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AEC RESEARCH AND DEVELOPMENT REPORT

OPERATIONAL SUMMARY OF THE HEAVY WATER COMPONENTS TEST REACTOR

OCTOBER 1961-DECEMBER 1964

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DU PONT

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ABSTRACT

The Heavy Water Components Test Reactor (HWCTR) is a pressurized reactor that is cooled and moderated with heavy water. The reactor is located at the Savannah River Plant, Aiken, South Carolina, and was operated for the Atomic Energy Commission by the Du Pont Company. It was designed for the purpose of testing, at actual operating conditions of flow, temperature, and pressure, candidate fuel assemblies and other reactor components for a heavy water power reactor fueled with natural or slightly enriched uranium.

This report describes the startup and operating history of the HWCTR from October 1961 to December 1964. The reactor facility was shut down in December 1964.

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OPERATIONAL SUMMARY OF THE HEAVY WATER COMPONENTS TEST REACTOR
OCTOBER 1961 - DECEMBER 1964

SUMMARY

The HWCTR was operated from October 1961 to December 1964 to test fuel elements and other reactor components of potential use in heavy water moderated and cooled power reactors. Operations were terminated and the facility placed in standby condition as a result of the decision by the U. S. Atomic Energy Commission to redirect the research and development work on heavy water power reactors to those that are cooled with organic materials. The facility is being maintained in a condition such that it could be reactivated in approximately six months.

Five months was devoted to the assembly and installation of reactor components and preoperation testing of equipment and systems. During the initial hydraulic tests with light water, a protective film of magnetite was formed on the surface of the process system. This film remained intact during the subsequent operation and accounted for the completely satisfactory performance of the large amount of carbon steel in the process system.

The total nuclear exposure in the HWCTR was 13,882 megawatt days (MWD). Thirty-six test assemblies containing tubular fuel of uranium metal or uranium oxide were irradiated, and the utility of this fuel for power reactors was successfully demonstrated. One assembly of tubular oxide elements reached an exposure of 17,500 MWD/Tonne. Ten failures of experimental fuel were experienced during this period. In each instance the failure was detected promptly, and the reactor was shut down before the process system became seriously contaminated.

Normal maintenance and minor modifications to equipment and systems resulted in satisfactory performance throughout the operating period.

The steam generators leaked through the tube-to-tube-sheet joints at rates between 5 to 15 pounds per day from the beginning of operations. The uncollected losses of heavy water from the rest of the system varied, but were as low as 8 pounds per day under best conditions. Because the HWCTR system consisted of standard components throughout, these results indicate that slight additional attention to leak tightness would produce a system from which D_2O losses would be sufficiently low to constitute only a minor item in the cost of operation.

The composite concrete and steel containment building leaked about 0.75% of the contents per day when the internal pressure was 5 psig. The provision of carbon beds to absorb halogen fission products and the isolated location of the facility made these leakage rates acceptable.

BRIEF DESCRIPTION OF FACILITY

Only a brief description of the facility is given in this report; a detailed description is in the Final Hazards Evaluation report, DP-600.[1]

The HWCTR is housed in a containment building 70 feet in diameter and 125 feet high. Approximately half of the building is below grade and is prestressed concrete. The upper half of the building is carbon steel. The building is designed to withstand an internal pressure of 24 psig and was tested pneumatically at 29 psig. The containment building houses the reactor and coolant systems, the charge-discharge mechanisms, and the reactor instrumentation. The control room and emergency power equipment are in a separate building. A photograph of the HWCTR Area is Figure 1. The containment building, shown in Figure 2, has an enclosed volume of 420,000 ft^3 and a free volume of 320,000 ft^3.

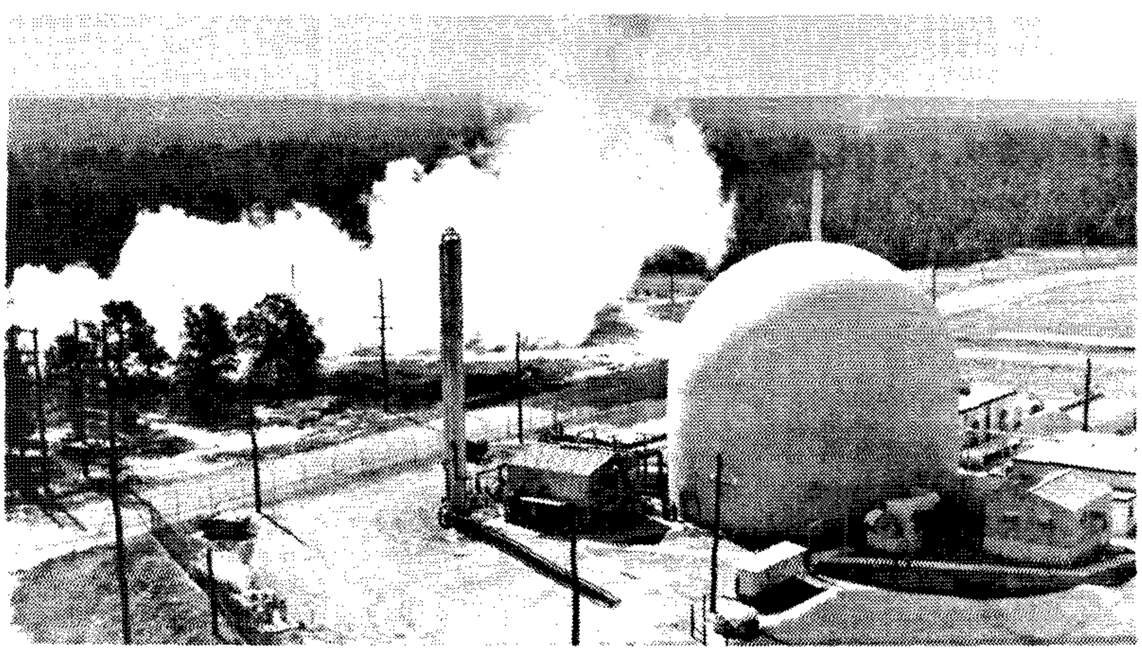


FIG. 1 THE HWCTR AREA

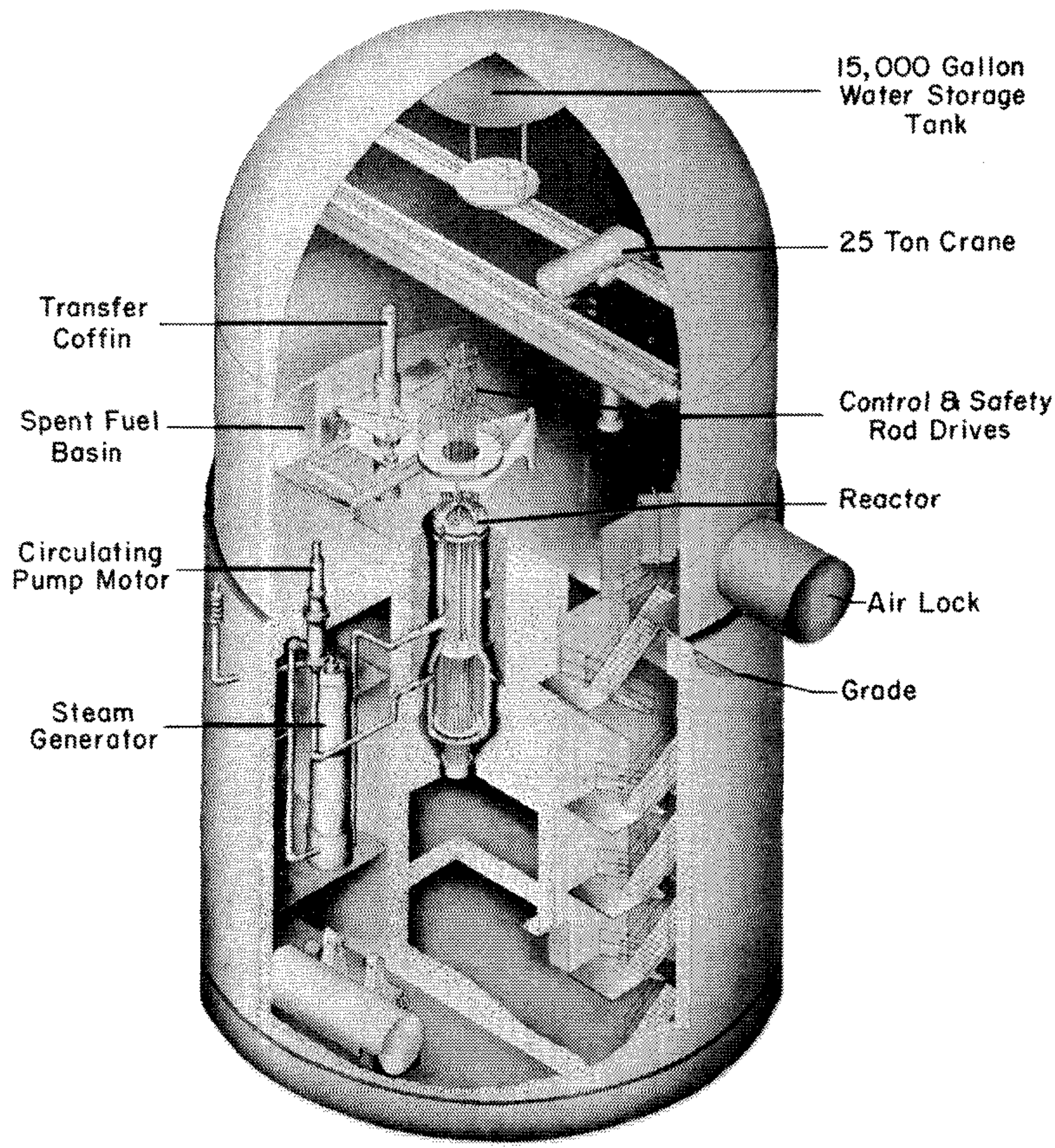


FIG. 2 CONTAINMENT BUILDING

The reactor vessel has an over-all height of about 30 feet, a maximum inside diameter of 7 feet, and a volume of 4000 gallons to the normal D_2O level. The shell and head are carbon steel plates and forgings, and all inside surfaces are clad with stainless steel, 0.25-inch nominal thickness. The design pressure of the vessel is 1500 psi at a temperature of 315°C. A cut-away drawing of the reactor is Figure 3.

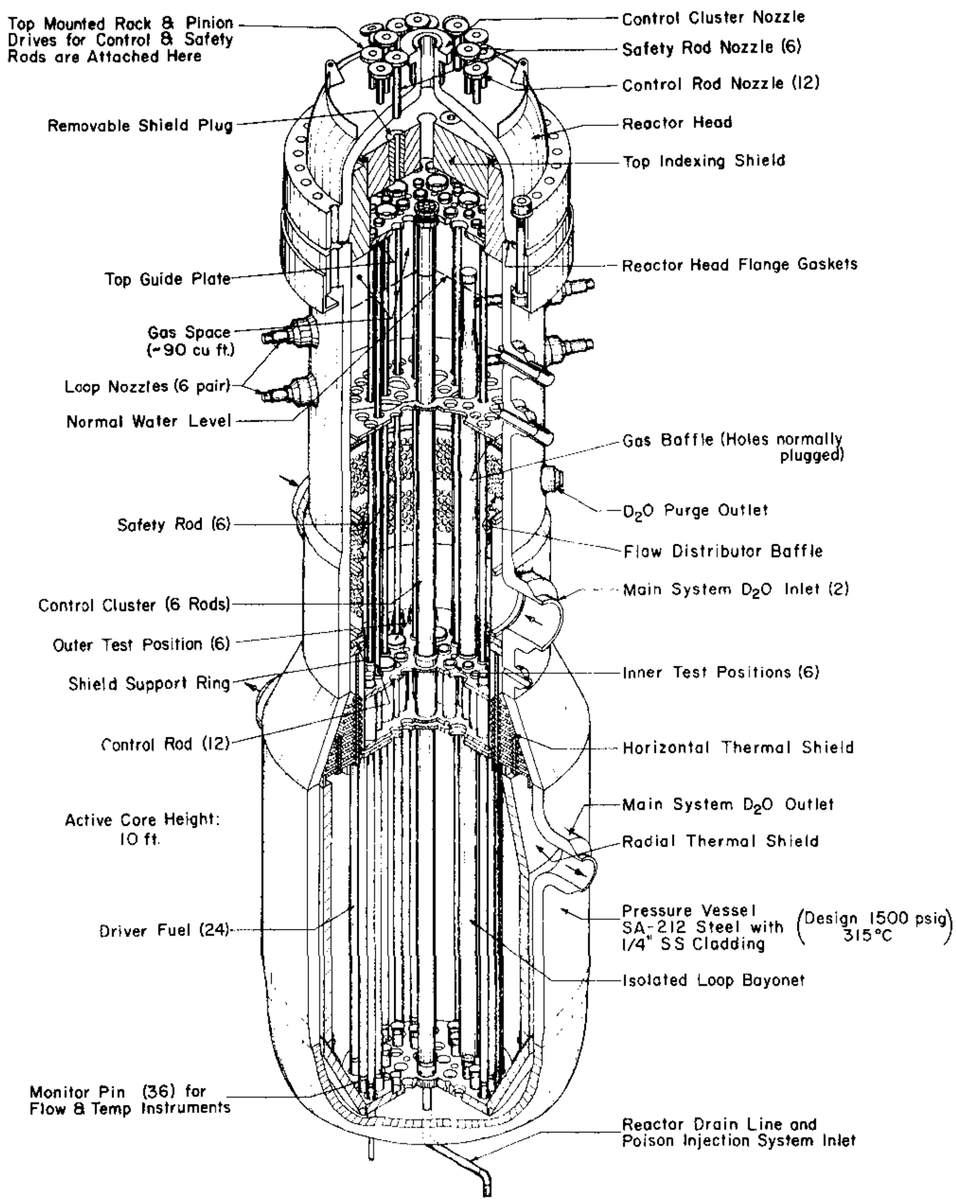


FIG. 3 CUT-AWAY VIEW OF THE REACTOR

The core consists of a central test region of 12 positions surrounded by a ring of 24 driver fuel positions. The arrangement of these fuel positions, and the locations of control rods, safety rods, and instrument thimbles are shown in Figure 4. Rack and pinion drive mechanisms for the control and safety rods are located above the reactor. The drive mechanisms are bolted to the reactor head and are lifted with it when the head is removed.

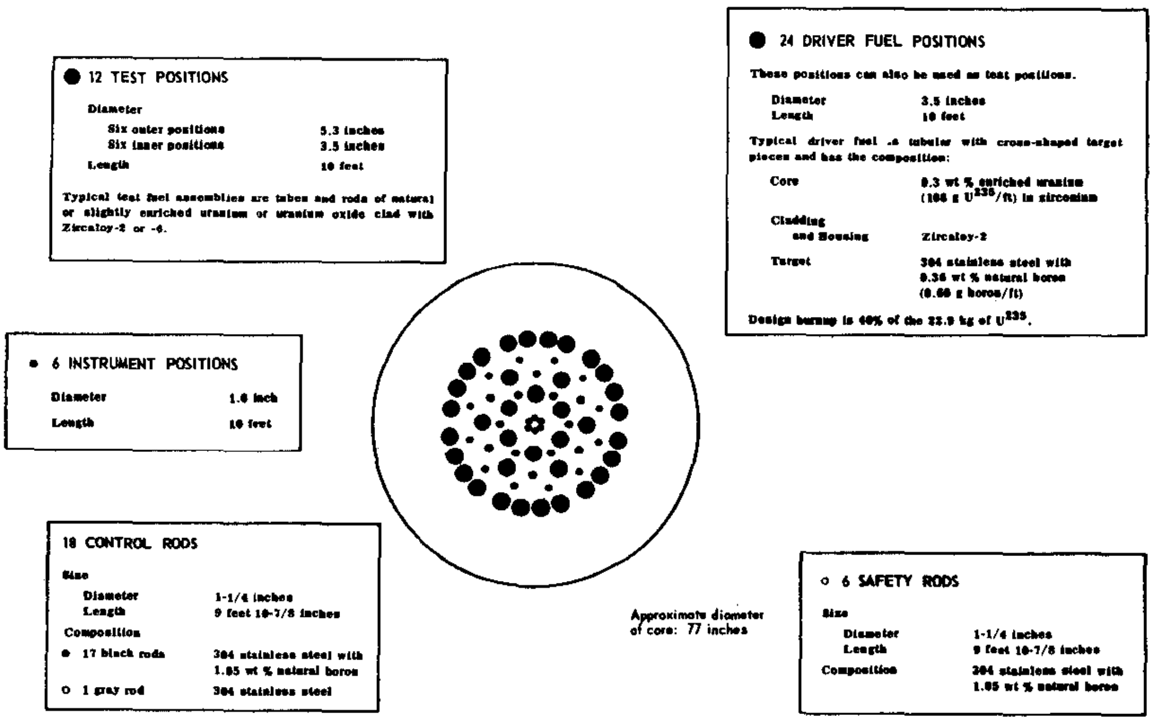


FIG. 4 THE HWCTR LATTICE

Heavy water to moderate and cool the reactor is pumped through two nozzles into the top section of the reactor vessel at about 10,000 gpm. It flows down through the fuel assemblies, up through the moderator space, and out to two coolant loops. The heavy water is cooled by boiling light water in a steam generator in each loop; the steam produced is discharged to the atmosphere. A schematic diagram of the high pressure coolant system and the low pressure purification-makeup system is shown in Figure 5. The moderator pD is maintained at 10.5 with lithium hydroxide to inhibit corrosion of the mild steel piping and equipment. The concentration of dissolved oxygen in the D_2O is suppressed by maintaining excess deuterium in the pressurizing gas. Ionic impurities are removed by LiOD mixed-resin beds.

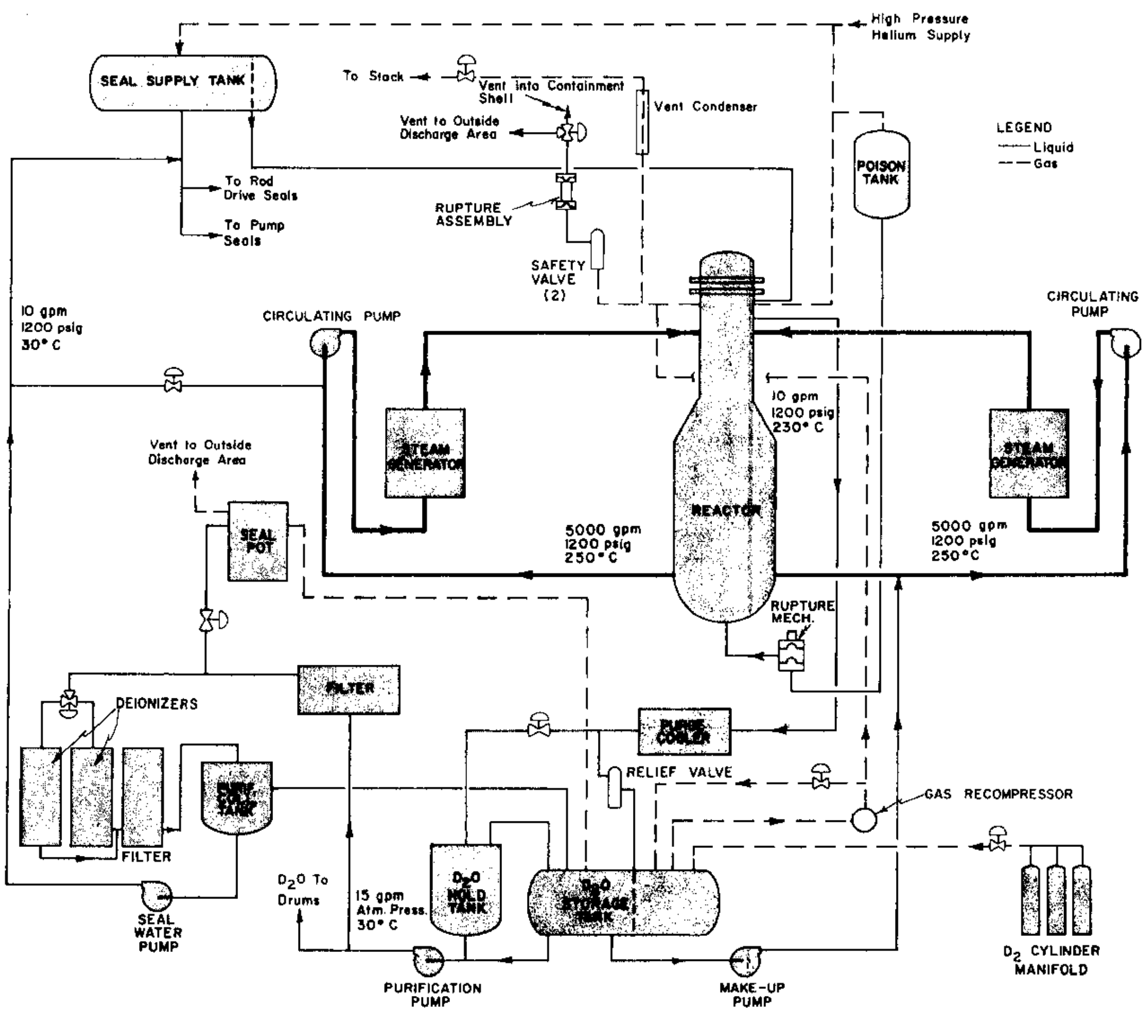


FIG. 5 MAIN COOLANT SYSTEM

Each outer test position can be fitted with a bayonet, a special re-entrant pressure tube that isolates the test position from the main coolant system. Two bayonets were installed initially, but one was removed after failure because of a vibrational problem. Each of these positions is cooled by a separate isolated coolant loop. All piping and equipment in the loop systems is stainless steel. A schematic diagram of the isolated coolant loops is shown in Figure 6.

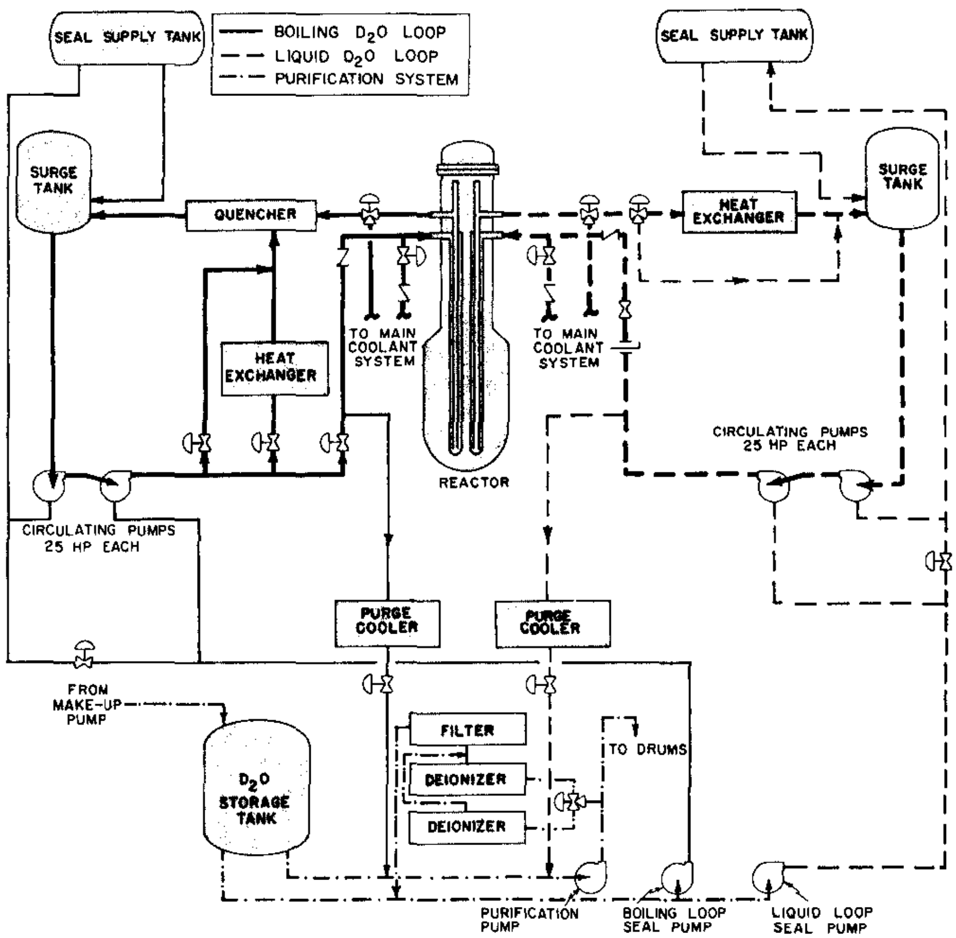


FIG. 6 ISOLATED COOLANT LOOPS

CHRONOLOGICAL SUMMARY

The bar chart in Figure 7 summarizes briefly the history of the facility. The timing and content of the three major periods in the lifetime of the facility, (1) precritical test period, (2) low power test period, and (3) power operation period, are described in this section. Results of tests are not reported except wherein they affected the schedule. Results of tests were reported briefly in monthly power reactor progress reports[2] as they occurred.

PRECRITICAL TESTING - OCTOBER 1961 TO MARCH 1962

Major construction of the facility was essentially completed by October 1961. As shown in Figure 7, five months were required for the

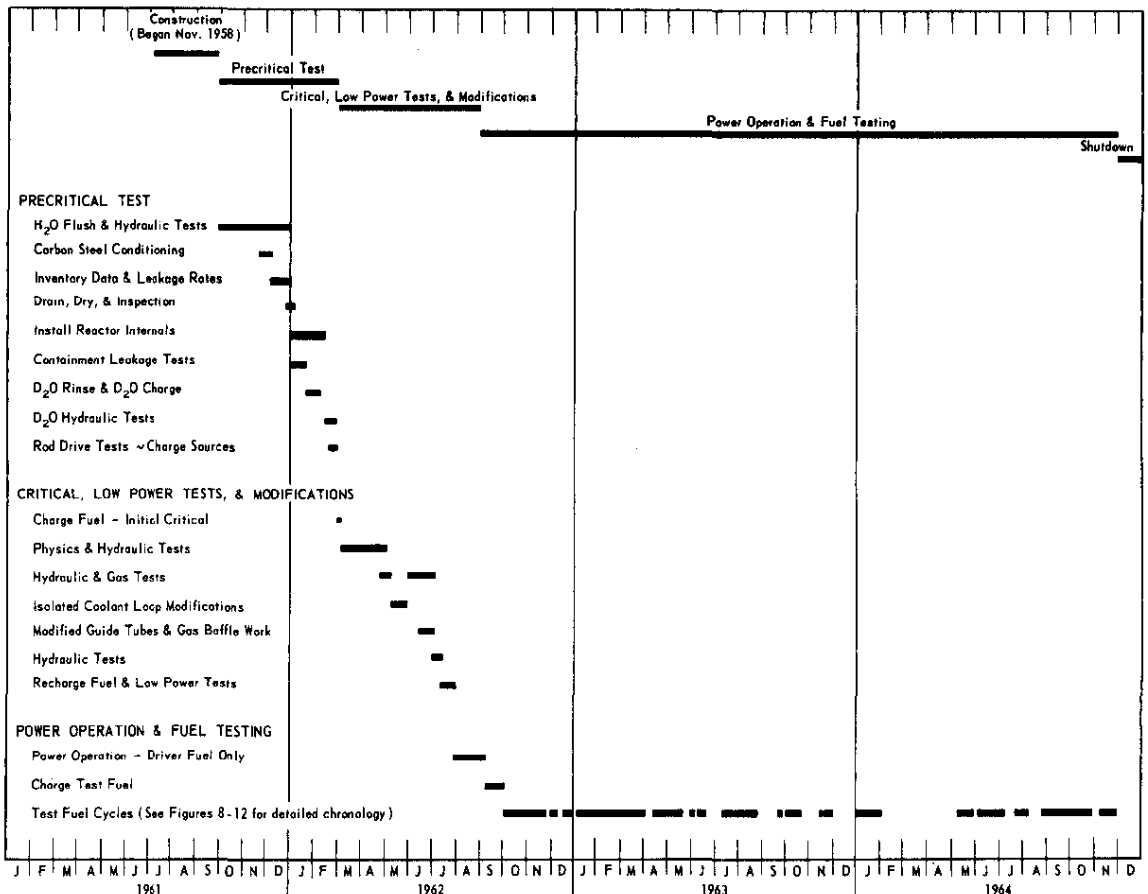


FIG. 7 OPERATING CHRONOLOGY OF THE HWCTR

final assembly and installation of reactor components, equipment run-ins, pressure testing, cleaning, and equipment and system performance tests. The system was cleaned and tested initially with light water. A protective film of magnetite (Fe_3O_4) was formed on the surface of all the carbon steel piping and equipment in the high pressure system. This was accomplished by operating at high pH (10.0 to 10.8) and high temperature (200 to 260°C) in the absence of oxygen for twelve days.

Following these tests, the high pressure system was drained, inspected, and vacuum dried to remove the light water from the piping-vessel complex. System pressure was reduced to 30-40 mm Hg absolute using two large vacuum pumps in parallel. Dew point measurements of the discharged air indicated a minimum moisture content of 0.0035 to 0.005 lb/lb of dry air. Only 150 ml of water was drained from low points in the high pressure system at the completion of a five-day vacuum drying period. The low pressure systems and isolated coolant loops were rinsed with D_2O. The rinse removed approximately 640 pounds of H_2O and left about 5 pounds of H_2O in the various systems. By February 11, 5603 gallons of 99.77 mole percent D_2O had been charged to the main storage tank for filling the reactor system.

A series of pressure tests and mechanical improvements were made to evaluate and to minimize leakage from the HWCTR containment shell. These tests began in December 1961 and continued through January 1962. In November 1960, after completion of the basic containment shell, the leakage rate was 0.56% of the building contents per day at 24 psig. A preliminary test, in early November 1961, showed a leakage rate of 8.4% per day at 24 psig. A series of twelve leakage tests, each followed by repair efforts, resulted in a final leakage rate of 0.66% per day at 24 psig internal pressure.

The reactor internal parts (top shield, guide plates, guide tubes, and isolated loop bayonets) were installed, and photographs were taken of pertinent inaccessible parts and assemblies for future reference. On February 6, after satisfactorily completing performance tests with the fuel transfer coffin, the charging of all reactor components except fuel was begun; these components included housing tubes, shield muffs, safety and control rods, and an internal gamma flux monitor. This work was completed by February 16, 1962.

Upon completion of the charging of reactor components on February 16, 1962, about ten days were devoted to general performance and hydraulic testing of the system. These tests included leak rate measurements, vibration measurements, pump performance, flow decay, and tests of the poison injection system. Except for minor mechanical and instrument difficulties, the only major deficiency was associated with the injection times and injection pressures of the poison injection system. Pending redesign of the system, the problem was resolved for the low power test period by specifying different injection pressures for the two cases where the reactor was pressurized and unpressurized.

At the end of February 1962, preparations were begun for the initial-critical test. These included: charging two Sb-Be neutron source rods, installing two BF_3 ion chambers in the reactor core, final calibration and testing of all safety circuits, and thorough cycling and testing of the safety and control rod drives.

INITIAL-CRITICAL, LOW POWER TESTS, AND MODIFICATIONS

The specific objectives for this period were: (1) to achieve the initial critical, (2) to measure the margin of control, (3) to measure the worth of the control and safety rods, (4) to measure the flux shapes in a variety of rod configurations and conditions, (5) to measure the migration area, and (6) to measure the static temperature coefficients for the lattice. Details and results of these tests are reported in DP-967.[3] The initial criticality was achieved at 11:31 PM on March 3, 1962. The reactor was critical for about 130 hours during March and April for these tests.

During the course of the low power physics tests, steady state and transient hydraulic characteristics of the system were determined at system pressures up to 1000 psig and moderator temperatures up to 240°C. During this period, minor difficulties were experienced with loose bottom fittings on safety rod guide tubes, broken actuators on rod drive switches, failed piston rings on the helium recompressors, and electronic noise in the log N-period circuitry. Repairs and redesign corrected adequately these difficulties.

Toward the end of the low power physics tests, a relationship between the operating conditions of the hydraulic system and nuclear reactivity was discovered. Investigation revealed that voids were formed in the core region of the reactor by aspiration of helium through the control and safety rods and by dissolution of helium from the moderator. The results of these investigations and the subsequent modifications made to the reactor system to eliminate these phenomena are described in DP-988.[4]

In brief, nuclear and hydraulic tests in late April and early May showed that helium was aspirated from the gas space in the upper part of the reactor tank into the core region through the cooling holes in the guide tubes and rods at a number of rod configurations and/or reactor water levels. Subsequent tests in a hydraulic mockup of these assemblies showed that the aspiration could be eliminated by slotting the guide tubes to provide a direct path for coolant flow to the rod with the rod at any elevation and by providing additional holes in the rod extensions and rods to provide a direct path for coolant flow through the rods.

Several tests were performed at high temperature and pressure conditions to investigate the problem of dissolution of helium from the moderator in the reactor core. These tests showed that the coolant in the neck section of the reactor was nearly saturated with helium. The subsequent pressure loss and temperature rise across the fuel could, at some conditions of power operation, result in the moderator in the core region being supersaturated with helium. Any subsequent collapse or reduction in voids caused by pressure increase or flow decrease would, in effect, add an increment of positive reactivity to the lattice. Measurements of the rate of helium transfer to the moderator from the gas space, with and without a temporary baffle in the neck section of the reactor vessel, showed that the baffle would reduce the equilibrium concentration of helium in solution to a point that void formation would not be possible in the reactor core at the anticipated reactor operating conditions.

The above tests and the subsequent design and component modification took place in May and June 1962. Part of the month of May was also devoted to completing some design modifications to the isolated coolant loops and the poison injection system.

A series of high temperature hydraulic tests was conducted between July 1 and 13, to evaluate the effectiveness of the modifications made to eliminate the formation of voids in the moderator. These tests indicated that the changes which had been made to the control and safety rods and guide tubes eliminated aspiration of helium into the moderator under all conditions and that no gas entrainment occurred as long as the water level was maintained above the new gas baffle in the neck section of the reactor vessel. Measurements of the helium solution rate and dissolved gas concentration showed that these values were sufficiently low to permit operation at design conditions without helium dissolution from the moderator anywhere in the system. The equilibrium gas concentration in the moderator in the neck section of the reactor vessel at operating temperature and pressure was about 60% of saturation.

The hydraulic tests in the boiling and liquid isolated loop were also completed during this period.

All driver fuel and low power test fuel were recharged to the reactor between July 14 and 17. A series of low power physics tests was then begun to verify by nuclear measurements that no voids existed in the moderator and to redetermine basic physics data which might have been affected by voids during previous tests. These tests confirmed that voids did not exist and furnished the necessary data for correcting previous test results. These tests comprised seventeen critical runs and were concluded on July 27, 1962, thus completing the second major period in the operation of the facility.

POWER OPERATION AND FUEL TESTING

Bar chart summaries of this period, from July 1962, through December 1964, are shown in Figures 8 through 12.

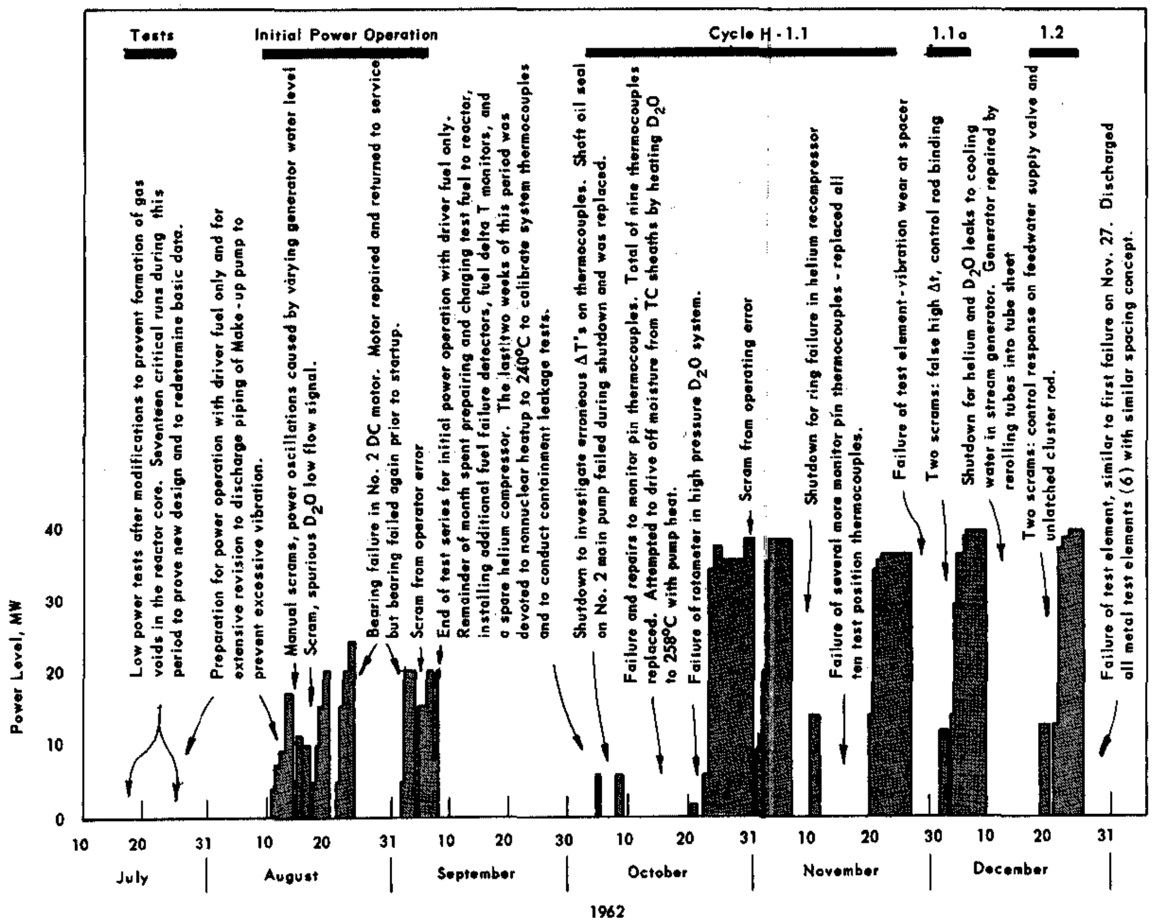


FIG. 8 OPERATING SUMMARY JULY - DECEMBER 1962

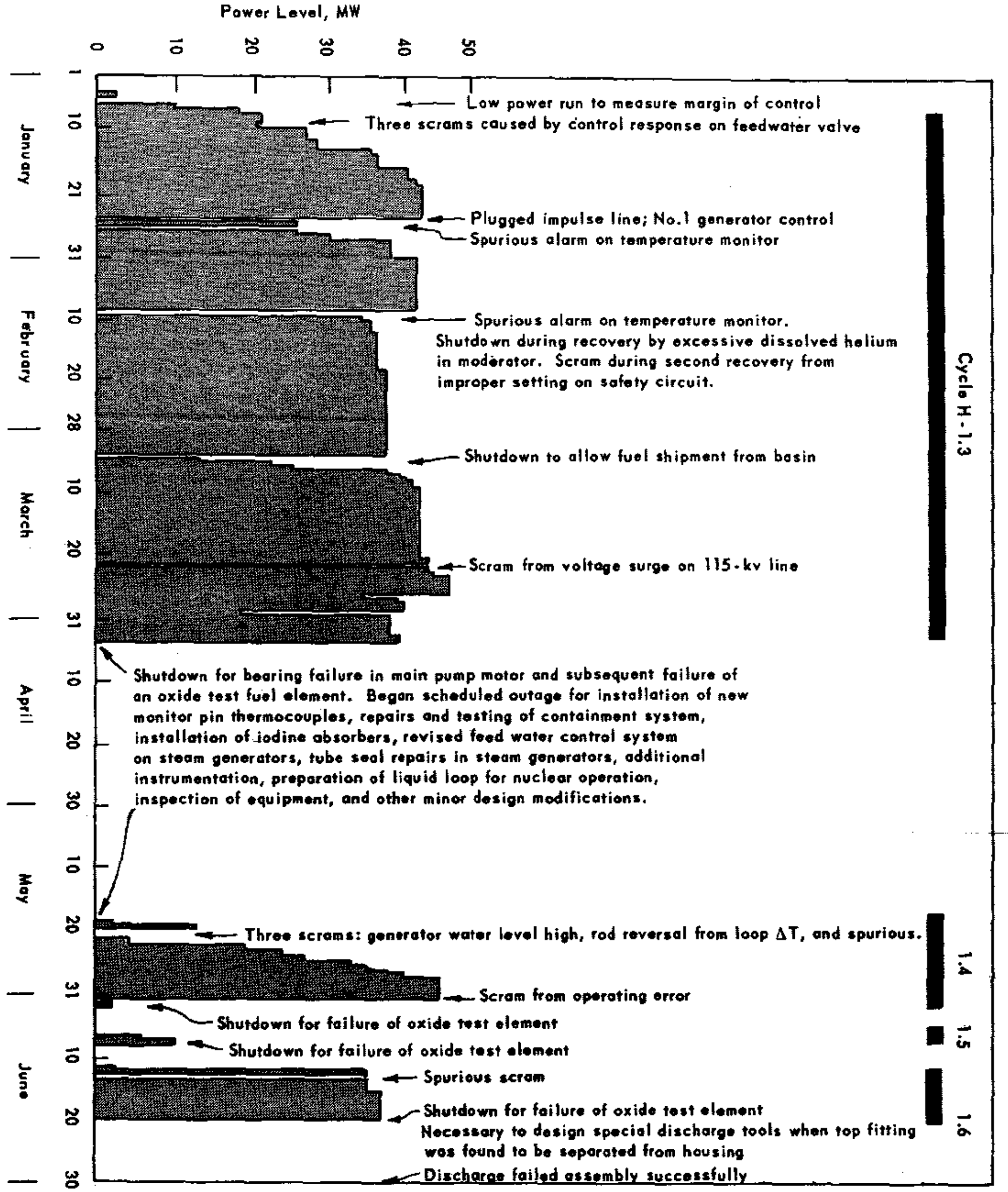


FIG. 9 OPERATING SUMMARY JANUARY - JUNE 1963

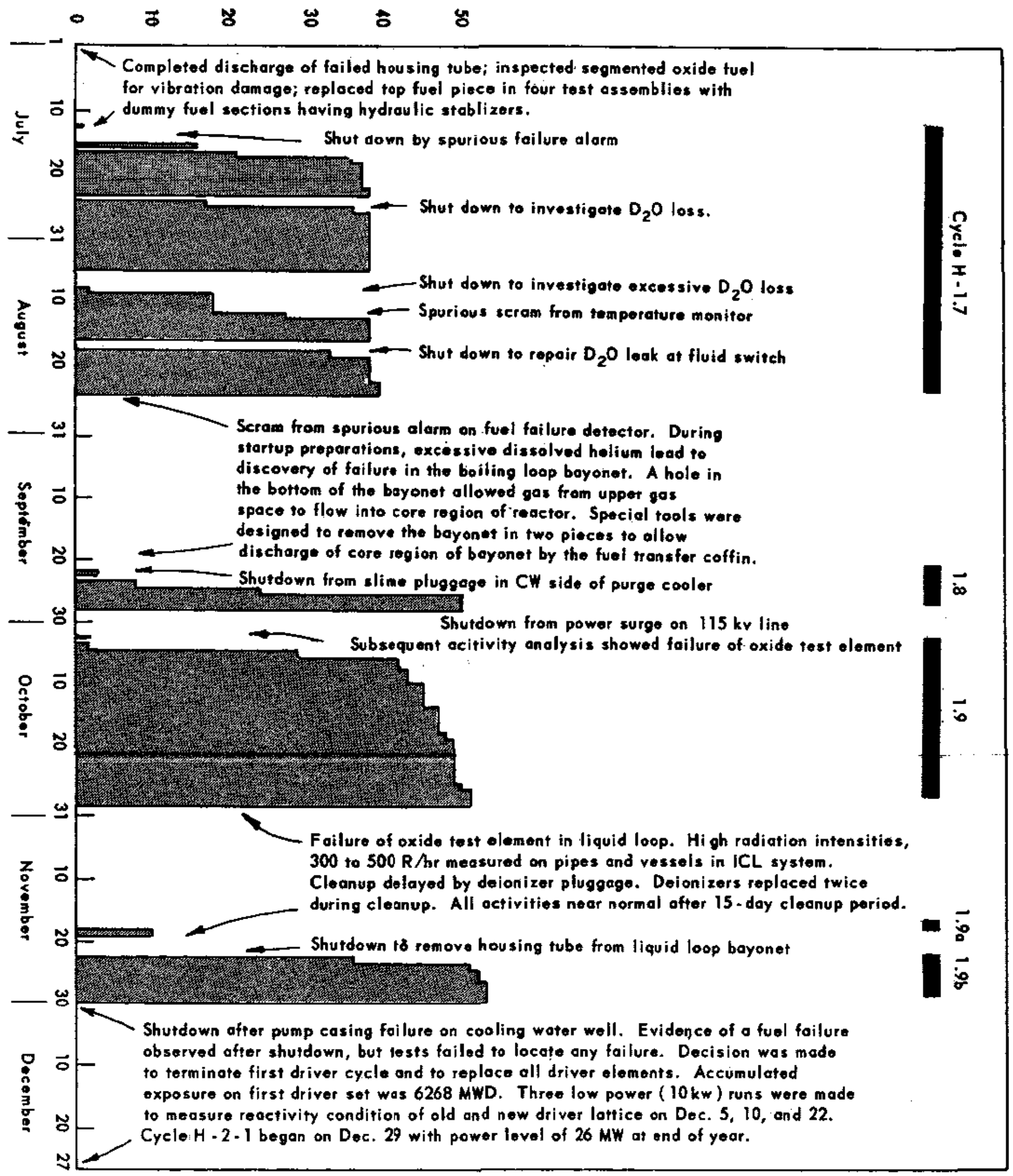


FIG. 10 OPERATING SUMMARY JULY - DECEMBER 1963

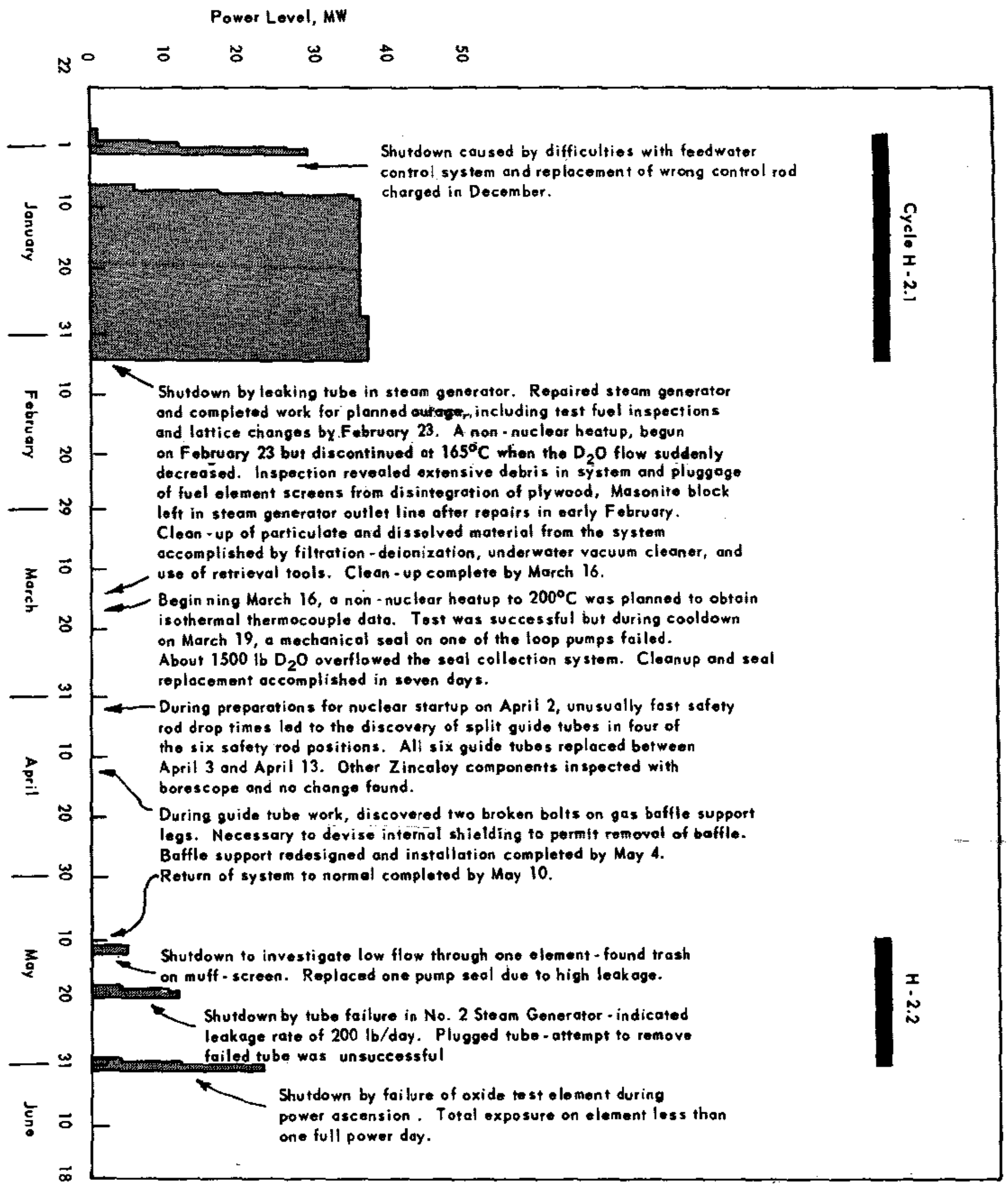


FIG. 11 OPERATING SUMMARY JANUARY - JUNE 1964

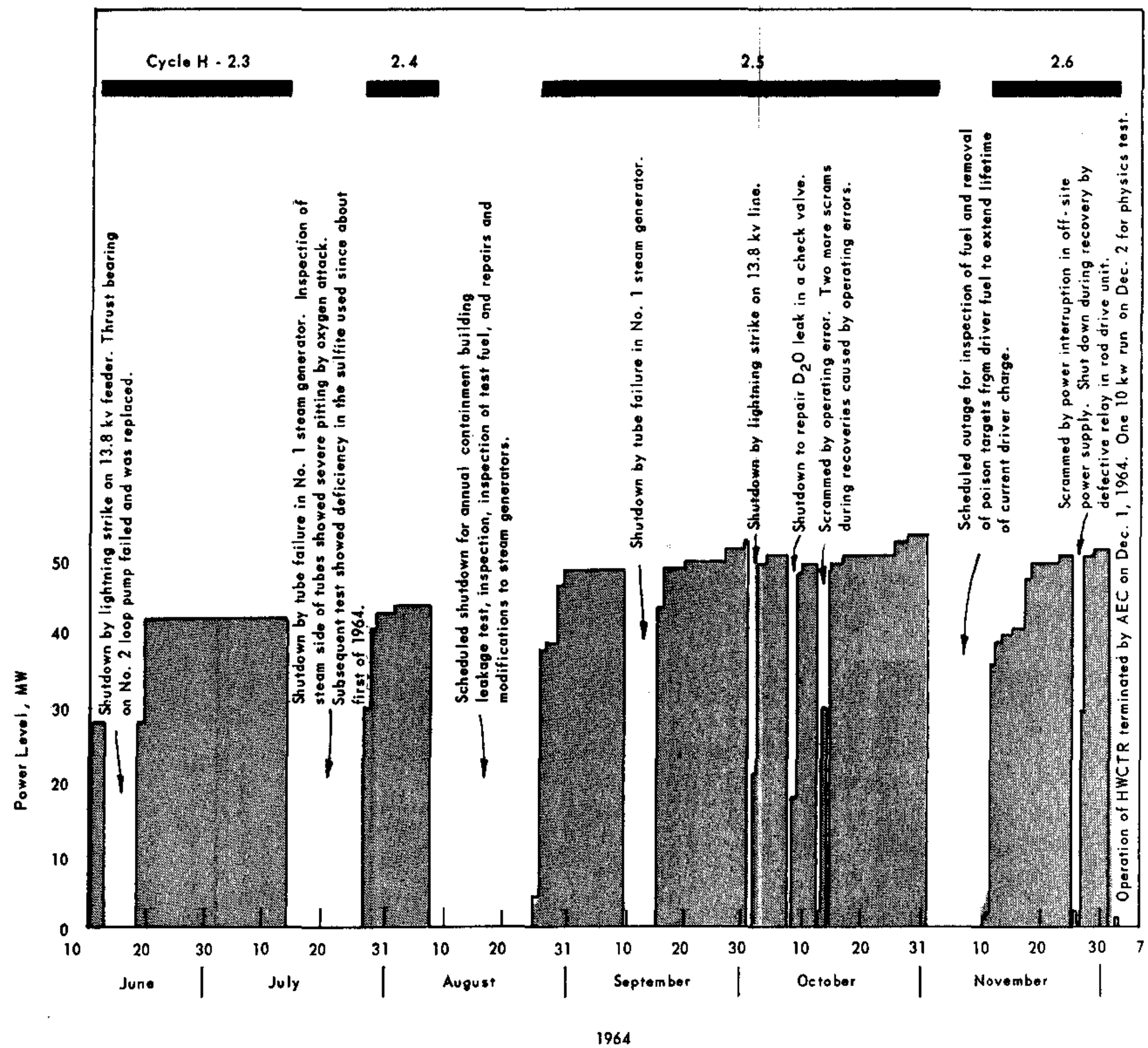


FIG. 12 OPERATING SUMMARY JUNE - DECEMBER 1964

During the remainder of July and until August 11, preparations were made for operation at power with driver fuel only. This period included four days for extensive revision to the discharge piping of a high pressure injection pump in the D_2O make-up system. Severe vibration in a long section of unbraced piping caused fatigue failures in two stainless steel tee connections; also, the tees contained stress-raising notches. Rerouting the piping, installation of liquid pulse traps (snubbers), and adequate bracing reduced the vibration to acceptable levels.

Initial-power operation was begun August 11, and the reactor was operated at about 60% innage until the end of the test period on September 7. The objectives of the power tests with driver fuel only were: (1) to establish that all equipment operated properly, (2) to demonstrate that the reactor could be operated safely within the established limits, (3) to determine the adequacy of operating procedures, and (4) to obtain technical data necessary for future operation.

The major cause of lost time during this test period was bearing failures in the DC motor on one of the main circulating pumps. The upper (outboard) ball bearing became damaged and seized, causing the outer race to rotate in its retainer in the end-bell. The initial cause of the trouble was not apparent but was presumed to be either a faulty bearing or some slight mis-seating during the initial installation. The motor was repaired and returned to service, but the upper bearing again failed after three days of service. Investigation showed that the lower (inboard) bearing was not properly seated on the shaft, causing undue thrust on the upper bearing.

On three occasions during the initial-power run, diverging oscillations occurred in the indicated water level in one of the steam generators and in the reactor power. The water level oscillated as much as ±4 inches and the power as much as ±25%. In each case the oscillations were stopped by adjustments of the level controller.

The test program with driver fuel only was concluded September 7, and during the remainder of the month preparations were made for the first irradiation of test fuel elements in the HWCTR. In addition to preparation and charging of test fuel assemblies, the following major work was accomplished:

1) Two new systems for the detection of fuel failures were installed. One type, a delayed neutron monitor, was installed on both the main system and the isolated coolant loops. The second type, a scanning liquid photoneutron monitor, was installed to monitor sequentially the activity in the effluent stream from each of the ten test fuel positions.

2) A spare compressor for helium was installed.

3) Performance tests were made on the primary D_2O relief valves, and additional bracing was installed.

4) The hydraulic system was operated, and moderator was heated to 240°C on pump heat to calibrate system thermocouples.

5) A ten-point continuous temperature monitor was installed for the test fuel positions in the main system.

Containment leakage tests were conducted during the last week in September. These tests indicated that the leakage had increased from 0.66% of the building contents per day, as determined in January 1962, to about 4% per day. Details of these tests are reported in DP-968.[5] The decision was made to permit nuclear operation with this leak rate while modifications and repairs to the containment system were studied. Operating limits were imposed, however, at a power of 45 MW and a total exposure of 4500 MWD for the fuel charge. These values were calculated to keep the quantity of radioactive material released from the building following a maximum credible accident within the values of the example given in the hazards evaluation report, DP-600.[1]

Nuclear operation with test fuel in the reactor was begun on October 5.

Test Fuel Cycles*

Figures 8 through 12 show, in brief detail, the reactor power level and the major causes of shutdowns and outage time for the period of power operation of the reactor, beginning August 11, 1962, and ending December 1, 1964. Figure 13 shows the reactor innage, by month, and the accumulated innage for this same period of operation. Reactor innage, as used in this report, is defined as the ratio of the time the reactor was critical to the available time in the period concerned, expressed as a percentage.

*The terminology H-1.1, H-1.2, etc, was given to consecutive reactor test cycles between shutdowns for test lattice changes during the life of the first charge of driver fuel. Starting with the second charge of drivers, the cycles were numbered H-2.1, H-2.2, etc.

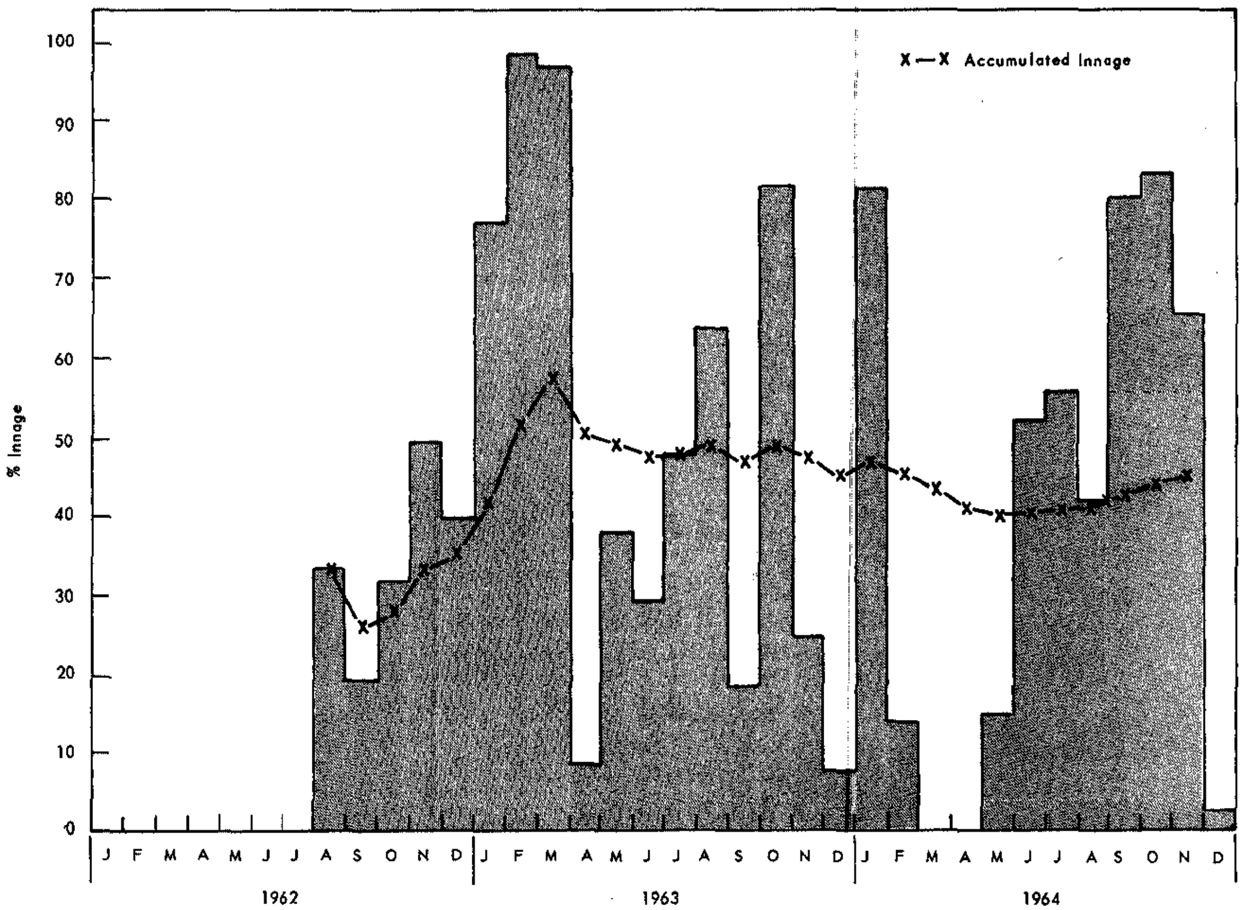


FIG. 13 HISTORY OF REACTOR INNAGE

Table I lists the starting and ending date for each reactor cycle and shows, by reactor position, the test fuel assemblies in each of the reactor cycles. Table II briefly describes each of the test fuel assemblies. The operating characteristics and performance of these test fuel assemblies were reported in progress reports,[2] and will not be reported here except where their performance affected the reactor schedule.

TABLE I

HWCTR Cycles and Test Fuel Positions

Cycle	Starting Date	Ending Date	Reactor Positions[a] 37	38[b]	39	40	42	55	56	57	58	59	60
H-1.1[c]	10/5/62	11/27/62	CANDU	--	TWNT-5	SOT-1-2	OT-1-2	TWNT-11	TWNT-14	TWNT-9	TWNT-12	TWO-1-2	TWNT-7[d]
H-1.1[c]	12/2/62	12/9/62	CANDU	--	TWNT-5	SOT-1-2	OT-1-2	TWNT-11	TWNT-14	TWNT-9	TWNT-12	TWO-1-2	TWNT-13
H-1.2	12/19/62	12/26/62	CANDU	--	TWNT-5	SOT-1-2	OT-1-2	SOT-2-3	TWNT-14[d]	TWNT-9	TWNT-12	TWO-1-2	TWNT-13
H-1.3	1/6/63	4/3/63	CANDU	--	OT-1-4	SOT-1-2	OT-1-2	SOT-2-3[d]	SOT-1-3	OT-1-3	OT-1-5	SOT-2-2	OT-1-6
H-1.4	5/19/63	5/31/63	CANDU	OT-3-2	ETWO-2	SOT-1-2	OT-1-2	SOT-5-2	SOT-1-3	SMT-1-2	OT-1-5	SOT-2-2[d]	OT-1-6
H-1.5	6/6/63	6/7/63	CANDU	OT-3-2	ETWO-2	SOT-1-2	OT-1-2	SOT-5-2	SOT-1-3	SMT-1-2	OT-1-5	OT-1-3	OT-1-6[d]
H-1.6	6/11/63	6/20/63	CANDU	OT-3-2	ETWO-2	SOT-1-2	ETWO-3	SOT-5-2	SOT-1-3	SMT-1-2	OT-1-5	OT-1-3[d]	OT-1-2
H-1.7	7/13/63	8/25/63	CANDU	OT-3-2	ETWO-2	SOT-1-2	ETWO-3	SOT-5-2	SOT-1-3	SMT-1-2	SOT-1-4	OT-1-7	SMT-1-3
H-1.8	9/22/63	9/28/63	CANDU	(e)	ETWO-2	SOT-1-2	ETWO-3	SOT-5-2[d]	SOT-1-3	SMT-1-2	SOT-1-4	OT-1-7	SMT-1-3
H-1.9	10/3/63	10/29/63	CANDU	SOT-7-2[d]	ETWO-2	SOT-1-2	ETWO-3	OT-1-2	SOT-1-3	SMT-1-2	SOT-1-4	OT-1-7	SMT-1-3
H-1.9[c]	11/18/63	11/19/63	CANDU	(f)	ETWO-2	SOT-1-2	ETWO-3	OT-1-2	SOT-1-3	SMT-1-2	SOT-1-4	OT-1-7	SMT-1-3
H-1.9[e]	11/23/63	11/29/63	CANDU	--	ETWO-2	SOT-1-2	ETWO-3	OT-1-2	SOT-1-3	SMT-1-2	SOT-1-4	OT-1-7	SMT-1-3
H-2.1	12/29/63	2/4/64	CANDU	SOT-6-2	--	SOT-1-2	OT-1-4	OT-1-2	RMT-1-2	SMT-1-2	SOT-1-4	OT-1-7	SMT-1-3
H-2.2	5/11/64	5/31/64	CANDU	SOT-6-2	EMT-2(SS)	SOT-1-4	OT-1-2	SOT-9-2[d]	RMT-1-2	SMT-1-2	SOT-1-2	SOT-6-3	SMT-1-3
H-2.3	6/9/64	7/13/64	CANDU[d]	SOT-6-2	EMT-2(SS)	SOT-1-4	OT-1-2	OT-1-7	RMT-1-2	SMT-1-2	SOT-1-2	--	SMT-1-3
H-2.4	7/26/64	8/7/64	--	SOT-6-2	EMT-2(Zr)	SOT-1-4	OT-1-2	SOT-6-3	RMT-1-2	SMT-1-2	SOT-1-2	--	SMT-1-3
H-2.5	8/25/64	10/31/64	TMT-1-3	SOT-6-2	SOT-8-2	TMT-1-2	SOT-8-3	SOT-6-3	SOT-1-4	OT-1-7	SOT-1-2	SOT-9-2	OT-1-4
H-2.6	11/10/64	12/1/64	TMT-1-3	SOT-6-2	SOT-8-2	TMT-1-2	SOT-8-3	EMT-2(Zr)	SOT-1-4	SMT-1-3	SOT-1-2	SOT-9-2	RMT-1-2

(a) Position 41 was occupied by the boiling-cooled isolated coolant loop and was never used for fuel heating.
(b) Position occupied by the liquid-cooled isolated coolant loop.
(c) H-1.0 started 3/3/62 and ended 9/7/62. Zero-power fuel for flux measurements; drivers only for initial power operation.
(d) Fuel element failure.
(e) Gamma heater tube.
(f) Housing tube.

TABLE II

Test Fuel Identification

Designation	Shape	OD, in.	ID, in.	Unit Length, in.	No. Units	Fuel
CANDU	19-rod bundle	0.600 each	-	$19\frac{1}{2}$	5	Natural UO_2 sintered pellets in Zircaloy-AECL
TWNT	2-conc-tubes	2.06 1.02	1.70 0.66	118	2	Two thin-walled tubes of unalloyed natural metal
SOT-1	Tube	2.06	1.47	14	7	1.5% enriched UO_2 vibrated and swaged in Zircaloy
OT-1	Tube	2.06	1.47	118	1	1.5% enriched UO_2 vibratory compacted in Zircaloy
TWO-1	Tube	2.06	1.70	113	1	Thin-walled tube of unalloyed natural uranium metal
SOT-2	Tube	2.12 2.15	1.16	$13\frac{3}{4}$	8	Natural UO_2 vibrated and/or swaged in Zircaloy
ETWO	Tube	2.06	1.70	120	1	Unalloyed 2.1% enriched metal tube
SMT-1-2	Tube	1.70	1.24	$11\frac{1}{4}$	10	Natural uranium alloyed with Fe, Al in Zircaloy
OT-3	Tube	2.06	1.47	118	1	Single tube of vibrated and swaged natural UO_2
SOT-6	Tube	2.54	1.83	$14\frac{1}{4}$	7	Natural UO_2 vibrated and swaged in Zircaloy
RMT-1	Tube	2.07	1.57	120	1	Unalloyed natural uranium in 60-mil Zircaloy cladding
EMT-2	Tube	2.06	1.70	37	1	3% enriched uranium alloyed with 1.5% Mo in Zircaloy
OT-9	Tube	2.54	1.83	$14\frac{1}{4}$	7	1.2% enriched UO_2 vibrated and swaged in Zircaloy
TMT-1	Tube	2.55	1.85	98	1	1.4% ^{235}U in thorium core in Zircaloy
SOT-8	Tube	3.67	2.99	13	7	1.2% enriched UO_2 vibrated and swaged in Zircaloy
SMT-1-3	Tube	1.70	1.24	$11\frac{1}{4}$	5	Natural uranium alloyed with Fe, Al, Si in Zircaloy
SOT-5	Tube	2.125 2.150	1.065 1.080	$14\frac{1}{4}$	8	Natural UO_2 vibrated and/or swaged in Zircaloy

Reactor Cycles H-1.1, 1.1a, 1.2 (Figure 8)

Reactor operation for cycle H-1.1 began on October 5, 1962, and terminated on November 27. Reactor innage during this cycle was 46%, and total reactor exposure was 715.8 MWD. The following five causes accounted for the 24.4 days of outage during the cycle.

1. Replacement of a defective mechanical oil seal in the No. 2 circulating pump took approximately three days. A defective O-ring around the stationary face of the seal caused the failure.

2. Operation was delayed for about fifteen days because of two deficiencies in the monitor pin thermocouples: (1) Water was found inside the thermocouple sheath. The source of water was not found but presumably was introduced in the manufacturing process. During temperature operation, this water leaked from the end of the sheath and wet the terminal connectors. (2) Loose or disconnected thermocouple junctions were found in five of six thermocouples that were examined. The junction had been made by silver soldering the thermocouple wires to a stainless steel cap and then welding the cap to the sheath. The heat from the welding caused the wires to loosen in the cap. The defective thermocouples were replaced, and new thermocouples of a better design were ordered.

3. The glass face in a rotameter in the high pressure system failed during nuclear operation of the reactor at 1 MW, 133°C, and 1200 psig. The rotameter was rated at 1500 psig and 490°C. The failure was attributed to mechanical stresses introduced during maintenance work on the rotameter a few days before the failure. Approximately 4100 pounds of D_2O was lost from the system, of which 3200 pounds was collected as degraded material. Cleanup and minor revisions to the piping to eliminate all glass rotameters from the high pressure system took 1-1/2 days.

4. Failure of the helium recompressor to maintain system pressure, and subsequent maintenance time, caused three days lost time. Performance tests with bronze and glass-filled "Teflon"* rings, carbon-filled "Teflon" rings, and glass-filled "Teflon" rings were undertaken. The glass-filled "Teflon" rings gave the most satisfactory performance and were installed on the fourth stage of the two compressors.

5. Approximately 0.5 day was lost from a scram caused by an operating error during maintenance work on instrumentation.

*Trademark of E. I. du Pont de Nemours and Co. for fluorocarbon resins.

Cycle H-1.1 was terminated on November 27, 1962, by the first fuel failure in the test program. The test element contained two thin-walled tubes of unalloyed natural uranium metal. Uranium released to the process system was quite small, and no contamination or cleanup problem resulted. The failed element was replaced by an identical assembly.

Reactor operation for cycle H-1.1a began on December 2. Reactor innage during this cycle was 93%, and total reactor exposure was 222.7 MWD. The lost time of approximately 0.5 day during this short cycle was caused by a scram from a spurious temperature signal and by binding in a control rod drive unit. The cycle was terminated on December 9, when analyses for tritium in the cooling water indicated a D_2O leak of about 5 lb/day from the No. 1 steam generator. Testing and attempted repairs to this generator took about 10 days. "Freon"* and soap bubble tests showed many leaks between the tubes and the tube sheet, mostly at porosities that existed in the seal welds. Eleven tubes were rolled to a depth of four inches in the tube sheet. Subsequent tritium analyses indicated that the leakage was about 2.5 lb/day.

During the shutdown for steam generator repairs, a natural uranium metal element, similar to the one that failed on November 27, was removed for inspection and was replaced by a UO_2 test element. Cycle H-1.2 was begun on December 19, and was terminated on December 26 by failure of the second test fuel assembly of uranium metal. Reactor innage during the cycle was 93%, and reactor exposure was 161.4 MWD. As out-of-pile investigations of the first failure indicated that the twisted ribbon spacer employed on these two failed assemblies was the most likely cause of failure, the remaining five test elements employing this spacer concept were discharged and replaced by UO_2 test elements. Two manual scrams accounted for the approximately 0.5 day lost time during the cycle. The first was required when the feedwater control system did not respond during the temperature ascension on December 19, and the second occurred when a cluster rod was found unlatched and in the full-in position.

After charging of the new test fuel was completed, a zero-power critical was conducted on January 4 to measure the reactivity worth of the test charge.

*Trademark of E. I. du Pont de Nemours and Co. for fluorinated hydrocarbons.

Reactor Cycles H-1.3, 1.4, 1.5, 1.6 (Figure 9)

Reactor cycle H-1.3 began on January 6, 1963, and terminated on April 3. Reactor innage during this cycle was 95%, and reactor exposure was 3095.8 MWD. Power operation during this cycle was interrupted by ten shutdowns, which account for the 3.5 days lost during the 87 day period. Causes for the shutdowns were:

1. Four shutdowns were necessary for work on the steam generator level control system. This instrument had given considerable trouble since the inception of operation. The original control system, a water level transmitter coupled directly to the feed-water valve resulted in power oscillations because of the close coupling between reactor power- moderator temperature - heat removal capacity of the generators. This system was modified in early January, after three shutdowns within three days, by the installation of a cascade (two-element) control system. In this instrument system, the level indication from the steam generator adjusted the set point of a constant flow controller on the feedwater valve. In this manner, the direct coupling between the level in the steam generator and the opening of the feedwater valve and the attendant oscillations were eliminated. This cascade system was not satisfactory for startup and shutdown because of the range of flows and temperatures encountered, but when used in combination with an option to revert to the original level control system, satisfactory operation was realized.

2. Three shutdowns were caused by spurious alarms on the temperature monitor. One was the result of improper setting of the alarm point; the other two were caused by external voltage fluctuations, such as switching motor-generator sets which reflected through the power supply for this instrument. Improved power supplies were installed for this instrument.

3. The remaining three shutdowns were caused by: excessive dissolved helium in the moderator during the scram recovery on February 8, need to transfer irradiated fuel assemblies from the basin in the containment building to an inspection facility at another site, and a voltage surge on the incoming power line when a Plant 115-kv power line broke.

The cycle was terminated on April 3, when a manual scram was initiated because of high temperature in a bearing in the No. 2 AC motor. The bearing failure was attributed to a combination of sludge in the oil reservoir and insufficient clearances between the rotor and the face of the upper thrust bearing.

Approximately three hours after the shutdown for the bearing failure, an appreciable release of radioactivity in the process water was observed. As the activity release did not occur until after normal process water flow had been stopped, some difficulty was experienced in locating the failed element. The failure was located by stopping all flow through the fuel assemblies, thus allowing the element to heat up from decay heat, then re-establishing a small flow through the elements and quickly sampling the fuel effluent from each test fuel assembly. Five such attempts were necessary before the failure was located.

After discharge of the failed element, it was decided to remain shut down for a series of scheduled tests, equipment modifications, and test fuel inspections. This scheduled work took place over the next forty-five days and is delineated briefly in Table III.

TABLE III

Major Shutdown Jobs, H-1.3-1.4 Outage (4/3 - 5/18/64)

	Major Jobs	Remarks
1.	Discharge and inspect three test elements	New test elements (2 U-metal and 1 UO_2) were installed
2.	Install 34 monitor pin thermocouples	Thermocouples were calibrated in place by non-nuclear heatup to 200°C
3.	Install iodine absorbers in containment building	Details on these absorbers are in references 1 or 5
4.	Inspect ICL pressure tube and charge first test element to this loop	This fuel element was not part of the fuel test program but was for startup of loop operation
5.	Repair leaking tube sheet in steam generators	Attempted repair of 39 leaks by rerolling tubes and peening seal welds
6.	Install new or improved instrumentation for flux monitoring and feedwater control	Included a third log N-period system, a new fission counter, and replaced temporary feed-water control system with permanent instrumentation
7.	Chemically clean shell side of steam generator	Successfully cleaned with 5% HCL-1% HF
8.	Inspect equipment, including rod drive assemblies, transfer coffin, and deluge tank	Deluge tank severely pitted. Cleaned mechanically and refilled with water treated to inhibit corrosion
9.	Functional test poison injection system	Satisfactory
10.	Annual leak rate test containment building	Final leak rate was 0.76% of building volume per day at 5 psig

Nuclear operation was resumed on May 19, and cycle H-1.4 was terminated 13.5 days later by failure of an oxide test element. Reactor innage during the cycle was 87.5%, and reactor exposure was 368.4 MWD. Four scrams occurred during the cycle: (1) two were caused by the new cascade level controller on the steam generator - minor adjustments were made in the gain and time constants of the controller network, (2) spurious low cooling water flow signal, and (3) an operating error in placing a scram instrument on-line after maintenance but before resetting the scram relay.

The test element failure on June 1 occurred during the temperature ascension following recovery from a scram on May 31.

The failed assembly was replaced and cycle H-1.5 began on June 6, but was terminated less than 14 hours later by failure of another oxide test element during the initial power ascension for the cycle. Residual activity in the moderator from the failure on June 1 obscured the positive identification until the reactor power had reached about 10 MW. Initial activity increases began with the reactor power below 1 MW.

Cycle H-1.6 began on June 11, but on June 20 operation was again terminated by the failure of an oxide test element. One scram, a spurious signal from a reactor relief valve, occurred during the second day of this cycle. Reactor exposure was 263.6 MWD, with an innage of 96% for the 8.3-day cycle.

When an attempt was made to discharge the failed oxide element, the top fitting of both the fuel piece and the housing were found to have failed. This assembly was charged to the reactor for the H-1.3 cycle in January 1963. It was discharged to the spent fuel basin in April 1963, to gain irradiation space for other test assemblies. Following the oxide tube failure on June 1, 1963, this assembly was recharged to the reactor to continue its irradiation, which began on June 6. During the recharging, the bottom fitting of the housing caught on a receptacle in the spent fuel basin. An inspection, after freeing the element, did not reveal any apparent damage, and it was charged without further incident. In retrospect, this incident was probably the cause of the failure of these top fittings. Special tools that could be inserted the full length of the assembly and gripped at the bottom were designed and fabricated during the following ten days. After testing in a complete dummy run of the discharge operation, the fuel piece and the housing were discharged, separately, without incident. This work was completed on July 5.

The broken top fitting on the fuel prompted inspections of other test assemblies of similar design. The fuel top fitting and top fuel piece of each of four segmented assemblies were replaced by dummy sections because some damage was noted. The damage was attributed to hydraulic vibration. Two full-length oxide assemblies were discharged to permit detailed inspections where wear marks were found in the cladding near the tops of the fuel pieces.

Reactor Cycles H-1.7, 1.8, 1.9, 1.9a, 1.9b (Figure 10)

Initial criticality for cycle H-1.7 was attained on July 13, and the cycle was terminated on August 25, after a reactor exposure of 1123.7 MWD. Reactor innage during the cycle was 80%. Seven shutdowns occurred during the cycle, to account for the 8.6 days of lost time. Causes of these shutdowns were: one shutdown at the beginning of the cycle to bring the dissolved oxygen in the moderator below the operating limit of 0.1 ppm; three shutdowns from spurious alarms; and three shutdowns to investigate excessive D_2O losses (40-50 lb/day).

Part of the investigation for sources of D_2O loss was accomplished by inspection of the system at 200°C and 1100 psig immediately following a nuclear shutdown. Many minor leaks were discovered under these conditions. Most of these leaks were from valve packing glands, monitor pin nut assemblies, and instrument fittings in the high pressure system.

During startup preparations, following the shutdown on August 25, a problem was encountered with dissolved helium in the moderator. The dissolved helium content was about 1.5 times its normal value, and changes in system pressure were followed by rapid changes in the dissolved helium content in the moderator. The source of the difficulty was a failure of the unused boiling loop pressure tube. Under full hydraulic flow, the pressure drop across the reactor core was sufficient to cause all of the D_2O in the pressure tube to flow through a hole that had been worn through the bottom of the pressure tube, thus connecting the gas space in the neck of the reactor with the core region of the reactor. The remainder of August and the first three weeks in September were required to discharge the failed pressure tube. Discharge of the pressure tube, which was about 26 feet in length, required that it be cut in half to allow discharge of the in-core portion by the shielded transfer coffin. Inspection of the pressure tube at an off-site facility a few weeks later showed that a ragged hole about 3/32 inch wide had been worn through the one-inch-thick bottom forging of the tube. The end bushing was found separated from the pressure tube and badly worn. Apparently the bushing had been vibrating and rotating over a long period of time, perhaps since its original installation. This vibratory motion of the bottom of the pressure tube had caused an empty fuel tube housing in the pressure tube to wear against the inside of the bottom forging until a jagged rib on the housing tube wore through the forging.

Advantage was taken of this extended outage to perform scheduled inspections of some of the test fuel assemblies and to prepare an experiment for the liquid-cooled isolated loop to permit measurement of the gamma heating rate in a loop position.

Nuclear operation for the H-1.8 cycle began on September 22, but was interrupted during the temperature ascension by an abnormal effluent temperature from the liquid loop purge cooler. Inspection showed that the shell side of this cooler was plugged with slime. The cooler was flushed with a chlorine solution. This was the first difficulty resulting from slime since sterilization of the CW system during the April shutdown. Operation was resumed on September 23, but was terminated on September 28, when a flux monitor trip occurred during an electrical storm. Loss of an off-site power line caused a voltage surge through the power supply to the area. Reactor exposure for this cycle was 136.4 MWD, with an innage factor of 63%.

Approximately 2-1/2 hours after reactor shutdown, a general rise in gas activity in the reactor building and in the process system indicated a fuel element failure. Sample analyses of coolant effluent from the various test assemblies, using the "no-flow" method (described on page 25), confirmed the failure and its location. The element was a natural uranium oxide tube that had attained a maximum specific exposure of 1620 MWD/ton. The element was replaced by October 2.

Cycle H-1.9 began on October 3, and except for one brief interruption shortly after startup to lower the dissolved oxygen content in the liquid loop moderator, continued until it was terminated on October 29 by failure of the test element in the isolated loop. Cycle innage was 96.5%, with a reactor exposure of 1106.9 MWD.

The element that failed was made up of short tubes of enriched and natural uranium oxide and was an exploratory test of high heat ratings. At the time of failure, element and reactor operating conditions were:

Reactor power, MW	53
Moderator temperature, ^{o}C	200
Element max spec power, MW/ton	49
max spec exp, MWD/ton	1075
max heat flux, pcu/(hr)(ft^2)	507,000
max $\int k d\theta$, w/cm	65

Failure instrument readings increased from normal to full scale within a few seconds after the failure. Activity releases and radiation levels in the reactor building were higher than from any previous or subsequent fuel failure at the HWCTR. A total of 6000 curies of Xe (133 and 135) and 0.012 curie of I-131 were released to the atmosphere via the ventilation exhaust stack. Several surveys of radiation from the liquid loop piping revealed "hot spots" up to 500 R/hr at 1 inch.

Maximum cleanup flow through the loop purification system was begun immediately and continued through most of November. Some activity breakthrough and pluggage of deionizers were experienced, requiring replacement of the deionizers on two occasions. Except for three pockets of high activity in low velocity areas or dead legs of the piping, loop activities were near normal by November 18.

To permit further cleanup of the loop during nuclear operation, an empty housing tube was charged to the pressure tube to define the normal flow channel. Nuclear operation was resumed on November 18, but was shut down on November 19 to remove the housing from the loop pressure tube. Results of the inspection of the boiling loop pressure tube that was discharged in September became available at this time and showed that the empty housing tube it had contained contributed to the failure. Although there was no evidence of similar vibrations in the liquid loop pressure tube, the housing tube was discharged as a precautionary measure.

Cycle H-1.9b began on November 23 and operation at powers up to 53 MW continued until November 30, when one of the cooling water wells failed. (The pump had been in service ~12 years.) The pump casing broke just above the pump bowl and allowed the casing to contact the rotating pump impeller. The pump was badly damaged and required replacement before resumption of nuclear operation.

Evidence of a fuel failure was observed immediately after shutdown on November 30. The radionuclides present in the moderator were different from previous failures in several respects. The amount of fission products Xe and Kr were much less than previous failures. The primary fission product was iodine. No ^{239}Np was detected, although all other failures that released appreciable iodine to the moderator also released ^{239}Np. These differences tended to exclude from consideration all fuel elements except the driver elements, which contained enriched uranium only. Several attempts to locate a failed driver or test element by sampling the effluent from each element in the reactor were unsuccessful.

A late delivery date of December 20 for the replacement well pump together with the possibility of a failed driver element led to the decision to terminate the first driver cycle. Accumulated exposure on the first driver charge was 6268 MWD, with a peak burnup of 1.826 atom % fission (goal was 2.0%). Total reactor exposure was 7728 MWD.

Three low-power critical tests were made to measure the reactivity condition of the old and the new lattice, and to measure the neutron flux distribution in several fuel elements. These tests were conducted (1) on December 5, with the lattice components existing at the end of the H-1.9b cycle; (2) on December 10, with the same fuel components as in Test No. 1, but with new control rods; and (3) on December 22, with new driver element and scheduled changes in the test fuel lattice.

Other major work items accomplished during this shutdown were: (1) discharge and examination of the Sb-Be source rods, (2) inspection of three test fuel assemblies, (3) in-core examination of the liquid loop pressure tube, (4) annual 1500 psig pressure test of reactor vessel, (5) continued efforts to repair tube-to-tube sheet seal welds; 59 seal welds were repaired by peening over the welds, (6) replacement of the failed CW pump, and (7) installation of the permanent model of the continuous temperature monitor. (Details of these tests are described in the reports listed in reference 2.) The new model had more stable and accurate electronic components and replaced a temporary model in use since September 1962.

Reactor Cycles H-2.1, H-2.2 (Figure 11)

Reactor operation for the H-2.1 cycle began on December 29, 1963, but was interrupted on January 1, 1964, by difficulties with the feedwater control system. During the subsequent startup, evidence of helium aspiration into the reactor core was observed. The center control rod cluster was found to contain a rod that had not been modified to prevent gas aspiration. New rods had been charged to the cluster during the December shutdown. The proper rod was charged and startup was accomplished on January 7. Operation continued until February 4, when indications of a D_2O leak into the cooling water of the No. 1 steam generator necessitated a shutdown. Because subsequent fuel changes were made, this terminated the H-2.1 cycle. Cycle innage was 84%, and total reactor exposure was 1038.5 MWD.

Because previous experiences had shown that locating a heat exchanger leak was facilitated if the vessel was dry, the shell side was vacuum dried under heat for about 3-1/2 days. With the shell side pressurized to 400 psig with nitrogen, a failed tube was located easily by installing solid rubber stoppers in one side of the U-tubes and stoppers with holes in the other ends of the tubes. The location of the leak along the tube was determined by pushing a plug into the tube until leakage reoccurred. Although the tube was relatively accessible from the secondary side, the hole could not be located visually. As the proper tube removal tools were unavailable at the time, plugs were welded in both ends of the U-tube.

In addition to the tube leak, 63 leaks were found in the tube-to-tube sheet seal welds. Of the 63 leaks, 43 had been observed during the repair efforts in December 1963, and 15 had been repaired by peening the welds at that time. Ten of the 63 leaks were peened shut.

After completion of other scheduled shutdown work, a nonnuclear heatup was begun on February 23 to obtain isothermal calibration data on system thermocouples. The test was discontinued on February 24, at 165°C moderator temperature, when the D_2O flow through the system suddenly decreased. Inspection showed that pieces of plywood, "Masonite"*, and plastic were caught on the fuel element screens.

The investigation showed that following repairs to the steam generator earlier during the month, a line block had been left unintentionally in the D_2O effluent pipe of the generator. When flow was resumed in the system, the plug was swept through the ten-inch line between the generator and reactor inlet and lodged against a venturi meter in the line. The plug was undetected during operation with cold flow and during the first part of the nonnuclear heatup. When the moderator temperature reached about 165°C, the plug partially disintegrated and fragments were swept through the venturi into the reactor. Most of the fragments settled out in the upper part of the reactor. Screens over each fuel assembly filtered out much of the debris. Smaller particles rendered the entire moderator turbid and orange in color. In addition, the moderator became contaminated with dissolved organics.

Since the incident occurred during nonnuclear operation, no direct hazard was involved. Assembly flow reductions were not so severe that flow was inadequate for shutdown cooling.

Most of the debris was removed from the system with the initial cleaning of the muff screens. Several subsequent screen cleanings were made after the reactor was operated at full flow for several hours. U-bolts that held the line block together were removed from between the tank wall and the inlet baffle with a magnetic probe. Additional material was removed with an underwater vacuum cleaner. The initial mass of the line block is compared with the mass recovered from the reactor in Table IV. An unknown amount of the debris collected on the deionizers in the purification system and is not included in Table IV.

TABLE IV

Debris Removed from System After Line Block Incident

Material	Original Mass in Plug,[a] grams	Mass Removed from System,[b] grams	Percent Removed
Plywood	480	306	64
"Masonite"	180.7	118	65
Tygon tubing	67.3	28.7	43
Steel bolts, nuts, washers	2 units	2 units	100

(a) Based on a similar plug of an original pair.
(b) Weight after drying. This weight does not include any allowance for (1) binder agent in original plywood, (2) differences in moisture content, or (3) loss due to charring.

*Trademark of Masonite Corp.

Contaminants were removed from the moderator by filtration of a 10 to 15 gpm purge stream for several days. The stream was passed first through a DOD resin bed which neutralized the normally basic moderator and rendered the dissolved organics filterable. The organics and the initial turbidity were then satisfactorily removed by one-micron filters.

The final phase of the cleanup was conducted during another non-nuclear heatup which began on March 16. Objectives of this test were to complete the thermal breakdown of any organic material remaining in the system and to collect the isothermal thermocouple data begun on February 23. The turbidity and organic content did not increase during the heatup to 200°C; however, during the cooldown on March 19, a pump seal in the isolated coolant loop failed, and the subsequent chromatographic analysis showed about 1000 ppm organics. The cleanup of this material was accomplished satisfactorily. By March 26, there had been no detectable turbidity, and the concentration of organics was <1 ppm. All purification equipment used during the cleanup was then replaced.

The capacity of the leakage collection system was exceeded during the seal failure on March 19. Approximately 1530 pounds of 100% D_2O at about 90% purity were recovered from the building pumps. D_2O loss was about 500 pounds. Inspection of the seal assembly showed some foreign debris, and slight scoring of the seal faces and the base of the throttle bushing. Exact cause of the failure was not established but was thought to be failure of the seal ΔP control valve. Cleanup and seal replacement were completed by March 27.

Preparation for resumption of nuclear operation, which included recharging fuel removed during February and March, was completed by April 2.

During routine drop tests of the safety rods on April 2, conducted as part of the nuclear startup, the drop times of one safety rod were unusually short and were followed by a loud mechanical noise when the rod bottomed. These symptoms indicated that normal mechanical and hydraulic snubbing were not occurring. Inspection of the in-core section of the Zircaloy guide tube showed that the tube was split and broken off where the tube diameter was reduced to provide hydraulic snubbing. Subsequent testing and inspection showed that one other guide tube was split, and all six guide tubes were replaced. Detailed examinations then revealed that two more of the guide tubes were split. All of the splits originated at the reduced diameter section of the guide tube.[6] Primary cause of the failures, after two years of use, was attributed to low-cycle fatigue caused by the strain pulse from the pressure cycle during rod drops at elevated temperatures. Residual stresses of 13,000 - 28,000 psi in the outer tubing surface contributed to the failures. Detailed results of this investigation are reported in DP-971.[6] Replacement of the guide tube was completed by April 13.

During the guide tube work, inspections in the reactor vessel revealed that two of eight bolts that held the support legs for the gas baffle had broken. Because the gas baffle had been installed in the neck of the reactor vessel after the initial low power tests showed gas entrainment in the core coolant, the baffle was not attached to the vessel wall but was mounted on four support legs that were bolted to the top of the flow distribution baffle (see Figure 3).

The support structure for the gas baffle was removed and replaced with one of increased strength and rigidity. This work involved removing nearly all of the reactor internals down to the reactor top shield; this was accomplished while the irradiated fuel remained in the reactor. Radiation up to a peak of 3.5 R/hr was reduced to 150 mr/hr, or less, by using a lead-lined basket from which the work was performed and by hanging lead strips around the inside wall of the reactor. Metallurgical examination of the failed 17-4PH stainless steel bolts showed that the failures were probably caused by low-ductility rupture from overstressing during installation, in combination with thermal stresses during operation.[7] This work was completed by May 4, and reinstallation of reactor internals was completed by May 10.

Nuclear operation was resumed on May 11. The total lost time caused by this succession of difficulties was 96 days. A program that included a slow power ascension was planned for the H-2.2 cycle to return the reactor to normal full power conditions. The objective was to obtain assurance at each of several increasing levels of power that the process system was clean of debris from the line block incident. Also, the radiation intensity in the reactor core at these intermediate powers was sufficient, during the scheduled operating times, to decompose extensively any pieces of organic materials that might have lodged in the fuel assemblies. The first power plateau, 5 MW, at 200°C moderator temperature, was attained on May 11. Approximately 19 hours after reaching this plateau, the reactor was shut down when fuel element ΔP (flow) measurements indicated an 8% flow decrease in one of the driver elements. The remains of a lead pencil were found on the muff screen above this element. No other debris was found on other muff screens or on the top shield.

Prior to the above shutdown, erratic leakage through the No. 1 main pump seal occurred, ranging from a few drops per minute to about 3/4 gpm. Because of the erratic leakage, the high leakage rate, and the long period of service on this seal (approximately 19,000 hours since its installation in January 1962), the seal was replaced. Examination of the seals revealed some circumferential scoring in the bronze and tungsten carbide faces.

Criticality was attained on May 17, but the reactor was again shut down on May 19, from a power of 12 MW when indications of a

200-lb/day tube leak occurred in the No. 2 steam generator. The generator was not heated and vacuum dried as during previous repairs. Both the primary and secondary systems were drained and the shell was pressurized with N_2. The leaking tube was located readily. The location of the leak along the tube was approximately adjacent to the location of a baffle plate but interference from a cooling water inlet sparger prevented observation of the leak. Attempts to remove the tube for inspection were unsuccessful as the cutting tool broke off inside the tube and could not be removed. The tube was plugged and seal welded.

Nuclear operation was resumed on May 29, and a power of 23 MW was attained on May 30. The H-2.2 cycle was terminated on May 31, by failure of an oxide test element. As the initial power ascension for the cycle was never consummated, this particular element, which was charged during the first part of May, had been in the reactor less than one full power day. Inspection showed the cause of failure to be a fabrication defect. During this 19.8-day cycle, innage averaged 22%, and total exposure was only 30 MWD.

Reactor Cycles H-2.3, 2.4, 2.5, 2.6 (Figure 12)

Nuclear operation for the H-2.3 cycle began on June 10, and reactor power was stabilized at 28 MW. This power level was scheduled for 48 hours as part of the resumption of operation following the line-block incident. A lightning strike on the 13.8 kv feeder line to the area caused an electrical power interruption and a reactor scram on June 12.

Following the electrical failure, the No. 2 pump in the isolated coolant loop could not be restarted. All six shoes of the upper thrust bearing were severely wiped, and the three shoes of the lower thrust bearing were lightly scored. The bearing was replaced and the thrust collar was refinished. The cause of the failures was not determined.

Operation was resumed on June 18, and continued at full reactor power of 42 MW until July 13, when the reactor was again shut down because of a tube failure in the No. 1 steam generator. Immediately following shutdown, system radioactivities indicated that a fuel element had failed, thus terminating the H-2.3 cycle. This cycle had operated with an 84.7% innage for an exposure of 1066.6 MWD.

In retrospect, an analysis of the records of the failure detection systems indicated that the failure may have occurred as a small defect during the first part of the H-2.3 cycle. Radionuclide and gamma spectrum analyses had indicated a neptunium peak as early as June 22, but, since the peak remained constant, the reactor was not shut down. Although the failure locating system showed four identifiable peaks from the same reactor position in the hour following shutdown, the indexing mechanism malfunctioned and indexed to an unknown point; hence, the identification of the failed position was unknown. Six no-flow heatup tests were run in attempts to locate the failed element. The results were not definitive but showed two elements as most suspect. These two elements were discharged and placed in isolated containers filled with uncontaminated water. During discharge, a gas activity monitor associated with the transfer coffin vent increased by a factor of 40 for one of the elements. Water samples from the isolated containers confirmed this result. The failed element was an oxide rod bundle being irradiated under the USAEC-AECL Cooperative Program.

Tritium analyses of the generator blowdown at the time of shutdown on July 13, indicated a maximum leak rate of 17 lb D_2O/day. Numerous attempts to locate the leak by pressurizing the shell with nitrogen were unsuccessful. The defective tube was located by adding "Freon" to the nitrogen and using a halide detector. The tube was plugged and seal welded.

As this was the third tube failure within a six-month period, a detailed inspection of the secondary side of both steam generators was made. Inspections during February 1964, following the first tube failure, had shown only minor pitting, with a maximum pit depth of about 15 mils. These pits appeared to be healed and no tubercles were present.

The July 1964, inspections revealed severe pitting corrosion. The maximum pit depth measured was 85 mils, and a number of pits 40 mils to 60 mils deep were found. The tube walls were originally 109 mils thick. These pits were covered with tubercles (corrosion products) indicating that the pits were active.

Treatment of the cooling water had always been entirely by chemical addition, using catalyzed sodium sulfite for removal of dissolved oxygen from the water supplied by deep wells in the area. A review of performance data showed that the sulfite residual had always been maintained greater than 10 ppm. However, the particular brand of sulfite in use since October 1961, had been changed early in 1964. Tests of the efficacy of the two brands of catalyzed sulfite showed a slow reaction time for the sulfite used during 1964, and it was concluded that this sulfite failed to remove dissolved oxygen from the feedwater with sufficient rapidity to minimize corrosion in the steam generators. Use of the original brand of sulfite was reinstituted. This problem is discussed in more detail in DP-964.[8]

Nuclear operation for the H-2.4 cycle began on July 26, and continued without interruption until terminated on August 7, for a scheduled outage to perform driver and test fuel inspections, to perform the annual containment leakage test, and to continue the inspection of and repairs to the steam generators. Exposure for the cycle was 471 MWD.

Inspection of the steam generators during the July shutdown had indicated that a possible cause of previous tube failures, besides the oxygen pitting attack, was the feedwater inlet sparger. This sparger was slightly bowed and pressed against the tube bundle in both generators. A redesigned sparger, smaller in diameter, was installed in both generators. A section of the tube that failed on May 19, was removed and inspected. This failure was caused by crevice corrosion under a support baffle.

Other work on the secondary coolant system accomplished during this outage included cleaning and sterilizing the miscellaneous heat exchanger system to reduce the bacterial count and slime problem that had periodically plagued this system.

The annual leakage rate test on the containment building was conducted on August 15 and 16. The observed leak rate was 0.63% of the building content per day at a pressure of 5 psig. This rate is about 17% less than that measured in 1963. A search for leak sites was conducted continuously during the test. Only one major leak was found and this was repaired. Details of this and other containment leak rate tests are reported in DP-968.(5)

Nuclear operation for the H-2.5 cycle was begun on August 25, and continued without interruption until September 9, when the reactor was shut down because of another tube leak in the No. 1 steam generator. Indicated leakage rate was 75 lb/day. The tube was located and plugged without difficulty. No further inspections were made of the secondary side of the generators because they were nearing their end-of-life and plans to order new tubes and tube sheets were in progress.

Operation was resumed on September 15, and continued until September 30, when the reactor was scrammed by noise spikes on the flux monitors. The noise spikes derived from a lightning strike on an intra-area electrical feeder line. Xenon override was unsuccessful and the reactor remained shut down for about 20 hours. Operation then continued at 51 MW until October 7, when the reactor was shut down to investigate a D_2O leak. The leak was indicated by an increase in the tritium activity in the containment building. The leak occurred at a body plug in a check valve and was repaired by torquing and seal welding the plug.

Operation was resumed on October 8, but was interrupted on October 12, when the reactor was scrammed during routine maintenance on a scram instrument. During the scram recovery, two more scrams occurred, caused by operational errors during adjustments to scram instruments.

The reactor was returned to power on October 14, and operated at 50-54 MW until the end of the H-2.5 cycle on October 31. The scheduled outage was to allow inspection of fuel components and to replace the poison targets in the driver fuel with Zircaloy flow guides to extend the lifetime of the charge.

The H-2.5 cycle operated at 86.5% innage over a 70-day period for a total reactor exposure of 2695 MWD.

Fuel inspections and target replacement were completed in a ten-day period, and operation of the H-2.6 cycle began on November 10. Operation for this cycle was continuous except for one interruption on November 25, when the reactor was shut down by an interruption in the electrical service to the area. During the recovery, a second shutdown was necessary to replace a defective relay in one of the control rod drives.

The intention had been to operate the H-2.6 cycle to the end-of-life of the second driver charge. This was predicted to occur about mid-December, if this innage were 100%. Operation of the HWCTR was terminated on December 1, 1964, as part of the curtailment by the U. S. Atomic Energy Commission of the development of D_2O-moderator reactors that are cooled with liquid D_2O. The H-2.6 cycle had operated at 99% innage for a reactor exposure of 853 MWD. An irradiation of flux mapping wires at 10 kw power was made on December 2, and then the reactor was shut down for deactivation.

Accumulated exposure on the second charge of driver fuel elements was 6154 MWD. Total lifetime exposure for the HWCTR was 13,882 MWD.

The remainder of December and January 1965, was used to place the HWCTR facility in a standby condition for possible future reactivation. All of the fuel assemblies and the neutron sources were removed from the reactor and stored at another Plant facility. Other reactor components, including control rods, safety rods, and corrosion coupons were left in the reactor core.

The high pressure part of the primary system was drained and then was vacuum dried to remove the D_2O. This system and the low pressure D_2O system were filled with nitrogen to minimize corrosion during the standby state.

The secondary coolant system was drained and left at ambient conditions. The shell side of the steam generators and purge coolers was filled with nitrogen.

A description of the standby condition of the HWCTR is reported in reference 9.

SYSTEMS AND EQUIPMENT

A detailed description of the systems and equipment for the HWCTR is given in reference 1. This section reports on the performance of these systems and equipment over the approximately three years operation of the facility. A brief description of the system or equipment piece and a schematic diagram are included here where such description aids the report on performance.

REACTOR ASSEMBLY (Figure 14)

Description

The reactor is a pressure vessel, approximately 30 feet high, made of carbon steel with 1/4-inch, type 304 stainless steel cladding inside. The vessel was manufactured per ASME specifications SA-212 for a pressure of 1500 psi at a temperature of 315°C. The walls vary in thickness from 3-1/2 to 5-1/2 inches. The vessel consists of a lower 7 feet-2 inch ID shell section, an upper 5 feet ID shell section, a conical transition section, a hemispherical welded-on bottom head, and an ellipsoidal bolted-on top head. The head flange is secured to the shell flange by thirty-two 3-1/4-inch studs, washers, and nuts. A double "Flexitallic"* gasket seal, with leakoff between the gaskets, is used between the head flange and shell flange. The control and safety rod drives are mounted above the reactor head on the top drive platform. Permanent internal components include: (1) top indexing shield plug, (2) shield plug support ring, (3) top guide plate, (4) gas baffle, (5) top shield and flow distribution baffle, (6) horizontal thermal shield, and (7) side thermal shield. The equipment on the top drive platform includes four bolt tensioners for reactor head bolt tensioning and four worm-gear jacks, driven by an air drill, for lifting the reactor head-rod drive assembly.

*Trademark of Flexitallic Gasket Co.

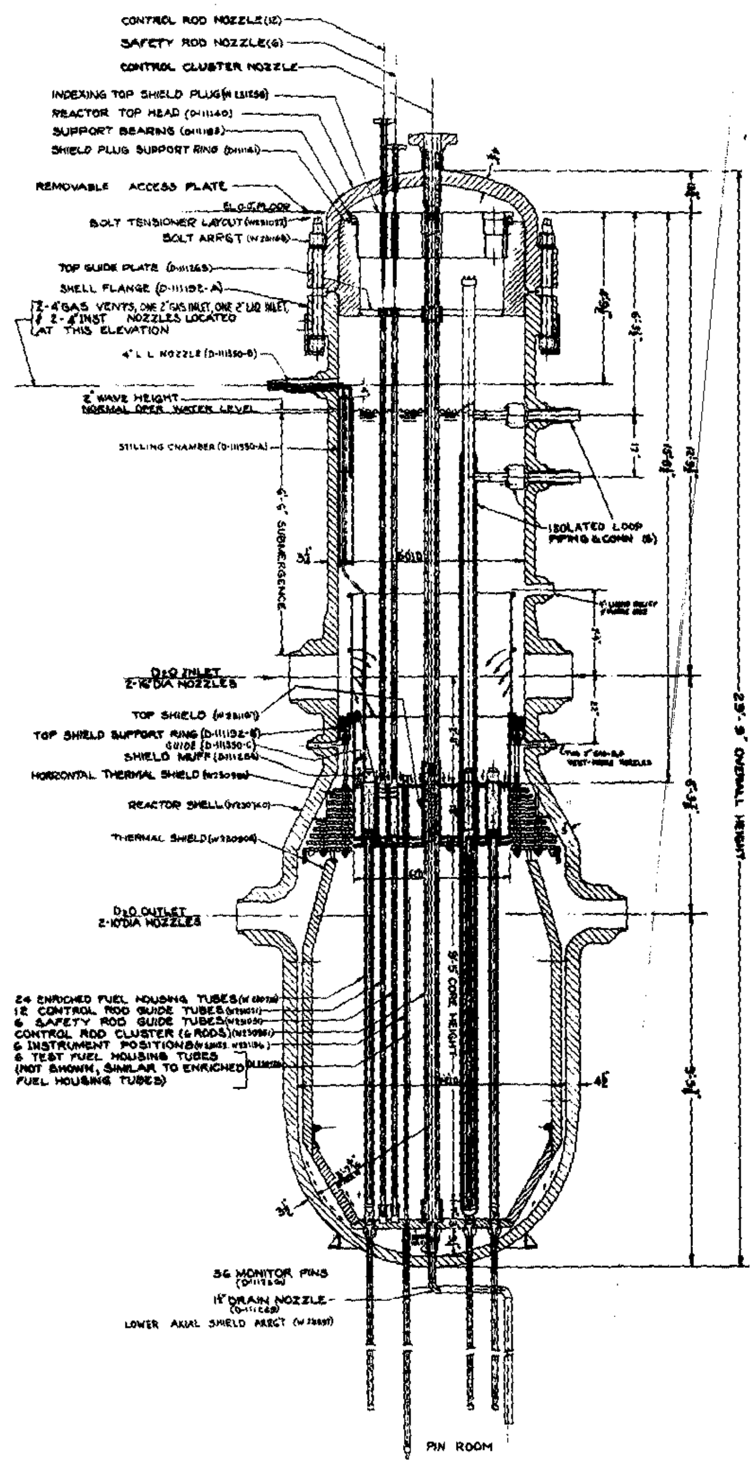


FIG. 14 REACTOR ASSEMBLY

Performance

Performance of the reactor vessel was satisfactory and caused no operating difficulties. The limitation that was imposed on the pressure and temperature of the vessel to ensure against brittle fracture, caused no operating difficulties or restrictions on desired testing conditions. Pressure was limited to atmospheric below 22°C and to less than 300 psig below 60°C.

Corrosion was monitored by metallurgical examination and by sets of corrosion coupons of the materials of construction. Sets of corrosion coupons were removed periodically during the three-year period and evaluated. Examination showed that the original magnetite film, formed during the light water testing following completion of construction, remained as a tightly adhering, uniform film. Results of these examinations, and evaluations of corrosion coupons are reported in DP-964.[8]

Troubles were experienced with two of the permanent internal parts of the reactor assembly. During the first few months of operation, the indexing shield plug sometimes jammed, because of very close tolerances. The plug was removed and 3-5 mils was machined from the outer diameter. Periodic metallurgical examination and dye-checking of the ball-bearing assembly and the support ring revealed no failures. Small spots of corrosion were sometimes found but caused no problems.

Failure of four 17-4PH stainless steel bolts in the support legs for the gas baffle, and failure of two other bolts in the gas baffle were discovered in April 1964. Failure of the support leg bolts was attributed to low-ductility rupture from a combination of thermal stresses and overstressing during installation. The other bolts failed by stress corrosion because of improper heat treatment. The support structure for the gas baffle was replaced with one of increased strength and rigidity. Details of this problem are discussed in progress reports.[7]

The performance of the double "Flexitallic" gasket seal, used between the head flange and shell flange, was nearly perfect. During the lifetime of the facility, the reactor head was removed about 100 times. Leakage never occurred past the outer gasket and occurred only once past the inner gasket. This single instance was caused by a defect in the inner gasket. This success was attributed to rigid inspection procedures to ensure good gaskets, and the use of new gaskets each time the head was replaced.

Minor troubles were experienced on a number of occasions with the Biach bolt tensioners when the heads of cap screws in the assembly sheared off and became engaged in the mechanism. The diaphragm seal assembly also leaked oil on several occasions. A modification to substitute an O-ring for the diaphragm seal assembly as well as to eliminate the cap screw shearing problem was designed but was not installed prior to the shutdown of the reactor facility.

MAIN CIRCULATING SYSTEM (Figure 15)

Description

The primary cooling complex for the reactor consists of two circulating systems each containing a pump, two ten-inch motorized valves, and a steam generator. The piping, fittings, and equipment in the main circulating system are carbon steel, with the exception of the stainless steel piping that penetrates the biological shield and connects to the reactor.

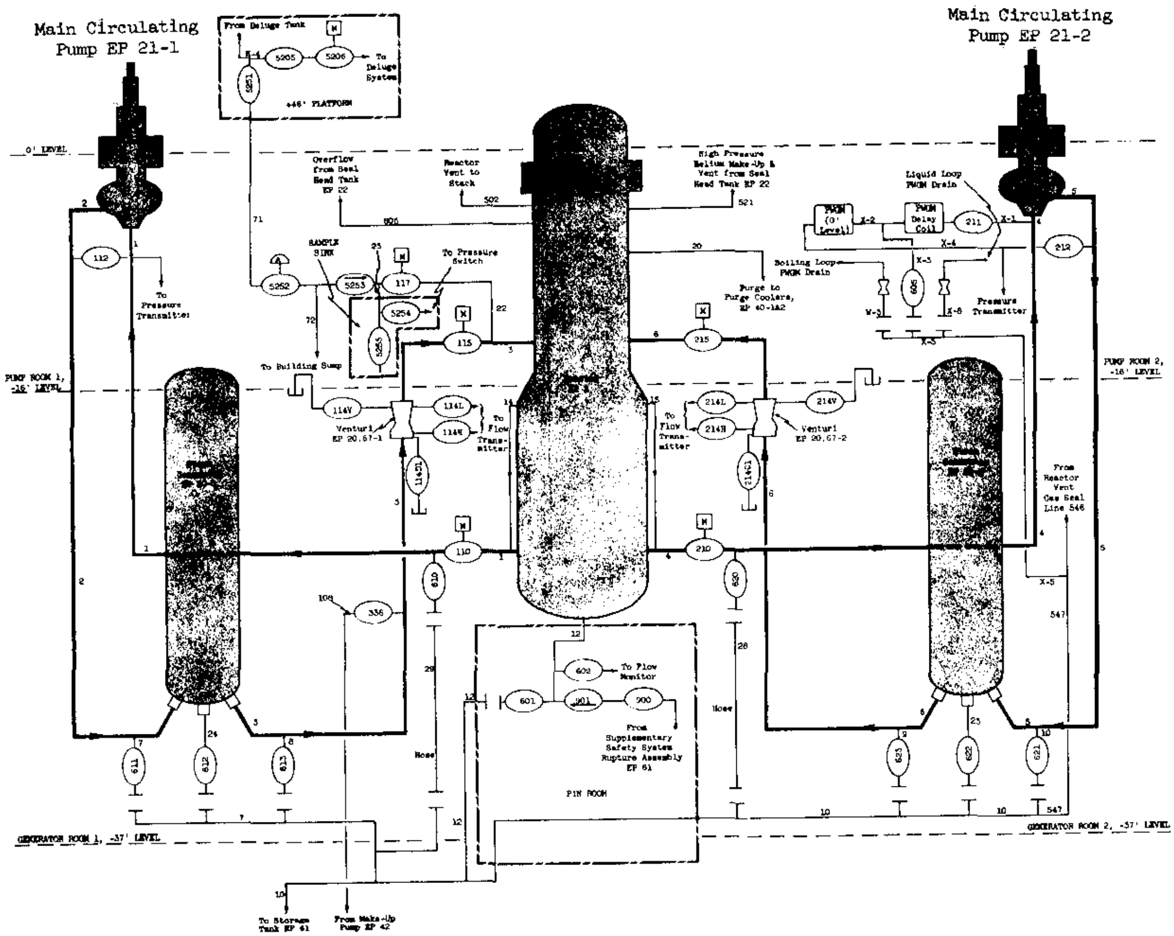


FIG. 15 MAIN CIRCULATING SYSTEM

At the time reactor operation was terminated, there were two known problems with the main pump, both of which were associated with the shaft D_2O seals.

Fretting corrosion. In May 1964, a groove, attributed to fretting corrosion, was found cut into the seal shaft sleeve on both pumps in the area of the anti-extrusion ring and the rotating face O-ring. A new shaft sleeve with a tungsten carbide plating in the problem area was installed in one pump. This pump was operated less than one month before the facility was shut down, so no evaluation of the new sleeve was possible.

When the seal assemblies on both pumps were replaced in May 1964, because of erratic and high leakage rates (0-3/4 gpm), examination of the lapped surfaces showed them to be in good condition and not to be the cause of the intermittent spurts of high leakage. Slight scoring, or grooves, was found in the bronze face with a depth up to 3-4 mils. Grooves in the tungsten carbide face were less than 1/2 mil deep. These seals had been in service since January and October 1962, for a total service life of about 19,000 and 13,000 hours, respectively. After finding the fretting corrosion groove, it must be assumed that the majority of the leakage was from bypassing the rotating face O-ring. If this assumption is correct and the hardened shaft sleeve corrects the fretting corrosion, seal life could reasonably be expected to be greatly increased over the above performance; certainly in excess of 25,000 hours.

Seal leakage on depressurization. One pump had a history of seal leakage following depressurization in the final six months of operation. The problem occurred only on one pump and only with shaft sleeves fabricated by the Plant shops. Vendor drawings of the shaft sleeve specified a finish of 32 RMS for the surface under the rotating face, whereas the vendor stated that the original sleeves were finished to better than 16 RMS. The shaft sleeve on this pump was replaced, with one having the proper finish about one month before termination of operation, hence evaluation was not complete. No leakage occurred during the one depressurization made during that period.

In addition to the seal parts, the pump impeller, wearing rings, volute, and other parts of the pump assembly were inspected. No sign of corrosion or cavitation was found. The wearing rings were measured and the dimensions were the same as the original dimensions.

The 10-inch Chapman valves performed without a single malfunction during the three years of reactor operation. Prior to nuclear startup of the facility, during the light water testing period, valve spindles in all four valves became severely scored by metal filings in the mechanism and were replaced. Leakage of D_2O past the valve packing into the leakage collection system became excessive during the final months of operation and plans were in progress to repack all four valves. The packing, a "Teflon" cup and cone type, had been installed in January 1962, when the spindles were replaced, hence had served during all high temperature, high pressure operation of the facility.

Performance of the steam generators was mentioned previously in this report and is reported in detail in DP-964.[8]

PURGE AND LEVEL CONTROL (Figure 16)

Description

During normal operation, D_2O was purged from the reactor through a 4-inch nozzle located about 5 feet below the normal D_2O level in the reactor. The rate of flow was controlled by the reactor level. Purge D_2O was passed through two purge coolers in series, then through a pressure reducing valve, and then into the low pressure purification system. Relief valves downstream of the purge coolers provided protection against reactor overpressure.

When the water level dropped below a preset level, a 1700 psig positive displacement pump injected D_2O at 30 gpm from a large storage tank into one of the main circulating systems.

The positive displacement pump was an Aldrich triplex inverted pump. The stuffing box gaskets were O-ring type wherein internal pressure sealed the joint. Telescoping bushings were used, with Style 835 (Universal Packing Co.) packing along the plunger rod to prevent leakage. This noncompressible packing was a buna-N compound with a hard phenolic core and with phenolic top and bottom adapters.

The purge coolers were multipass type heat exchangers with four D_2O passes and one H_2O pass per cooler. The coolers were carbon steel and contained 49 carbon steel 3/4-inch U-tubes.

Performance

The Aldrich triplex pump operated satisfactorily with only routine maintenance. Minor leakage of D_2O past the plunger packing occurred;

however, it was collected and was of no concern. Excessive vibrations occurred in the discharge piping from this pump during the first four months of operation. Cracks developed in two forged stainless steel reducing tees in the discharge piping. Both fittings failed at stress-raising sharp shoulders where the body size was reduced. Additional bracing, minor alterations to the piping configurations, and the addition of a hydraulic pulsation damper overcame the vibration.

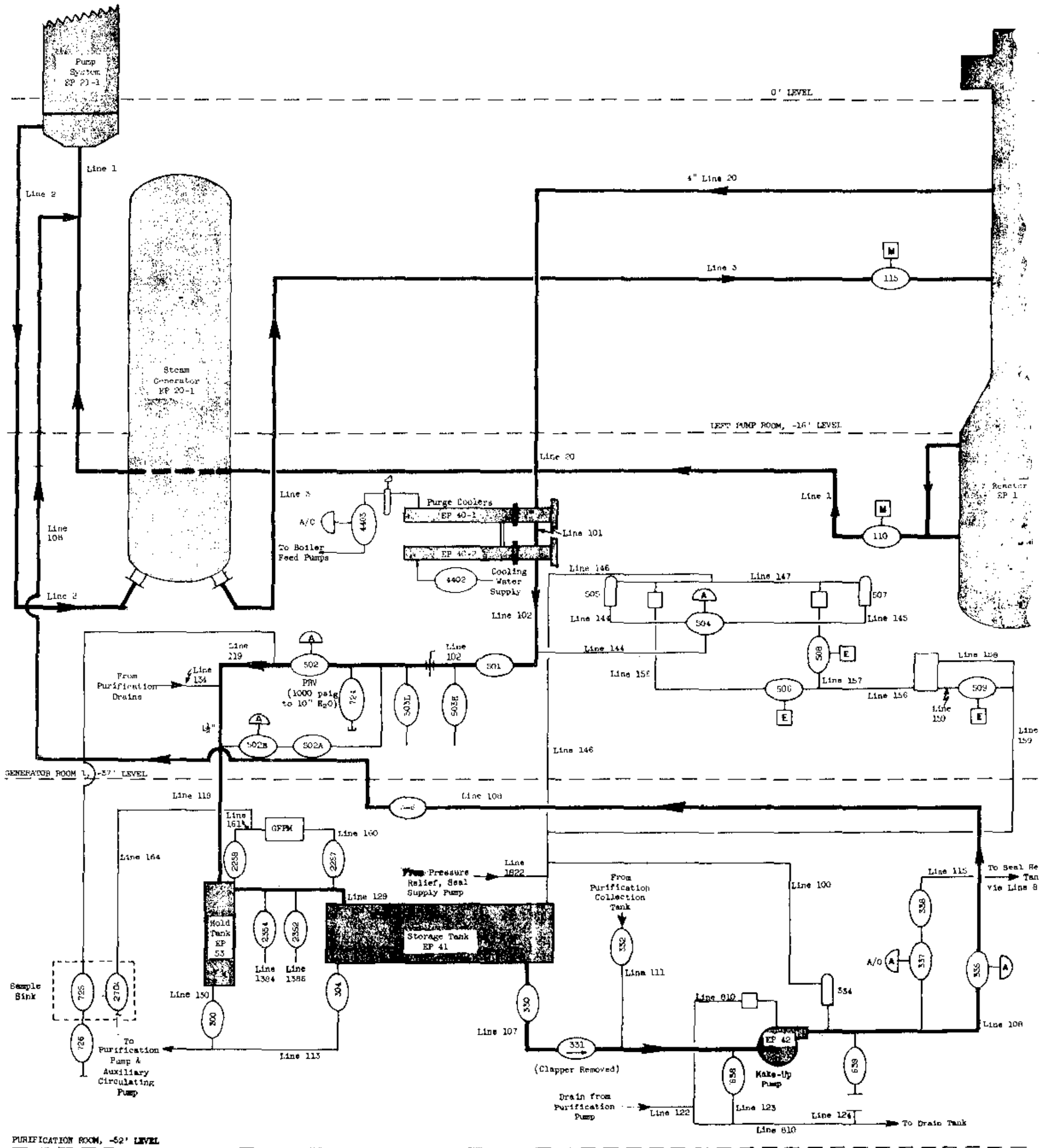


FIG. 16 PURGE AND LEVEL CONTROL SYSTEM

One of the purge coolers leaked D_2O into the cooling water system at the rate of 0.5 to 1 lb/day during the last two years of operation. The second purge cooler periodically showed leakage indications of about 0.5 lb/day, which was about the lower level of detection. No attempts to locate or repair the leakage point were made since there was no appreciable change in the leakage rates. As inspections of the shell side of the tubes revealed very minor corrosion, it was assumed that the leakage was at the tube-to-tube sheet welds. Corrosion performance of the purge coolers is reported in DP-964.[8]

Two problems were experienced with the D_2O relief valves during the initial testing period. These valves were designed to pass 150 gpm of D_2O at 100°C, with an upstream pressure of 1320 psig and a maximum back pressure of 10 psig, and were to be leaktight at pressures up to 95% of the relief setting. Bench tests, tests in a hydraulic loop, and functional tests in the reactor system all showed that these valves could not reliably be expected to relieve within ±50 psig of any desired pressure. To provide an adequate pressure margin below the set point of pressure relief valves located in the gas space of the reactor, set at 1480 psig, normal operating pressure was limited to 1200 psig maximum with the liquid relief valves set at 1300 psig.

The second problem experienced with these valves was severe vibration of the valves and associated piping when they were functionally tested in the system. Additional bracing and support reduced the vibration to tolerable levels.

A simplified pressure relief system was being designed which circumvented the problems with the original equipment. The simplification was made possible by the more reasonable requirements of Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code, than the requirements used in the original design, which were based on Section VIII of the Code, Unfired Pressure Vessels.

MAIN PURIFICATION SYSTEM (Figure 17)

Description

The purification system removed particulate and dissolved ionic impurities from the D_2O. Additionally, the system included means for controlling the pD of the D_2O between 10.2 and 11.2. The equipment consisted of a holdup tank, a canned rotor pump, two deionizers, a 5-micron filter, and a collection tank. Piping and space were allowed for a prefilter before the deionizers, but this was not used except in the cleanup period following completion of construction.

After the purge D_2O was cooled and its pressure was reduced to 10-inches H_2O (see previous section on Purge and Level Control), the water entered a 55-gallon, carbon steel holdup tank. The water was pumped at a rate of 15 gpm from the holdup tank by a stainless steel centrifugal pump. The water passed through a mixed bed deionizer containing lithium and deuteroxide resins. A second deionizer, piped in parallel, was a mixed bed of DOD resins and was used intermittently to keep the pD below 11.2. The D_2O then passed through a 5-micron afterfilter and flowed into a 48-gallon collection tank. From the collection tank, about 10 gpm was returned to the high pressure system. Make-up D_2O could be added to the system via the collection tank or removed from the system by a separate line from the discharge from the purification pump. With the exception of the holdup tank and the main storage tank, all purification piping and equipment was stainless steel.

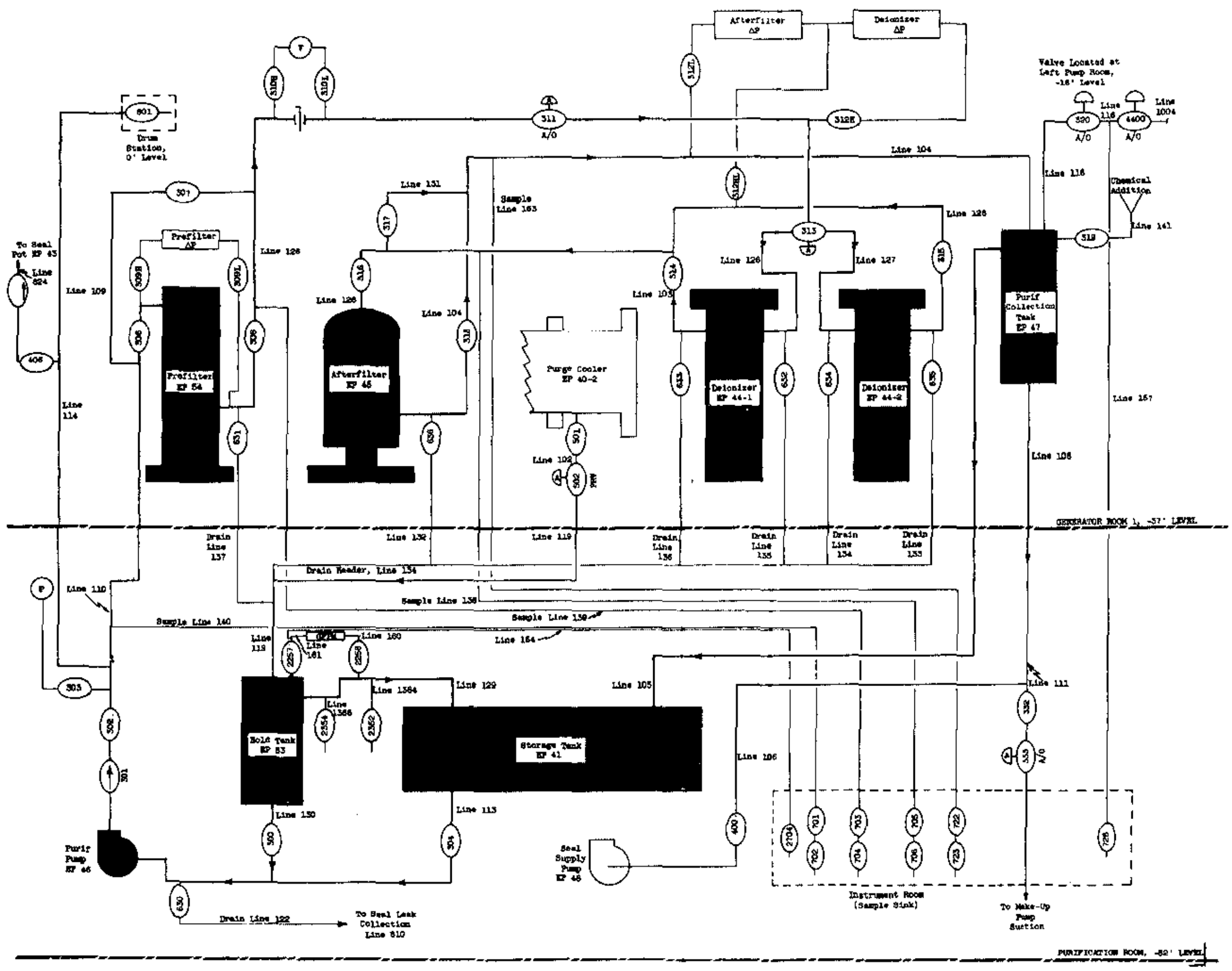


FIG. 17 MAIN PURIFICATION SYSTEM

Performance

Except for occasional pluggage of pump strainers by resin fines and a single bearing failure in the canned Chempump, performance of this system was satisfactory and trouble-free. Corrosion in the purification system was monitored by coupons and visual inspections. These data are reported in DP-964.[8]

The pD of the moderator-coolant was controlled at 10.7 ±0.5 by exchange through a mixed resin, Li^+OD^-, ion exchanger. The rate of pD change was slow and positive, typically about 0.25 pD unit per month. As necessary, the pD was lowered by diverting the flow through the D^+OD^- mixed resin bed. The ion exchanger beds had a resin volume of 5 ft^3 and normal use was about one year (throughput of about 8 million gallons).

Dissolved oxygen was controlled by the addition of deuterium gas to the helium in sufficient amounts to maintain a concentration of 10-20 std cc D_2/kg D_2O in the high pressure coolant system. During nuclear operation, the dissolved deuterium combined radiolytically with the dissolved oxygen to give a dissolved oxygen concentration of less than 5 ppb.

The chemical control of the moderator-coolant and the performance of the purification system are reported in DP-964.[8]

SEAL WATER SYSTEM (Figure 18)

Description

The seal water system consisted mainly of: a seal supply pump, a seal head tank, a seal leakage collection system, a catch pot to remove oil that could be present in seal leakage from either the main or loop pumps, a drain tank, and a drain tank pump.

The system provided a continuous supply of filtered and deionized D_2O for the seals on the main circulating pumps and the shaft seals on the control and safety rod drives. The seal pump delivered water to a supply header with the seal head tank, located in the top of the building, floating on the header to provide additional head and a small (20-minute) seal water reserve. The seal pump pumped about 9.6 gpm from the purification collection tank. A level control system on the seal head tank diverted the water not required for seal supply back into the main circulating system.

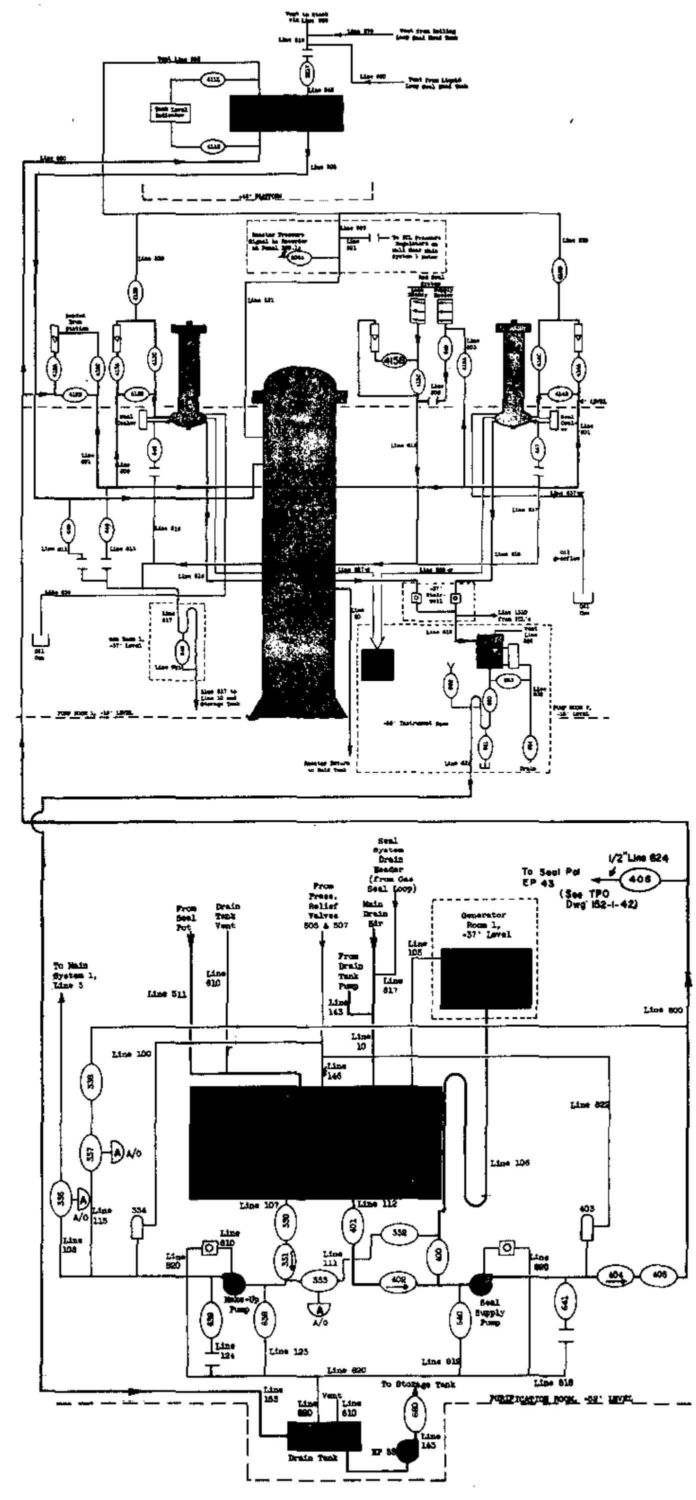


FIG. 18 SEAL WATER SYSTEM

Performance

Corrosion performance for this system is reported in DP-964,[8] and seal performance for the rod drive system is discussed in DP-971.[6]

Mechanical operation of the seal supply pump was satisfactory, except for leakage past the plunger packing. The pump was an Aldrich triplex inverted pump designed for direct horizontal flow. Maximum design pressure was 1700 psig and normal operation was 9.6 gpm at 280 rpm against a discharge pressure of 1200 psig. "Universal" packing 835 was used as a water seal around the plunger. Normally on depressurization, the packing would leak; however, on resumption of pressure operation the leakage would usually stop. On about ten occasions during the three years of operation, the leakage persisted after pressurization and it was necessary to replace the plunger packing. This problem was never satisfactorily resolved. Construction of the pump and stuffing box was quite similar to that of the make-up pump where packing leakage was nearly nonexistent. The main difference in the two pumps was in usage; continuous for the seal pump and infrequent for the make-up pump.

The capacity of the seal leakage collection system, particularly the catch pot, was inadequate to handle leakage from a gross seal failure and modifications to enlarge the capacity of the system were in progress when the HWCTR facility was shut down. On three occasions the seal catch pot overflowed:

1) Steel chips were left in a pump seal after reassembly following a routine inspection for wear. When seal flow was started, leakage exceeded the capacity of the catch pot and water overflowed through the vent line on the pot.

2) During shutdown, while performing a gas aspiration test that required rapid depressurization of the reactor, the seal cavity on one pump gas bound and blocked flow to the seal. Subsequent leakage when the seal warped from overheating overflowed the catch pot system.

3) While the system was cooling following a nonnuclear heatup, the seal ΔP control valve failed, causing loss of seal flow to the seals of one loop pump. Subsequent seal leakage overflowed the catch pot and the oil collection drum.

Other parts of the seal water system functioned satisfactorily during the three years of operation.

CONTROL AND SAFETY ROD SYSTEM

The reactor contained 18 control and six safety rods, each driven by an electric motor through a rack and pinion gear. Racks, pinions, and bearings were located inside individual pressure housing penetrated by means of floating ring labyrinth seals. The drives were mounted on the reactor vessel head. Safety rods had electromagnetic clutches and fell by gravity into the reactor when it was scrammed. Between initial-critical on March 3, 1962, and termination of operation on December 1, 1964, the reliability and performance of the rod drives was very good. Seal leakage was well within design limits. Inspection of seals and control rods showed no evidence of crud buildup or stress corrosion cracking of type 17-4PH stainless steel parts.

A complete description of the rod drive systems and a detailed record of their performance and malfunctions is reported in DP-971.[6] None of the difficulties experienced with the HWCTR rod drive system ever prevented the reactor from shutting down promptly or involved adding reactivity at an unsafe rate.

COOLING WATER AND STEAM SYSTEM (Figure 19)

Description

Secondary cooling water was supplied originally to the HWCTR from three wells rated at 350, 400, and 500 gpm. These wells were constructed with carbon steel casing and screens during 1951. The screen on the 500-gpm well failed in late 1963 and was replaced early in 1964 by a new deep well rated at 1500 gpm. The electric motor drive for the 400-gpm well was augmented by a direct coupled gasoline engine for emergency use in the event of a complete power failure.

Water from the wells was delivered to a common header from which it was routed to the cooling system and to a 150,000-gallon overhead storage tank. The overhead storage served as a reservoir for domestic and fire use as well as a backup for the reactor cooling system in the event of loss of well supply.

Water from the cooling header was routed to a carbon steel standpipe (67 feet high x 3 feet diameter). Lines from four chemical additive pumps injected sodium sulfite for oxygen control, trisodium phosphate for pH control, and sodium hypochlorite for bacterial control, into the water before it entered the standpipe. The standpipe provided a gravity pressure head for the cooling water system. From the standpipe, cooling water flowed to the miscellaneous heat exchangers and coolers in the reactor system. The effluent water from the miscellaneous heat exchangers discharged into a common header which terminated at the inlet to the head tank for the boiler feedpump. "Versene"*, for boiler scale control, was added to the feedwater at this point. Two boiler feedpumps supplied water to the steam generators in the primary coolant system. During reactor startup the steam generators were fed water by the head available in the standpipe, until the steam pressure reached about 25 psi, sufficient to drive the boiler feed pump turbines. Steam from the generators passed through control valves to a muffler and was released to the atmosphere. During reactor shutdown periods, the reactor was cooled by operating the steam generators as water-to-water heat exchangers, using the elevation head provided by the standpipe as the driving force for the water. Prior to nuclear startup, eductors were used to lower the water level in the steam generators to the normal operating level.

Major equipment pieces in these systems are described below:

- Standpipe - 1/4-inch carbon steel tank, 67 feet high x 3 feet diameter.
- Chemical Mix Tanks (2) - stainless steel, 575 gallon, two-compartment tank with a mixer in the upper compartment.
- Chemical Feed Pumps - four Madden Metriflow diaphragm metering pumps, 60 gph at 100 psig discharge pressure, used for sulfite and phosphate additions.
  -Three Wallace and Tiernan Dual Head Diaphragm pumps, 20.8 gph at 125 psig back pressure, used for "Versene", hypochlorite, and sulfite additions.
- Instrumentation - Hays Analyzer, Model 625, for dissolved O_2; conductivity; pH.
- Steam Turbines (2) - Terry, type ES.
- Boiler Feed Pump (2) - Worthington, 2-1/2-UNQ-10, four-stage centrifugal pump.

*Trademark of Bersworth Chemical Co.

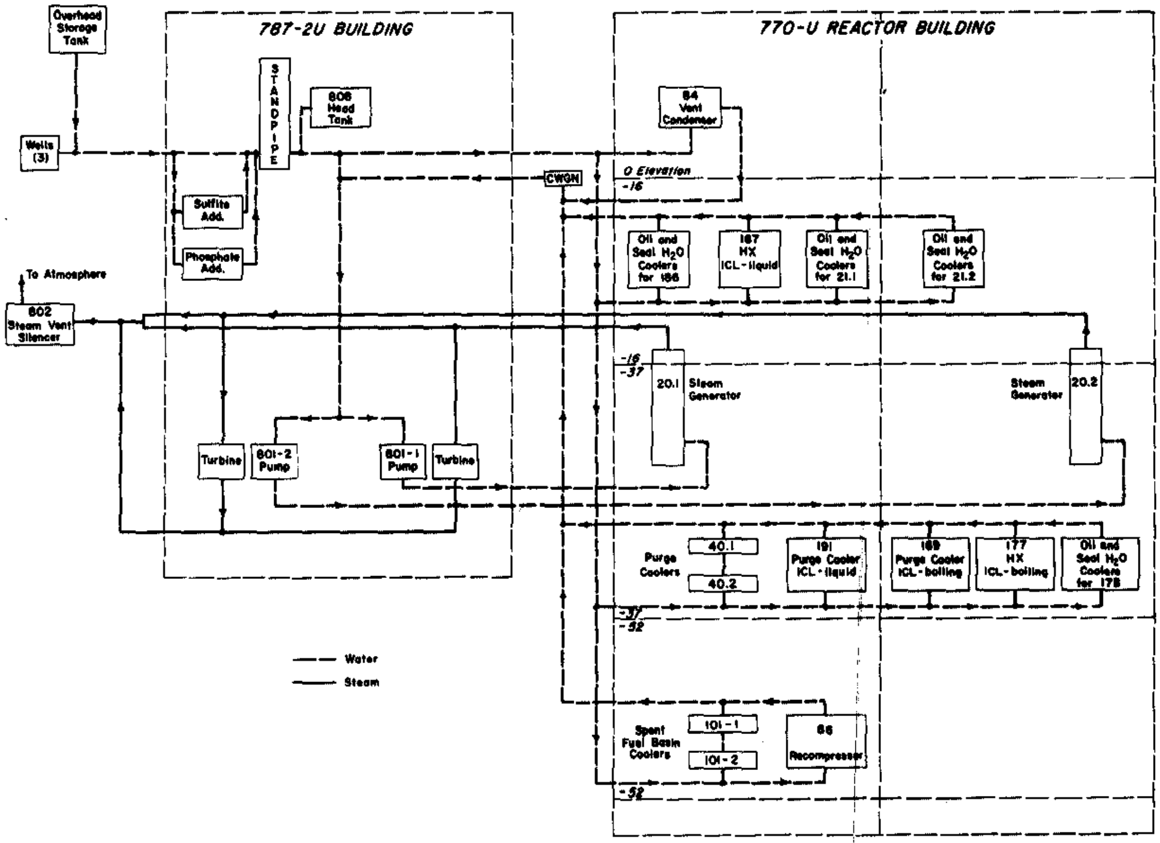


FIG. 19 COOLING WATER AND STEAM SYSTEM

Performance

Performance of the three original wells at the HWCTR site deteriorated during the three-year operation of the facility, ending with the collapse of the carbon steel screen in one well in late 1963. The remaining two wells were operable when operation was terminated, but they required frequent cleaning to maintain their pumping capacities. Based on inspections and performance of similar-aged wells at the Savannah River Plant, the remaining lifetime of these two wells, without replacement of the present casing and screens, is limited. The new well, 730 feet deep, used spiral wound stainless steel screen and is considered to be the primary source of cooling water.

Inspections and thickness measurements on the mild steel standpipe indicated that it was corroding at a rate of about 7.4 mils/year. Maximum pit depth after three years of operation was 50 mils; original thickness was 1/4 inch.

Among the seven chemical feed pumps, five worn bearings and three failed diaphragms were replaced in over three years of operation.

The only difficulty experienced with the steam turbines was the tendency of the overspeed trip and constant speed governors to bind from lack of normal movement. Inspection after termination of operation showed that the casings were in good condition and uniformly covered with an adherent film of magnetite. Most of the turbine blades had been slightly eroded by moist steam.

A single bearing failure occurred in each of the boiler feed pumps in three years.

The performance of the mild steel in the steam generators was of particular interest. The shells of these vessels were ASME-SA-212-B steel and the 3/4-inch tubes were ASME-SA-210 with 109-mil minimum wall thickness.

As shown by the preceding equipment description, the treatment of the secondary cooling water was entirely by chemical addition, that is, without deionization or deaeration. Catalyzed sodium sulfite was added to scavenge oxygen; trisodium phosphate was added for pH control; and a solution of the sodium salt of "Versene" was added to prevent scale formation on heat transfer surfaces. Sulfite analyses were performed every eight hours and were the primary assurance of oxygen control.

Use of the system began in October 1961. An inspection in May 1963, showed the tubes and shell to be in good condition with a maximum tube pit depth of 5 mils. An inspection in February 1964, showed general corrosion of 3-5 mils and a maximum pit depth of 15 mils.

Inspection of the steam generators in July 1964, subsequent to a tube failure, revealed much more severe pitting attack on the tubes than would have been predicted from the results of the February inspection. Pit depths ranged as high as 85 mils. Tubercles were evident on and around the pits. The accelerated corrosion was attributed to oxygen attack and was ultimately traced to a change in quality of the sodium sulfite. A sulfite residual greater than 10 ppm had been maintained consistently throughout the period. However, the particular brand of catalyzed sulfite that had been used since October 1961, was changed early in 1964. Purchase specifications for the two commercial brands of sulfite were identical. No specification of the reaction time of the sulfite with dissolved oxygen was contained in the purchase specifications. Laboratory tests of the efficacy of the two brands of catalyzed sulfite confirmed the slow reaction time of the sulfite used from early 1964 to July 1964.

Figure 20 illustrates the two types of behavior encountered with the catalyzed sulfites. Either the reaction proceeded almost immediately or it required several minutes induction time. The two brands of sulfite are identified as sulfite "B", used prior to February 1964, and sulfite "A" used from February 1964, to July 1964. In Table V, the scavenging action is rated "positive" if less than 300 ppb oxygen remained, and "negative" if more than 300 ppb remained after the one-minute reaction time.

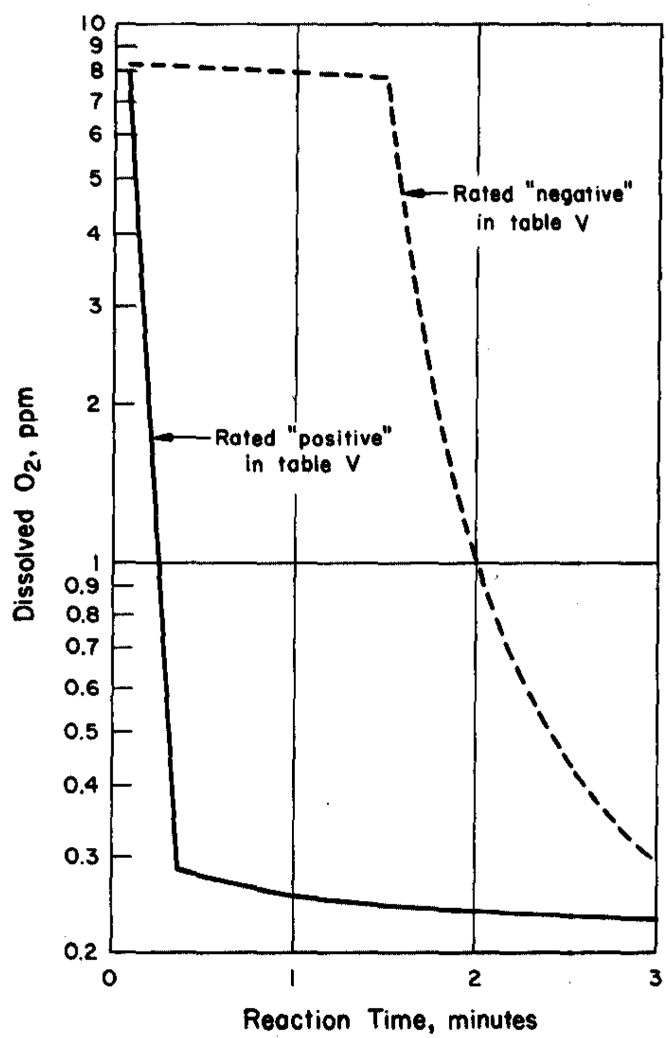


FIG. 20 REACTION RATES BETWEEN CATALYZED SODIUM SULFITE AND DISSOLVED OXYGEN

TABLE V

Efficacy of Catalyzed Sodium Sulfite for Oxygen Removal from HWCTR Well Water

(a) No Na_3PO_4 (c) >300 ppb O_2 at 1 minute

(b) 30 ppm Na_3PO_4 (d) <300 ppb O_2 within 1 minute

	Deionized Water (8.5 ppm O_2)		Well A (6.8 ppm O_2)		Well B (3.8-4.4 ppm O_2)		Well C (8.5 ppm O_2)	
	(a)	(b)	(a)	(b)	(a)	(b)	(a)	(b)
Sulfite "A" (pH 10.5)	(c)	(d)	(d)	(d)	(d)	(c)	(c)	(d)
Sulfite "B" (pH 8.4)	(d)	(d)	(d)	(d)	(d)	(d)	(d)	(d)
Modified sulfite "A"								
pH 9.4 with NH_4Cl	(c)		(d)	(c)	(c)	(c)		(d)
pH 8.5 with H_2SO_4			(d)	(d)	(c)	(c)		
0.3% Co^{2+} added	(d)	(d)	(d)	(d)	(d)	(c)	(d)	(d)

Sulfite "B" was consistently effective, but sulfite "A" was not. The performance of sulfite "A" was not improved by lowering the pH with NH_4Cl or H_2SO_4 to make it similar to sulfite "B". However, the addition of 0.3% Co^{2+} as a catalyst made sulfite "A" as satisfactory as sulfite "B".

On the basis of these data, the use of sulfite "B" was reinstituted and the in-line dissolved oxygen analyzer was installed. An inspection in December 1964, showed that all of the large tubercles that were present in July 1964, had been dissolved by the "Versene" and that the rigorous oxygen control had arrested the corrosion.

Further information on the performance of the secondary cooling system is contained in DP-964.[8]

PROCESS GAS SYSTEMS (Figure 21)

Description

Helium was used to provide an inert gas atmosphere in process system vessels containing D_2O, and to pressurize the reactor to prevent boiling at reactor operating temperature.

Helium was transported to the area by a gas cylinder trailer. The trailer was filled from a helium tank car transported by rail to the SRP site. The trailer carried 30 cylinders with a total volume

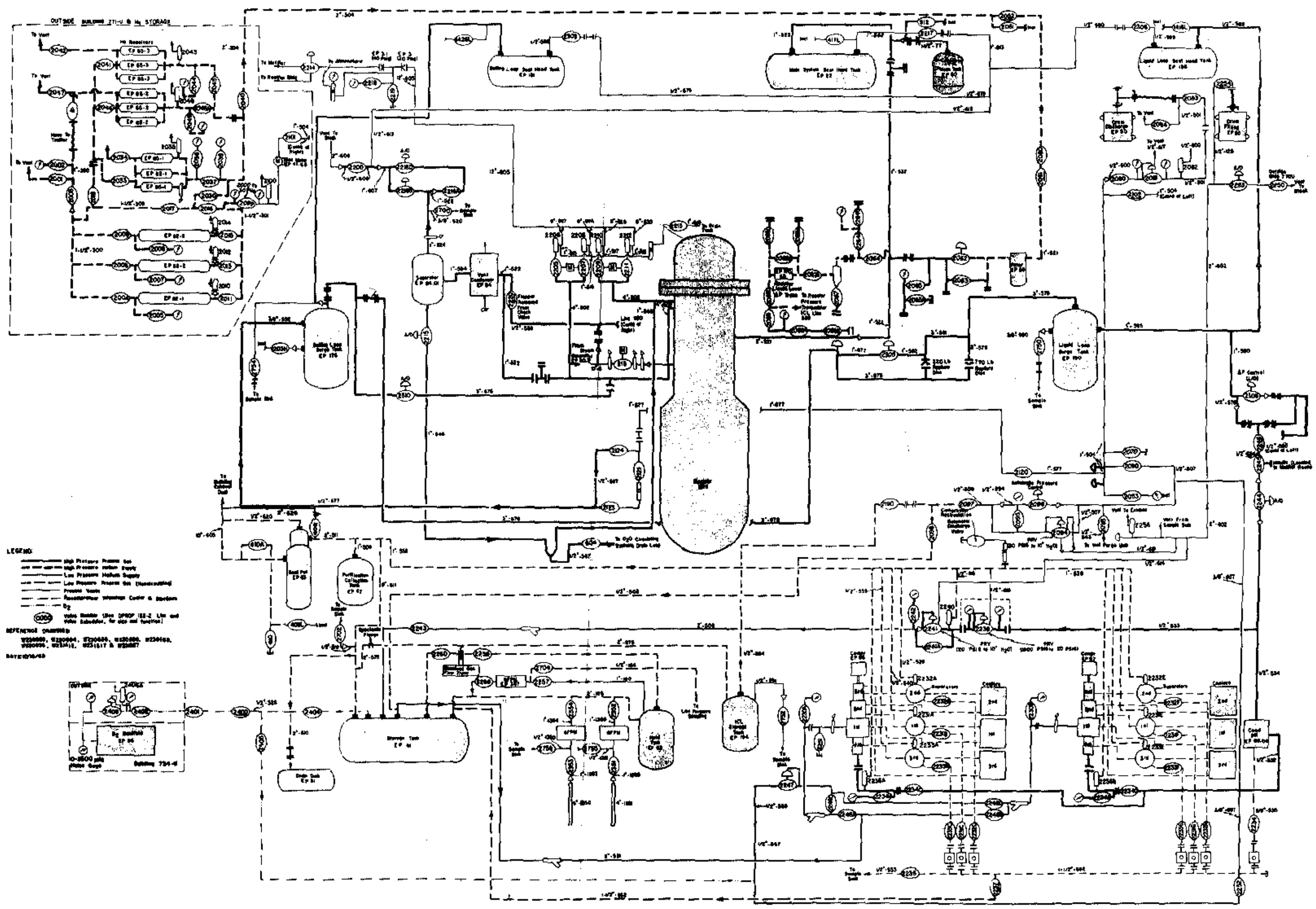


FIG. 21 HWCTR GAS SYSTEM

of 265 ft^3 and a design pressure of 3000 psig. Helium was stored in the reactor area in a low pressure storage bank capable of storing about 7000 scf at 450 psig and in a high pressure storage bank capable of storing about 8700 scf at 2000 psig.

Helium was supplied to the low pressure header in the reactor building from the low pressure storage banks. The low pressure header supplied helium to the drum station, to the D_2O storage tanks, to make-up lines associated with the gas compressor, and to helium purges for tank level instruments. It was possible to supply the low pressure header directly from the cylinder trailer or the high pressure storage bank in an emergency.

The high pressure receivers provided a means of adding helium directly to the reactor for initial pressurization, and in the event of a system leak or a gas compressor failure.

Following a shutdown during which the system was depressurized, helium was added directly to the high pressure system from the high pressure receivers until reactor pressure was 600 psig. The high pressure header was then valved out and the gas compressors were used to raise the system pressure to 1200 psig. Make-up helium was added to the system from the low pressure header to the storage tank on demand of a pressure controller that sensed the pressure in the reactor gas space. The discharge of the gas compressor passed through a pressure regulator valve that directed the gas to the high pressure system until demand was satisfied; then the gas was recycled to the storage tank. Reactor pressure was thereafter controlled indirectly by controlling the pressure in the storage tank. During reactor operation, process gas was constantly removed from the high pressure system by the D_2O purge stream and by leaks and an equivalent amount returned from the storage tank - gas compressor system.

As shown in Figure 21, the discharge of the gas compressor could be routed either directly to the reactor or to the liquid loop, and subsequently to the reactor by proper prepositioning of spool pieces. An automatic ΔP controller, which operated a valve in the gas line connecting the liquid loop to the reactor, also permitted operating the liquid loop at a higher pressure than the reactor.

The two compressors were Ingersoll-Rand, four-stage, positive displacement compressors. The units used "Teflon" glass-filled and "Teflon" carbon-filled piston and wear rings to eliminate the need for oil lubrication. The two compressors were identical except for the drive motors; one, driven by a 10 hp motor at 220 rpm, had a rated capacity of 8 scfm at 1600 psig discharge pressure, the other, driven by a 15 hp motor at 300 rpm, had a rated capacity of 11 scfm at 2000 psig discharge pressure.

A vent condenser and separator were located in the reactor vent line to condense and recover D_2O from the gas purge from the system during depressurization.

Equipment for adding deuterium gas to the high pressure system during pressurization or to the storage tank during reactor operation is shown in Figure 21.

Performance

In the original design, only one gas compressor was installed in the system. During the first six months of operation it was out of service because of low capacity on seven or eight occasions. The principal cause of the troubles was wear of the third and fourth stage piston rings. In August 1962, the second compressor was installed in parallel to ensure sufficient capacity during nuclear operation. An experimental program involving the use of various types of piston and wear rings resulted in satisfactory performance. Ring performance over the final two years of operation is summarized below.

1) Fourth stage piston ring (glass-filled "Teflon", step cut) - average 2000-hour-life before replacement.

2) Third stage piston ring - external (glass-filled "Teflon", step cut) - approximately 4000 hours of service.

3) First and second stage wear rings - (carbon-filled "Teflon", step cut) - The exact life of this wear ring was not established; however, it was in excess of 8000 hours. The ring was replaced before excessive wear occurred as such wear would cause major physical damage to the piston and cylinder.

4) All other piston and wear rings - (carbon-filled "Teflon", step cut) - These were replaced on a preventive maintenance basis at 8000 hours; no excessive wear was found.

All other parts of the gas system performed satisfactorily with only routine maintenance.

ELECTRICAL SYSTEMS (Figures 22 and 23)

Description

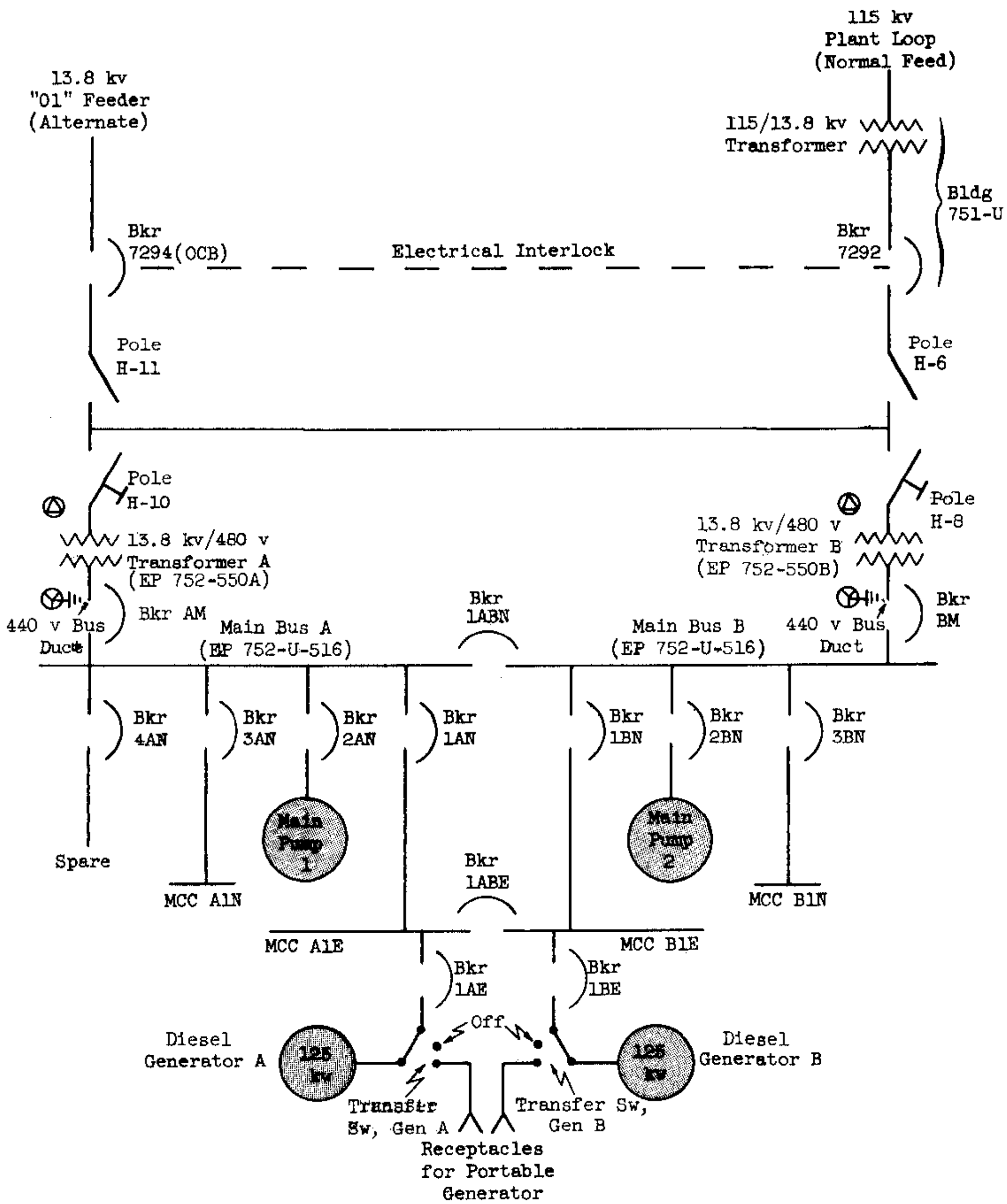
Two primary sources of power provided normal and alternate electrical service to the HWCTR area. The normal source was a 13.8-kv line from a 5000-kva primary substation. The alternate source was a 13.8-kv tap off a supply feeder to a nearby area. The normal source supplied the regular power requirements of 1250 kva; the alternate source, which was used only in emergency or during planned outages of the normal source, supplied emergency requirements of 600 kva.

Electrical equipment in the area was divided into three categories; normal, emergency, and essential. Examples of equipment supplied from the normal bus (breakers 3AN, 2AN, 4AN, 3BN, and 2BN on Figure 22) were main pump AC drives building lighting circuits, compressors, and fuel transfer equipment. Equipment supplied from the emergency bus included reactor inlet and outlet valves, emergency lighting circuits, and motor generator sets. All equipment fed from the emergency buses restarted automatically following a power interruption on the normal 13.8-kv line, the load being transferred automatically to the alternate 13.8-kv source. In addition, two 125-kv diesel generators provided backup power to the emergency buses if both 13.8-kv sources were interrupted.

Interruption of power to certain equipment could not be tolerated. This equipment was classed as essential service and included: (1) DC power to DC motors on the main coolant pumps, safety rod clutches, scram and reversal circuits, and interlock circuits, and (2) AC power to control and safety rod drives and to control room instrumentation. DC power to this equipment was supplied from AC-DC motor generator sets. If power to the AC motor in the AC-DC motor generator set(s) were lost, a 60-cell-battery bank supplied power to the DC bus until AC power was restored. In addition, the battery bank drove the DC generator as a motor and the AC unit performed as an alternator, supplying AC power to the rod drives and control panel instruments.

The emergency diesel generators were manufactured by the Waukesha Motor Company. Each unit was designed to operate at 1800 rpm and deliver 125 kva of 3-phase, 60-cycle electricity at 440 volts, with a 0.80 power factor. The engines were six-cylinder four-cycle diesels. The generators were single-bearing, wye-connected units with a current capacity of 204 amperes at 440 volts.

The motor-generator sets were designed by the Hertner Electric Company. For AC to DC conversion, each set was rated at 25 kw at 129 to 140 volts DC with ±1% voltage regulation. For DC to AC conversion, each set produced 15 kw of 3-phase, 60-cycle power at 440 volts.



NOTE: Breakers AM, BM, and 1ABN are key-interlocked so only two can be closed at the same time.

Breakers 1AN, 1AE, and 1ABE are key-interlocked so that breakers 1AN and 1AE must be open before 1ABE can be closed.

Breakers 1BN and 1BE are similarly key-interlocked with breaker 1ABE.

FIG. 22 PRIMARY ELECTRICAL DISTRIBUTION

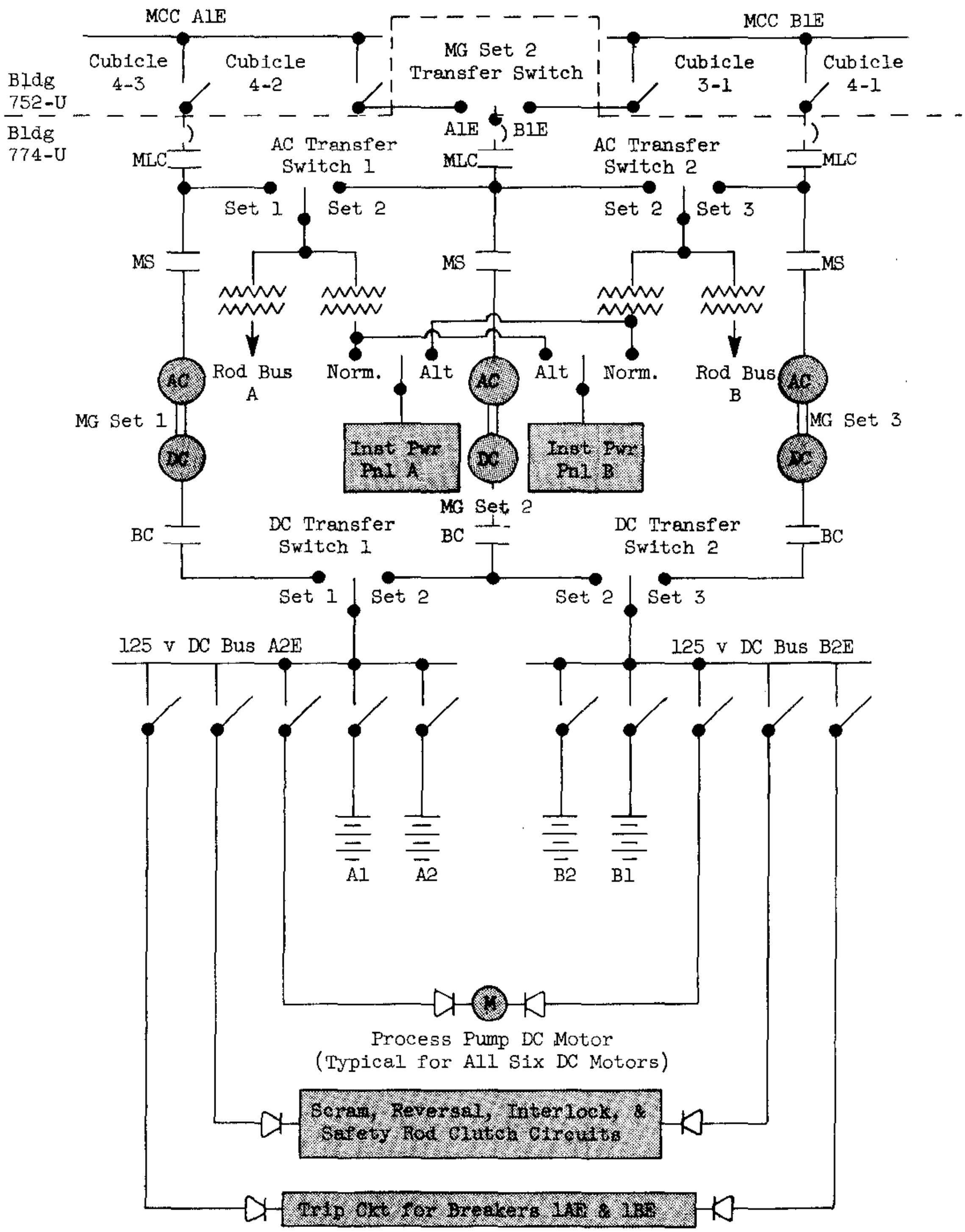


FIG. 23 CONTROL POWER DISTRIBUTION

Performance

Two types of problems occurred with the electrical system at the HWCTR. One type was associated with the control system components, i.e., relays, starters, and contactors; the second type was caused by an insufficient degree of isolation from power surges on the 13.8-kv power lines. Performance of the equipment pieces, e.g., motors, generators, and transformers, was excellent, with no major difficulties or failures.

The HWCTR is located in an area with a moderately high incidence of electrical storms. On a number of occasions each year, lightning strikes the 13.8 kv plant loop. On a few occasions, these power disturbances caused reactor scrams to be initiated by the nuclear instruments even though their power supplies have filters, voltage regulators, and in some cases, isolation transformers. The starting and stopping of large electrical equipment in the area, such as air compressors, also caused spikes on the nuclear instruments when the instruments were operating on a low current range (10^{-12} ampere). Modifications to further isolate the power supplies from electrical noise were not economically justifiable at the time they were considered.

The problem with the electrical controls was mainly failure of the motor generator sets to transfer properly from AC-DC operation to DC-AC operation upon loss of normal AC power. Upon loss of normal AC power to the synchronous motor, the DC generator transfers to a DC motor and the synchronous motor converts to a synchronous generator. The transfer time was approximately two seconds. The synchronous generator then supplied instrument and rod drive power during the period of emergency operation. The failure of normal power to the motor generator set was detected by a single phase undervoltage relay set to drop out at a decrease of 20% of normal voltage and to initiate the transfer. During the sequence of relay action, the inertia of the AC machine normally maintained sufficient voltage to keep the Motor Starter (MS) contactors (Figure 23) energized. On one occasion following a lightning strike on the 13.8 kv feeder, and on several occasions during shutdown test work, the MS contactor dropped out before the motor generator sets could transfer to emergency DC drive. The dropping out of the MS contactor disconnects the motor generator set from the instrument power bus. Design modifications to improve the control action of this circuit were completed but had not been installed at the time of termination of operation of the HWCTR.

INSTRUMENTATION

The description and performance report on individual instruments is beyond the scope of this report because of the large number of instruments involved. Most of the instrumentation was commercial product and was similar or identical to that used in other reactor systems. In general, its performance was good when coupled with a rigorous preventive maintenance program. Spurious scrams from the nuclear instruments, caused by electrical noise signals, were numerous in the first six months of operation but steadily decreased as the number of dirty connectors, insulators, and poor wiring connections were decreased.

The instruments used for the detection of fuel failures were designed and manufactured at the Savannah River Plant; hence, a description and report performance are given in the following sections.

Fuel Failure Detection Systems Description

Four systems for failed element detection were employed in the HWCTR. Except for the Scanning Liquid Photoneutron Monitor, the detection systems employed in the isolated coolant loops were identical with those used in the main system. The four systems and the sampling and activity monitors are described briefly in the following sections.

1. Low Energy Gamma Monitor (LEGM). Figure 24 presents a schematic diagram of the electronic and flow system for the LEGM. A sample of the coolant from the process stream was directed from the discharge of the circulating pump, passed through a 15-minute delay coil, and then presented to the probe scintillator crystal in four passes across the probe top.

The instrumentation scanned and recorded continuously the activity in the energy range from 0.03 Mev to 0.3 Mev, thus encompassing fission product ^{239}Np (0.1 Mev) and ^{133}Xe (0.08 Mev). The 15-minute delay coil prevented these low energy levels from being masked by short-lived products, such as ^{16}N (7.4-second half life), ^{19}O (29-second), ^{17}F (66-second), and ^{27}Mg (10-minute).

2. Gas Fission Product Monitor (GFPM). Figure 25 presents a schematic diagram of the electronic and flow system for the GFPM. A 10-gpm purge stream from the reactor vessel was cooled to 30°C and then depressurized into a purification hold tank. A 100-cm^3/min sample of the gases evolved in this tank was pumped through an electrostatic precipitator that consisted of a cylindrical tube maintained positive with respect to a central wire electrode. The gas stream was then returned to the low pressure gas system.

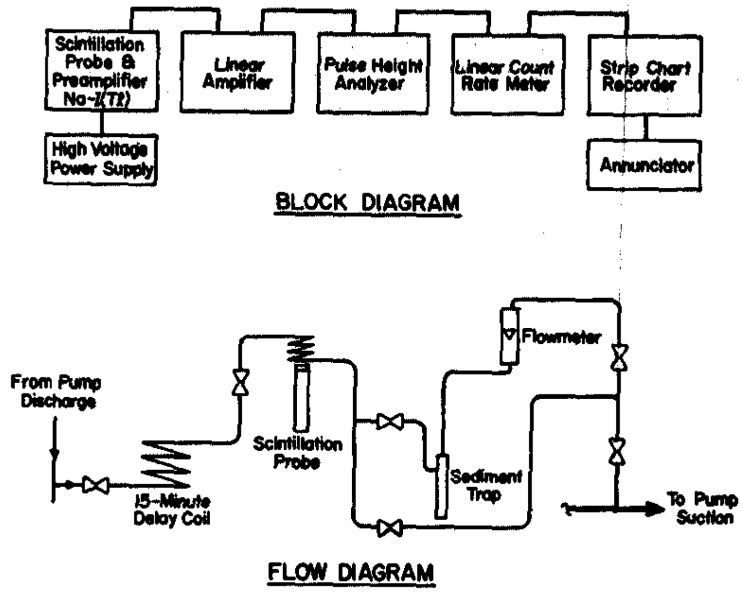


FIG. 24 LOW ENERGY GAMMA MONITOR (HWCTR TEST REACTOR)

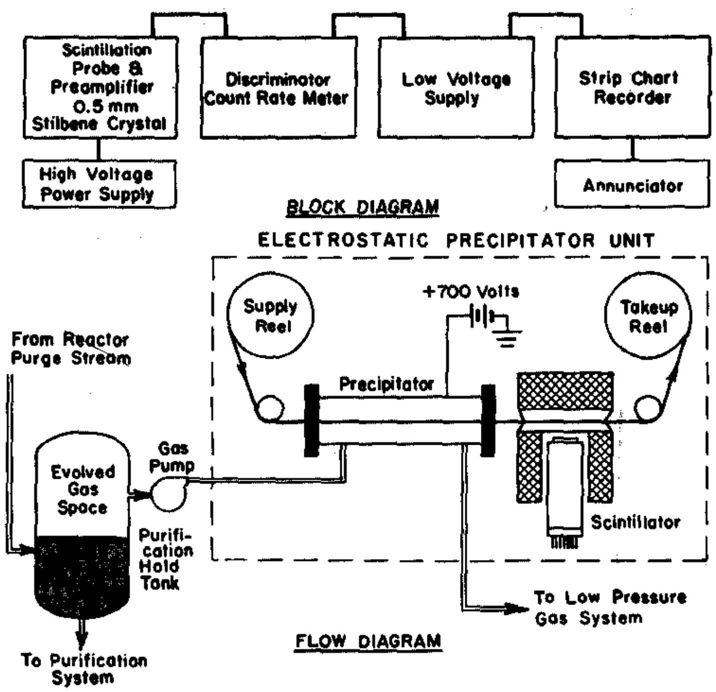


FIG. 25 GAS FISSION PRODUCT MONITOR

As radioactive gas atoms decayed within the cylindrical tube, their daughters were electrostatically precipitated onto the moving wire. If the daughter atom was radioactive, its activity was detected as the wire passed in front of a detector. The radioactive daughters of fission product xenon and krypton were detected in this manner.

The wire electrode was a 0.0055-inch diameter wire and moved through the cylindrical chamber at 2 inches/min. Five inches beyond the precipitator was a stilbene-crystal beta detector attached to a 6655-photomultiplier.

The central wire was used only once and then stored on a takeup reel. The wire on the supply reel was about 12.5 miles in length and lasted for about nine months of continuous operation. A tension-activated switch initiated a signal in the event of wire breakage.

3. Delayed Neutron Monitor (DNM). Figure 26 presents a schematic diagram of the electronic and flow system for the DNM. A sample of coolant from the discharge of the circulating pump was passed through

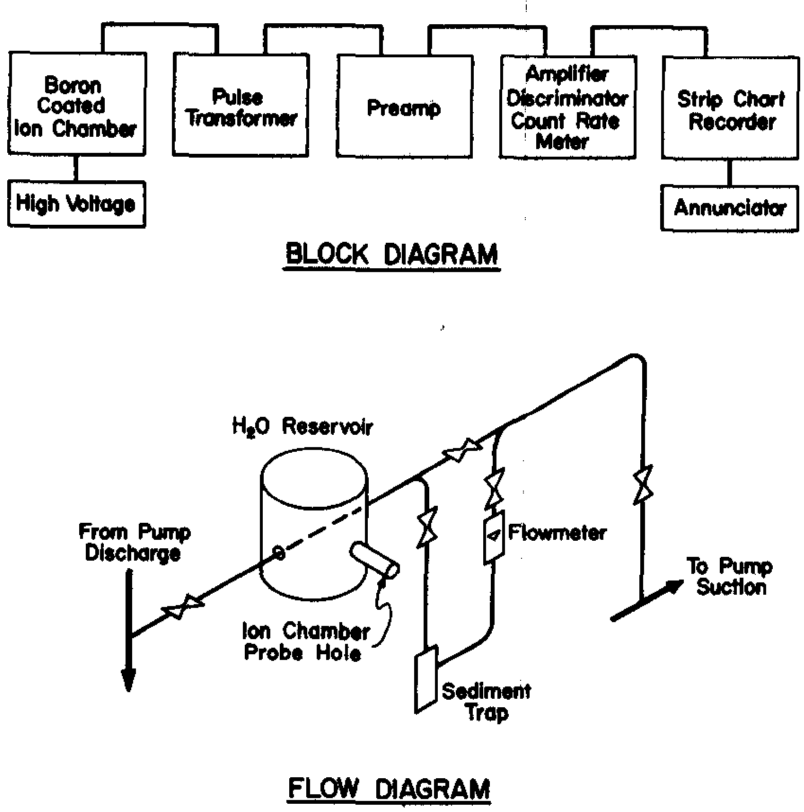


FIG. 26 DELAYED NEUTRON MONITOR

an H_2O reservoir. A boron-coated ion chamber in the reservoir detected the presence of delayed neutrons from ^{87}Br (54-second half life) and ^{137}I (22-second). The H_2O served both as a coolant and as a thermalizing medium between the sample line and the ion chamber. Transient time to the detector was about 60 seconds, thus minimizing photoneutron activity from short-lived nonfission product gamma emitters (i.e., ^{16}N) acting upon the D_2O in the sample line. From the reservoir, the sample was directed to a sediment pot, through a flowmeter, and then returned to the process stream at the pump suction.

4. Scanning Liquid Photoneutron Monitor (SLPM). Figure 27 presents a schematic diagram of the electronic and flow system for the SLPM. This system served as the only locator-detector among the failure detection systems. Samples of the effluent from each of the ten

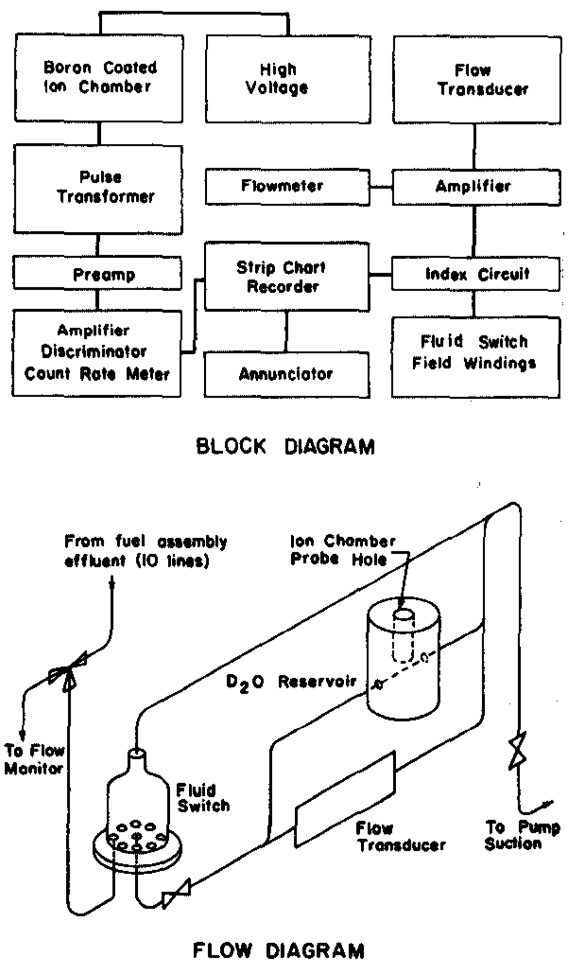


FIG. 27 SCANNING LIQUID PHOTONEUTRON MONITOR

test fuel positions were selected sequentially by a multiport valve and directed through a reservoir of pure D_2O. A boron-coated ion chamber in the reservoir detected photoneutrons caused by fission product gamma activity through the reaction $D(\gamma,n)H$. The D_2O thus acted both as a photoneutron source and as a thermalizing medium. Sample transient time to the reservoir was about 60 seconds; hence some delayed neutrons were also detected by the probe.

The mechanical arrangement of the multiport valve permitted a continuous sample from each of the test assemblies to pass through the valve with the sample from one of the ten positions diverted through the D_2O reservoir. A flow transducer, connected across a section of the sample line, measured the differential pressure as an indication of flow through the sample line. The combined samples were then returned to the suction of the circulating pumps.

The signal from the flow transducer was utilized to automatically re-index the fluid switch after each scan. The same signal was also displayed on the strip chart recorder as an indication of the position being measured. The scanning time for the fluid switch was adjustable between 4 to 15 seconds per position.

5. Cyclic Air Sampling Monitor (CASM), Stack Gas Activity Monitor (SGAM). Although the primary purposes for these monitors were not fuel failure detection, they are listed here for the sake of completeness. The CASM and SGAM proved very useful in monitoring the leaktightness of the high pressure water and gas systems. As there was always some leakage associated with a high pressure system, these instruments reflected any increase in activity in the moderator or gas system, and hence served as a backup to the fuel failure detectors. As each of the four failure detection systems depended upon full hydraulic flow conditions to deliver a sample to the detector, the CASM and SGAM instruments were the only quick means of failure detection when the reactor was shut down and at reduced hydraulic flow.

Fuel Failure Detection Systems Performance

Data obtained from these instruments for the ten test fuel failures experienced at the HWCTR are given in Table VI. The initial set of data in the table are the normal, full power readings. Response of the instruments to a fuel failure was quite prompt; in every case except failure No. 8, the reactor (if it were operating) was shut down before significant contamination of the system had resulted.

Failure No. 8, which occurred in the liquid cooled isolated loop, was very sudden and an estimated several grams of uranium was released although the reactor was shut down within a few seconds following the failure. Radioactivities in the loop returned to near normal after about two weeks of cleanup flow through the loop purification system.

TABLE VI

Response of Failed Element Detectors

Failure	Type Fuel	Instrument[a]	Background, counts/sec	Max Response, counts/sec	% Increase in Signal	Estimate of Released U	Condition of Reactor at Time of Failure
1	Natural U metal	LEGM	160	>2500	>1460		Operating at full power
		GFPM	Instrument	Inoperable			
		DNM	18	59	228	0.1-12 g	
		SLPM	5	13	160		
2	Natural U metal	LEGM	2150	3350	56		Operating at full power
		GFPM	42	65	55		
		DNM	23	37	61	0.04-4 g	
		SLPM	Instrument	Inoperable			
3	Natural U oxide	CASM	$3x10^{-13}$ amp	$3x10^{-10}$ amp	10^5	Gaseous	Shut down, for 4-1/2 hours
		SGAM	$15x10^{-13}$ amp	$3.4x10^{-9}$ amp	$2.3x10^5$		
4	Natural U oxide	LEGM	190	4270	2140	Gaseous	Startup in progress
		GFPM	15	165	1000		
		DNM	No response				
		SLPM	No response				
		CASM	$3x10^{-13}$ amp	$1.6x10^{-9}$ amp	530,000		
		SGAM	10^{-15} amp	10^{-8} amp	10^9		
5	1.5% enriched uranium oxide	LEGM	262	1700	550		Power ascension in progress
		GFPM	250	3000	1100	Gaseous	
		DNM	No response				
		SLPM	3	9	200		
		CASM	10^{-12} amp	10^{-10} amp	10,000		
		SGAM	10^{-11} amp	$4x10^{-9}$ amp	40,000		
6	1.5% enriched uranium oxide	LEGM	282	564	100	Gaseous	Operating at full power
		GFPM	54	95	76		
		DNM	No response				
		SLPM	No response				
7	Natural U oxide	LEGM	800	2400	200	Gaseous	Shut down for 2-1/2 hours
		GFPM	300	>3000	>900		
		CASM	$1.4x10^{-12}$ amp	$1.6x10^{-11}$ amp	1040		
		SGAM	$4x10^{-10}$ amp	$1.7x10^{-9}$ amp	410		
8	Natural U oxide	LEGM(L)[b]	88	45,000	51,000		Operating at full power
		DNM(L)	4	66	1500		
		GFPM(L)	75	3,000	3900	Several grams	
		LEGM(M)[b]	650	875	35		Failure occurred in one isolated loop
		DNM(M)	51	>300	>490		
		GFPM(M)	120	3000	2400		
		CASM	$5x10^{-13}$ amp	$3x10^{-10}$ amp	60,000		
		SGAM	$3x10^{-12}$ amp	$1x10^{-8}$ amp	330,000		
9	1.2% enriched uranium oxide	LEGM	750	3900	420		Power ascension in progress
		GFPM	15	100	570	Gaseous	
		DNM	8	11	40		
		SLPM	7	21	200		
10	Natural U oxide	LEGM	1100	2200	100	Gaseous	Shut down for 3 hours
		SGAM	$4x10^{-10}$ amp	$1x10^{-9}$ amp	150		
		CASM	$1x10^{-12}$ amp	$3x10^{-11}$ amp	2900		

(a) Normal background
LEGM 200-300 counts/sec
GFPM 20-60 counts/sec
DNM 30-50 counts/sec
SLPM 5-15 counts/sec
CASM 10^{-13} to $2x10^{-12}$ amp
SGAM 4 to $7x10^{-12}$ amp

(b) (L) denotes loop detector
(M) denotes main system detector

CONTAINMENT BUILDING (Figure 2)

Description

The portion of the containment building above grade level is a 70-foot-ID x 30-foot-high steel cylinder, topped by a steel hemispherical dome. The steel plate in the cylindrical portion is 3/4-inch thick, the plate in the dome is 3/8-inch thick. The plates are carbon steel, ASTM Specification A-201-57T, Grade B per A-300, Class I. Pipe penetrations and forgings through the steel shell are A-333, Grade C, and A-350, Grade LF1, respectively.

The over-all height of the containment structure is 125 feet; 60 feet is below grade. The below-grade portion is reinforced concrete, with the concrete shell prestressed, using 1-9/16-inch-diameter steel cables tensioned to 135,000 psi.

The steel shell is anchored to the concrete foundation by 328 high-strength 1-1/2-inch-diameter anchor bolts. These bolts were pretensioned so that compression at the joint remained at the design pressure of 24 psig. The steel shell is insulated on the outside with 2-1/2 inches of "Polystyrene-Styrofoam"*.

A complete description of the containment system is given in DP-600(1) and DP-968(5).

Performance

<u>Leakage Rate</u>. Tests conducted to measure the gas leak rate from the containment building are described in detail in DP-968.(5) A brief summary of the leak rate results and their relationship to hazards evaluation are given here.

The initial measurement of the leak rate was made in November 1960, immediately after the containment shell was assembled. The conduits and pipes that penetrate the shell were covered with temporary seals because most of the wiring and piping had not been installed. The test was conducted at internal building pressures of 29 and 24 psig. The measured leak rate at 24 psig was 0.56% of building volume per day, well below the 1% value used in safeguards analyses. With one exception, all subsequent tests were conducted at a building pressure of 5 psig, and the measured leak rate was converted to the equivalent rate at 24 psig. All leak rates given in this report are based on the equivalent rate at 24 psig.

* Trademark of Dow Chemical Co.

Tests made in December 1961, and January 1962, just prior to nuclear startup, revealed that a large number of conduit seals were leaking An extensive program of conduit repair was successful in reducing the leak rate from 8.5% of building volume per day to 2.5%. Some leakage was observed through the concrete outer wall below grade level, but no quantitative appraisal of this leakage could be made. In the final test of the series, the floor at the -52' elevation was covered with water. A leak rate of 0.7% of building volume per day was measured.

No further testing was conducted until September 1962, just prior to power operation. At this time, the measured leak rate was 8.1% with the basement floor dry and 4.7% with the floor flooded. The repair of a number of leak sites was completed, and leak rates of 4.2% and 4.1% were measured with the basement floor dry, then flooded. Approval to operate the reactor was obtained from the AEC, but the maximum power was restricted to 45 MW, and the maximum exposure to 4500 MWD.

The consequences of the maximum credible accident were re-examined for the situation where iodine absorbers were present in the 770-U Building. The results of the analysis showed that, with iodine absorbers in operation, an acceptable leak rate would be 7.6% of the building volume per day (at 24 psig) with the basement floor flooded. Iodine absorber units were installed in May 1963.

The leak rate measured in May 1963 was 2.9% of building volume per day, with the basement floor dry.

A review of the HWCTR experience shows that continued effort is required to maintain an acceptably low leak rate.

Corrosion of External Surface. Inspections of the exterior surface of the containment building revealed that significant pitting corrosion is occurring in the presence of moisture that has entered through defects in the insulating system (including coatings). The maximum pit depth measured was 45 mils. The containment shell is constructed of 3/4-inch-thick carbon steel, and is covered with adhesive-bonded insulation. Primer paint was applied to the shell prior to the installation of the insulation.

A corrective program was considered to remove the insulation, clean the steel surface, and add new insulation. However, the uncertainties as to the most suitable insulating material and to the prospects for future operation, combined with the observed low corrosion rate indicated that no corrective action should be taken at that time. Periodic inspections of the building have been continued.

D_2O LEAKAGE DETECTION AND LOSSES

A program to monitor and to control D_2O leakage from the process system was pursued actively throughout the 3-year period of HWCTR operation. The materials and components used in the construction of the high pressure system were of standard industrial quality. The HWCTR did not contain special design features that might be included in a full-sized D_2O power reactor system to minimize D_2O losses. A power reactor would have specially designed equipment to minimize and to collect D_2O leakage, and perhaps a ventilation system that would permit the recovery of much of the D_2O that evaporates in the process areas.

Descriptions of measured D_2O leak rates and the methods of detection are given in reference 2.

The lowest leak rate attained during operation at full power conditions was 14 lb/day (7.3% of the moderator inventory per year), measured over a 23-day interval in January 1964. Steam generator and purge cooler losses accounted for 5.5 lb/day of the total loss; 8.5 lb/day were unaccounted for. Leak rates from 20 to 40 lb/day were measured during other periods of operation, with up to 15 lb/day of D_2O leaking through the generators.

Leaks occurred in the steam generators from the beginning of operation. Initially, the leakage occurred at defective tube sheet welds. Early in 1964, the shell side of the tubes was attacked by oxygen corrosion when oxygen control of the feed water was lost. In four separate incidents from February to September, tubes developed holes and were plugged. Other deep pits were observed that almost penetrated the tube wall. Retubing of the generator will be necessary, if the facility is reactivated. During the retubing, special attention should be given to the tube-to-tube-sheet seal welds.

D_2O leakage from the steam generators and purge coolers was detected by a cooling water gamma monitor system and by tritium analyses of cooling water samples. The gamma monitor was on-line, and provided continuous monitoring. Results of the tritium analyses were available approximately 2 hours after sampling. At the time the reactor was shut down, leak rates as low as 1 lb/day could be detected, and changes as low as 0.5 lb/day could be observed from the tritium analyses. The moderator tritium content at shut down was approximately 400 μc/ml.

Other sources of D_2O leakage were valves, flanges, joints, monitor pins, and rotating drive shafts. A program to tighten flanges, valve packings, and other mechanical fittings was set up on a routine basis. Valves, flanges, and pipe joints were wrapped with "Teflon" bags containing leak detectors. Low radiation zones were patrolled routinely to observe for leakage. Following scheduled shutdowns, high radiation zones were entered before cooldown to observe for leakage. A leak collection system to recover leakage from monitor pin connections was installed.

The total D_2O loss at HWCTR was 21,970 pounds, from January 1962 through December 1964. The average monthly rate was 610 lb. Wide variations occurred in the reported monthly loss rate obtained from inventories. The causes were inventory uncertainties and occasional spills in which a high, real loss was experienced.

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DP-232	DP-395	DP-485
DP-245	DP-405	DP-495
DP-265	DP-415	DP-505
DP-285	DP-425	
DP-295	DP-435	etc. to DP-965
DP-315	DP-445	
DP-345	DP-455	
DP-375	DP-465	
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