler Building

Contractors

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T. F. Hanley Corporation	General Construction Contractor - Modifications within Containment Vessel
R. Acton Company	Contractor - Concrete and Reboiler Building
Emerson Comstock Company	Electrical Contractor

[a]Now a member of the Particle Accelerator Division
[b]Terminated

REFERENCES

1. "The Experimental Boiling Water Reactor," ANL-5607, May 1957.

2. P. A. Lottes, et al., "Lecture Notes on Heat Extraction from Boiling Water Power Reactors," ANL-6063, October 1959.

3. C. R. Breden, et al., "Soluble Poisons for Control," ANL-5244, November 1955.

4. E. A. Wimunc and J. M. Harrer, "Hazards Evaluation Report Associated with the Operation of EBWR at 100 MW," ANL-5781 Addendum, Appendix I, December 1959.

5. J. M. West, et al., "Hazard Summary Report on the Experimental Boiling Water Reactor (EBWR)," ANL-5781, November 1957, p. 115.

6. R. A. Mattson, "The Determination of Soluble-Poison Concentration in H_2O," Thesis, Michigan College of Mining and Technology, 1957.

7. D. D. DeFord and R. S. Braman, "Study of Neutron Absorptiometry and Its Application to the Determination of Boron," Anal. Chem., Volume 30, No. 11, November 1958, pp. 1765-72.

8. R. J. Armani, "The Continuous Monitoring of Boric Acid Concentration by Neutron Absorptiometry," ANL-6439, October 1961.

9. J. A. Thie, "Fluid Poison Control of Boiling Water Reactors," Nucleonics, Volume 16, No. 5, May 1958, pp. 82-3.

10. T. L. Kettles and N. Balai, "EBWR Reactor Pressure Vessel Revisions for Instrumentation and Increased Steam Output," ANL-6162 (to be published).

11. R. Avery, et al., "EBWR Core IA Physics Analysis," ANL-6305, February 1961, pp. 32-3.

12. V. M. Kolba, "EBWR Test Reports," ANL-6229, November 1960, p. 173.

13. C. F. Reinke and R. Carlander, "Examination of Irradiated EBWR Core I Fuel Elements," ANL-6091, July 1960.

14. J. A. Thie, "EBWR Physics Experiments," ANL-5711, July 1957.

15. R. W. Deutsch, "Computing 3 Group Constants for Neutron Diffusion," Nucleonics, Volume 15, No. 1, January 1957, p. 47.

16. R. Sher, "Effective Resonance Integrals in Th and U," Proceedings of Third Annual Meeting, Am. Nuc. Soc., June 1957.

17. "Reactor Physics Constants," ANL-5800, August 1958.

18. H. J. Amster, "A Compendium of Thermal Neutron Cross Sections Averaged over the Spectra of Wigner and Wilkins," WAPD-185, January 1958.

19. J. A. Thie, et al., "Void Measurement in a Boiling Water Reactor," Nuclear Science and Engineering, Volume 11, No. 1, September 1961.

20. C. R. Breden, "Water Chemistry and Fuel Element Scale in EBWR," ANL-6136, November 1960.

21. R. F. S. Robertson, "Tests of Defected Thoria-Urania Fuel Specimens in EBWR," ANL-6022, June 1960.

22. V. C. Hall, Jr. and R. H. Leyse, "Radiation Levels in EBWR," Nucleonics, Volume 19, No. 3, March 1961, p. 80.

23. W. R. Smith, General Electric Co., Vallecitos Boiling Water Reactor, private communication, December 11, 1958.