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DESIGN REPORT FOR NMSR PRESSURIZED WATER LOOP AT ORR

I. T. Dudley
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DESIGN REPORT FOR NMSR PRESSURIZED WATER LOOP AT ORR

I. T. Dudley D. E. Tidwell D. B. Trauger

ABSTRACT

A pressurized water loop constructed in the Oak Ridge Research Reactor for use in the Maritime Reactor Program is described. Irradiation testing of pressure vessel material specimens and several sets of fuel pins for the NS "Savannah" Nuclear Merchant Ship Reactor, NMSR, has been completed. Operating conditions for the fuel tests nominally were 500°F and 1750 psig.

The loop has capability for operation at 650°F, 2500 psig, with a maximum water flow rate of 80 gpm, and heat removal in excess of 150 kw. Design features include convenient access for insertion and removal of test specimens, efficient use of equipment cubicle space, and reliable operation. The loop is entirely constructed of 300-series stainless steel. A bypass purification and sampling system affords continuous water-chemistry control, and provision has been made for attachment of special equipment for chemistry studies.

Fuel and material test specimens are irradiated in the space provided by two tubes, 1.5 in. ID by 24 in. long, in positions A-1 and A-2 of the ORR core. The maximum unperturbed thermal-neutron flux is approximately 7.5×10^{13} neutrons/cm^2·sec and the peak gamma heat about 5 w/g with the ORR operating at 30 Mw.

INTRODUCTION

Early in 1958 the Oak Ridge National Laboratory was requested by the Maritime Reactors Branch of the AEC to construct a pressurized-water loop in support of the Maritime Ship Reactors Program. Specific objectives of the test program were the evaluation of fuel elements and fuel-element design features for NS "Savannah" replacement cores. Extension of the program was anticipated to include fuel irradiations appropriate to other pressurized-water reactors of interest for Maritime service. Duplication of a then existing loop in the MTR (Materials Test Reactor) was considered; however, a detailed study revealed that extensive design changes would be required. Accordingly, a new design was undertaken including an extensive study of pressurized-water technology. Loops installed in various test reactors at Chalk River, the MTR, and those proposed for the ETR (Engineering Test Reactor) at the NRTS (National Reactor Testing Station) were studied; and the supporting resources of the sponsors, Westinghouse Atomic Power Division, Argonne National Laboratory, Knolls Atomic Power Laboratory, Atomic Energy of Canada, Ltd., and United Kingdom Atomic Energy

Authority were utilized. A new loop design was evolved which embodied the best features of the existing facilities and included several improvements.

The MTR and ETR were considered as possible locations for this loop; however, the ORR (Oak Ridge National Laboratory Research Reactor)[1] was selected because of its favorable arrangement of reactor and pool facilities, availability of equipment space and the convenience of location. Lattice positions A-1 and A-2 were chosen for the in-pile positions and all equipment required to maintain the desired operating conditions was located within a cubicle in the basement of the ORR building.

Design of this test facility was begun in February 1958, and procurement of equipment was started in the latter part of that year. Installation was complete and the facility was ready for operation by the scheduled date, December 1, 1959. It has operated successfully since that date with only minor alterations.

An analysis of the completed installation[2] shows the following cost distribution, excluding overhead: engineering and design, 25.8%; fabrication and installation, 25.2%; materials, 49.0%.

OBJECTIVES

The test loop provides a facility in which fuel and other materials can be irradiated under conditions simulating those for a pressurized-water reactor. The specific test loop design objectives were the provision of a facility for:

1. prototype testing of fuel pins proposed for use in the first nuclear-powered merchant ship (NS "Savannah"),

2. development and testing of other fuel element designs,

3. development and testing of structural and control materials of interest in pressurized-water reactors,

4. studies of water-chemistry and activity buildup in pressurized-water systems,

5. other studies of a basic or applied nature which require an environment combining radiation and high-temperature water.

DESIGN DATA

	Test Loop Design Criteria	Typical Experimental Test Conditions	NMSR[a] Operating Conditions
Pressure, psig			
Loop	2500 (max)	1750	1750
In-pile tube	2250 (max)		
Temperature, °F[b]			
Loop	650 (max)		
In-pile tube inlet		500	
Mean		508	508
In-pile tube outlet	625 (max)	516	
Fluid flow			
In-pile tube (U-tube), tube), in., ID		1.5	
Effective flow area, in.^{2}/pin		0.393	0.185
Velocity, first pass, fps		10.00	9.66
Flow rate, gpm	0–80[c]	40	
Fuel pins, in.			
UO_2		Swaged powder	Pellets 0.4265 × 0.5 long
SS clad thickness		0.035	0.035
Clad outer diameter		0.5	0.5
Active length		16.5	66.0
Overall-length		18.0	
Pins per test		6 (3 each leg)	
Pin configuration[d]		0.612 in. × 60° pitch	0.612 in. × sq pitch
Heat generation rate, kw			
Fuel pins at NMSR av, 7.8 w/g		18	
Fuel pins at NMSR max, 34.6 w/g		77	

DESIGN DATA (continued)

	Test Loop Design Criteria	Typical Experimental Test Conditions	NMSR[a] Operating Conditions
Heat generation rate, kw			
Gamma heat, 10 w/g (in-pile tube)		67.0	
Total	150.0[e]	144.0 (max)	
Heat exchanger capacity (water-cooled), kw	150.0[e]		
Electric line heater capacity, kw	60		
Loop piping	1-1/2 IPS sched-80 347 SS		
Water chemistry			
pH range	6.5–11.0	7.5–8.5	7.5–8.5
Hydrogen content, ppm	0–4	3.6	3.6
Total max allowable solids, ppm		5	5
Max oxygen content, ppm	0.05	0.05	0.05
Max chlorine content, ppm	0.1	0.1	0.1
Flow rate through purification system, % of total flow	0–1.25	0.20–1.00	0.20–0.25

[a] Nuclear merchant ship reactor.

[b] Loop temperature may be varied by heat exchanger and electric heater controls.

[c] Flow rate may be varied by main throttle valve control.

[d] Test pin dimensions may be varied within limitations of in-pile tube.

[e] Heat exchanger rated at 150.0 kw with inlet temperature of 300°F.

DESCRIPTION

The test loop is designed to recirculate water at a maximum of 2500 psig, 650°F, and at flow rates to 80 gpm. The in-pile section for use in the NS "Savannah" tests is designed for operation at a maximum of 2250 psig and 625°F. Circulation of water through the loop provides forced convection cooling of the test specimens and in-pile section pressure pipe.

The test facility consists of an in-pile section, a main loop heat exchanger, canned-rotor pumps for circulation, electric heaters for temperature control, a surge tank, a purification system, an out-of-pile test section, a water sample station, a dump tank, and a water makeup system. All process equipment except the in-pile section, sample station, dump tank, and makeup system are located within the shielded equipment cubicle. Instrumentation and controls are provided for the recording and readout of data and for automatic control of equipment. The entire facility is conveniently arranged for a minimum of operating and maintenance requirements. Flow sheets and drawings describing both the functional and physical arrangement of equipment are presented in Figs. 1–9.

The primary system was constructed entirely of 300-series stainless steels (principally type 347). This type of steel was selected because it is "stabilized" and resists intergranular corrosion at weld joints without annealing.

A battery-powered motor-generator set was provided to supply power temporarily for afterheat removal during outages of the normal power supply. This MG set will also supply emergency power for two other in-pile experiments when they are put into operation.

In-Pile Section

The in-pile section, which is located in lattice positions A-1 and A-2 of the reactor, consists of a U-tube made of 1.5-in.-ID pipe. The maximum unperturbed thermal neutron flux within the in-pile test section occurs in the A-2 leg and is approximately 7.5×10^{13} neutrons/cm^2·sec at 30 Mw operation of the ORR. The peak thermal neutron flux in the A-1 leg is approximately 4×10^{13} neutrons/cm^2·sec for the same ORR power.

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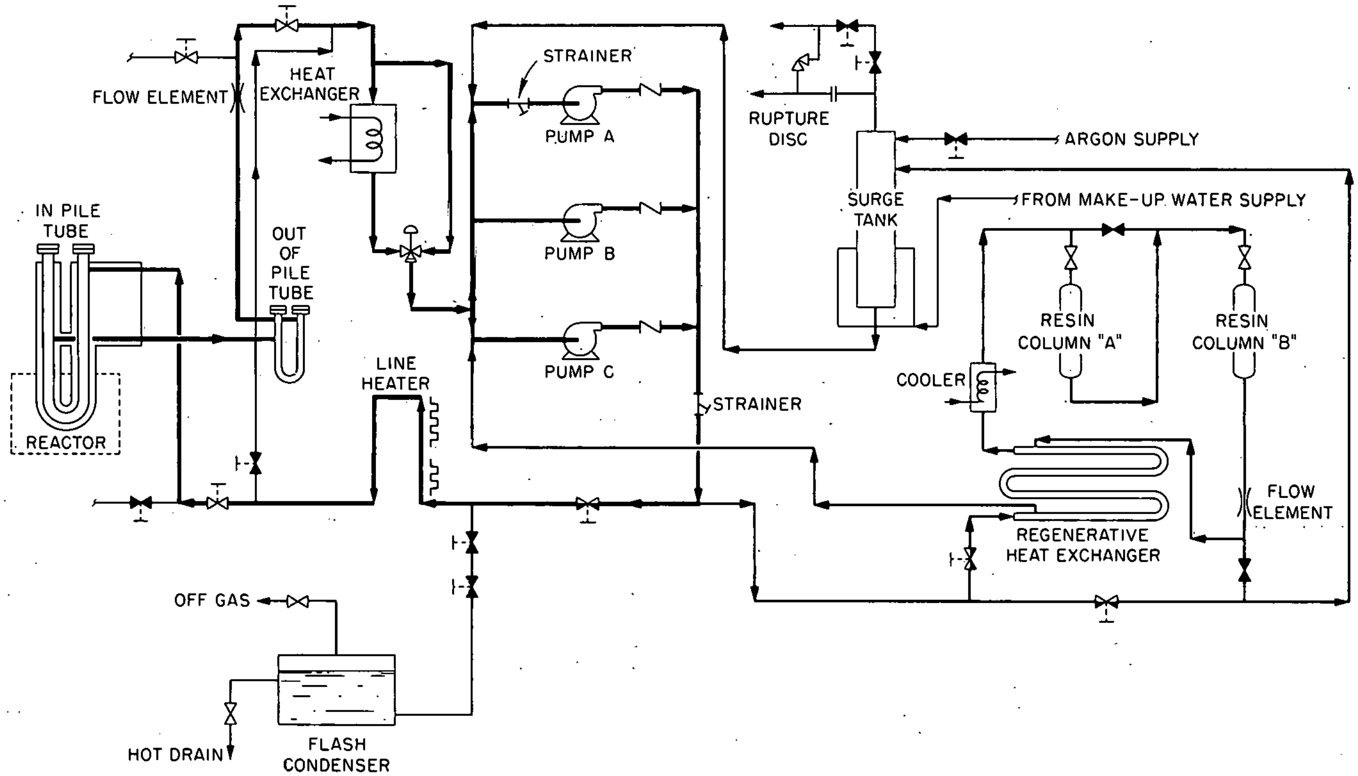

Fig. 1. NMSR Pressurized-Water Loop at the ORR – Simplified Flow Diagram.

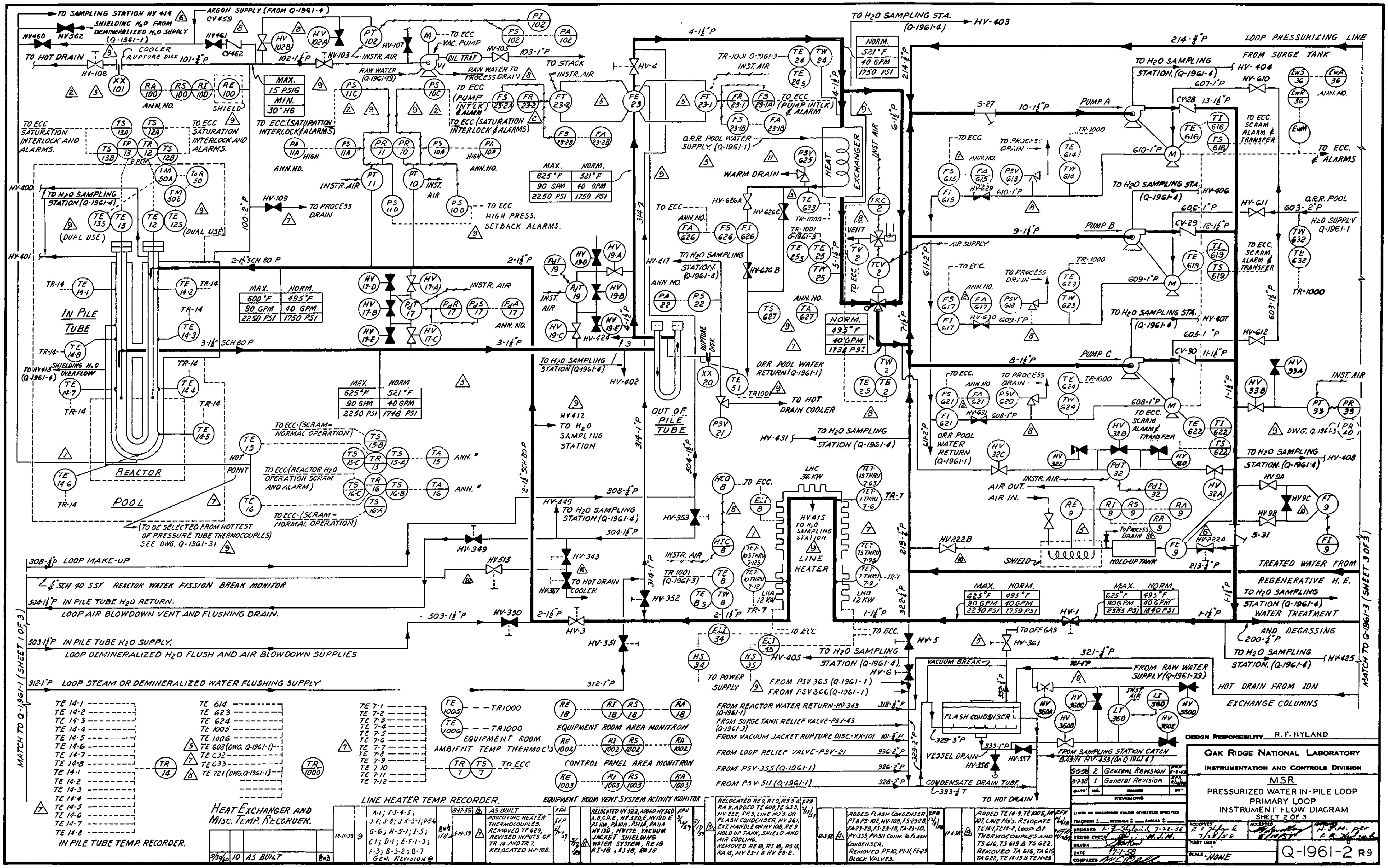

Fig. 2. Flow Diagram – Main Portion of Loop.

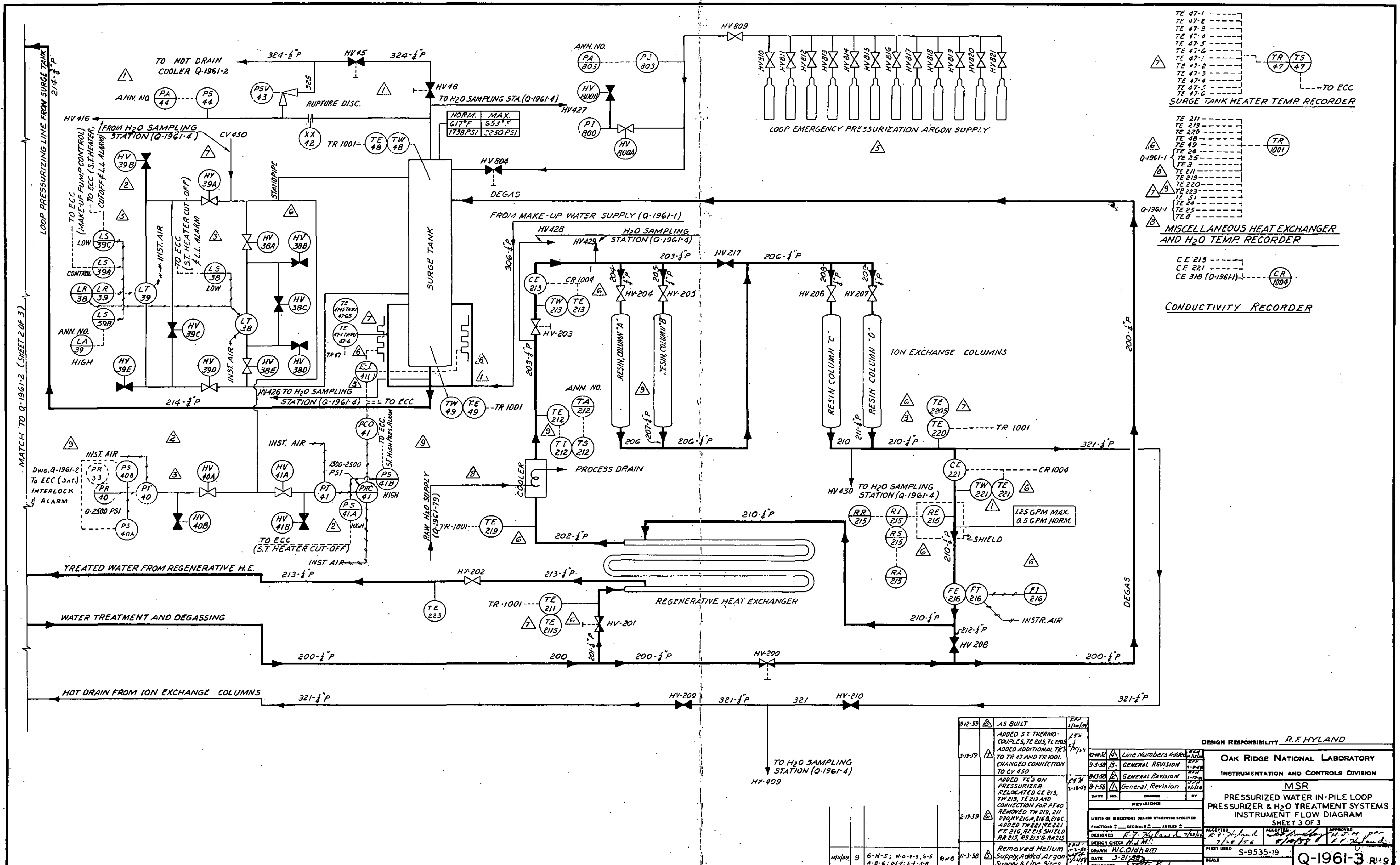

Fig. 3. Flow Diagram — Surge Tank and Purification System.

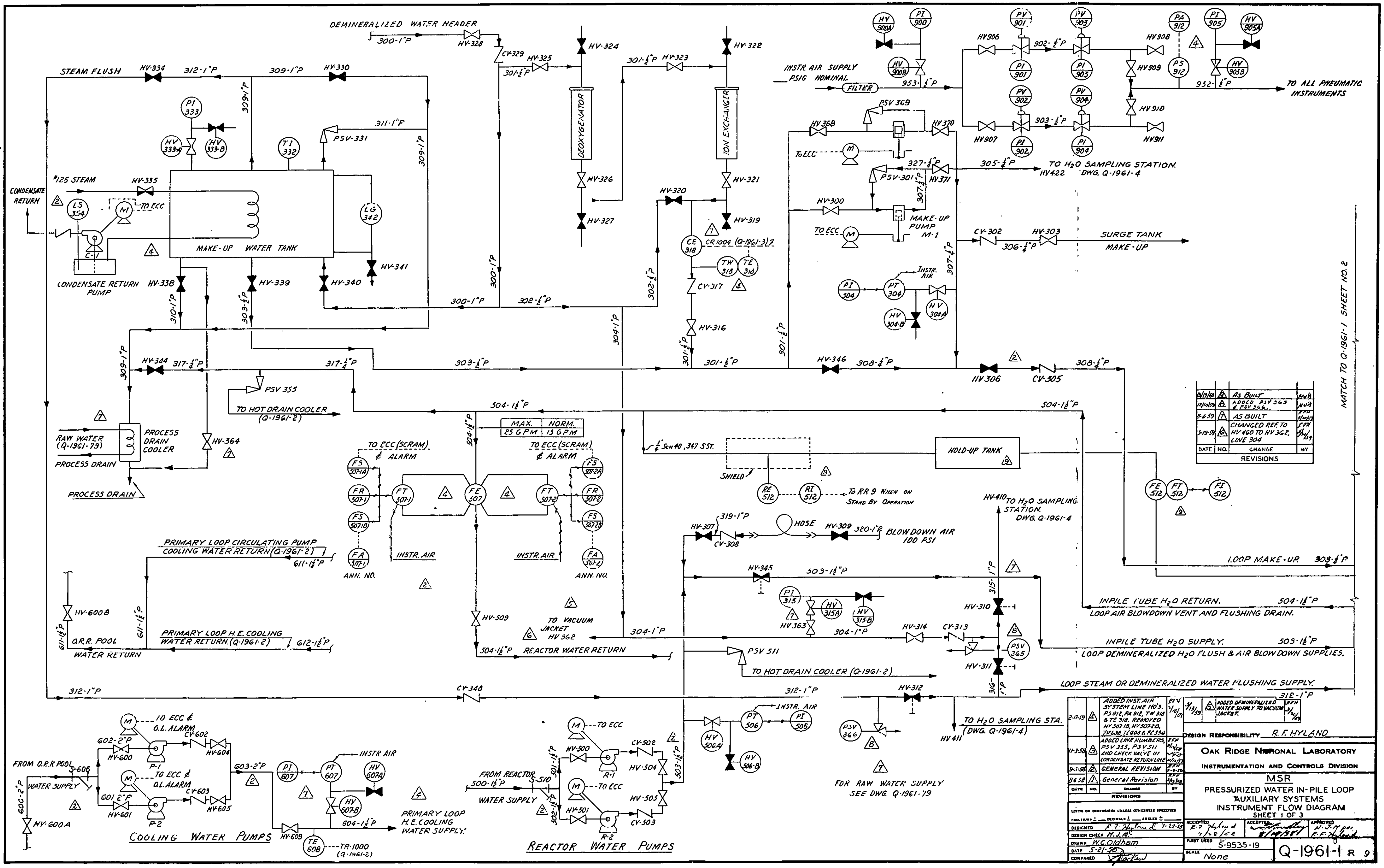

Fig. 4. Flow Diagram — Makeup System.

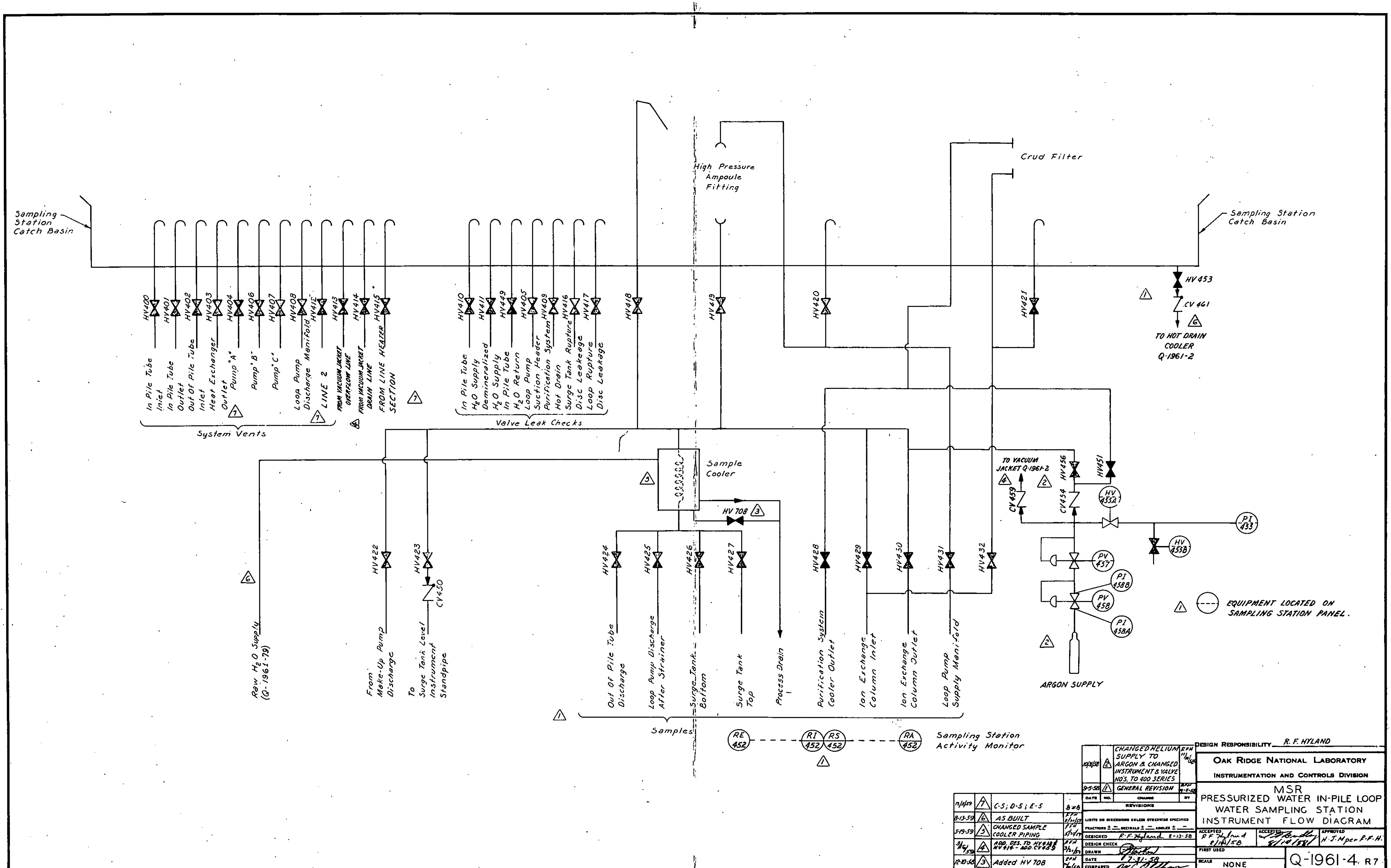

Fig. 5. Flow Diagram — Sample Station.

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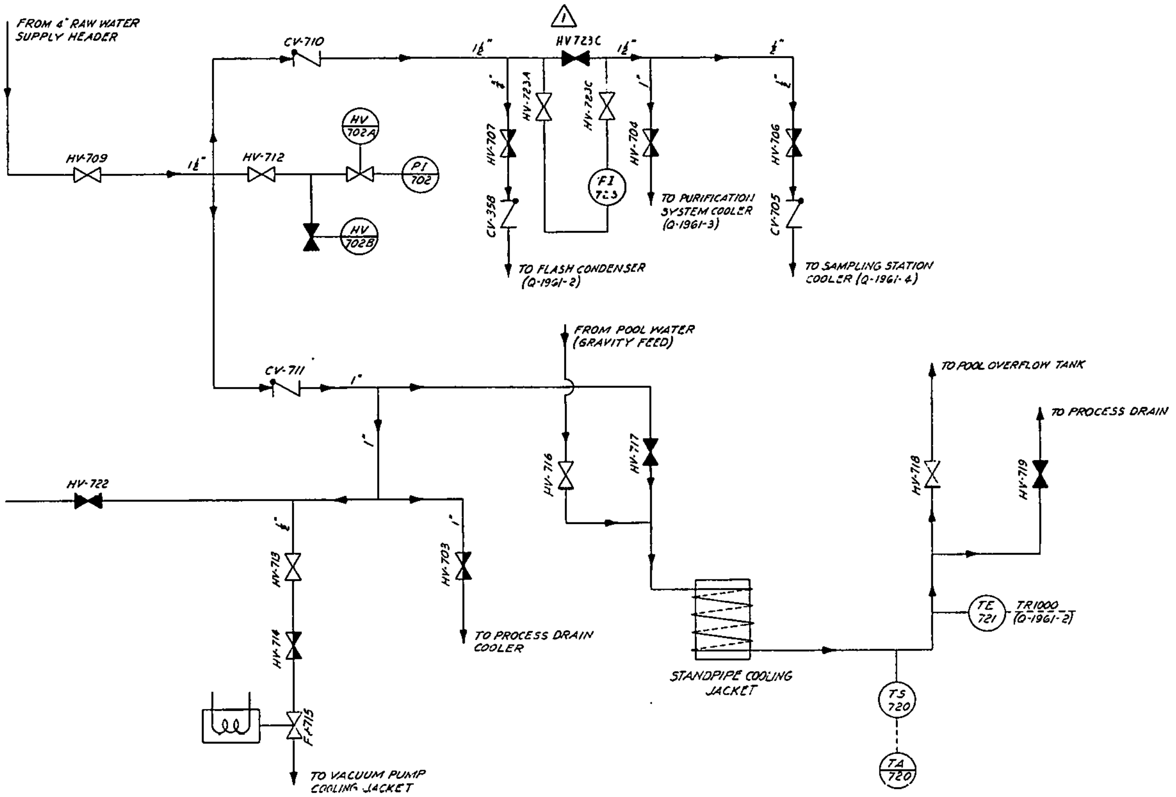

Fig. 6. Flow Diagram — Raw Water System.

Fast neutron flux values (above 1 Mev) are approximately a factor of 5 lower than the respective thermal flux values. These values are extrapolations of measurements made at low power and at 20 Mw operation of the ORR.[3] The extrapolations take into consideration the increase in reactor power and the addition of a large experimental facility in core position B-1. Test specimens are accommodated in both legs and are positioned axially in the peak flux by specimen holders. The in-pile section provides space for the simultaneous irradiation of six experimental fuel-pin test specimens of the initial design. These specimens are 0.5 in. in diameter, 18 in. long, and are assembled in units of three. Fuel specimens or material specimens of other configurations may be tested within the space available by use of appropriately designed holders. Test specimens and U-tubes are cooled by the flow of loop-system water.

The in-pile section piping enters the reactor vessel through a flange located directly over lattice positions A-1 and A-2. A stainless steel jacket encases the pressure piping for secondary containment and thermal

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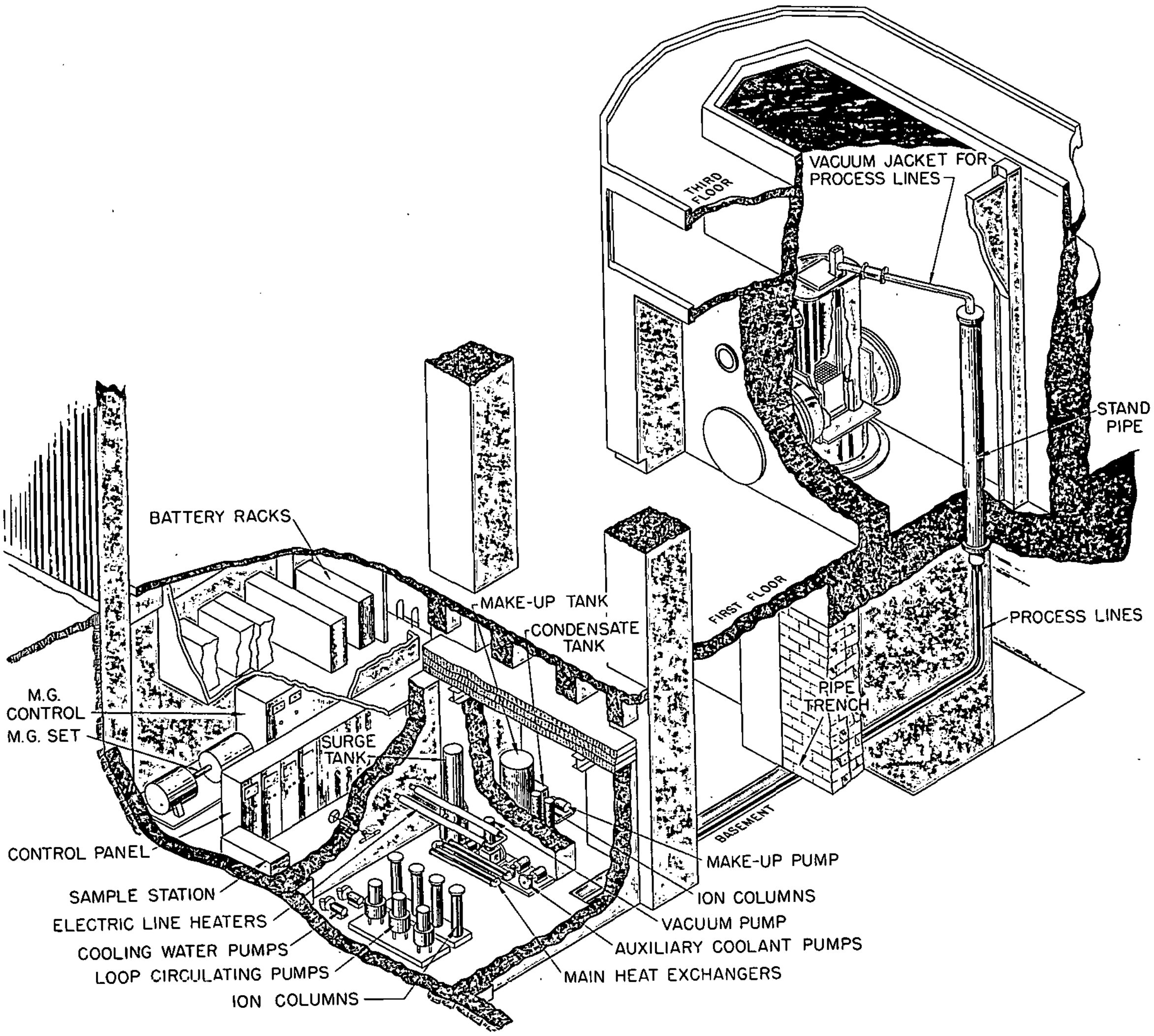

Fig. 7. Cutaway View of the NMSR Pressurized-Water Loop in the ORR.

insulation. An aluminum sleeve fits over the in-pile section vacuum jacket within the core and conforms to the configuration of ORR fuel elements so that reactor cooling-water flow rates are not altered by this installation.

Each leg of the in-pile section extends above the reactor vessel re-fueling flange, where an O-ring-sealed breech-lock closure is provided for the insertion and removal of test specimens. Reactor pool water is

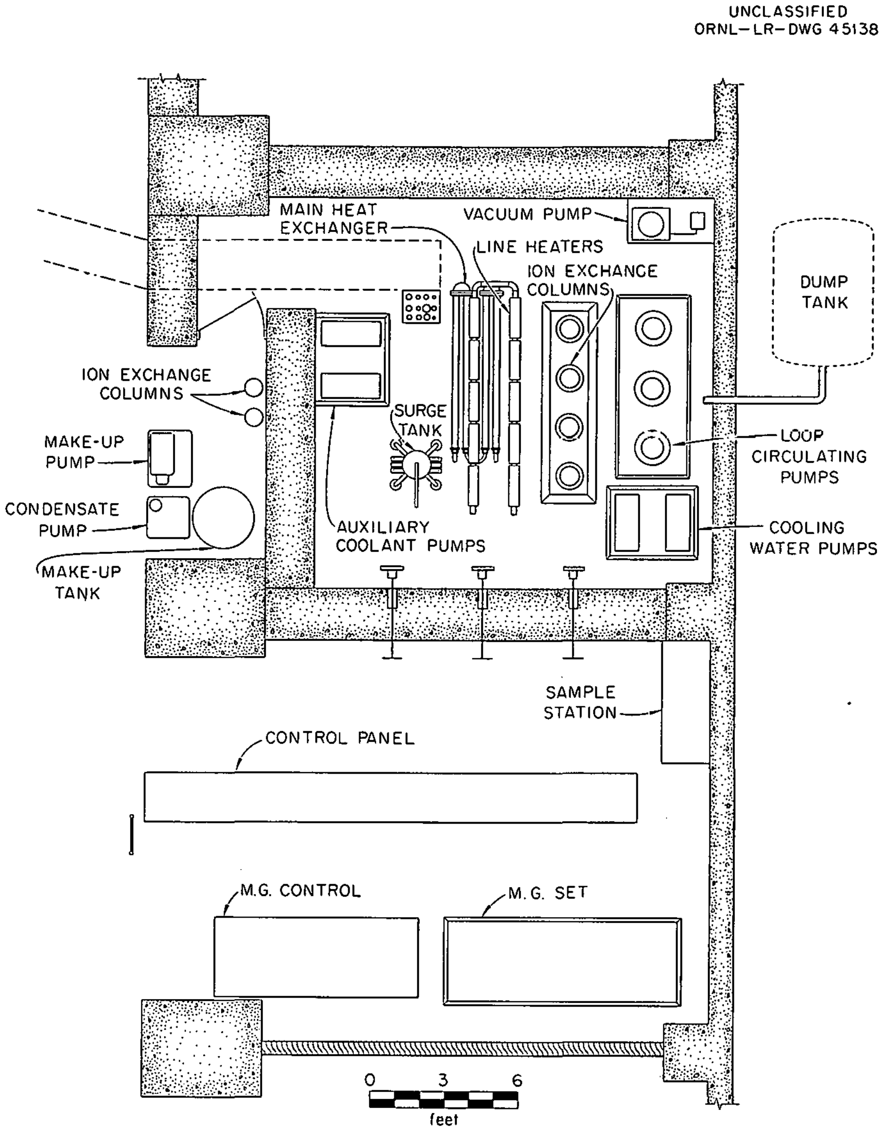

Fig. 8. Plan View of Equipment Cubicle and Operating Area for NMSR Pressurized-Water Loop at the ORR.

9 ft deep over the test specimens during removal; thus no additional shielding is required.

Design of the in-pile section is shown in Figs. 10 and 11 which illustrate the arrangement of concentric piping for insulation and secondary containment.

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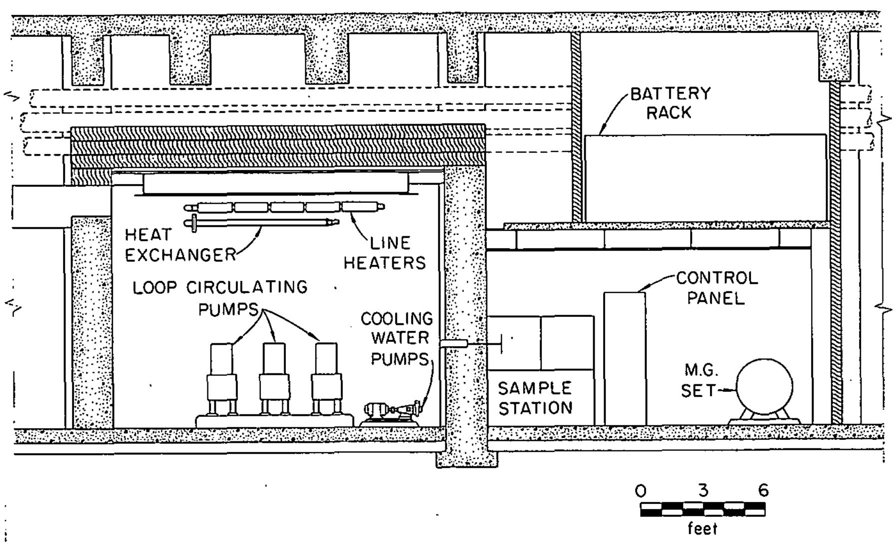

Fig. 9. Elevation of Equipment Cubicle and Operating Area for NMSR Pressurized-Water Loop at the ORR.

Pressure Tubes

Design pressure and temperature of the in-pile section were set at 2250 psig and 625°F. Combined thermal and pressure stresses were limited to values specified by ASME Boiler and Pressure Vessel Code.[4] Stress analyses made on the in-pile portion of this loop indicated that type 316 stainless steel met code requirements. Other materials such as types 304 and 347 stainless steel could not be used because of their lower allowable stress values. Thermal stresses induced by gamma heating were of considerable importance in the design of the in-pile section. A conservative value of 10 w/g for the peak gamma heating was assumed in the design criteria; measured values for gamma heating in this area of the reactor have shown the heating rate to be approximately 5 w/g for 30-Mw ORR operation.[5] For thermal stress considerations it was assumed that all gamma heat in the pressure tubes would be removed only by the loop water flowing over the inner surfaces.

In order to use the maximum inner diameter of a standard size pipe whose wall thickness could be adjusted by machining the exterior, 1-1/2-in. sched-80 pipe was chosen. It was purchased with an actual inside diameter of 1.510-1.490 in. along a minimum length of 36 in. from one end. For that portion within the reactor core, the wall thickness was chosen by optimizing for the minimum combined pressure and thermal stress as shown

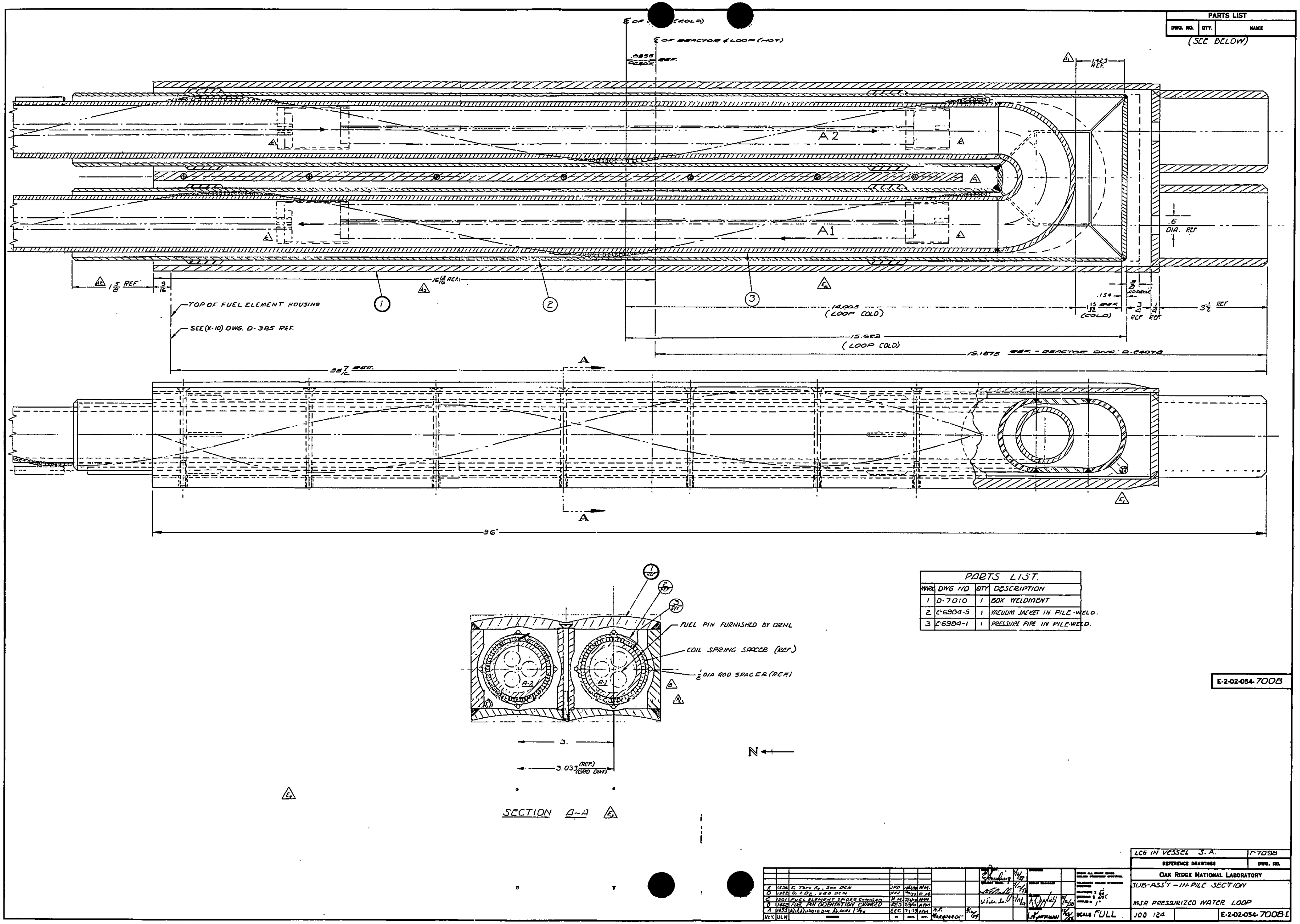

Fig. 10. Assembly – Lower End of In-Pile Section.

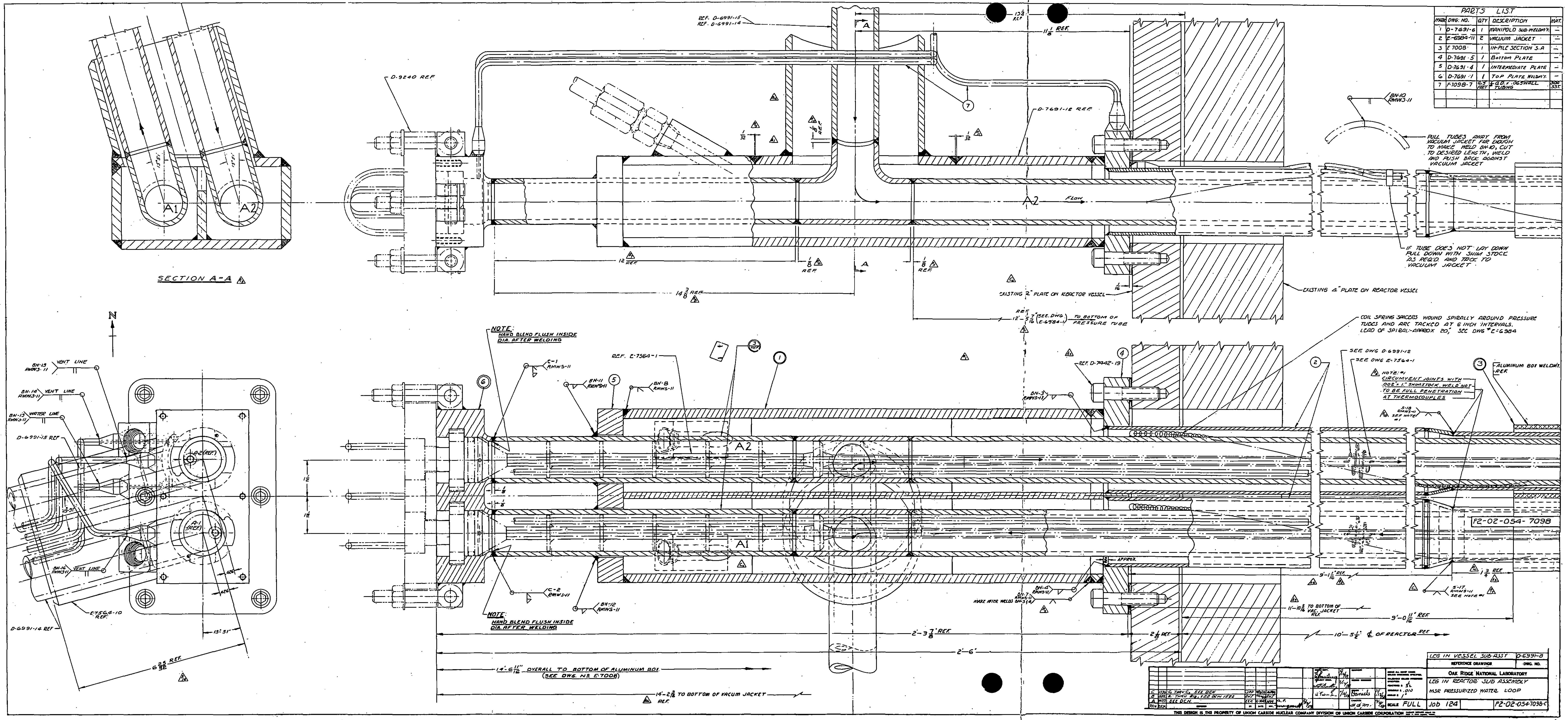

Fig. 11. Assembly — Upper End of In-Pile Section.

in Fig. 12. The nominal wall thickness is slightly greater than the optimum to avoid exceeding the allowable pressure stress for type 316 stainless steel when tolerances are considered.

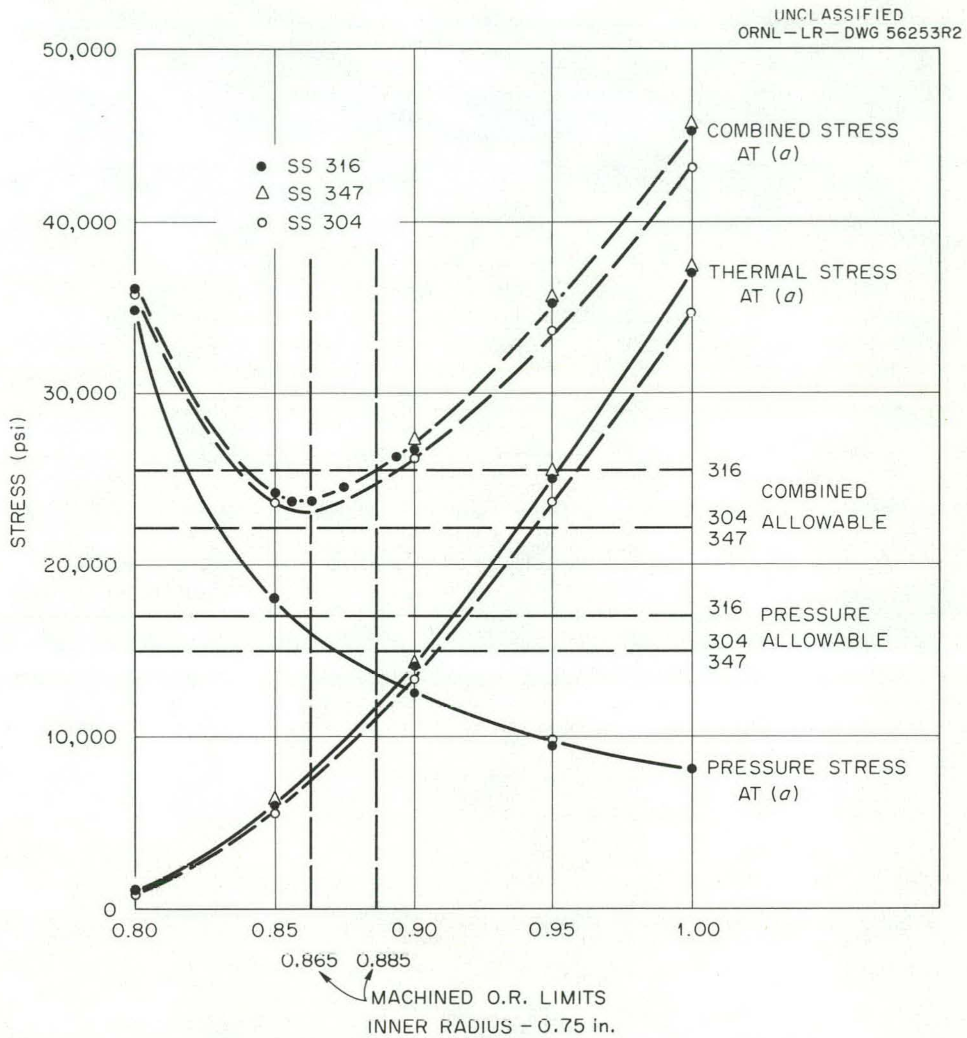

Fig. 12. Effect of Wall Thickness on Stresses at Inside Radius for Various Stainless Steel Pipes.

A 1-1/2-in. sched-80, 180° return bend was chosen for joining the tubes. Since the gamma heat is much lower at the edge of the reactor core, it was not necessary to thin this part to the same wall thickness as that of the tubes. This return bend was purchased with a center-to-center dimension of 3 ± 1/64 in. (see Fig. 13).

Specimen Holders

Test specimens are positioned vertically in the in-pile pressure tubes by a 3/8-in.-OD tube suspended from each pressure seal closure. Spiral

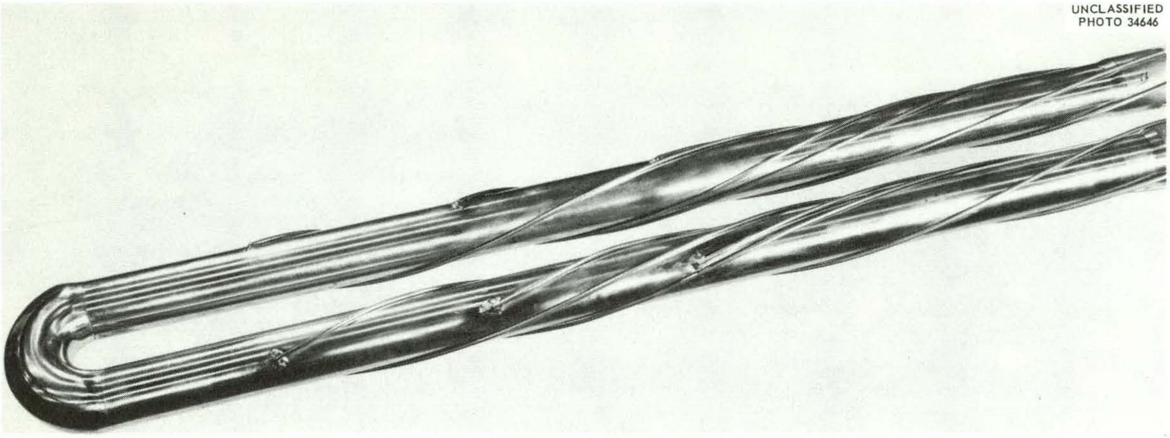

Fig. 13. In-Pile Pressure Tube with Thermocouple Attached.

springs are mounted on this tube to minimize lateral vibrations induced by the flow of loop water. Fuel test specimens of the NS "Savannah" type are furnace-brazed together with two ferrule spacers, and the assembly is then tack-welded to a radial spacer at each end. The upper radial spacer is welded to the 3/8-in. suspension tube (see Figs. 14–16).

The assembly of test specimens is removed from the in-pile section by lifting after disengaging the bayonet latch on the pressure seal closure.

Vacuum Jacket

A jacket installed over the high-temperature piping provides thermal insulation and secondary containment. The jacket is evacuated to reduce heat transfer by thermal convection currents in the gas annulus and to increase the sensitivity of leak detection. This jacket is connected to a vacuum pump via the pool piping vacuum jacket and a 2-in. line.

Two small lines are tied to the in-pile vacuum jacket to provide a blow-down drain, water fill, water-level indication, and gas purge. On loss of coolant flow through the in-pile pressure pipe, the reactor cooling water may be used as a heat sink by increasing heat transfer from the pressure piping to the vacuum jacket on filling the vacuum system with helium or water. Water additions of course can be made only when the pressure-pipe temperatures are low.

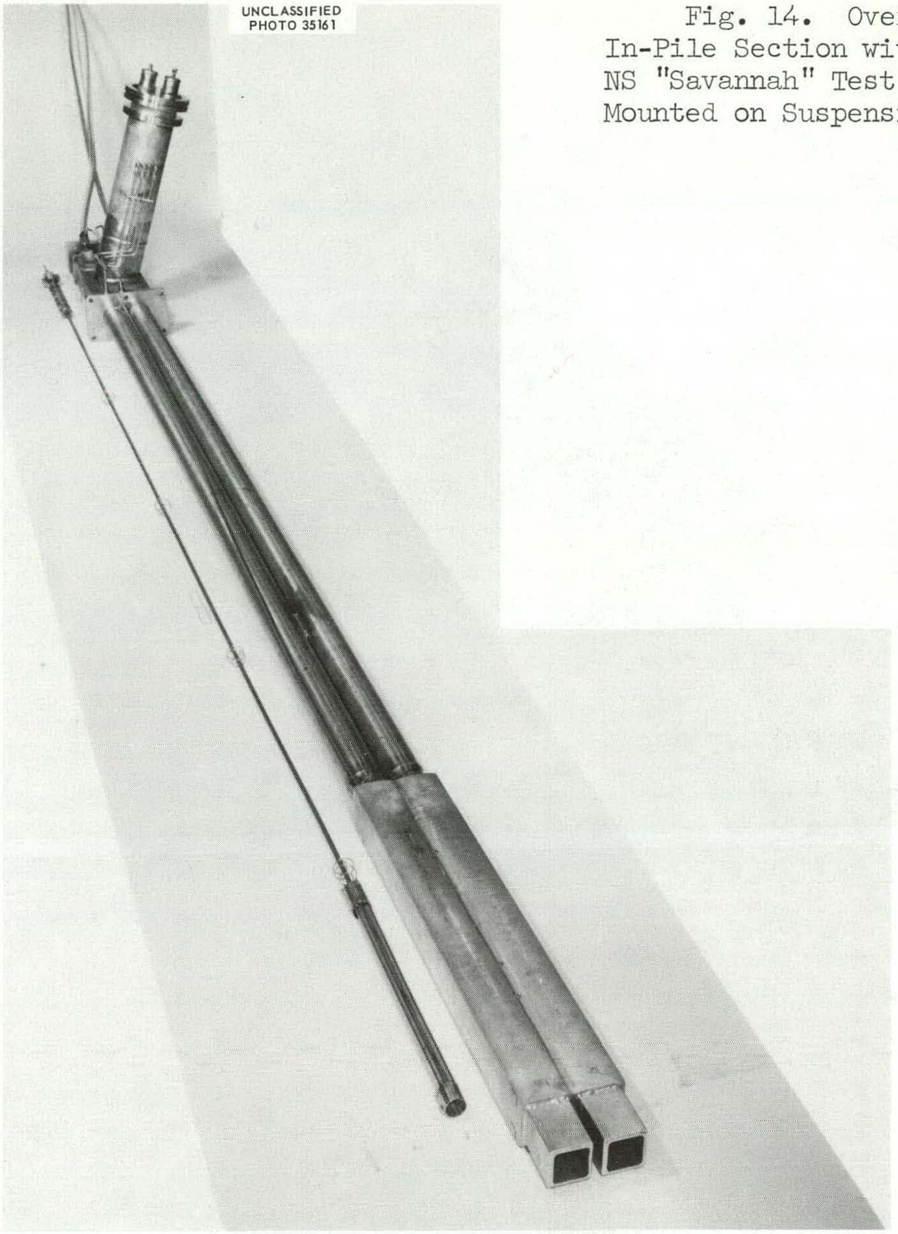

Fig. 14. Overall View of In-Pile Section with One Set of NS "Savannah" Test Specimens Mounted on Suspension Assembly.

Pressure Seal

The breech-lock closure is sealed with a neoprene O-ring which is limited to a maximum operating temperature of 300°F. The seal region is cooled by direct contact with the reactor pool water in which it is immersed. Baffles are installed between the seal and flowing high-temperature water within the loop piping (see Fig. 10). These baffles serve to reduce heat transfer to the seal region and to minimize the loop heat losses

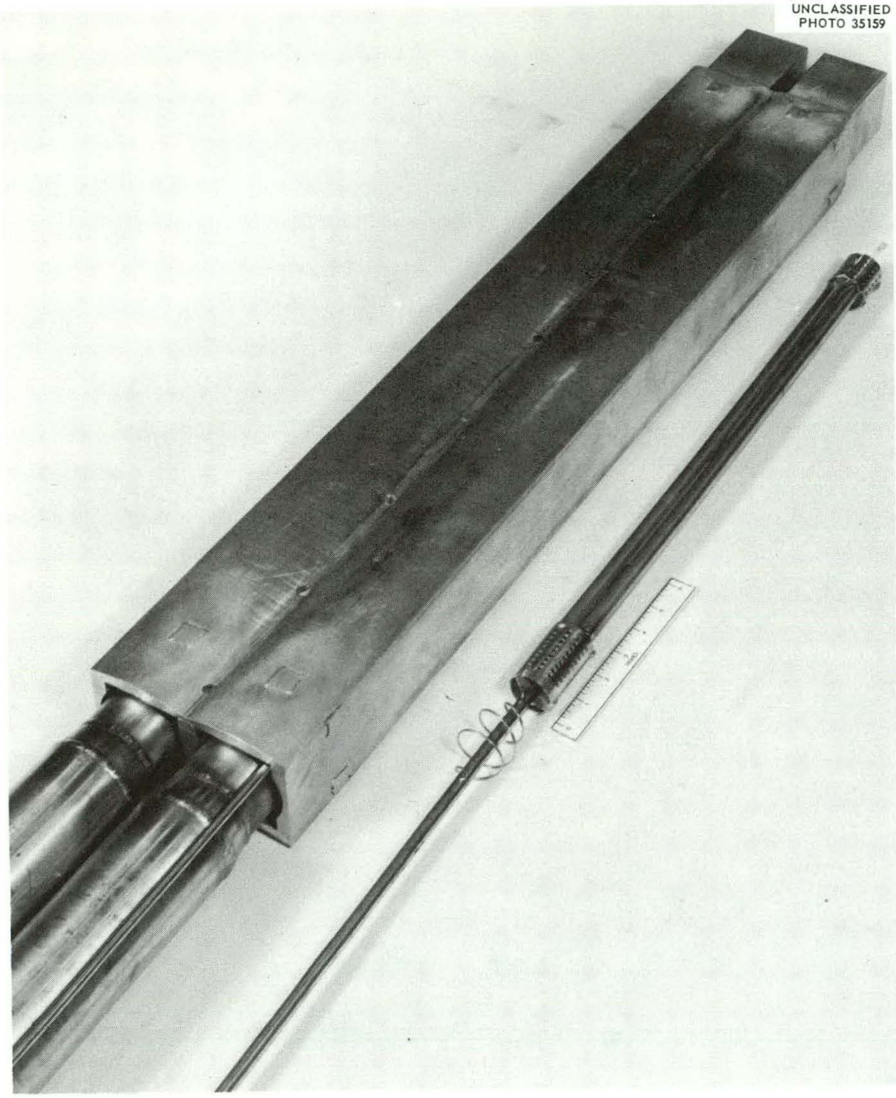

Fig. 15. In-Pile Section Assembly Showing the Aluminum Filler Box and a Set of Dummy NS "Savannah" Test Specimens.

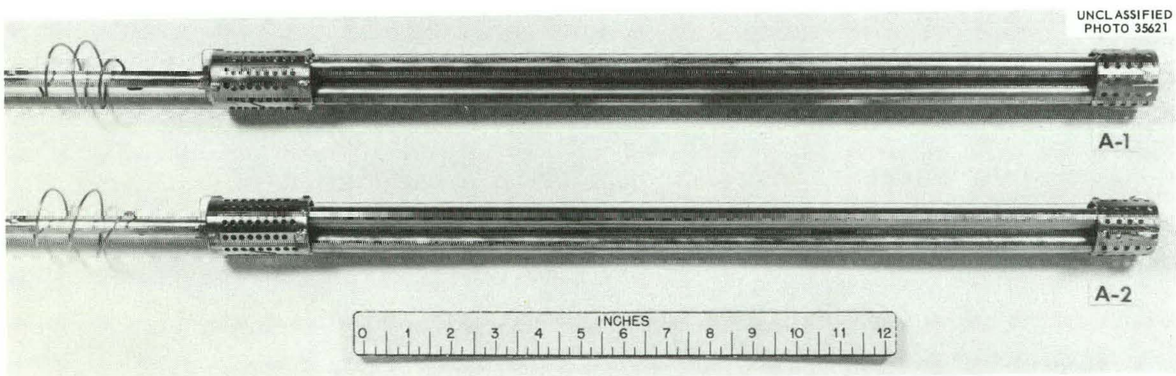

Fig. 16. NS "Savannah" Fuel Test Specimens (Swaged) Assembly.

at this point. Loop temperature is reduced to 100–125°F when the pool water level is lowered below the elevation of the O-ring seals, both to protect the seals and to reduce hazards from high temperature piping and steam for personnel working on top of the reactor vessel.

Thermocouples

Grounded junction type chromel-alumel thermocouples in stainless steel sheaths with magnesium oxide insulation are used for measurement of in-pile structure and water temperatures. Thermocouples, 1/4 in. OD are located in the upper end of each leg of the U-tube to measure the inlet and outlet water temperatures. New thermocouples are installed with each experimental fuel assembly. The sheaths penetrate the pressure system through welded joints in the O-ring-sealed closures on top of the in-pile section. These welds are backed up by Teflon packing joints (see Fig. 17).

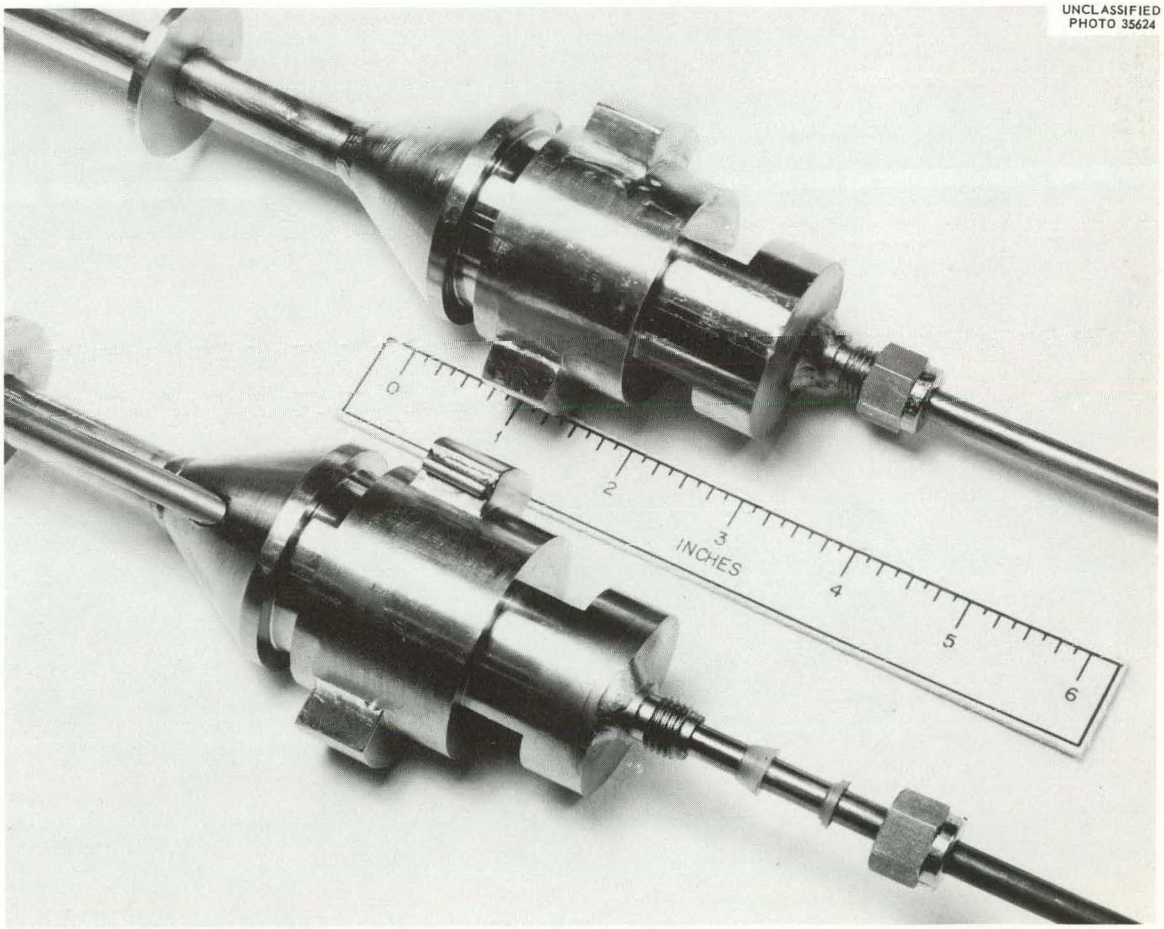

Fig. 17. Upper Ends of Test Specimen Supports and Pressure Seal Closures Showing Thermocouple Pressure Seals and Backup Packing Glands.

Surface temperatures of the pressure tubes within the reactor core are monitored by five 1/8-in. OD thermocouples on each leg of the U-tube (see Fig. 13). They penetrate the vacuum jacket through Conax seals located above the reactor vessel (see Fig. 18).

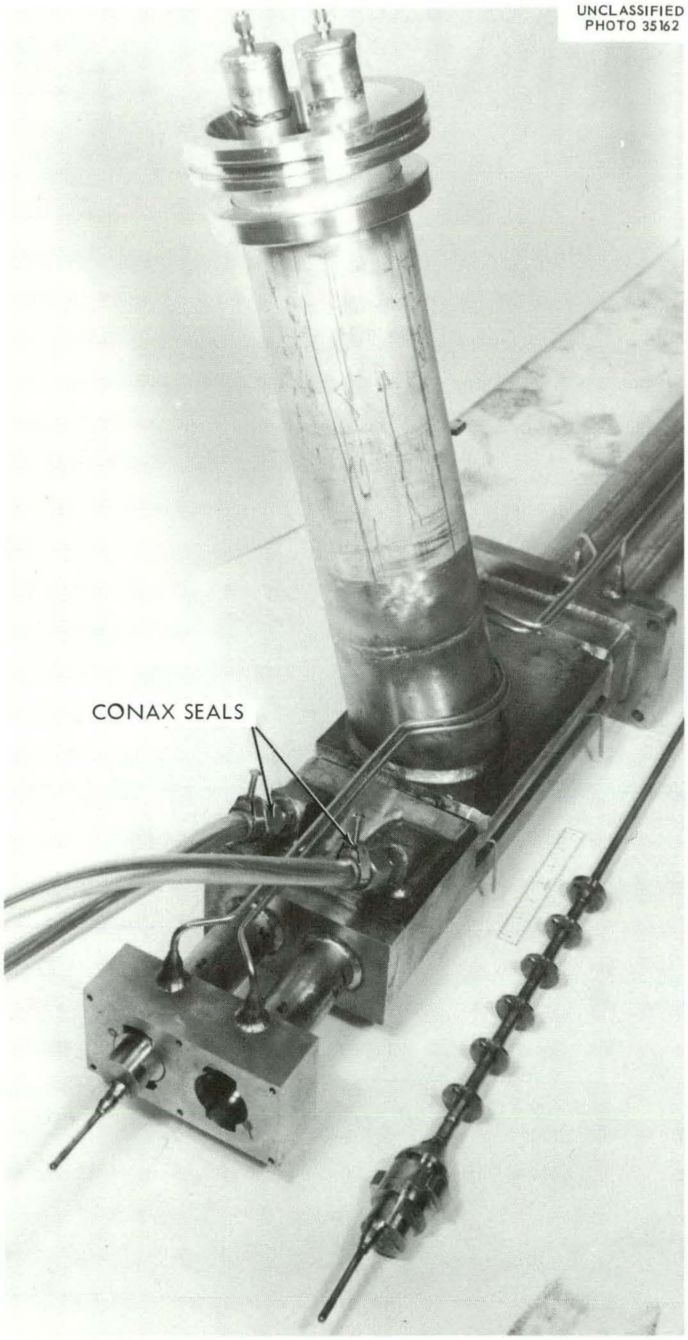

Fig. 18. Upper End of In-Pile Section During Fabrication.

Out-of-Pile Test Section

The out-of-pile test section which is immediately downstream of the in-pile section and located in the equipment room is provided for specimen corrosion studies. Control specimens may be exposed to an environment similar to that of the in-pile specimens except for the absence of the heat flux and radiation. This test section is similar to the in-pile test section except that it is positioned horizontally instead of vertically. Access to each leg of the out-of-pile test section is by means of "Flexitallic" gasketed blind flanges.

Pressure Piping

The major portion of the loop pressure piping 1-1/2-in.-IPS, sched-80 type 347 stainless steel, with butt-type weld joints made in accordance with ORNL specification PS-13. All of the pressure piping and fittings were given dye-penetrant and ultrasonic inspection before installation. Weld joints were inspected by dye-penetrant and radiographic methods. The design meets the requirements of the ASA Piping Code (using a corrosion allowance of 0.025 in.) for conditions of 2500 psig and 650°F.

Socket-weld joints were used on 1/4-in. tubing in areas of low operating temperatures. High-pressure (30,000 psi) autoclave fittings were used on 1/4-in. tubing connections within the sample station.

In-Pool Piping

System pressure piping connected to the in-pile test section penetrates the pool floor through the south standpipe (see Figs. 7 and 19). The pipes are contained within a 5-in.-IPS, sched-40 pipe vacuum jacket which serves as a thermal insulator and secondary containment. An O-ring-sealed sleeve in the jacket is provided for access to the pressure-pipe weld joints. The pressure pipes are to be cut at this location whenever the in-pile section is removed.

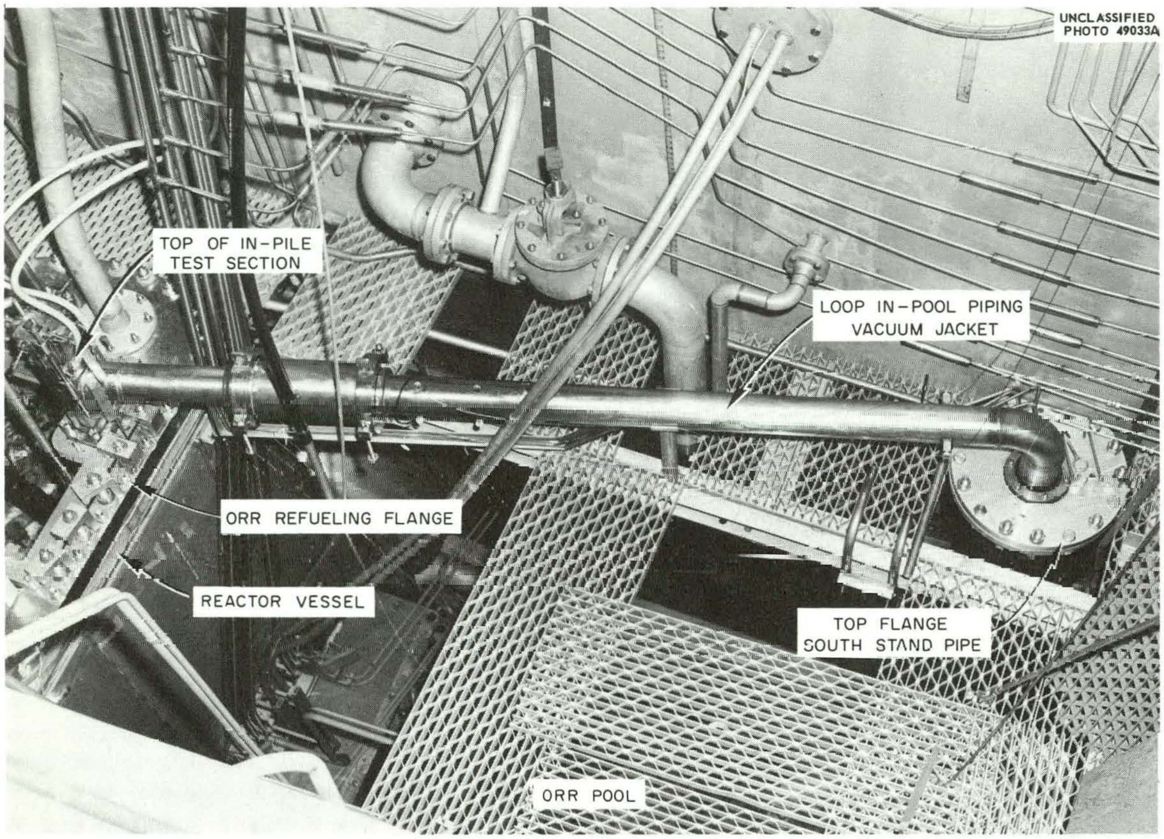

Fig. 19. View of ORR Pool Showing In-Pile Section and In-Pool Piping.

Differential thermal expansion in both the horizontal and the vertical standpipe runs was accomodated in the standpipe to prevent transmission of excessive forces to the reactor structure. Thermal expansion of the 5-in. jacket within the standpipe is downward since it is anchored only at the upper flange joint. Expansion of the pressure pipes in this section is upward relative to the 5-in. jacket, since they are anchored to the jacket at the lower end. The net upward expansion of the pressure pipes is accommodated by flexing the 11 ft horizontal runs.

Thermal expansion of the horizontal 5-in. jacket is provided for by deflection of the concentric sleeve in the upper portion of the standpipe (see Fig. 20).

Pressure-System Valves

All 1-1/2-in. valves, including the blending valve, were supplied by the Annin Company. They are equipped with Teflon packing, cooling fins, and Flexitallic gaskets in the body flange joints. All valves were welded to the piping with butt-type joints.

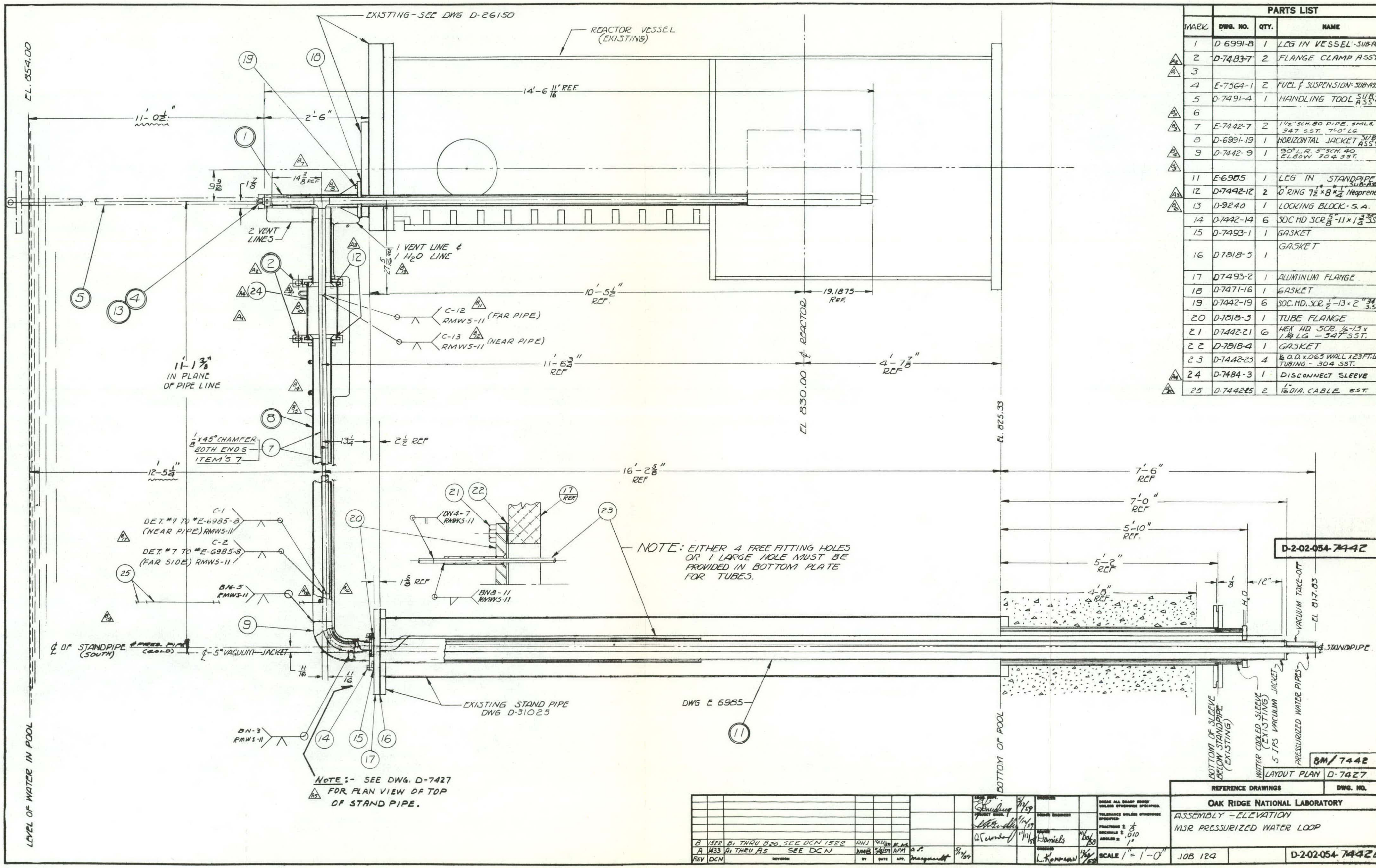

Fig. 20. Elevation of In-Pile Section and Pool Piping.

Modified autoclave valves were installed in the 1/4-in. pipe lines at high-temperature locations. Autoclave valves were used exclusively on instrument lines and sample lines. All of the sample line valves were located at the sample station.

Heat Exchanger

A water-cooled heat exchanger is used for removing heat generated within the system by fission and gamma heating. It may also be used for emergency cooling. The capacity of the heat exchanger is based on 150 Kw heat transfer with the system operating at a temperature of 300°F. The unit has excess capacity at higher operating temperatures. A direct water-to-water heat exchanger was chosen to reduce space requirements and to simplify controls. Two Griscom-Russell size No. C-72 heat exchanger units were selected and installed in series overhead near the equipment cubicle ceiling (see Fig. 21). Cooling water for these units is taken from and

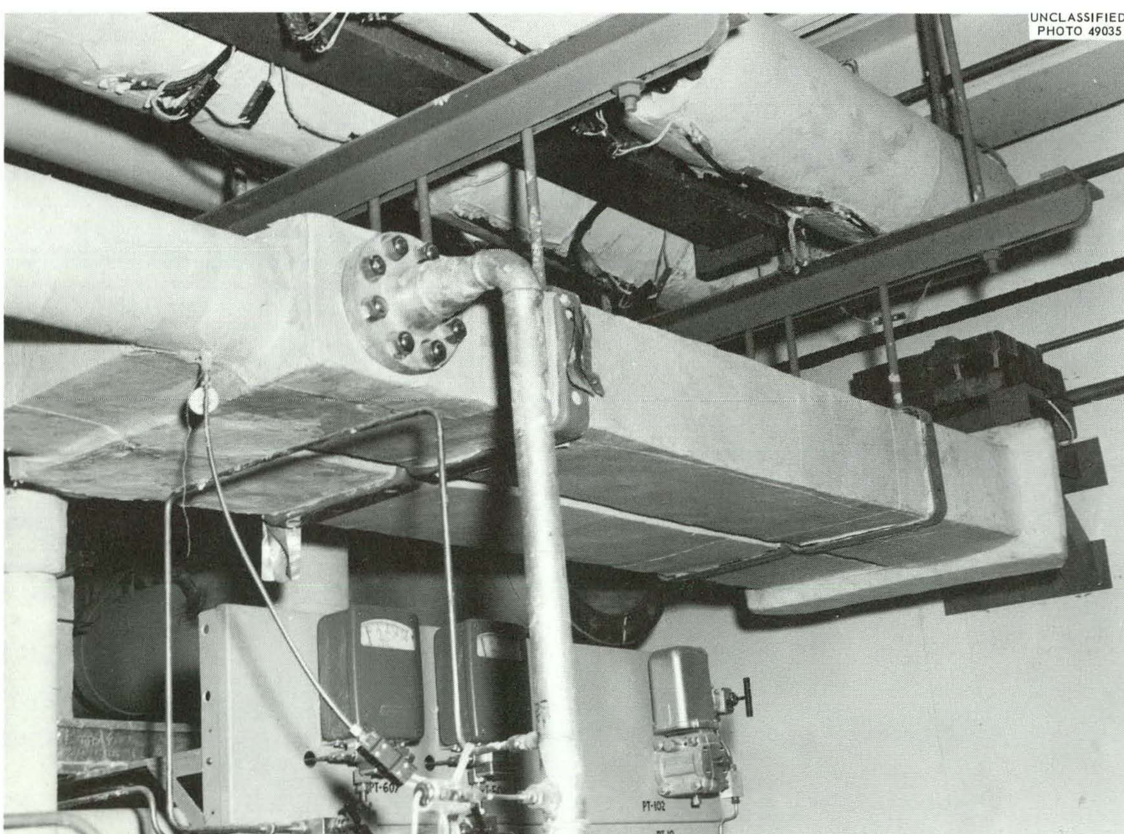

Fig. 21. Water-Cooled Heat Exchanger Mounted Overhead.

returned to the ORR pool by a system containing one operating pump and one standby pump. These pumps are arranged for automatic shifting of operation to the standby unit. Refer to Appendic C for the control details. Water enters from the pool at approximately 100°F, and is limited to temperatures less than 200°F on leaving the heat exchanger. It is not practical to control the heat exchanger capacity by varying the pool water flow rate in view of the limitation on its outlet temperature. The capacity of the heat exchanger is therefore controlled by a heat exchanger bypass line and blending valve in the loop system. This is an automatically controlled, pneumatically operated valve supplied by the Annin Company.

Main Circulating Pumps

The three main circulating pumps are canned-rotor water-cooled, Westinghouse Model A-150D. They are installed in parallel, with one normally operating and two in standby (see Figs. 22 and 23). Each pump is equipped with a check valve on its discharge. This arrangement permits rapid shifting of operation to the standby units. Cooling water for these pumps is

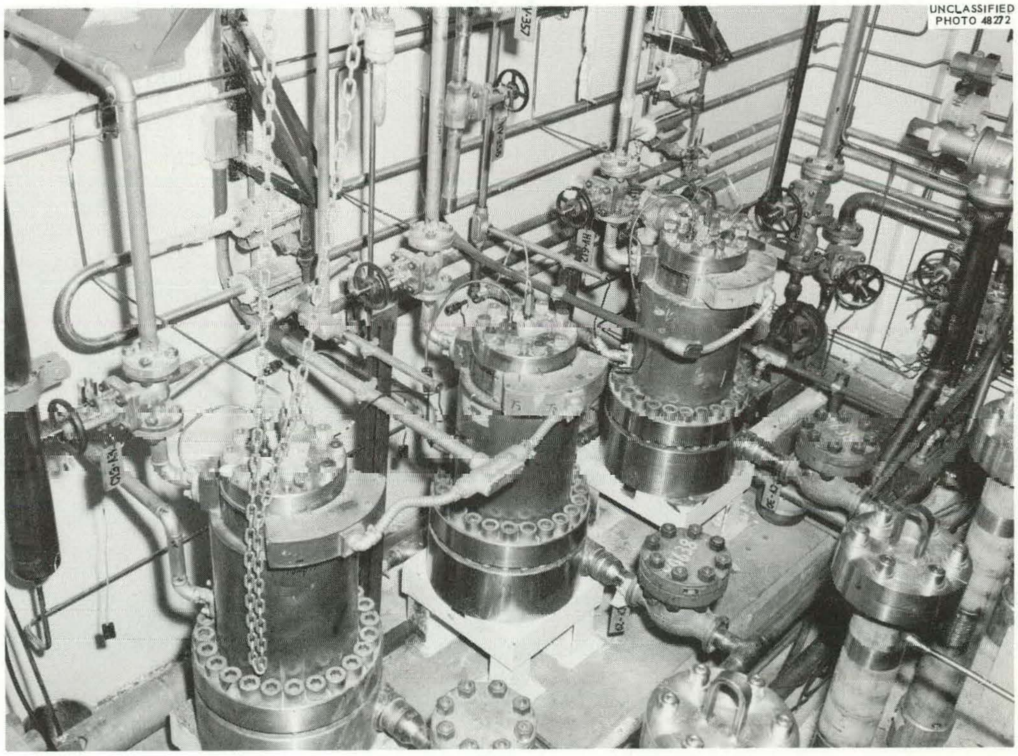

Fig. 22. Main Circulating Pumps.

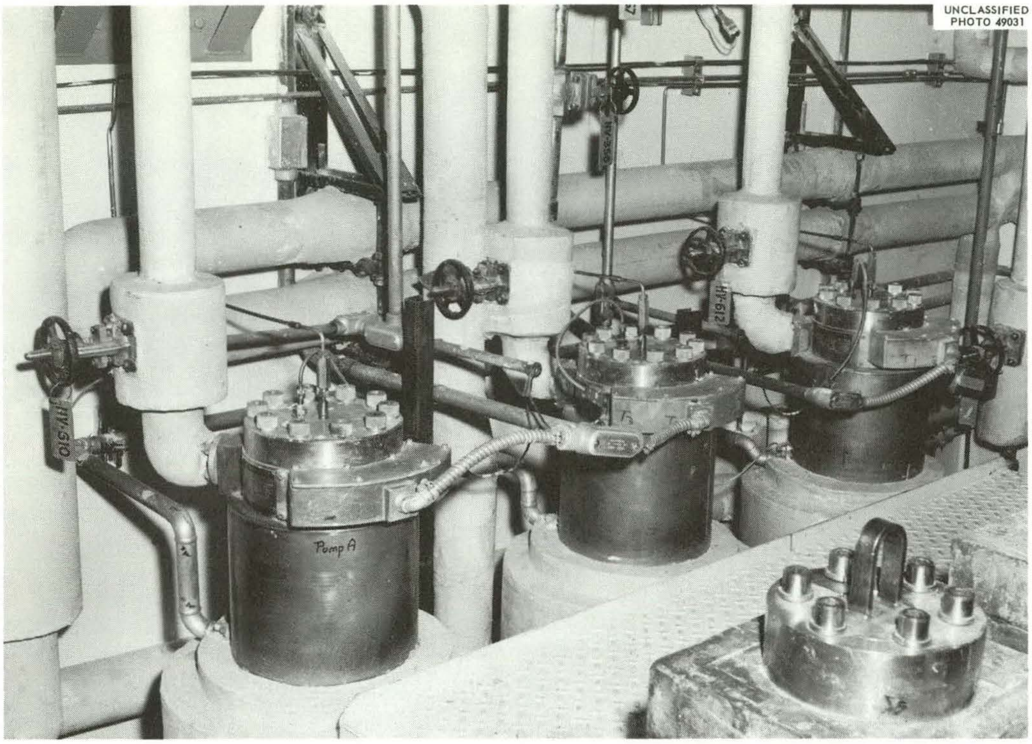

Fig. 23. Main Circulating Pumps with Installation Complete.

taken from the reactor pool and is supplied by the system used for cooling the main heat exchanger. Each pump will provide the design flow rate of 80 gpm (see Fig. 24).

Line Heaters

The line heater consists of a section of 2-in. pipe machined to a close tolerance on the exterior, with clamshell-type electrical heaters clamped to its surface. The total capacity of the three groups of line heaters is 60 kw. Two groups produce 12 kw each and are manually controlled by on-off switches; a third produces 36 kw and is manually controlled by a pneumatic-positioner-operated Variac.

The heaters are semicylindrical and consist of tubular electrical heaters embedded in iron castings. Each semicylindrical section has a heating capacity of 3 kw. The inner cylindrical surfaces were machined for close contact with the outer pipe surface.

Heat is added to the loop by these heaters in controlled amounts to establish and maintain the desired recirculating water temperature.

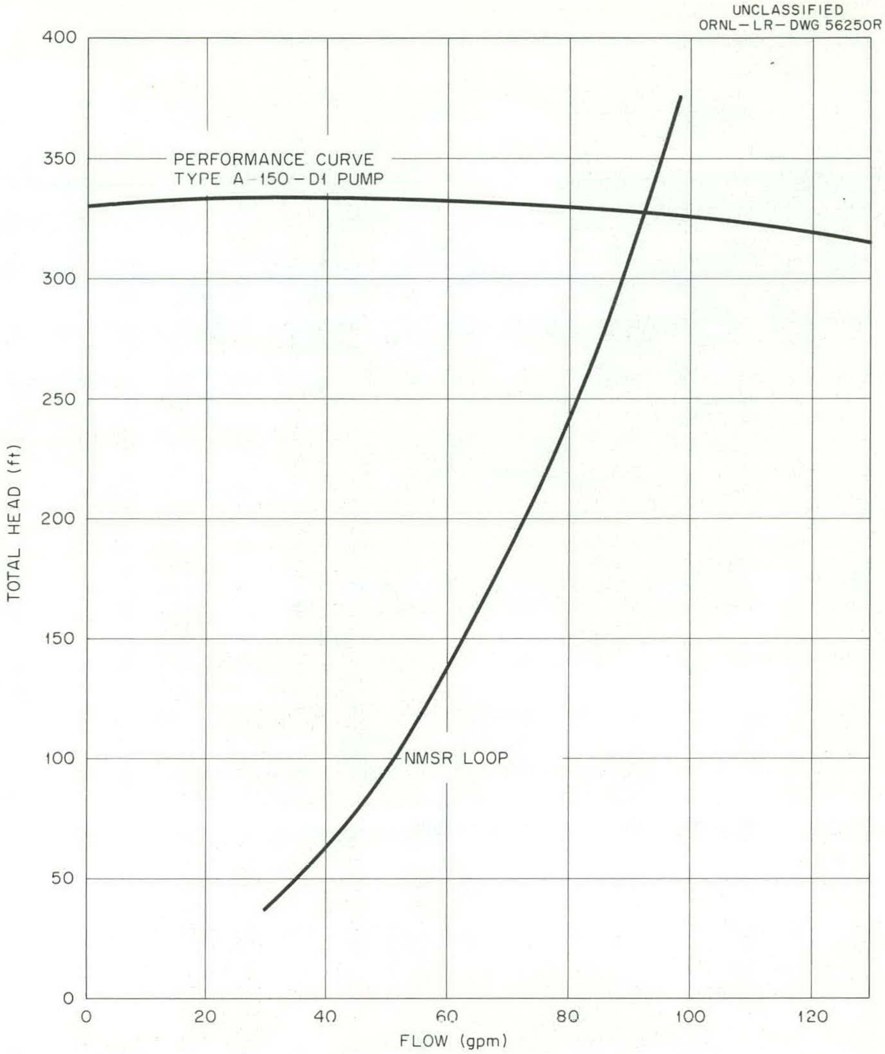

Fig. 24. Total Pressure Drop vs Flow.

Surge Tank

The surge tank is required to absorb volume changes during heating or cooling of the system and also to provide and maintain the desired system pressure (see Fig. 25). The volume of the surge tank is adequate to accommodate changes in system water volume from contraction (such as that encountered during emergency cooling of the entire system from normal operating conditions). The surge-tank heaters are clamped to four pipe legs which extend from the sides and re-enter the tank at the bottom. The heaters are of the clamshell type, identical to the line heaters, and have a total capacity of 24 kw.

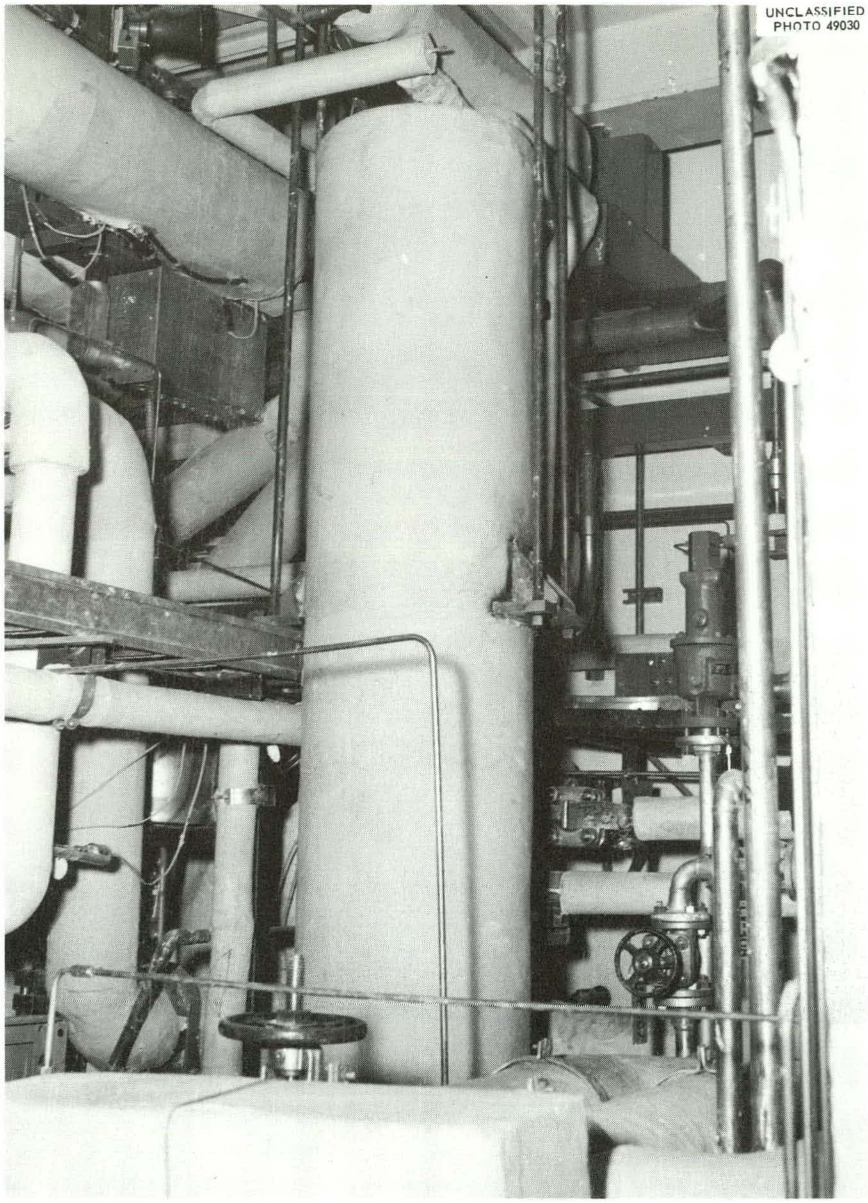

Fig. 25. Surge Tank and Supports.

Purification System

An ion exchange type of purification system in a small parallel flow circuit is provided for control of water purity in the loop (see Figs. 3 and 26). A portion of the recirculating loop water is diverted from the system and passes through a regenerative-type heat exchanger and a water-cooled heat exchanger before entering the ion exchange columns. This water must be cooled to approximately 95°F in order to prevent damage to the ion

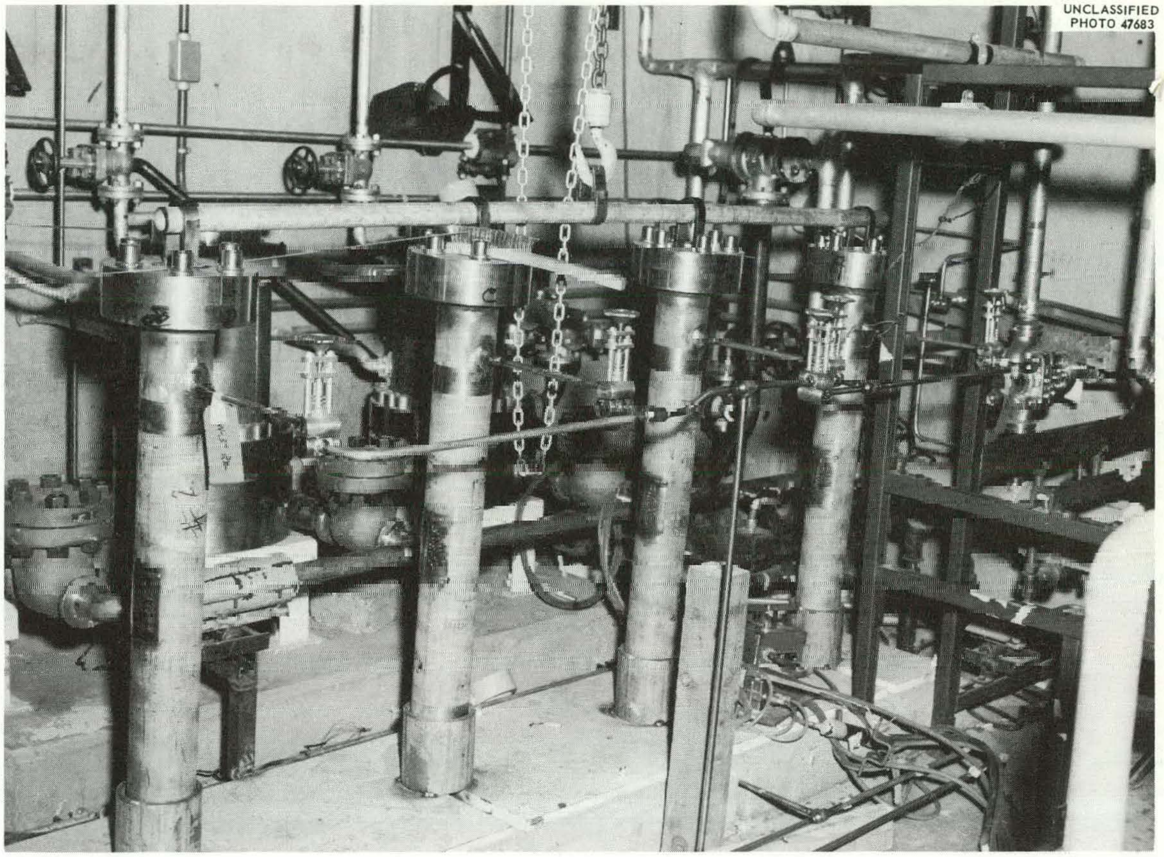

Fig. 26. Purification-System Ion Exchange Columns.

exchange resin. Water leaving the ion exchange columns passes through the regenerative heat exchanger to regain some heat before being returned to the loop. Conductivity cells are located in the lines before and after the ion exchange columns to measure their efficiency. These cells are Industrial Instrument Corporation Model CED-300 with a constant of 0.1.

Filtration of solids from the loop water is also a function of the ion exchange columns. Induced activity in these solids causes a buildup of activity in the columns as the solids accumulate. Shielding is provided by 4 in. of lead brick stacked around each column plus 4 in. of lead within the top of each column.

The ion exchange columns were charged with a mixture of Rohm and Haas Company nuclear grade Amberlite resins No. Xe-77 and Xe-78. These resins have performed very well and have not required changing. The mixed resin is contained in removable cartridges, and its regeneration is not intended.

A special transport cask is provided for cartridge removal and transfer to the burial grounds.

Particle Collectors

In addition to the ion exchange system, there are two full-flow strainers to collect foreign particles. One strainer, located at the inlet to one of the pumps, is always used at startup for collection of extraneous particles. Another strainer is located in the line downstream from the pump discharge header for protection of the in-pile section.

Degasifier System

A bypass degasifier system is provided for removal of dissolved gases from the loop water. This system is used mainly during startup and after additions of large quantities of makeup water. A portion of the water is taken from the loop and fed through a degasifier nozzle located in the vapor phase of the surge tank. Gas is released by vaporization of the small water droplets on exposure to the high temperature steam. Degassed water returns to the circulating system through the pressure-equalizing line which connects the surge tank to the system. Periodically the accumulated gases are vented from the surge tank.

This system can be utilized for rapid reduction of loop pressure by its cooling action on the water within the surge tank.

Makeup System

The makeup system provides purified water for loop filling and replacement of losses during operation. Demineralized water, furnished by the ORR utilities system, is used for makeup. This water enters the makeup system through ion exchange columns which contain a sulfite deoxygenating resin and a mixed-bed ion exchange resin. The condition of the resins is monitored by a conductivity cell on the outlet of the columns and by periodic water analysis. The water is stored in the makeup tank, where it is heated to 300°F and degassed by boiling off about 10% of its volume. The makeup tank has sufficient capacity to fill the entire loop with degassed water.

Makeup water is injected into the system by either or both of two Milton Roy Company 18-8 stainless steel Simplex controlled-volume MSI-53-74-5 pumps. These pumps are also used for hydrostatic pressure testing of the loop (see Fig. 27). Only one pump had been installed when the photograph was made.

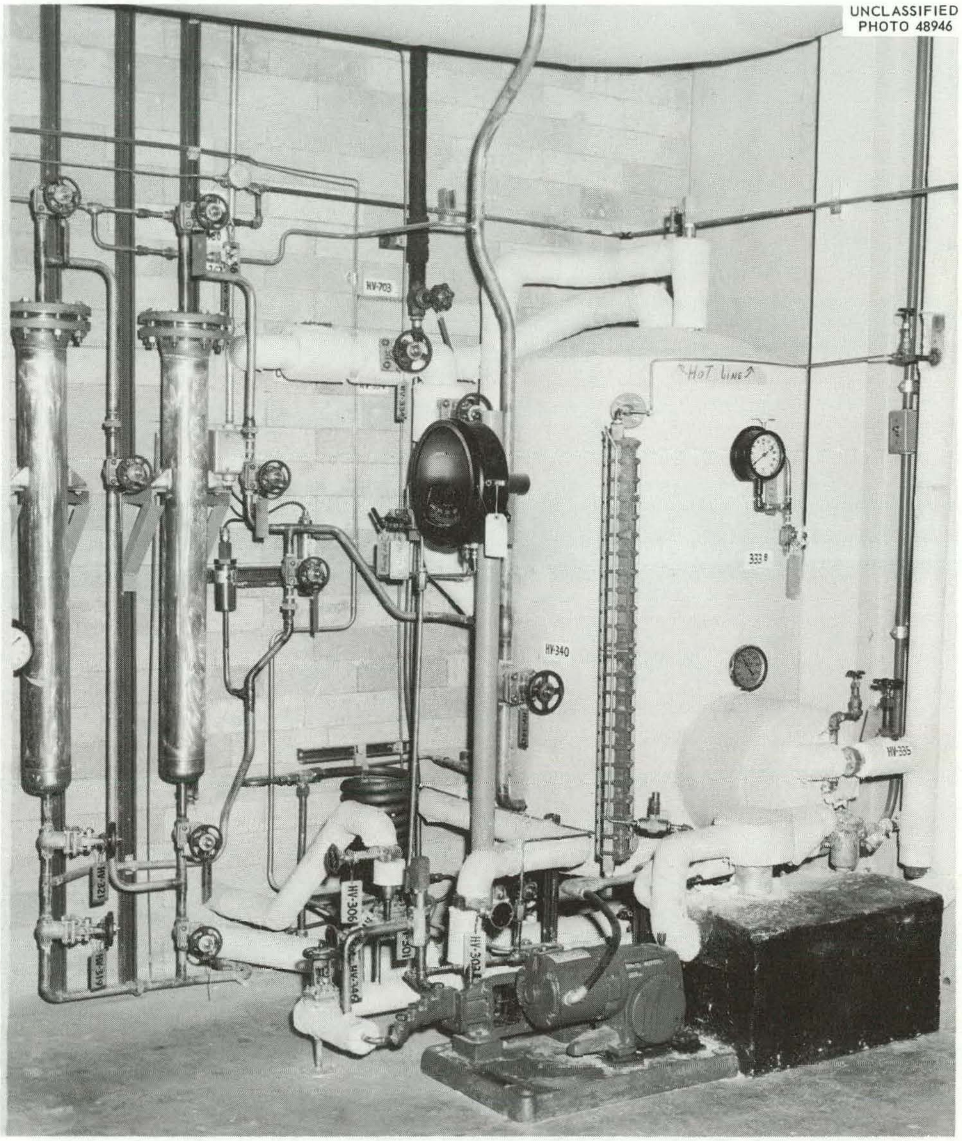

Fig. 27. View of Makeup System.

Steam from the makeup tank may be used for flushing the loop before filling with water. This operation minimizes loop venting and degassing operations. A compressed air supply is provided for blow down of the loop during draining operations. These two features have had very little use since operation of the loop began.

Sampling System

The sample station provides a means for obtaining water samples during operation from various points in the system such as the main pump inlet manifold, in-pile section outlet, surge tank, ion exchange columns, etc.

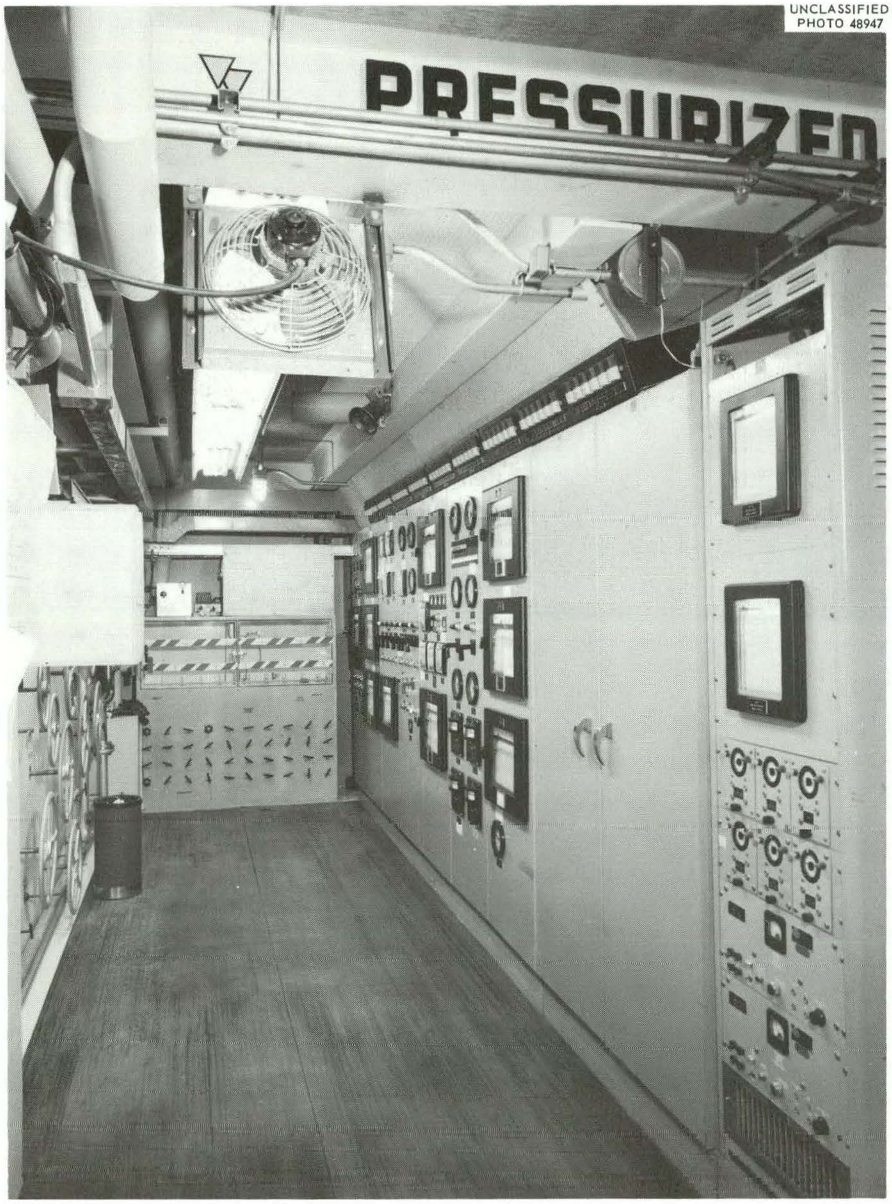

Fig. 28. Operating Area — Sample Station in Background.

(see Figs. 2–5). All of the normal sample lines which can contain water over 200°F run through a small cooler before entering the sample station. If high-pressure and high-temperature samples are desired, the sample cooler must be deactivated and the sample taken in a special high-pressure sample bomb. One special high temperature sample line has been added for filtration experiments. The high-pressure bombs are also used for the addition of special materials for controlling the loop water chemistry. Water is circulated through the bomb by the pressure difference obtained between the sample points and the pump suction. The bomb can be replaced by experimental devices for water chemistry studies.

The entire sample station is enclosed, and air which sweeps through the enclosure is drawn into the equipment-room exhaust duct (see Fig. 28).

Loop Reactor Cooling Water System

The loop reactor cooling water system is provided for use in the event that recirculation of the loop water cannot be maintained. Reactor cooling water is contained in a large recirculation system used principally for cooling the reactor core and other reactor parts. The loop reactor cooling water system takes this water from the upstream side of the reactor, passes it through a booster pump, the loop in-pile test section and flow meter, and returns it downstream of the reactor. It is necessary to shut down the reactor and isolate the loop in-pile test section from the rest of the loop before switching over this mode of operation. The loop must be at low temperature and pressure when operating with this system. The loop reactor cooling water system flow instrumentation can, on low flow, switch operation from the operating booster pump to the standby booster pump. If this fails to restore flow, a reactor scram is initiated.

Instrumentation and Controls

Instrumentation for the loop is designed with three specific objectives: (1) protection of the reactor and loop from damage at all times, (2) as fully automatic control of the loop as is practical, and (3) indication or recording of all necessary data. The instrument panel contains instruments for the control, measurement, and monitoring of pressure, temperature, flow, surge-tank water level, dump-tank water level, pump power, water conductivity, heater current, battery voltage, and radiation intensity at various locations. This panel is arranged for convenient operation of the valve extension handles during observation of associated instruments (Figs. 28 and 29).

Controls for the battery-powered motor-generator set are separate from those of the loop. The motor-generator instruments and controls are on a panel located directly behind the loop instrument panel.

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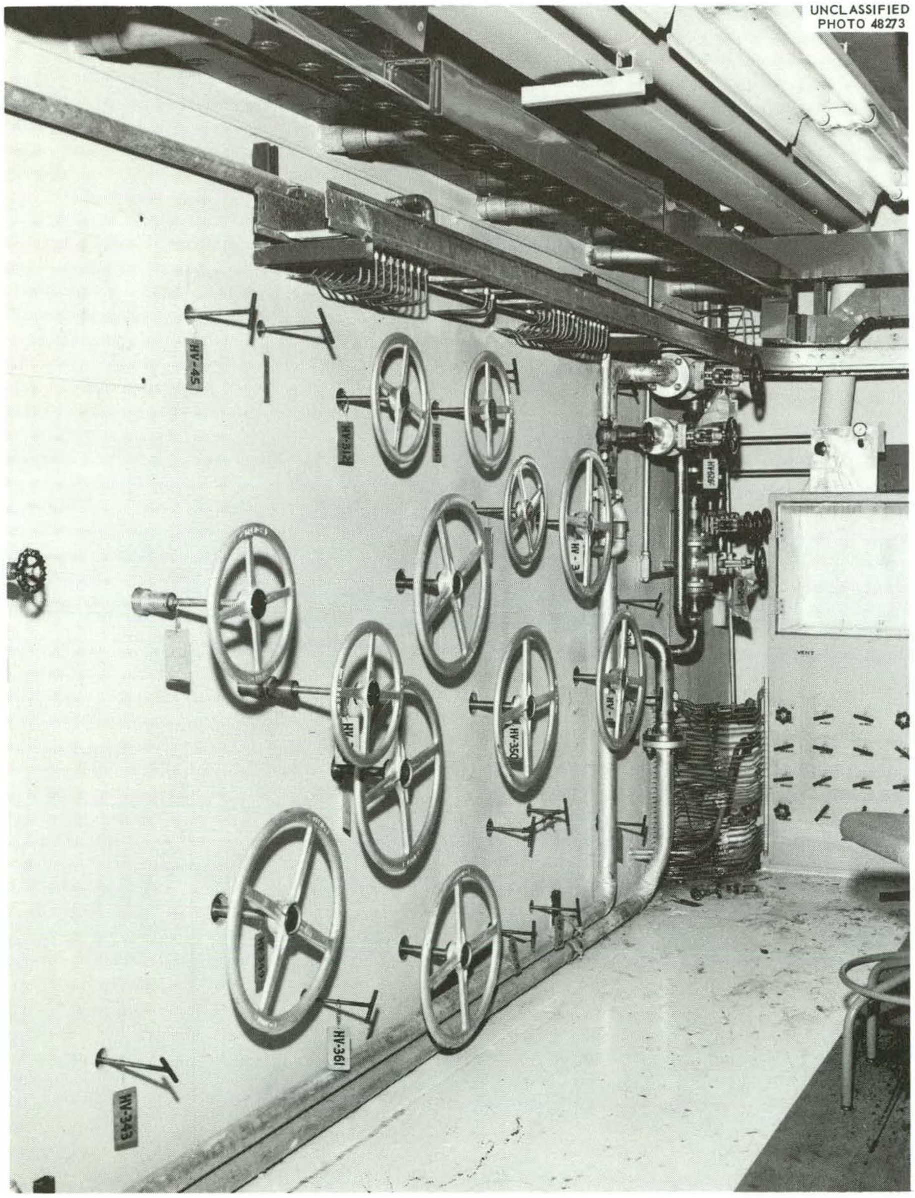

Fig. 29. View of Valve Extension Handles.

Pressure Control

The loop pressure is maintained and controlled by steam pressure within the surge tank. This head of steam is formed by the electrical heaters strapped to pipe legs on the bottom of the surge tank. Pressure is sensed by a Foxboro pressure transmitter and is automatically maintained by a Foxboro pressure controller which varies the power to the heaters by a pneumatic-positioner-operated Variac. Current to the heaters is measured by ammeters. Pressures in the surge tank, in-pile section, and pump discharge manifold are monitored by pressure transmitters. The in-pile pressure monitoring system includes dual transmitters and reactor shutdown safety circuits.

Pressure gages monitor the pressure of the process water, demineralized water, flushing steam, and makeup water. There are differential pressure (d/p) indicators and recorders on the panel board with which the pressure drop across the pumps, in-pile section, out-of-pile section, and surge-tank water level are measured.

Temperature Control

Loop water temperature is controlled by an automatically operated by-pass around the water-cooled heat exchanger and by manually operated electric heaters mounted on a section of the loop piping. The three-way blending valve in the by-pass line is pneumatically operated and controlled by a Brown Air-O-Line controller. For this control, the temperature is sensed by a thermocouple located immediately downstream from the junction of the heat exchanger outlet and bypass line. Loop water temperatures also are measured at the in-pile inlet and outlet, heat exchanger inlet and outlet, pump inlet and discharge manifolds, and at the outlet of the line-heater section. The in-pile inlet and outlet temperatures are tied into the reactor safety circuits, and overtemperature will produce a reactor setback.

Ten thermocouples are located on the outer surface of the in-pile pressure tube. The two thermocouples having the highest readings are tied into the reactor safety circuit. Overtemperature indication by either will produce a reactor setback.

Thermocouples are mounted on the pipe surface beneath each line heater and overtemperature of any one will activate a switch to turn off all line heater power.

Flow Control and Measurement

Circulation of water must be maintained in the loop at all times when the reactor is operating to cool the test specimens and in-pile tube. In order to maintain the required flow with adequate safety, three pumps are installed in parallel, with one operating and the other two in standby status. Flow is controlled by a hand-operated flow control valve, measured with a Venturi, and recorded. Duplicate d/p cells for measuring the head across the Venturi are required for safety. The d/p cells were made by Moore Products Co. and the two pen recorder was made by the Foxboro Co. Two pressure switches are operated from each of the flow signals emitted by the d/p cells. One pressure switch produces an alarm and the other initiates transfer to the first standby pump followed by a reactor scram if flow is not recovered within 2 sec. Operation of the standby pumps is further controlled by an automatic rotary switching system as described in Appendix C.

When operation is with reactor cooling water, the flow is measured by an orifice plate and two d/p cells. For safety, the system of reactor cooling-water flow measurement, signal transmission, recording, and pressure switches for the scram circuit is similar to that of the main water flow system except for lower pressure ratings of the d/p cells. Two loop reactor cooling-water pumps are installed in parallel, one serves in standby.

Flow rates of cooling-water for the main loop pumps and heat exchanger are measured by rotameters.

Electrical System

The power supply for portions of the electrical system is arranged to provide power from a battery-powered motor-generator set and the ORR emergency diesel generator set in addition to that from the normal distribution system. Loss of the normal power supply will cause an automatic switchover to the battery-powered supply. After the ORR diesel generator is started and up to the proper speed, an automatic switchover is made to continuously charge the batteries. By this arrangement the instruments and controls, main circulating pumps, cooling-water pumps, surge-tank heaters, and loop reactor cooling-water pumps are assured of a continuous power supply.

Surge-Tank Liquid Level

The surge-tank liquid level is indicated and recorded on a two-pen Foxboro recorder. The signal to the recorder is received from two Moore d/p cells. The high head side of each d/p cell is connected to the top of the surge tank through a standpipe which serves as the reference leg. The reference leg is air-cooled and maintained full at all times to provide a constant water level and density. Pressure to the low head side of the d/p cells comes from the bottom of the surge tank and varies with the level of the water in the tank. Corrections are made in the level readings to compensate for density differences between the reference and measuring legs.

Radioactivity Monitoring System

Radioactivity is monitored inside the equipment-cubicle ventilation duct, vacuum header to pool piping and in-pile section, sample station, and operating control area. Fission break monitors are installed on the ion exchange column outlet, pump discharge manifold, and reactor cooling-water line from the in-pile test section.

Activity in the loop control-panel area is monitored by a Monitron (vibrating condenser electrometer with a large ionization chamber) which produces an alarm at activities above 2 mr/hr. Another Monitron is mounted in the equipment cubicle, and activities above 7 mr/hr are indicated by a red light located at the equipment-cubicle entrance.

Activity at the sample station is monitored by a G-M tube which actuates an alarm when the activity exceeds 10 mr/hr above the normal background. The equipment-cubicle exhaust duct is monitored by a G-M tube which will activate an alarm at 2 mr/hr above normal background. A G-M tube is also mounted on the vacuum header from the in-pile section and in-pool piping vacuum jackets. It is set to activate an alarm at 2 mr/hr above normal background.

Holdup tanks are provided for N^{16} decay on the pump discharge header system and reactor cooling-water return line. The ion exchange columns provide the necessary delay time for N^{16} decay in the other system. G-M tubes are used in each system and are set to activate alarms at activities of 20 mr/hr above normal background.

No automatic loop controls or reactor safety circuits are associated with the radiation monitoring equipment. All radiation monitors are fed to indicators equipped with alarms. Both fission break monitors on the loop proper are read on recorders, and the fission break monitor on the reactor water line is read on an indicator.

Safety Features

Annunciator and Automatic Control Circuits

Automatic control circuits initiate corrective actions to rectify critical abnormalities when they occur and, if necessary, initiate reactor scram or setback actions. The abnormal conditions which will cause annunciation and corrective actions are as follows:

1. Low Main Loop Water Flow. — When flow is below normal for more than 2 sec the reactor is scrammed and the heat exchanger turned on to full capacity (see "Main Loop Pumps — Appendix C").

2. Low Pump Cooling-Water Flow on Two of Three Main Pumps. — When flow to two of three pumps is below normal for more than 3 sec the operating cooling-water pump is automatically stopped, the standby cooling water pump started, the reactor is set back, the heat exchanger turned on to full capacity, and all line heaters turned off (see "Cooling-Water Pumps — Appendix C").

3. Approaching Steam Saturation Conditions. — This occurs when the loop water conditions approach saturation during a decrease in loop pressure or increase in loop temperature. Setpoints are established for specific experiments before reactor startup.

(a) High water temperature at inlet or outlet of in-pile test section will each produce a reactor setback.

(b) Low pressure will produce a reactor scram.

4. High In-Pile Tube Temperature. — When the temperature of the in-pile tube exceeds a preset point, the reactor is set back.

5. Low Loop Reactor Water System Flow. — This station is operative only when the in-pile tube is being cooled by reactor water. Should low loop reactor water flow exist for more than 2 sec the reactor will be scrammed (see "Loop Reactor Water System Booster Pumps — Appendix C").

6. High Ion-Column Temperature. — Alarm only.

7. High System Pressure. — When excessively high pressure exists, the surge tank heaters are automatically cut off. They are automatically turned on again when the pressure returns to normal. A reactor setback is initiated if the pressure reaches a higher limit.

8. Surge-Tank Water Level Control. — The makeup pump is turned on and off automatically to maintain the water level. If it drops below the desired operating level, the surge-tank heaters are automatically turned off. At a high surge-tank level the makeup-pump control circuit is broken to prevent addition of water to the system.

9. Main Loop Pump Overload. — An indication of electrical overload on the operating pump will produce an alarm and transfer operation to a standby pump.

10. High loop Water Activity. — Alarm only.

Relief Valves

There are six relief valves in the system. They are located on (1) the surge tank, for protection in the event of a failure of the automatic pressure control circuit, (2) the in-pile tube, for protection in the event this portion of the system is isolated from the remainder of the system and there is sufficient heat generation in the in-pile tube to cause excessive pressure, (3) the makeup tank, for protection against overpressurization by steam, (4) the makeup pump, to prevent excessive pressure in the pump if operated while the discharge valves are closed, and (5) two reactor cooling-water lines, to prevent overpressurization of these lines in case of misoperation of the main loop block valves.

The pressure relief valves on the surge tank and the in-pile section are installed downstream from rupture disks. These disks are to prevent leakage through the relief valves during operation at normal pressures.

A pressure switch with an overpressure alarm is tied into the system between each rupture disk and relief valve.

Pressure relief valves are installed in the cooling-water system to provide individual protection for the main-loop heat exchanger and for each main pump.

Radiation Protection

Loop-operating personnel are protected against radiation from the water activity by the inclusion of all loop equipment within a cubicle which has walls with the equivalent shielding of barytes concrete 2 ft thick. Piping within the pool is shielded by 12 ft of water, and the piping which passes beneath the basement floor is shielded by two steel plates, each 1 in. thick, plus 6 in. of lead brick. See Fig. 30 for the pipe trench leading to the equipment room.

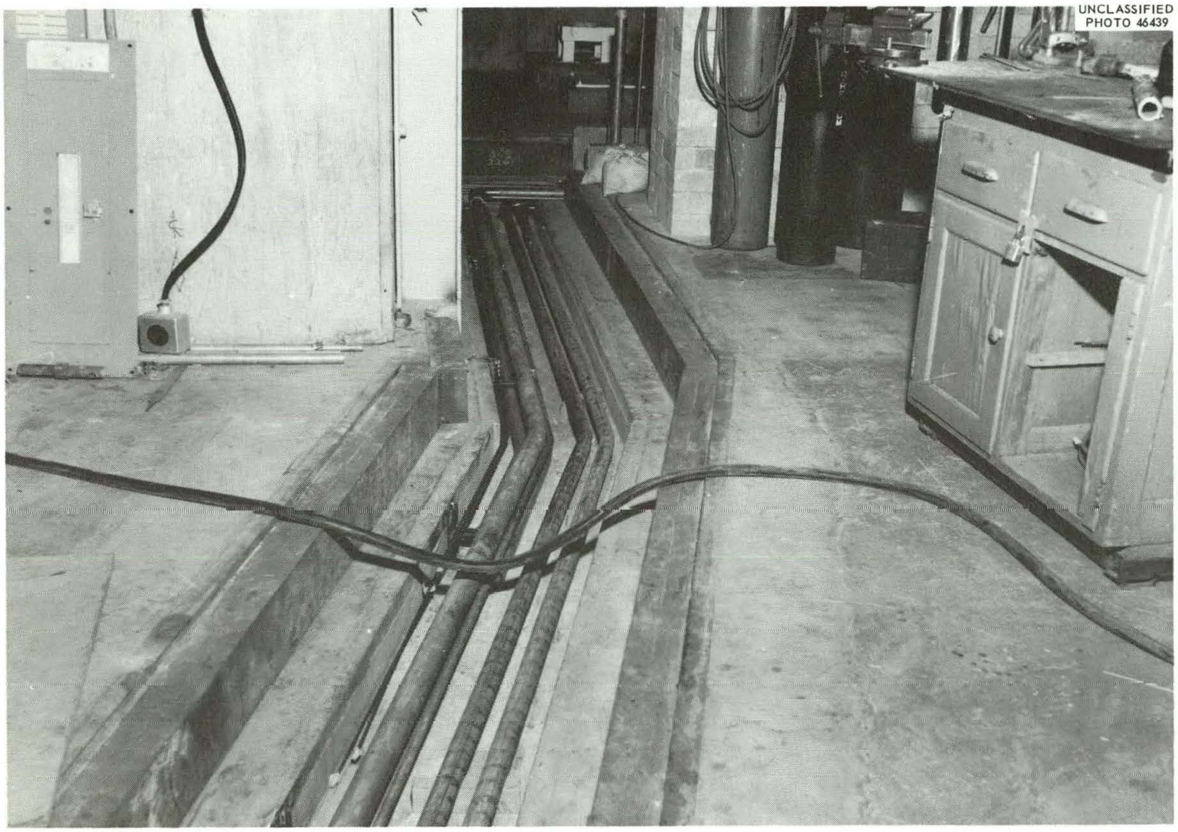

Fig. 30. Pipe Trench Facing Pressurized-Water Loop Equipment Cubicle.

The shielded equipment cubicle is vented to the reactor-stack exhaust system through a 15-in. duct. Air flows from the equipment room through the duct at all times at a rate of 1000 cfm. A 5-in. duct is provided to exhaust air from the sample station.

Test-Specimen Charging and Discharging

When a test specimen is removed from the in-pile test section, that portion of the loop is isolated from the rest of the system by block valves. The breech lock closure locking device is removed from the top of the in-pile section, and a remote-handling tool is attached to one of the O-ring-sealed closures. Test specimens which are attached to the closure by a specimen holder are withdrawn from the in-pile section by disengaging the closure breech lock and moving the assembly upward.

Shielding for this operation is provided by the pool water. Specimens may be stored in the pool until ready for inspection within the hot cell.

Out-of-pile specimens may be withdrawn in a similar manner, except that they should be drawn into a container such as a length of pipe to prevent the spread of contamination. The only activity of these specimens is that due to deposition of radioactive corrosion products from the system.

OPERATIONAL EXPERIENCE

The loop has been in operation since December 1959, with only minor modifications. A fission product monitor was added to the reactor cooling-water system, a flow element was installed in the main-loop fission product monitoring system, a standby makeup pump was installed, and the battery-powered emergency motor-generator circuits were reworked for increased reliability.

During the period that the loop has been in operation, the most objectionable difficulty has been the occasional malfunctioning of electronically operated temperature-monitoring safety instruments. Reactor shutdowns followed because all safety controls are designed for fail-safe operation. The frequency of these malfunctions has been greatly reduced by exercising special preventative maintenance procedures and modifying some components within the instruments.

The line heaters, surge-tank heaters, main circulating pumps, cooling-water pumps, loop reactor-water pumps, vacuum pump, and heat exchanger have all operated satisfactorily. All valves have proven adequate and reliable. The three main circulating pumps have accumulated a total of more than 15,000 hours without incident. The loop leakage is practically zero, and water makeup is required only for replacement of the amounts removed by sampling.

Instrumentation and Controls

Instrumentation of the loop has proven to be adequate and generally reliable, with one exception. The main-loop fission product monitoring system, as originally designed, was dependent upon N^{16} activity-level changes for flow indication. Crud buildup in the system piping at the gamma monitor produced sufficient background to mask these changes. A flowmeter has since been installed to indicate flow through the system.

The automatic control features designed into the loop have operated very satisfactorily, with only routine maintenance being performed. The automatic-control sensitivity of the heat exchanger bypass blending valve has been the one requiring the most attention. This is attributed to the fact that the blending valve is nearly closed during operation of the loop at low power. The valve was originally sized in favor of minimum pressure drop for emergency cooling rather than optimum temperature-control sensitivity.

Radiation and Contamination Levels

Radiation levels within the equipment cubicle and at the sample station have been exceedingly low. During normal operation the equipment cubicle activity is about 50 mr/hr. Contamination within the equipment cubicle and at the sample station has also been exceedingly low. This is attributed to the excellent leak-tightness of the system. Generally, during reactor shutdowns, the aisles of the equipment cubicle may be entered without special clothing. The low contamination level at the sample station is attributed to good sampling techniques and measures taken to prevent the spread of contamination.

Experiments

All experiments performed to date have been in support of the Maritime Reactors Program. These experiments involved both swaged and vibratory compacted UO_2 fuel clad in type 304 stainless steel and were irradiated at nominal NS "Savannah" reactor operating conditions of 500°F and 1750 psig. Charpy impact specimens of the NS "Savannah" reactor vessel steel have also been irradiated in this test loop.

The procedure for charging and discharging test specimens from the loop, described above ("Test-Specimen Charging and Discharging"), has proven to be very satisfactory. A special containment vessel and associated remote tools and equipment for use in the removal of ruptured fuel specimens have been designed and fabricated since the loop first went into operation. There has been no occasion for their use.

Water Chemistry

The sample station and the sampling techniques employed have proven satisfactory. Maintenance of water chemistry at the desired levels is practical, and the analytical techniques are adequate, with one exception. Determination of oxygen content is being done by the Winkler method, with 0.05 ppm as the lower detection limit. The target oxygen level in the loop has been reduced from 0.05 to < 0.01 ppm for some experiments. Therefore, a more sensitive method of oxygen determination is necessary. An oxygen analyzer capable of determinations at the level of one part per billion has been ordered. The new analyzer will consist of a flowmeter, an inlet ion exchange column, a conductivity cell, a thallium column, a second conductivity cell, and a cleanup ion exchange column. Dissolved oxygen in the water will combine with the thallium to form thallous hydroxide, which is a strong electrolyte. Conductivity of the water will be measured at the inlet and outlet of the thallium column. The difference between these two conductivity measurements will determine the thallous hydroxide concentration and, in turn, the oxygen concentration in the water. The inlet ion exchange column will lower the background conductivity of the water, and the outlet ion exchange column will remove the thallous hydroxide from the water before it returns to the main loop. The new analyzer will be installed in the loop purification system as a side steam. This will provide a more precise and continuous analysis of oxygen concentration of the loop.

Special water-chemistry studies are being made to determine the specific activities and the chemical compositions of filterable and nonfilterable impurities in the loop water. From these studies it is hoped that an improved filter and/or ion exchanger can be developed for the removal of low-level impurities. Water conductivity measurements presently show ~0.015 micromho/cm at 37°C, indicating very low concentrations of ionic impurities. This is a particularly advantageous feature of the test loop when used for water-chemistry experiments.

ACKNOWLEDGMENTS

A. L. Boch, H. C. McCurdy, and H. W. Savage contributed extensively to the establishment of criterion and conceptual design. The following people assisted in the mechanical design and construction: A. A. Abbatiello, C. B. Brown, P. A. Gnadt, A. P. Marquardt, and G. W. Renfro. Design, construction, and installation of the instrumentation and controls were by R. F. Hyland and G. W. Greene. Portions of the design calculations were made by J. A. Conlin, W. S. Harris, and J. W. Miller. Stress calculations were made by S. E. Moore, and the loop-water-activity and fission break monitor design calculations in Appendixes A and B were made by H. N. Culver and F. E. Gillespie. D. W. Cardwell maintained a program of cost evaluation and schedule status through the construction phase which contributed substantially to completion of the loop as scheduled and within the estimated cost.

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Appendix A

RADIOACTIVITY IN WATER

Radioactivity Due to Clad Defect

Activity may be introduced into the water loop as a result of small defects in the cladding of the fuel element. An estimate of the amount of activity may be made by using Eq. (1). This equation uses an escape rate coefficient, ν, which is based on a 5-mil-diam defect in the cladding material. The values of ν used in Eq. (1) are based on data from Westinghouse experiments with UO_2:[6]

$$A_L = \frac{(3 \times 10^{10})\ \gamma \nu P (10^6)\ k}{(3.7 \times 10^{10})\ \lambda V_T} = \frac{8.12 \times 10^5\ \gamma \nu P k}{\lambda V_T}\ \mu c/cm^3\ , \qquad (1)$$

where

A_L = specific activity in the loop, $\mu c/cm^3$,
γ = fission yield, atoms/fission,
ν = escape rate, sec^{-1},
λ = decay constant, sec^{-1},
P = power generation per rod, w,
V_T = system volume, cm^3,
k = time correction factor = $(1 - e^{-\lambda t})$.

An estimate of any of the fission products may be made by using Eq. (1). A tabulation of the important gamma-emitting nuclides is given in Table A.1. This list includes all the fission products having gamma energies of 0.5 Mev or greater and fission yields of 0.1% or greater. The activities listed in Table A.1 are for a power generation of 12,800 w/rod, which is the maximum case. Since most of the fission products in the table are short-lived, the activities for six-month and one-year operation will be the same for all the nuclides except Y^{90}, Y^{91}, Zr^{95}, Nb^{95}, and Ba^{137}.

Activity in Water Due to Corrosion-Product Activation

The background activity due to the corrosion of activated nuclides in the core may be estimated from the following expression:

$$N_b^W = \frac{CA_c\ N_a^c\ \sigma_a\ \phi}{V_T\left(\lambda_b + \frac{CA_c}{W_c}\right)\left(\lambda_b + \frac{\alpha}{T}\right)}\ . \qquad (2)$$

Table A.1. Specific Activity in Water Due to a Clad Defect*

Nuclide	Half-Life	Specific Activity, 6-month Exposure ($\mu c/cm^3$)	Specific Activity, 12-month Exposure ($\mu c/cm^3$)
Br^{84}	30m	1.77×10^{-2}	
Br^{87}	55.6s	1.33×10^{-3}	
Kr^{87}	78m	1.84×10^{0}	
Kr^{88}	2.77h	5.49×10^{0}	
Rb^{88}	17.8m	5.88×10^{-1}	
Rb^{89}	15.4m	6.60×10^{-1}	
Y^{90}	64.5h	2.86×10^{-5}	5.3×10^{-5}
Sr^{91}	9.7h	2.13×10^{-1}	
Y^{91m}	51m	2.20×10^{-5}	
Y^{91}	58d	7.93×10^{-2}	8.81×10^{-2}
Y^{92}	3.6h	2.35×10^{-4}	
Y^{93}	10h	6.95×10^{-4}	
Y^{94}	16.5m	1.92×10^{-5}	
Zr^{95}	63d	4.46×10^{-1}	5.19×10^{-1}
Nb^{95}	35d	8.39×10^{0}	1.15×10^{1}
Te^{129}	72m	1.94×10^{-2}	
Te^{131}	24.8m	1.92×10^{-2}	
I^{131}	8.05d	1.20×10^{2}	
I^{132}	2.4h	2.26×10^{0}	
Te^{133}	2m	3.21×10^{-3}	
I^{133}	20.8h	2.87×10^{1}	
I^{134}	52.5m	1.42×10^{0}	
I^{135}	6.68h	8.39×10^{0}	
Xe^{135m}	15.6m	1.75×10^{-1}	
Xe^{135}	9.13h	2.12×10^{1}	
I^{136}	86s	1.59×10^{-2}	
Ba^{137m}	2.6m	3.30×10^{-6}	6.35×10^{-6}
Cs^{138}	32m	1.65×10^{0}	
Ba^{139}	85m	9.09×10^{-3}	
Ba^{140}	12.8d	2.07×10^{0}	
La^{140}	40.2h	2.72×10^{-1}	
La^{141}	3.7h	1.93×10^{-3}	
La^{142}	74m	7.79×10^{-3}	
Ce^{143}	32h	1.06×10^{-2}	

*Activities are for a power generation of 12,800 w/rod.

Since $\lambda_b > CA_c/W_c$,

$$N_b^W = \frac{CA_c \; N_a^c \; \sigma_a \; \phi}{V_T \; \lambda_b \left(\lambda_b + \frac{\alpha}{T}\right)} \; . \tag{3}$$

Therefore the water activity is expressed as

$$A_b^W = \frac{CA_c \; N_a^c \; \sigma_a \; \phi \; (10^6)}{V_T \left(\lambda_b + \frac{\alpha}{T}\right)(3.7 \times 10^{10})} \; \mu c/cm^3 \; . \tag{4}$$

Since the activities in the NMSR have been computed according to this equation, the activities of the various nuclides in the water loop may be approximated by the following:

$$\frac{(A_b^W)_L}{(A_b^W)_S} = \frac{\left[\dfrac{CA_c \; N_a^c \; \sigma_a \; \phi}{V_T \left(\lambda_b + \frac{\alpha}{T}\right)}\right]_L}{\left[\dfrac{CA_c \; N_a^c \; \sigma_a \; \phi}{V_T \left(\lambda_b + \frac{\alpha}{T}\right)}\right]_S} \; . \tag{5}$$

A comparison of the parameters for the NMSR and the water loop is shown in Table A.2. Equation (5) becomes

$$\frac{(A_b^W)_L}{(A_b^W)_S} = \frac{\dfrac{(1.42 \times 10^3)(5 \times 10^{13})}{10^5 \; (\lambda_b + \alpha/T)_L}}{\dfrac{(4.45 \times 10^6)(8 \times 10^{12})}{(3.38 \times 10^7)(\lambda_b + \alpha/T)_S}} = 0.677 \; \frac{\left(\lambda_b + \frac{\alpha}{T}\right)_S}{\left(\lambda_b + \frac{\alpha}{T}\right)_L} \; . \tag{6}$$

Since there is a material difference in the core, the activity of various nuclides may be expressed as

$$(A_b^W)_L = 0.677 \; (A_b^W)_S \; F \; \frac{\left(\lambda_b + \frac{\alpha}{T}\right)_S}{\left(\lambda_b + \frac{\alpha}{T}\right)_L} \; , \tag{7}$$

where F is the ratio of parent nuclides in the different stainless steels. The resulting activities are shown in Table A.3. The use of type 316 stainless steel results in the additional activities due to Mo^{99} and Mo^{101}. These are estimated from Eq. (5) as

$$A_b^w = \frac{(1.93 \times 10^{-11})(1.42 \times 10^3)\ N_a^c\ \sigma_a\ (5 \times 10^{13})\ 10^6}{(10^5)(3.7 \times 10^{10})(\lambda_b + 1.3 \times 10^{-4})}$$

$$= \frac{3.7 \times 10^{-4}\ N_a^c\ \sigma_a}{\lambda_b + 1.3 \times 10^{-4}} .$$

For Mo^{99},

$$A_b^w = \frac{(3.7 \times 10^{-4})(4.47 \times 10^{18})(4.5 \times 10^{-25})}{1.3 \times 10^{-4}} = 5.7 \times 10^{-6}\ \mu c/cm^3 ;$$

for Mo^{101},

$$A_b^w = \frac{(3.7 \times 10^{-4})(1.82 \times 10^{18})(2.0 \times 10^{-25})}{9.21 \times 10^{-4}} = 1.46 \times 10^{-7}\ \mu c/cm^3 .$$

Table A.2. Comparison of NMSR and NMSR-Loop Parameters

Parameter	Symbol	NMSR	Water Loop
System volume, cm^3	V_T	3.30×10^7	10^5
Surface area in-pile, cm^2	A_c	4.45×10^6	1.42×10^3
Average thermal-neutron flux	ϕ	8×10^{12}	$< 5 \times 10^{13}$
Purification rate	α/T	3.73×10^{-5}	$\sim 1.3 \times 10^{-4}$ ($\sim$0.5%)
Cycle time, sec	T	27.1	38
Clad material in core		304 SS	316 SS
Corrosion rate	C	5 mg c/m^{-2} $month^{-1}$ (7)	5 mg c/m^{-2} $month^{-1}$

Table A.3. Fission Product Activities in Loop Water Due to Corrosion

Nuclide	$(A_b^w)_{S\,\infty}$	F	$\dfrac{\left(\lambda_b + \frac{\alpha}{T}\right)_S}{\left(\lambda_b + \frac{\alpha}{T}\right)_L}$	$(A_b^w)_L$
Co^{60}	3.68×10^{-4}	1.0	0.287	7.15×10^{-5}
Fe^{55}	1.49×10^{-2}	0.94	0.287	2.68×10^{-3}
Fe^{59}	3.14×10^{-4}	0.94	0.287	5.74×10^{-3}
Cr^{51}	2.16×10^{-2}	0.9	0.287	3.77×10^{-3}
Ni^{65}	1.22×10^{-4}	1.27	0.55	5.77×10^{-5}
Mn^{56}	1.91×10^{-2}	1.0	0.545	7.05×10^{-3}
Mn^{54}	2.25×10^{-4}	0.94	0.287	4.1×10^{-5}
Co^{58}	1.26×10^{-3}	1.27	0.287	3.1×10^{-4}
Mo^{99}				5.7×10^{-6}
Mo^{101}				1.46×10^{-7}

Appendix B

FISSION BREAK MONITOR DESIGN

Halogen-filled Geiger counters are used to detect radioactivity in the water loop resulting from small fuel cladding defects. Because the loop is at high temperature, the halogen-filled Geiger tubes are cooled by flowing air around them. These tubes are used as integrated current detectors; when operated at 900 v, 10 mr/hr of Co^{60} gamma radiation corresponds to approximately 1 μa of current. The logarithmic-type count rate meter has a scale of 0.1 to 100 mr/hr.

The most important radioactive isotope having a long half-life and also a large fission yield is I^{131}. The activity of I^{131} is 120 μc/ml, which corresponds to 4.45×10^6 dis sec^{-1} ml^{-1}. The relative intensities of gamma rays in the spectrum of I^{131} are as follows: 80.1 kev (7.8), 284 kev (8.1), 364 kev (100), 514 kev (1.5), 637 kev (11.6), 722 kev (2.4). Eighty-six per cent of the beta disintegrations from I^{131} go to the 364-kev level of Xe^{131}.[8]

The estimated reading from a 3/8-in., sched-40 pipe containing 120 μc/ml with a lead shield having a 1/2- by 6-in. slit is 20 mr/hr on the logarithmic count rate meter. The details of this calculation are given below.

For a G-M counter 4 in. from an I^{131} solution having 4.45×10^6 dis sec^{-1} cm^{-3},

$$\text{count rate} = S\ \Omega/4\pi\ \epsilon\ e^{-ux}\ V\ ,$$

where

S = source strength, dis sec^{-1} cm^{-3},

Ω = solid angle of counter from point source 4 in. away,

ϵ = counter efficiency,[9]

V = volume of source, cm^3,

e^{-ux} = attenuation of 0.364-kev gamma ray from I^{131}.

(The counter is a cylinder with a radius of 0.31 in. and a height of 4 in.)

$$\begin{aligned}\text{count rate} &= (4.45 \times 10^6)(1.15 \times 10^{-2})(10^{-3})(0.86)(19.3)\\ &= 850 \text{ counts/sec}\\ &= \text{monitor reading of } \frac{850}{39.5} = 21.5 \text{ mr/hr}\ .\end{aligned}$$

The calibration of the logarithmic count rate meter was made with a Co^{60} source. The source at 68 cm away gives 10 mr/hr, and the logarithmic count rate meter was calibrated at this point. Also, the counter when placed 68 cm from the source gives 395 true counts/sec. There is a 4% dead-time correction at this radiation level (see Fig. B.1).

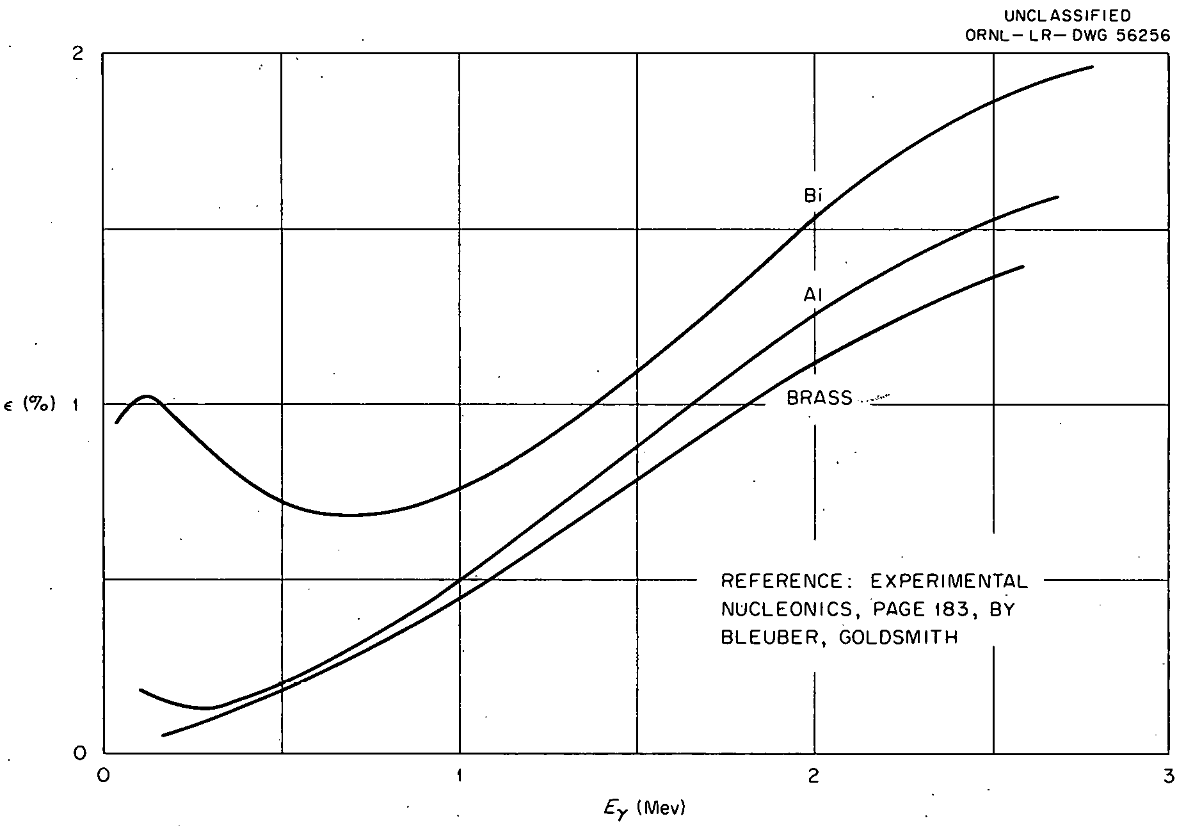

Fig. B.1. Gamma-Ray Sensitivity of Cylindrical G-M Counters.

Appendix C

PUMP CONTROLS

Main Loop Pumps

In the following discussion the main operating pump will be referred to as pump A, the first standby main pump as pump B, and the second standby main pump as pump C. The general mode of operation is as follows:

In the event of low main-loop water flow, pump B will be started and pump A stopped. If pump B does not resume normal flow in 2 sec, pump C will be started and pump B stopped, the reactor will be scrammed, the line heaters turned off, and the heat exchanger turned on to full capacity. If pump C does not resume normal flow in 2 sec, a switch back to pump A is attempted. The high temperature of an operating pump will transfer operation to a standby pump. The following features are provided:

1. An interlock to prevent starting of any pump initially unless cooling-water flow is normal in all three pumps.
2. Control may be manually taken away from any operating pump by operating the "start" switch of a standby pump, provided cooling-water flow to all pumps is normal (see No. 3).
3. Because the main pump motors can operate 2 to 4 min without cooling water, the pumps are not stopped in the event of low cooling-water flow. If the cooling-water pumps fail to maintain flow to the main pumps, the reactor is set back, the heat exchanger is turned on to full capacity, and the line heaters are turned off. If an excessively high temperature develops in the main operating pump, the operator either manually transfers control to one of the cold standby pumps or allows the automatic circuit to perform this transfer. A momentary-contact interlock bypass switch for cooling-water flow is provided in the control circuit to facilitate the manual transfer.
4. Lockout, in the automatic switching circuit, of the two standby pumps while the operating pump is on.
5. A circuit which is independent of the automatic switching circuit for testing individual pumps.
6. Immediate return of control to the operating pump in the event of a momentary (less than 15-sec) control-power interruption.
7. Lockout of all main pumps if a control-power outage exceeds 15 sec. (This prevents automatic starting of any pump when the control power is turned on initially.)
8. In a low loop-flow situation, the pumps will continue to switch every 2 sec in the sequence A, B, C, A, etc., until flow is restored or until pump control power is turned off.
9. Pump automatic switching circuit may be shut off manually only by turning off main pump control-power switch.

The specific mode of operation is as follows:

1. Water flow in the loop low enough to reach setpoint No. 1: the annunciator trips.

2. Water flow in the loop low enough to react setpoint No. 2:
   (a) The annunciator trips.
   (b) Timing action starts for contacts in low loop-water flow (scram) annunciator circuit.
   (c) Timing action starts for contact in scram makeup and dropout circuits.
   (d) Timing action starts for contacts in heat exchanger blending-valve maximum cooling circuit.
   (e) Timing action starts for contacts in loop-heater contactor circuits.
   (f) Pump B starts; pump A stops.

3. If flow recovers within 2 sec, the following occurs:
   (a) Low loop-water flow (setpoint No. 1) annunciator clears and may be reset.
   (b) Timing action ceases for contacts in low loop flow (scram) annunciator circuit, scram makeup and dropout circuits, loop-heater contactor circuits, and heat exchanger maximum cooling circuits. Original time intervals are reset on relays, and all circuits return to their condition prior to the low-flow situation.

4. If flow does not recover, the following occurs after 2 sec:
   (a) Contacts operate in scram makeup and dropout circuits — reactor scrams.
   (b) Contacts in low-flow (scram) annunciator circuit open — annunciator trips.
   (c) Time-delay contacts in loop-heater contactor circuits and heat exchanger maximum cooling circuit operate, loop heaters are shut off, and heat exchanger turned on to full capacity.
   (d) Pump C is started; pump B is stopped.

5. If flow still does not recover, the following occurs after an additional 2 sec: switch back to pump A is attempted.

6. Overload with pump A running:
   (a) Pump A motor overloads open — pump stops and flow fails.
   (b) Main pump overload annunciator trips.
   (c) Action (switch to pump B) is identical to that under low loop-water flow above.

7. Main control power failure with pump A running:
   (a) All relays drop. Reactor scram called for after 2 sec, but reactor has already scrammed due to action of non-time-delay relays.
   (b) Heat exchanger turned on to full capacity and line heaters turned off.
   (c) If power does not recover within 15 sec, pumps will not restart automatically.
   (d) If power does recover in 15 sec, pump A is restarted automatically.
8. High pump-motor temperature with pump A running:
   (a) Pump B is started; pump A is stopped.
   (b) High main pump temperature annunciator trips.
   (c) Pumps will continue to switch in the sequence A, B, C, A, etc., as pump high-temperature switches open until a previously opened temperature switch is encountered. In this case the pump which has the open switch will run until shut off manually.

Loop Reactor-Water System Booster Pumps

In the following discussion the operating loop reactor-water system booster pump is referred to as R1 and its standby pump as R2. The mode of operation is as follows:

1. Low loop reactor-water system flow produces the following:
   (a) Low flow (setpoint No. 1) annunciator trips.
   (b) Timing action starts for contacts in low flow (setpoint No. 2 — scram) annunciator circuit.
   (c) Timing action starts for contacts in scram makeup and dropout circuits.
   (d) Standby pump R2 starts; operating pump R1 stops.
2. If flow recovers within 4 sec:
   (a) Low flow (setpoint No. 1) annunciator clears and may be reset.
   (b) Timing action ceases for contacts in low flow (setpoint No. 2 — scram) annunciator circuit and scram makeup and dropout circuits. Original time intervals are automatically reset.
   (c) Standby pump R2 remains on.
3. If flow does not recover in 4 sec:
   (a) Low flow (setpoint No. 2 — scram) annunciator trips.
   (b) Relay contacts in reactor scram makeup and dropout circuits operate. Reactor scrams.
   (c) Standby pump R2 stays on.

4. An overload with operating pump R1 running:
   (a) Motor overload opens — pump stops and flow fails.
   (b) Pump R1 overload annunciator trips.
   (c) Remaining action same as for low reactor-water flow above.
5. Control-power failure with pump R1 running results in the following:
   (a) All relays drop — reactor scrams.
   (b) If power recovers within 15 sec, operating pump R1 is restarted automatically.
   (c) If power does not recover within 15 sec, pump must be restarted manually.

Additional reactor-water pump control features are as follows:

1. Control may be taken away from either pump by operating the start switch of the other pump.
2. A pump testing circuit is provided for both pumps which is independent of the automatic switching circuit.
3. After the operating pump is turned on and the automatic switching circuit set up, the pumps may be shut off only by turning the reactor-water pump control-power switch to the "off" position.

<u>Cooling-Water Pumps (Pool Water for Cooling Main Loop Pumps and Main Loop Heat Exchanger)</u>

Cooling water flow rates through each of the three main circulating pump motors are monitored by flowmeters equipped with flow switches. The signals from these switches control specific safety and cooling water pump transfer actions. In the following discussion the operating cooling-water pump is referred to as P1 and the standby cooling-water pump as P2. The general mode of operation is as follows:

1. Low cooling-water flow from any main circulating pump:
   (a) Low main circulating pump cooling-water flow annunciator trips.
   (b) Main circulating pump starting circuit is broken by flow switches.
2. If two or more cooling-water flow switches open, the following happens:
   (a) The standby cooling-water pump P2 starts; the operating pump P1 stops.

(b) Timing action starts for contacts in the setback makeup and dropout circuits, the heat exchanger maximum cooling circuit, the line-heater circuit, and the low cooling-water flow (setback) annunciator circuit.

3. If flow recovers within 3 sec, and two out of three flow switches close, the following occurs: timing action ceases for contacts in the setback makeup and dropout circuits, heat exchanger maximum cooling circuit, line-heater circuit, and low cooling-water flow (setback) annunciator circuit. Original time intervals are reset on all relays, and all circuits return to their condition prior to the low-flow situation.

4. If flow does not recover and two or more flow switches remain open, the following occurs after 3 sec:

   (a) Contacts operate in setback makeup and dropout circuits — reactor setback is initiated.

   (b) Low cooling-water flow (setback) annunciator trips.

   (c) Contact opens in heat exchanger maximum cooling circuit — heat exchanger is turned on to full capacity.

   (d) Contacts open in line-heater circuits; all heaters are turned off.

5. Overload with operating pump P1 running:

   (a) Motor overloads open — pump stops and flow fails.

   (b) Cooling-water pump overload annunciator trips.

   (c) Remaining action same as low cooling-water flow condition above.

6. Control-power failure with operating pump P1 running:

   (a) All relays drop — reactor scrams.

   (b) If power recovers within 15 sec, operating pump P1 will be restarted automatically.

   (c) If power does not recover within 15 sec, pump P1 must be restarted manually.

Additional cooling-water pump control features are as follows:

1. Control may be taken away from the operating pump by operating the "start" switch of the standby pump.

2. A pump testing circuit is provided for both pumps which is independent of the automatic switching circuit.

3. After the operating pump is turned on and the automatic switching circuit is set up, the pumps may be shut off only by turning the cooling-water pump control-power switch to the "off" position.

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