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NATIONAL REACTOR TESTING STATION
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SPECIAL POWER EXCURSION
REACTOR TESTS

SPERT IV FACILITY

by

R. E. Heffner, E. L. Morris, D. T. Jones,
D. R. Seaman, and M. K. Shane

PHILLIPS
PETROLEUM
COMPANY

Atomic Energy Division
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SPERT IV FACILITY

ABSTRACT

The Special Power Excursion Reactor Test IV (Spert IV) reactor is a swimming-pool type nuclear research reactor which has been constructed to provide a facility for conducting reactor kinetic behavior and safety investigations especially in the field of reactor stability. The facility has been designed to incorporate a wide range of flexibility in flow rates and direction of flow, and has a heat removal capacity of 1 Mw thermal in addition to the heat sink capacity of the reactor pools. This report describes the engineering features of the reactor and supporting process equipment as constructed at the National Reactor Testing Station.

SPERT IV FACILITY

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SPERT IV FACILITY

I. INTRODUCTION

The Spert (Special Power Excursion Reactor Test) Project, operated by Phillips Petroleum Co., was established as a part of the United States Atomic Energy Commission's (AEC) Reactor Safety Program in 1954, and is directed toward experimental and theoretical investigations of the kinetic behavior and safety of nuclear reactors. The Spert IV facility described herein has been constructed to broaden the Spert safety program in the area of reactor stability investigations, to provide a prototype for safety tests of swimming-pool type reactors, and to provide a versatile facility for testing a variety of core types and sizes.

General objectives of the Spert IV facility design are:

(1) To provide a highly flexible facility in which reactor power excursions on typical large static pool reactors could be performed, and experimental information gathered on the resulting behavior of the reactor. One of the initial objectives is to investigate reactor stability phenomena associated with water systems.

(2) To incorporate in the facility sufficient auxiliary equipment to permit simulating the operating conditions of various types of large pool reactors (forced circulation, static pools, heat removal for low power operation etc.) and low pressure (up to 100 psi) pressurized reactors.

(3) To provide a versatile facility for economical and relatively rapid non-destructive investigations of various prototype core designs including large core configurations.

(4) To provide a facility in which the effects on reactor behavior of the parameters associated with various reflector materials could be investigated.

The majority of the experimental studies are to be conducted from low initial powers and will involve a relatively small total energy release. However, provision for operation for a limited time at power has been incorporated in the design to permit investigations at conditions which duplicate, as nearly as feasible, those nuclear and hydraulic conditions normally found in large pool and low pressure reactors. Power operation also is required to obtain experimental data on the effect of initial power on reactor behavior and on the transient response and hydraulic stability of the over-all coolant system.

The basic components of the Spert IV facility are two reactor tanks and a coolant recirculation system. In conjunction with the recirculation coolant loop, sufficient heat exchanger capacity is installed to permit temperature control of the reactor pool water at reactor powers up to 1 Mw.

All auxiliary equipment needed to support the reactor facility is installed as a part of the facility.

The reactor and the major plant equipment are remotely operable, with the controls located in the control center building approximately 1/2 mile from the reactor.

The conceptual design of the facility was prepared by Phillips Petroleum in 1959. Architectural and engineering design were accomplished by B.D.Bohna and Company under contract to the AEC. Design and procurement of the fuel, the core support structure, the control rod drives and the reactor control system were assigned to Phillips Petroleum. Construction was accomplished by a lump sum contract with Technical Contractors, Inc., as the prime contractor. Construction of the facility was completed in August 1961.

This report describes the Spert IV facility and its auxiliaries generally, and is intended to provide Spert personnel, and other engineers and scientists actively engaged in the reactor kinetics and safety program, with a convenient reference on the engineering features of the reactor and supporting equipment.

Appendix A contains a tabulation of the design data for the main facility components and significant auxiliary equipment.

Appendix B contains a description of the first operational core to be investigated in this facility.

Appendix C contains a summary of pertinent engineering calculations.

II. PLANT SITE AND BUILDINGS

1. SPERT SITE

The Spert site is located within the boundaries of the National Reactor Testing Station (NRTS) approximately 50 miles west of Idaho Falls, Idaho. The location of the site with respect to other NRTS installations is shown in Figure 1.

A general plan of the Spert site is shown in Figure 2. The reactor areas have been arranged in a semicircle of approximately 1/2-mile radius from the control center and about 1/2 mile from each other. Spert I is approximately northwest of the control center. The other three reactor areas are spaced at approximately 60° increments clockwise and the entire site is encompassed by a three-strand barbed wire perimeter fence.

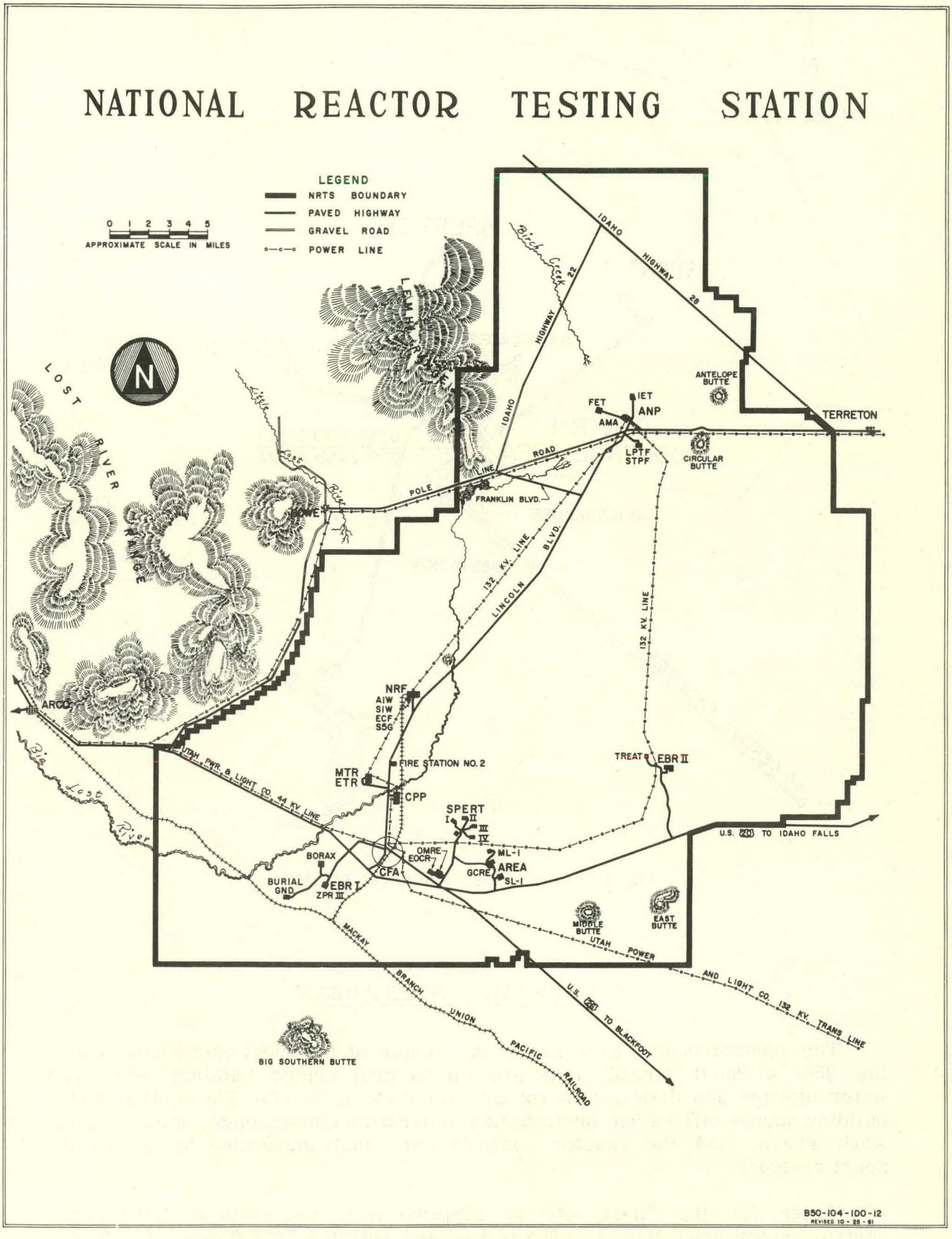


Fig. 1 Map of National Reactor Testing Station (NRTS-61-6356).

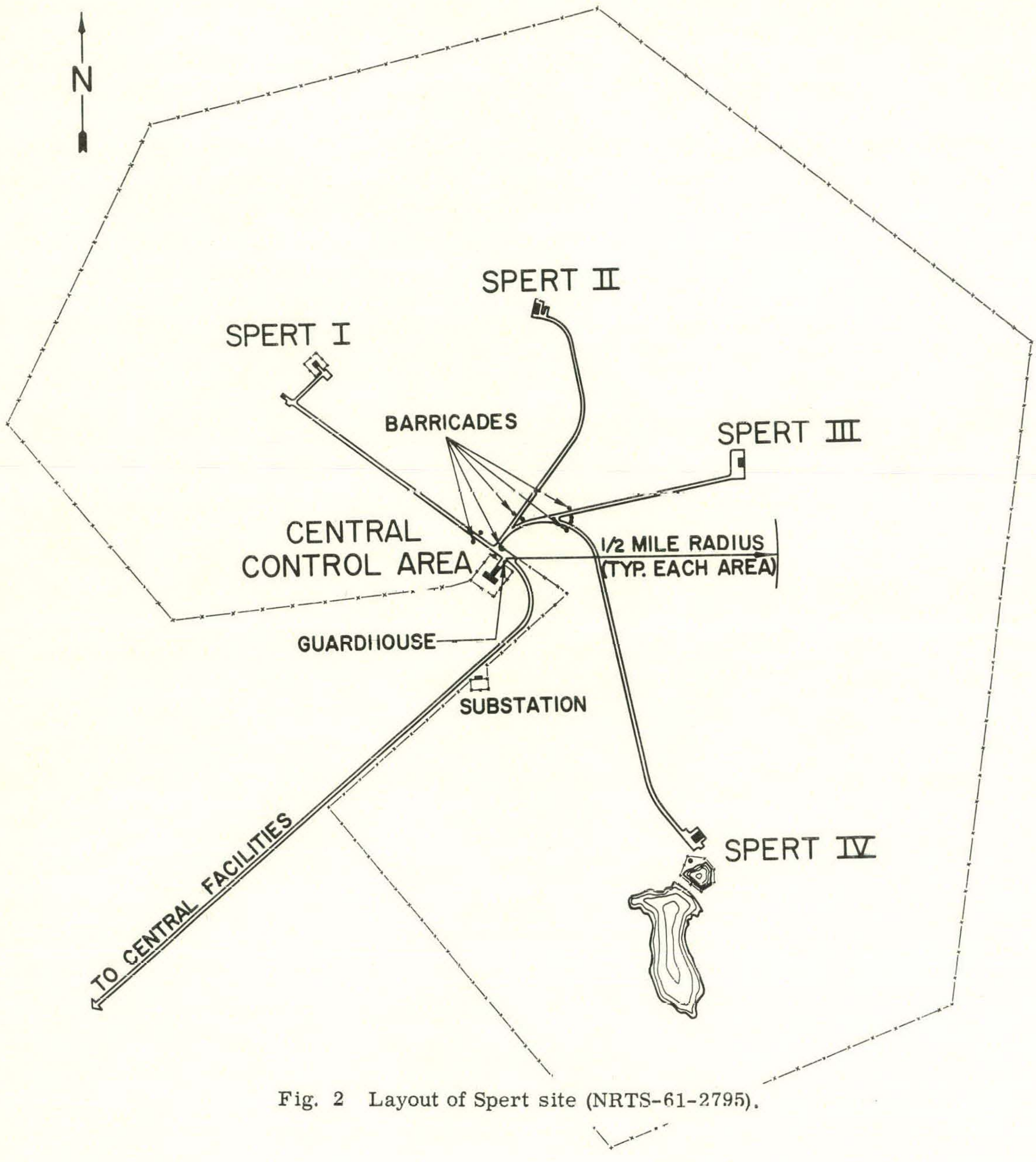


Fig. 2 Layout of Spert site (NRTS-61-2795).

2. CONTROL CENTER AREA

The control center area forms the center of the Spert operations. Within the 250- x 250-ft fenced area are the control center building and the raw water storage and distribution equipment for the Spert site. The control center building houses offices and laboratories, a darkroom, instrument and mechanical work areas, and the reactor controls and instrumentation for all of the Spert reactors.

Water for the Spert site is supplied from two wells located near the control center area. Well 1 is 653 ft deep and well 2 is 1217 ft deep. A 400-gpm deep well pump on well 1 and a 550-gpm deep well pump on well 2 supply water to the two ground level storage tanks. A total capacity of 75,000 gal of ground level storage is available. An automatic level control maintains the tank levels by intermittent operation of the pumps.

Water is distributed to all areas by two 400-gpm booster pumps which, in conjunction with a pressure control valve, maintain a line pressure of about 72 psi. A 750-gpm fire water pump supplies extra capacity if the water demand exceeds the capacity of the booster pumps.

Electrical power is supplied to the control center area and reactor areas from 13.8-kv feeders from the Spert substation. Power to the substation is obtained from the 132-kv NRTS distribution loop.

3. SPERT IV AREA

The Spert IV reactor is located approximately 1/2 mile southeast of the control center. A general layout of the area is shown in Figure 3. A three-strand barbed wire fence with warning signs is a part of the Spert area perimeter fence, and limits access to this area. Normal access is limited to the paved road leading from the control center guardhouse. The paved road is provided with a barricade and warning lights near the guardhouse to restrict use of this road during reactor operation.

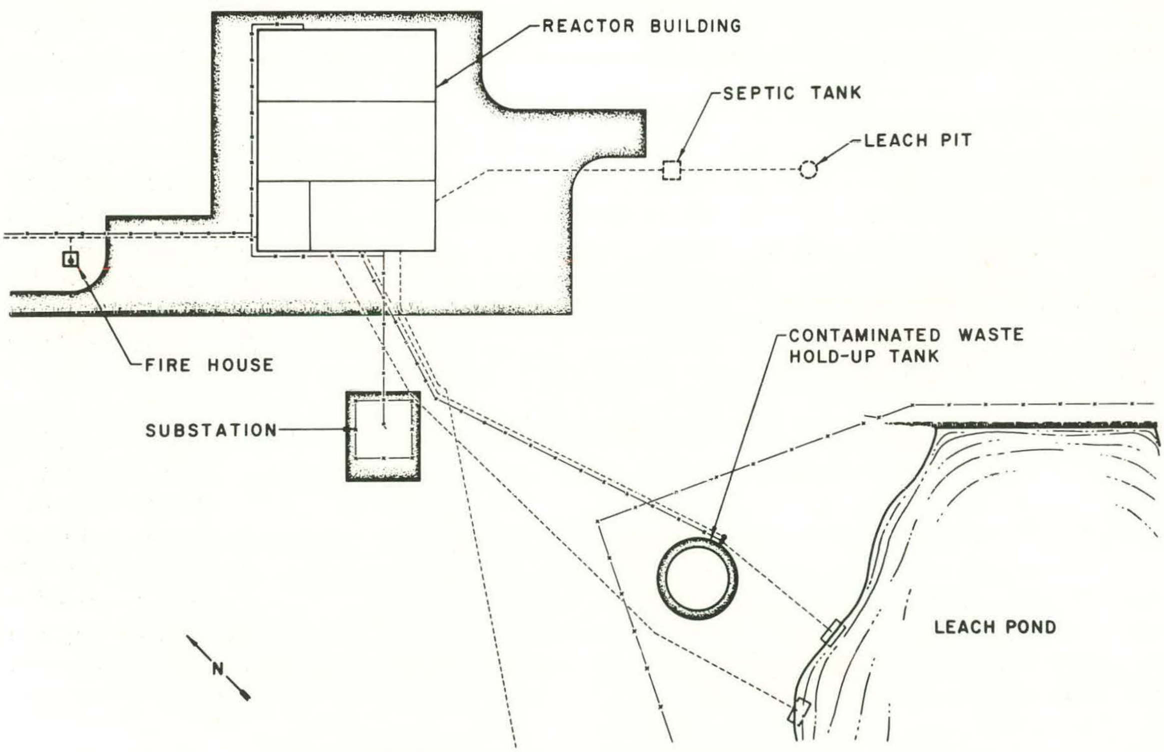


Fig. 3 Layout of Spert IV area (NRTS-61-1994).

Padlocked gates in the perimeter fence allow traffic to pass through the area for electrical power line maintenance during reactor shutdown periods. Use of these gates is under the direct control of Spert management.

Power for the reactor area substation is obtained from the 13.8-kv Spert substation. Located west of the reactor building is the Spert IV substation containing one 500-kva, 3ϕ, 13.8-kv, 480-v transformer and a 25-kva, 1ϕ, 13.8-kv, 120/240-v isolation transformer. Power from the isolation transformer is used for instrument power and experimental use. One 3ϕ, 480-v, 120/240-v transformer for lighting and other 120/240-v power is located in the reactor building and is served by the 13.8-kv, 480-v transformer.

Reactor control power is obtained in the reactor building and connected to the control console in the control center by cables laid on a pea gravel bed and covered with 4 in. of pea gravel. The gravel is used to promote drainage around the cables, discourage rodent damage, and to prevent sunlight or other damage to the cables.

Instrument signal coaxial cables are laid in a similar fashion in a parallel run at least 15 ft from the power cables.

Raw water is pumped to the reactor building though a 6-in. schedule-40 steel pipe. Because of lava bedrock on or near the surface, the line is not buried except at road crossings and in the control center area. The unburied portion is covered with a minimum of 5 ft of dirt for frost protection. Water is supplied at the reactor building at approximately 50 psig.

A 26-ft-diameter x 16-ft-high welded steel tank with a 61,000-gal capacity is used for hold-up of contaminated water waste. This tank is located 170 ft south of the reactor building. The tank is uninsulated and is equipped with immersion heaters to prevent freezing. The total volume of one reactor pool can be stored in this tank.

A leaching pond is located approximately 100 ft south of the contaminated water hold-up tank. This pond is used for the waste hold-up tank effluent and plant chemical waste. Its capacity is approximately 428,000 gal, which is equivalent to the capacity of both pools plus 328,000 gal. The normal percolation rate is 1 in. in 16 hr. Twenty-four inches of rip-rap stone is placed on the pond side of the leaching pond dyke to the top of the dyke, and to a height of 3 ft on the reverse slope of the dyke. Figure 4 is a photograph of the leaching pond taken prior to plant operations.

Waste raw water from the heat exchanger and other uncontaminated waste is diverted to a 6-million-gal capacity pond called "Spert IV Lake" located south of the leaching pond. The capacity is based on a 400-hr full-power (1 Mw) operation of the reactor during a period when the percolation rate is negligible. An overflow weir permits excess water to flow to the desert should the lake become full. The lake side of the dyke is rip-rapped with 24 in. of stone to within 1 ft of the top. The reverse slope of the dyke is rip-rapped up 3 ft from the toe. Figure 5 is a photgraph of the Spert IV lake prior to plant operations.

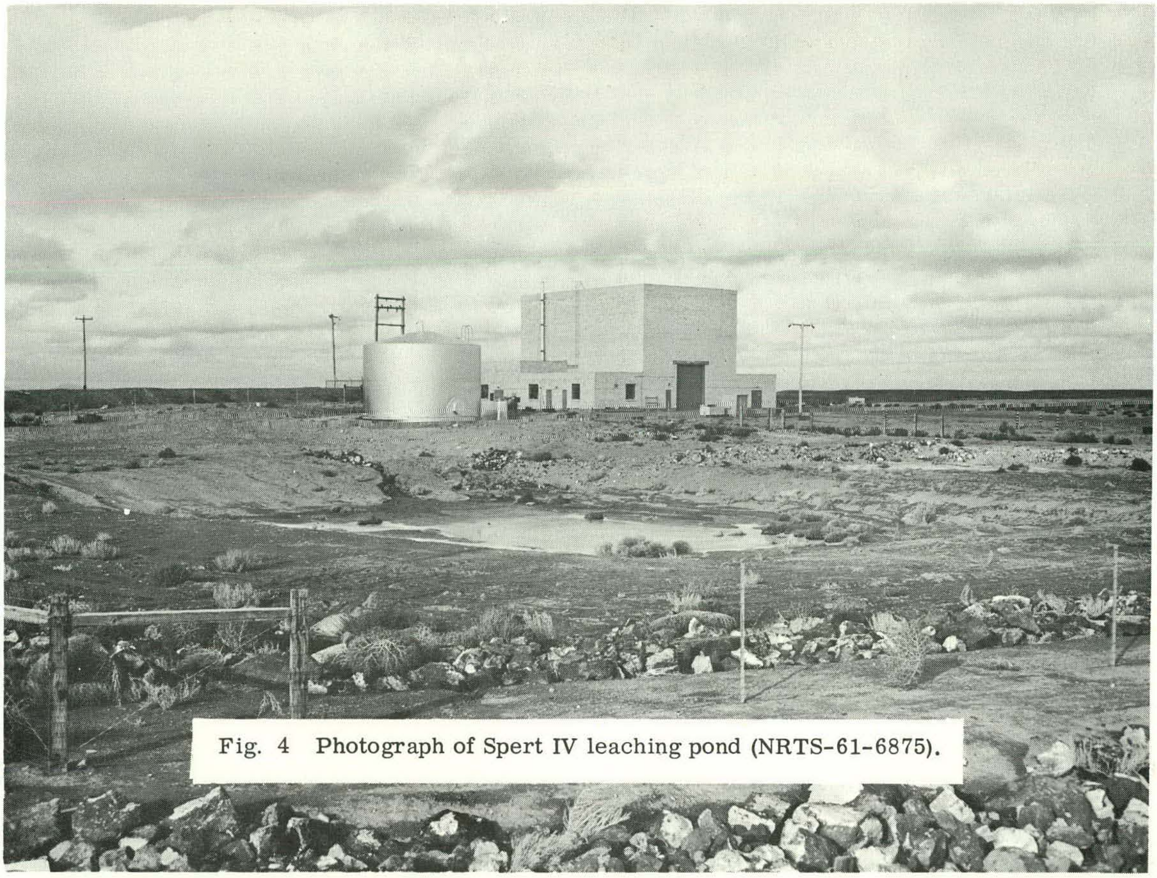


Fig. 4 Photograph of Spert IV leaching pond (NRTS-61-6875).

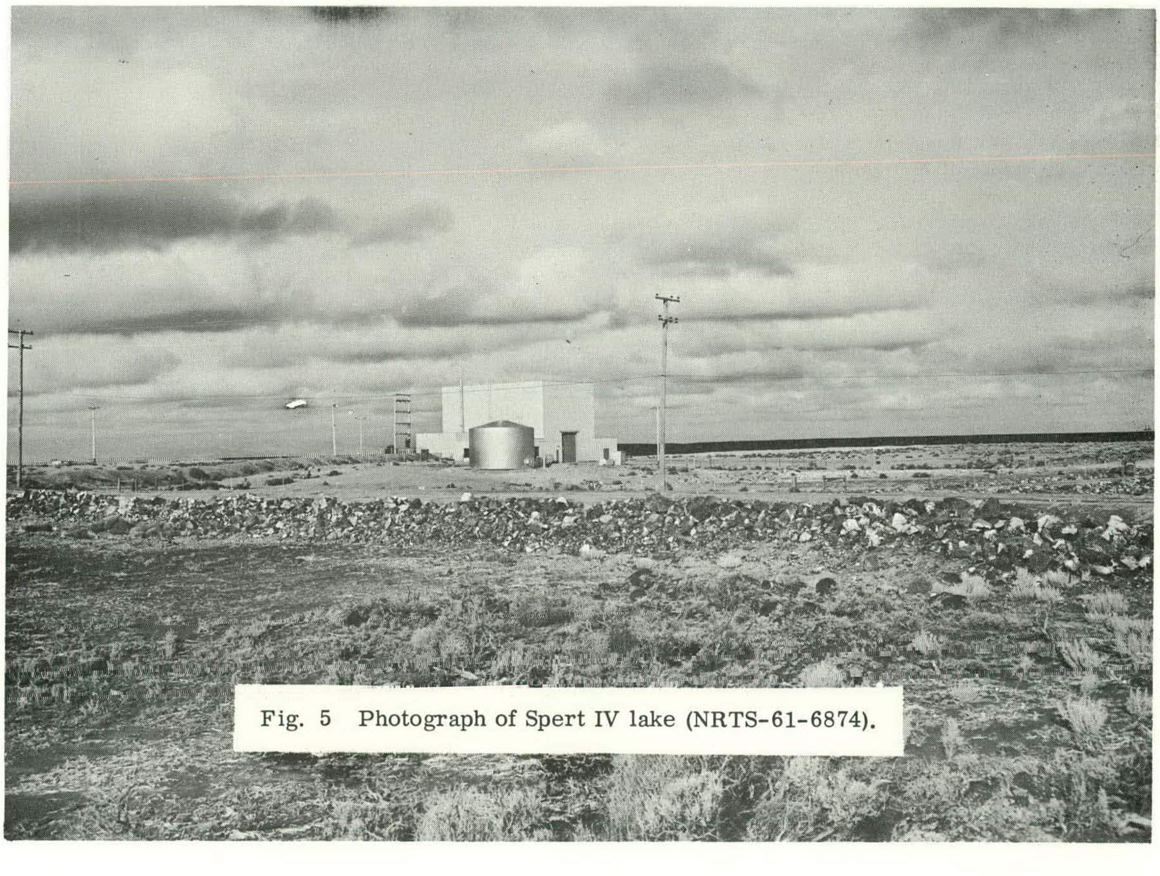


Fig. 5 Photograph of Spert IV lake (NRTS-61-6874).

4. SPERT IV REACTOR BUILDING

4.1 General

The reactor building consists of a high-bay main reactor building housing the reactor proper and process equipment, and two low-bay wings. A reactor building pictorial, plan and section are shown in Figures 6, 7, and 9 respectively, and a photograph of the building is shown in Figure 8.

4.2 Main Reactor Building

The main reactor building is a 73-ft-long x 48-ft-wide x 44 1/2-ft-high steel-girdered pumice-block structure with a 27-ft-deep full basement. A 12-ton overhead traveling crane spans the length and width of the main reactor building.

Hatches are provided in the main floor for access to the principal equipment in the basement. Equipment installed in the basement includes the 76-in.-diameter x 18-ft-high deionized water storage tank, the two coolant pumps and their associated piping, the heat exchanger, the sump pump and the air compressor with its air receiver.

Allowable floor loading in the basement is 2500 lb/sq ft. Allowable floor loading for the main floor is 750 lb/sq ft, except for the floor area south of the pools which has an allowable floor loading of 2500 lb/sq ft for supporting heavy equipment associated with loading and shipping shielding casks.

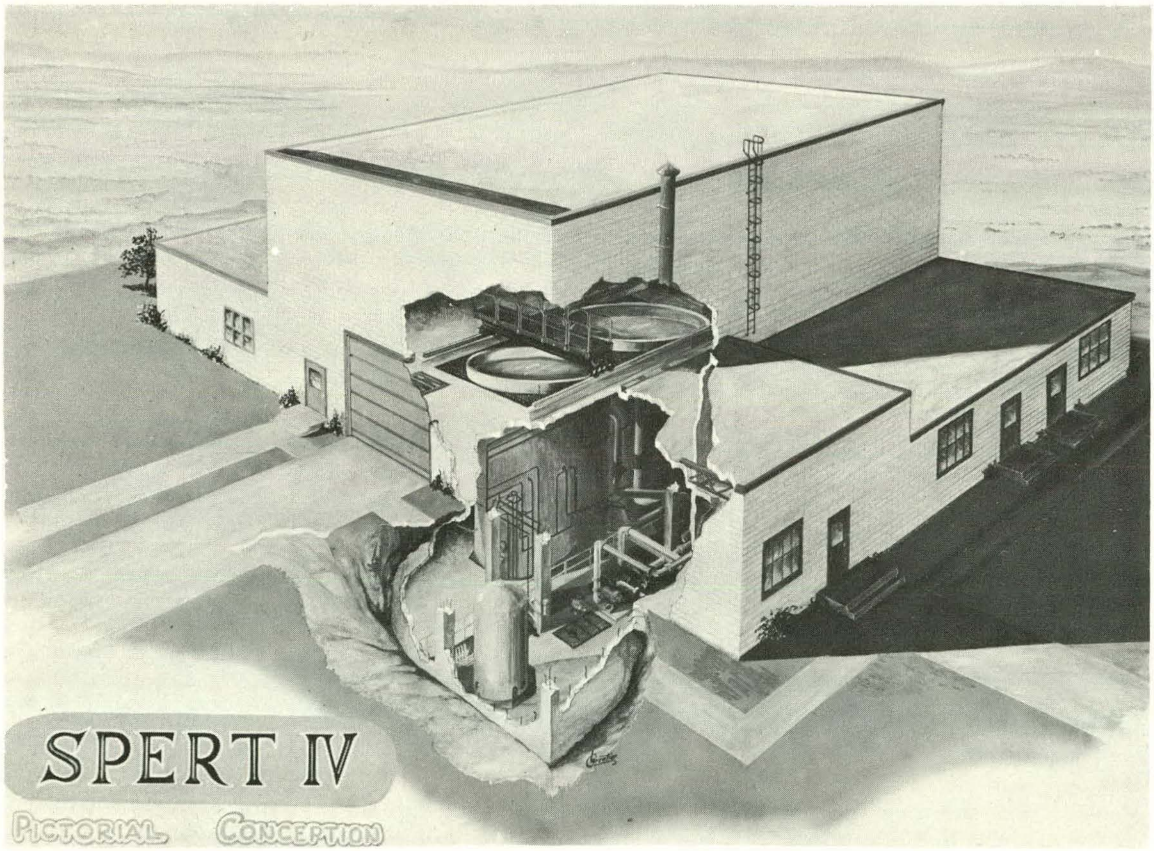


Fig. 6 Pictorial of Spert IV building (NRTS-60-4264).

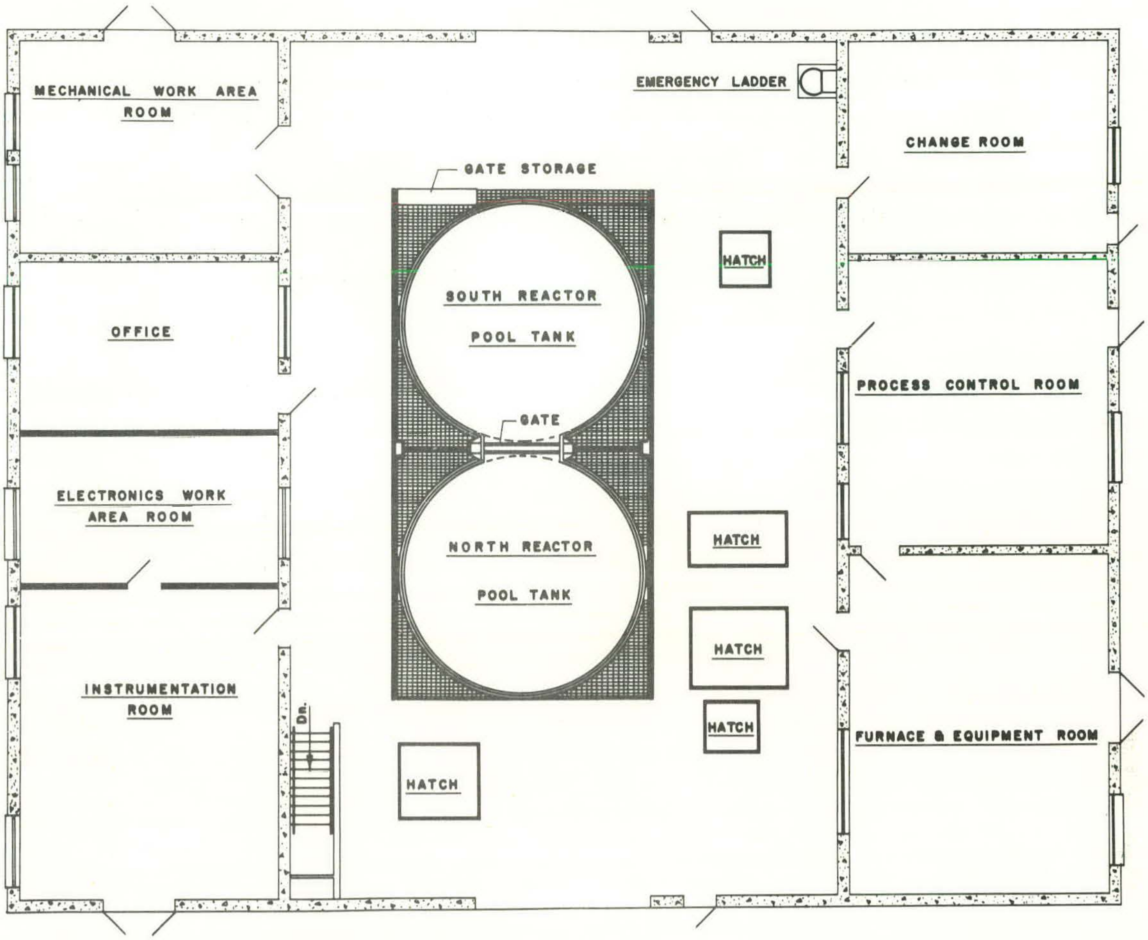


Fig. 7 Reactor building plan (NRTS-61-2774).

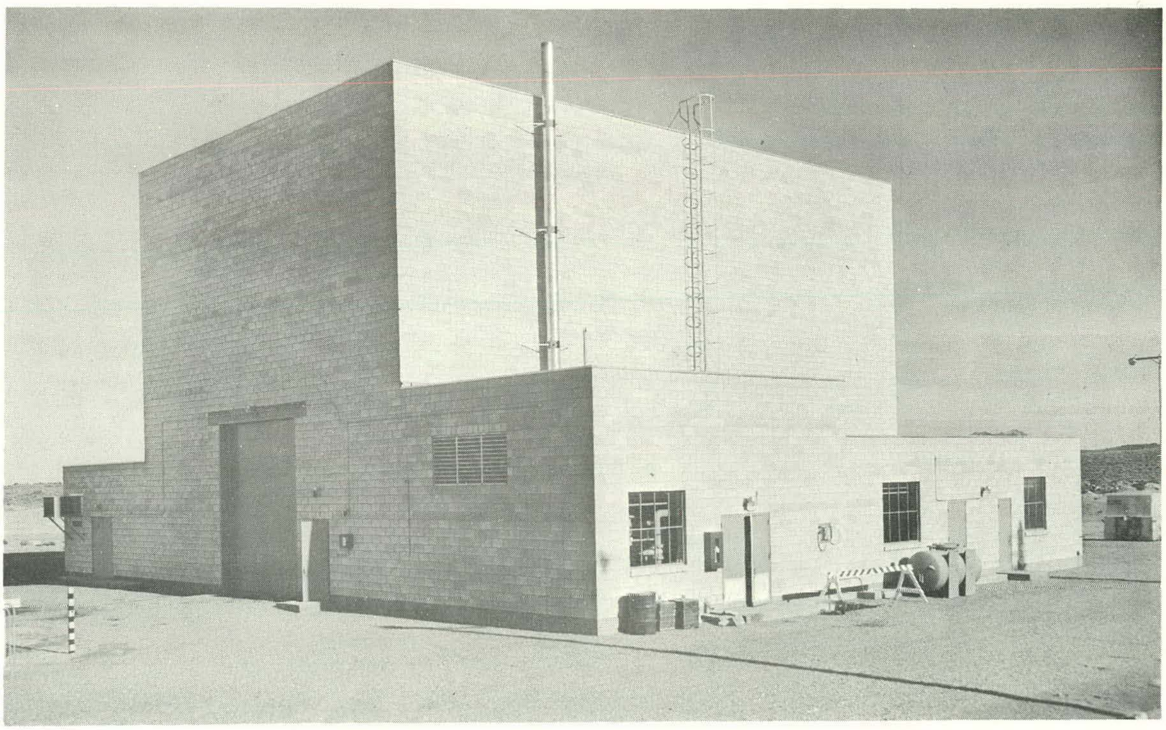


Fig. 8 Photograph of Spert IV building (NRTS-61-6379).

4.3 East Wing Building

The east wing is a 22-2/3-ft-wide x 73-1/3-ft-long x 13-ft-high pumice-block building with no basement. The permissible floor loading is 250 lb/sq ft.

Housed in the east wing building are a mechanical work area, an office, an electronics work area and an instrumentation room. The instrumentation room is served with an air refrigeration unit to prevent overheating of the electronic equipment associated with the reactor experiments.

4.4 West Wing Building

The west wing building is a 22-2/3-ft-wide x 73-1/3-ft-long pumice-block building. Roof height is 13 ft except for the area over the utility room where a 17-ft roof height has been provided for equipment clearance. Allowable floor loading in the west wing is 250 lb/sq ft. There is no basement under the west wing. This wing houses the change room, the process control room, and the furnace and utility room containing heating and ventilating equipment, water softener, demineralizers and the steam boiler.

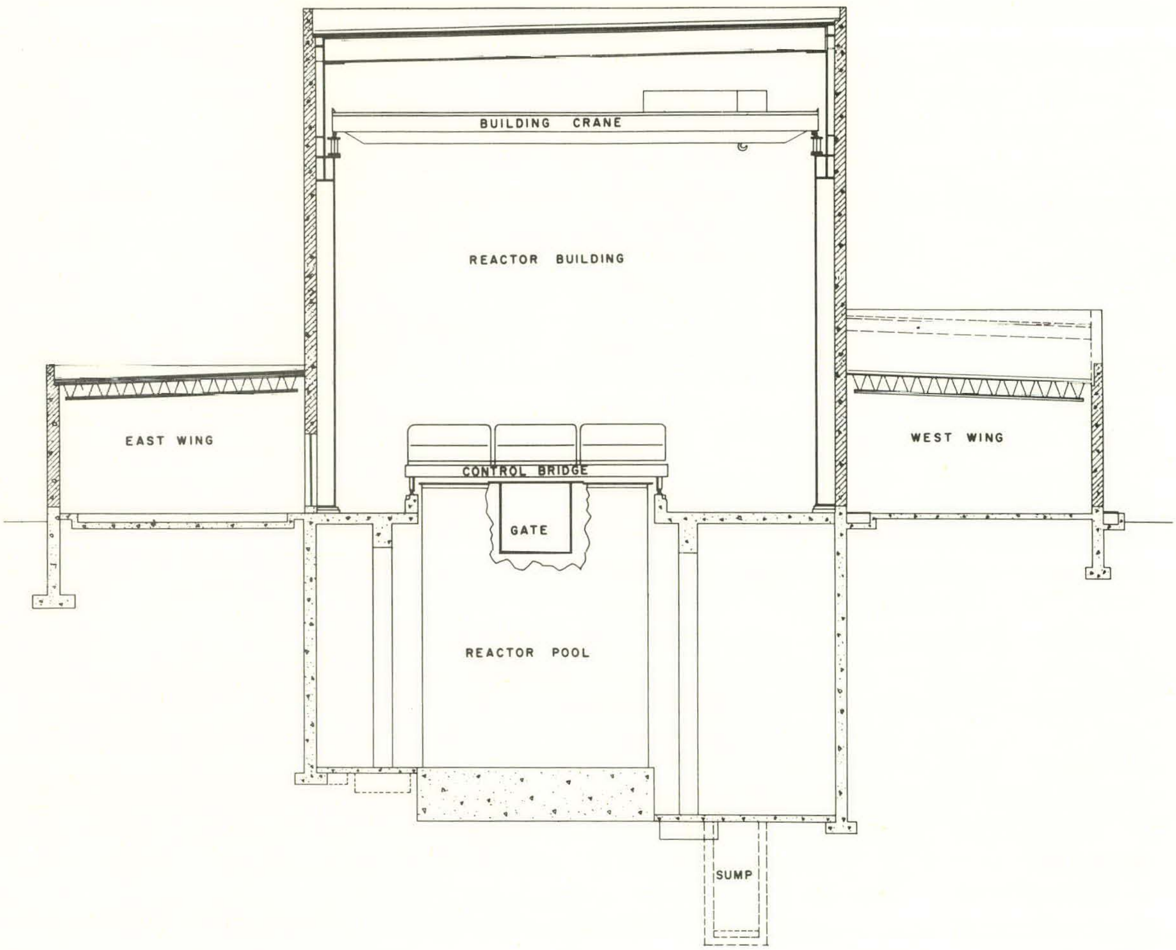


Fig. 9 Reactor building section (NRTS-61-1993).

III. REACTOR POOLS

The design criteria for the reactor pools were as follows:

(1) Pressure - 25-ft hydrostatic head plus a 50-psig surcharge.

(2) Photography - provision should be made for viewing the side of the test core with a television camera or motion picture cameras.

(3) Water temperature - variable from 50°F to 130°F.

(4) Piping connections - because of the requirements for up-flow, down-flow and/or simple recirculation, numerous nozzles at various elevations are required. The additional future need for pressure vessel connections to the pool piping requires that future piping modifications be possible without extensive contruction work such as would be necessary if concrete pools were used.

(5) Interconnection of the reactor pool with a storage area is required to permit storage and handling of hot fuel and other radioactive equipment under water. The interconnection also was to serve as a flow channel to permit utilizing the storage area as a heat sink for short power experiments generating more heat than could be removed by the heat exchanger. This interconnection must permit draining either tank for work while the other tank remains full.

As a result of a design feasibility and economic study, the architect-engineer recommended the present design consisting of two 20-ft-diameter by 25-ft-deep stainless-steel tanks joined together by a common 6- x 6-ft removable gate. Although either tank may be used for nuclear testing, the South tank is to be used primarily for hot fuel storage and the North tank is to be used primarily for nuclear testing.

The design criteria for the Reactor Pools specified pressure surges of 50 psig of 0.1-sec duration without specifying exactly the rate of loading. Considerable discussion and correspondence about this specification were generated on the basis of studies conducted by Bohna and Co., the architect-engineer. All concerned were made aware of the complexity of the problems of stress analysis for shock conditions. Much study of similar, but not identical, problems is recorded in the literature but neither the literature surveyed nor the various consultations with experts provided quantitative relationships from which to design for a shock condition. Accordingly, the architect-engineer made a recommendation to base the pool design on a static load "in excess of 50 psig". This recommendation was accompanied by calculations showing that with a 5/16-in. pool wall the hoop stress for 50 psig plus hydrostatic head would be 23,200 psi, about 77% of the yield strength of type 304 stainless steel. Weld efficiency effect and safety factor must be added to the 23,200 psi.

After further discussions during the Title I review in which many factors, such as the absence of personnel from the building during operation, costs, method of operation, the uncertainty as to the attenuation of pressure surges in the water between the reactor core and the pool walls, etc., were discussed, it was decided to use a 5/16-in. tank wall. The decision was based on judgment rather than on established codes. Adherence to the ASME Boiler Code, for example, would dictate the use of 1/2-in.-thick walls, in the absence of any shock effects. Part of the basis for the decision and a factor which influenced the evaluation of the problem greatly was the stated intention to conduct the experimental work in a sequence such that the pool walls would be subjected to pressure surges at successively higher values. Thus, any possible over-stress would be detected by pressure and strain-gage measurements and a final safe upper limit of pressure loading established. The actual pressure limit may be more or less than 50 psig, depending upon the data gathered. The use of accurate strain-detection devices is thus mandatory, and it must be clearly recognized that, depending upon the data thus accumulated during operation, a maximum allowable limit on the predicted strain resulting from a given experiment will have to be imposed. It must be recognized also that, at least as far as is known, the predicted nature of the pressure transient expected in any experiment can at best be only approximated. Therefore, some amount of uncertainty will be involved each time a new experiment is devised. Clearly, the same analysis applies to any reasonable pool wall thickness. The specifications require 100% X-ray of all vertical seams to provide the best possible integrity of pool construction. The use of clad carbon steel was avoided because of greater uncertainties about the effect of impact.

On occasion, experiments probably will be conducted with the nuclear assemblies off-center in the pool--for example in a position near the viewing-port windows. This is certain to be a less favorable condition with regard to the load capacity of the tank walls than would be the case for a pressure surge originating in the center of the pool. Again this is a situation where a quantitative stress analysis cannot be presented. It is of special importance, therefore, to conduct such experiments cautiously, with careful review of strain information at each successively higher pressure surge.

Subsequent difficulties in designing a 6-ft gate to connect the pools further de-rated the upper 6 ft of the tanks to 10-psig static surcharge.

The tanks rest on concrete foundations in the basement with the tank top reinforcing angle elevated 2-1/3 ft above the main floor of the reactor building.

The tanks were fabricated in place using AISI type 304 stainless steel. All shell welds were 100%-radiographed and all other welds were 100%-tested by a dye penetrant method. The bottoms of the tanks and the first 2-ft-high course of the shell were fabricated from 5/8-in. plate. The remainder of the shell is 5/16 in. thick with a 2-1/2- x 2-1/2- x 5/16-in. angle stiffener welded to the top course. A 3-1/2-ft-square x 1-1/2-in.-thick center plate is installed in the North tank to provide a stable reference surface in the bottom of the pool. Some buckling and movement of the floor with respect to the center is expected beyond the central stiffened portion.

Each tank contains two 12-in. coolant nozzles. These nozzles are at elevations 5 ft, 5 in. and 20 ft, respectively, from the bottom of the tanks. Depending on valving, flow can be directed either direction through these nozzles. In addition to the reactor coolant nozzles, each tank is equipped with four scupper drains located 24 ft, 1 in. above the bottom, thus effectively limiting the pool depth to 24 ft. Each tank also is served by one 3-in. demineralized-water fill nozzle, one 3-in. raw-water emergency fill nozzle, a 2-in. drain line, and 3/4-in. instrument connections.

The North tank is provided with a 16-in. coolant nozzle located in the center of the tank. Depending on the valving at the coolant pump this nozzle may be either the reactor inlet or reactor outlet nozzle. The flange face of this nozzle is accurately positioned to permit future installation of either a reactor pressure vessel or coolant piping. The center line of this nozzle is the reference point for the reactor centerline. The flange on this nozzle contains 16 slots to match a standard 150-lb flange drilling template.

The North tank also has three viewing ports, each consisting of a 16-in. nozzle covered by a manually operated slide gate and a 1-in.-thick x 18-in.-diameter glass window. The slide gate helps protect the glass from pressure surges at times when observation is not required and provides a means of shutting off water loss in case the glass should be broken. The handwheels for operation of the slide gates are located on the periphery of the North tank on the main reactor floor level.

Hold-down bolts are provided in the floor of each tank for fastening experimental equipment or instrumentation independent of the core structure. These bolts are set in the concrete foundation for the tank and are seal-welded to the bottom of the tank. The South pool is provided with ten 1-1/2-in.-diameter bolts and the North pool has fourteen 1-1/2-in. bolts.

The 6- x 6-ft pool-connecting gate is retained by a built-up yoke and tie bar. The yoke consists of 1/2-in. AISI type 304 plate welded to each tank and backed up with twin 12-in. structural-steel channels. The yoke surrounds the bottom and two sides of the 6- x 6-ft opening between the tanks and is tied together at the top with a 3-1/2-in.-diameter removable tie bar.

The gate is built up of 5-in. 6061-T6 aluminum wide flange and channel structural shapes covered with 1/4-in. 6061-T6 aluminum "skin" The gate is sealed to the yoke by an inflatable neoprene tube of oval cross section. Maximum air pressure to effect the seal is 125 psi. Leak rates of less than 1 gal/hr have been attained using this seal.

IV. CONTROL BRIDGE

The control bridge is a device used for mounting the control rod drives over the core, for supporting the reactor core structure and for accurately repositioning the reactor core and drives after moving for maintenance. Although the first core will be located in the center of the North pool, the design of the control bridge is such that the operating core may be placed in either pool or at any pre-selected location in the pool.

The working bridge proper is fabricated from two 21-in.-wide flange structural-steel girders, 22 ft long and placed four ft center to center. The ends of these girders are mounted on trucks and rails for transport along the length of the pools. The bridge also may be moved by the main building crane.

Elevation and leveling of the bridge is accomplished by means of four geared jacks which lift the bridge and trucks from the rail. An indicator dial on the jack actuator permits repositioning within 0.001 in. in elevation. Vertical adjustment range for the bridge is 5 in.

Movement of the bridge along the rails is accomplished by a manually operated chain drive mechanism.

Design live loading of the bridge is 2500 lb vertical load on each bridge member with a maximum of 0.010 in. deflection.

Open-type cable raceways are provided along the inner face of each support girder. The bridge floor is covered with removable aluminum decking to permit relocation of the control rod drives for different core configurations and for access to the area beneath the bridge. Removable handrails and stairways complete the basic bridge structure.

V. COOLANT SYSTEM

1. GENERAL DESCRIPTION

Deionized water in the coolant system serves as the heat transfer medium to remove heat generated in the reactor core and reject it through the heat exchanger to the waste-cooling water. The coolant system consists of two reactor pools, two coolant pumps installed in parallel, the heat exchanger, and associated piping which is so arranged as to provide flexibility in choosing flow patterns. The process flow sheet is shown in Figure 10.

The most common flow pattern, illustrated schematically in Figure 11, utilizes the North pool as the reactor tank. Coolant enters the core through a 16-in. flanged inlet nozzle centrally located in the bottom of the North pool. The coolant flows up through the core and leaves the reactor through the lower 12-in. nozzle. The coolant enters the pumps and discharges into a 10-in. header. The coolant then enters the 10-in. Dall flow-measuring tube and discharges into a 12-in. line which enters the heat exchanger. After passing through the heat exchanger, the coolant flows in a 12-in. line to a 16-in. tee and into the reactor to complete the circuit.

The design heat removal capacity of the heat exchanger is 4.05×10^6 btu/hr and the maximum design operating pressure, temperature, and flow are 90 psi through the heat exchanger tubes, 130°F, and 5000 gpm, respectively. After absorbing reactor heat in the heat exchanger, the waste water is routed directly to the Spert lake.

Three of the more commonly used flow patterns, in addition to that shown in Figure 11, are shown schematically in Figures 12, 13 and 14. A list of possible flow patterns is presented in Table I. Detailed descriptions of the coolant system and its component parts follow.

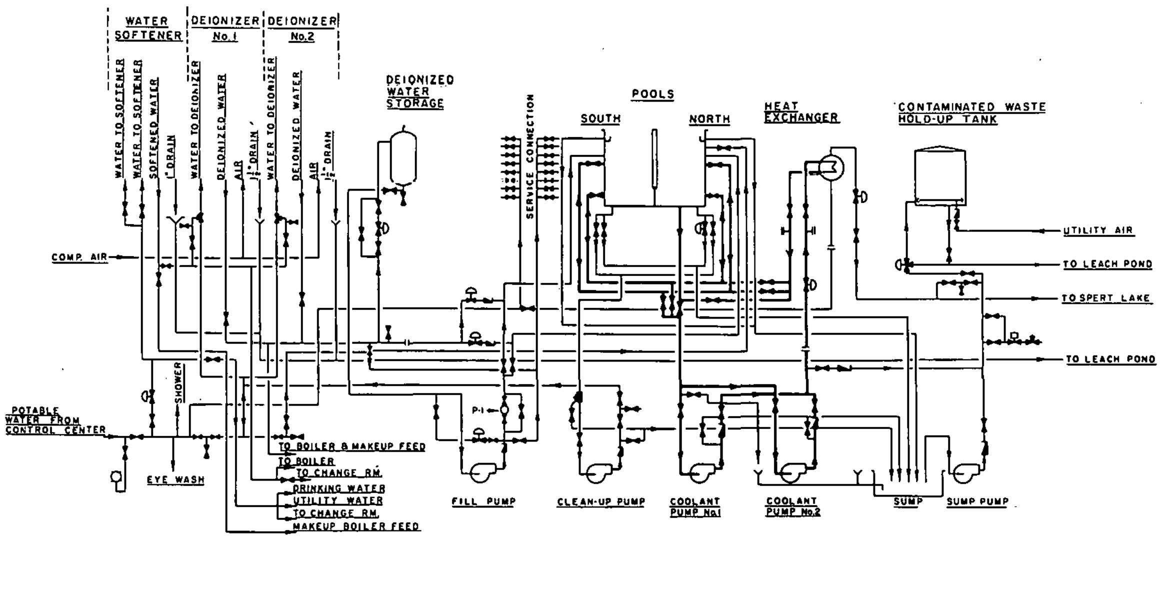
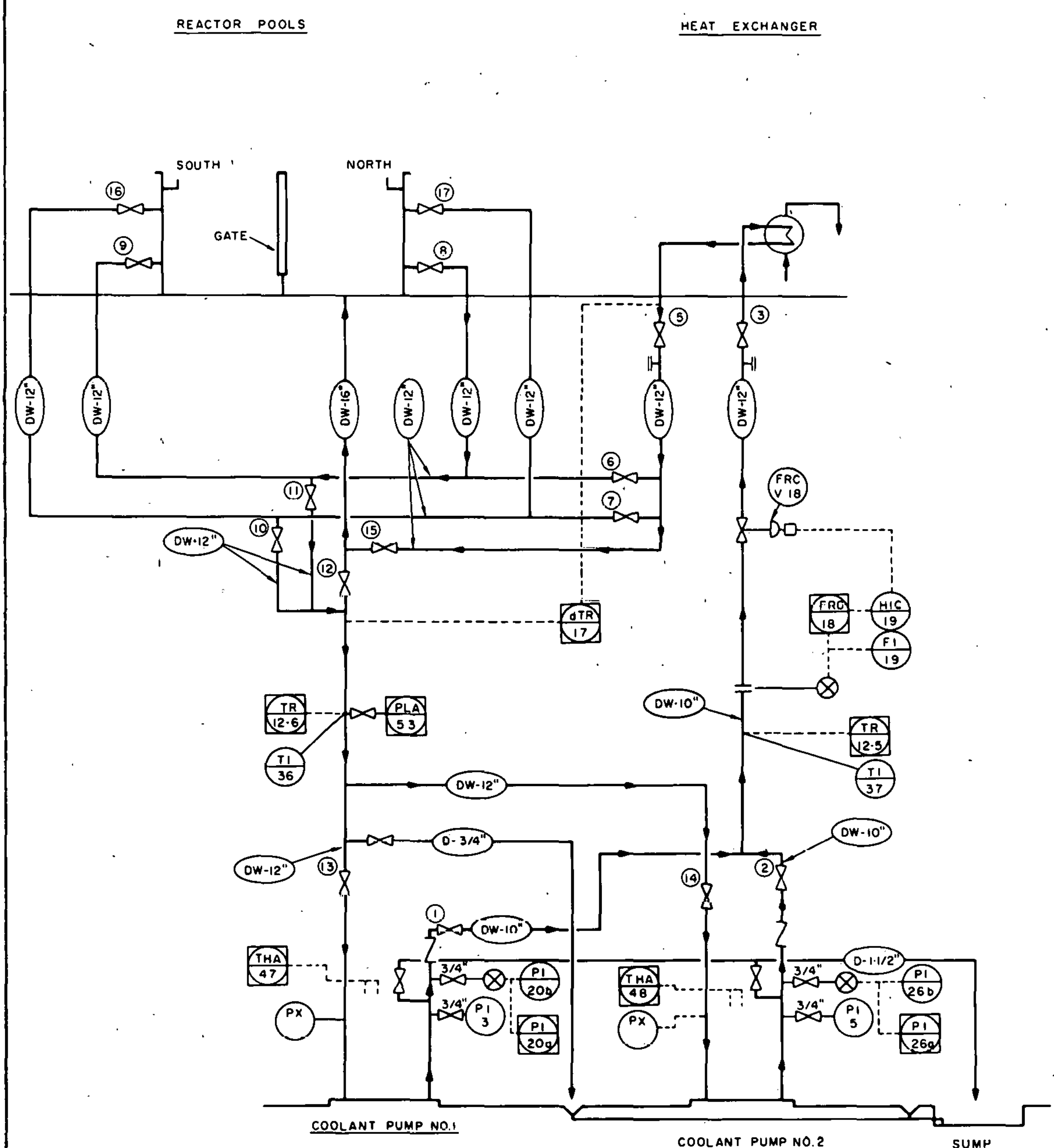
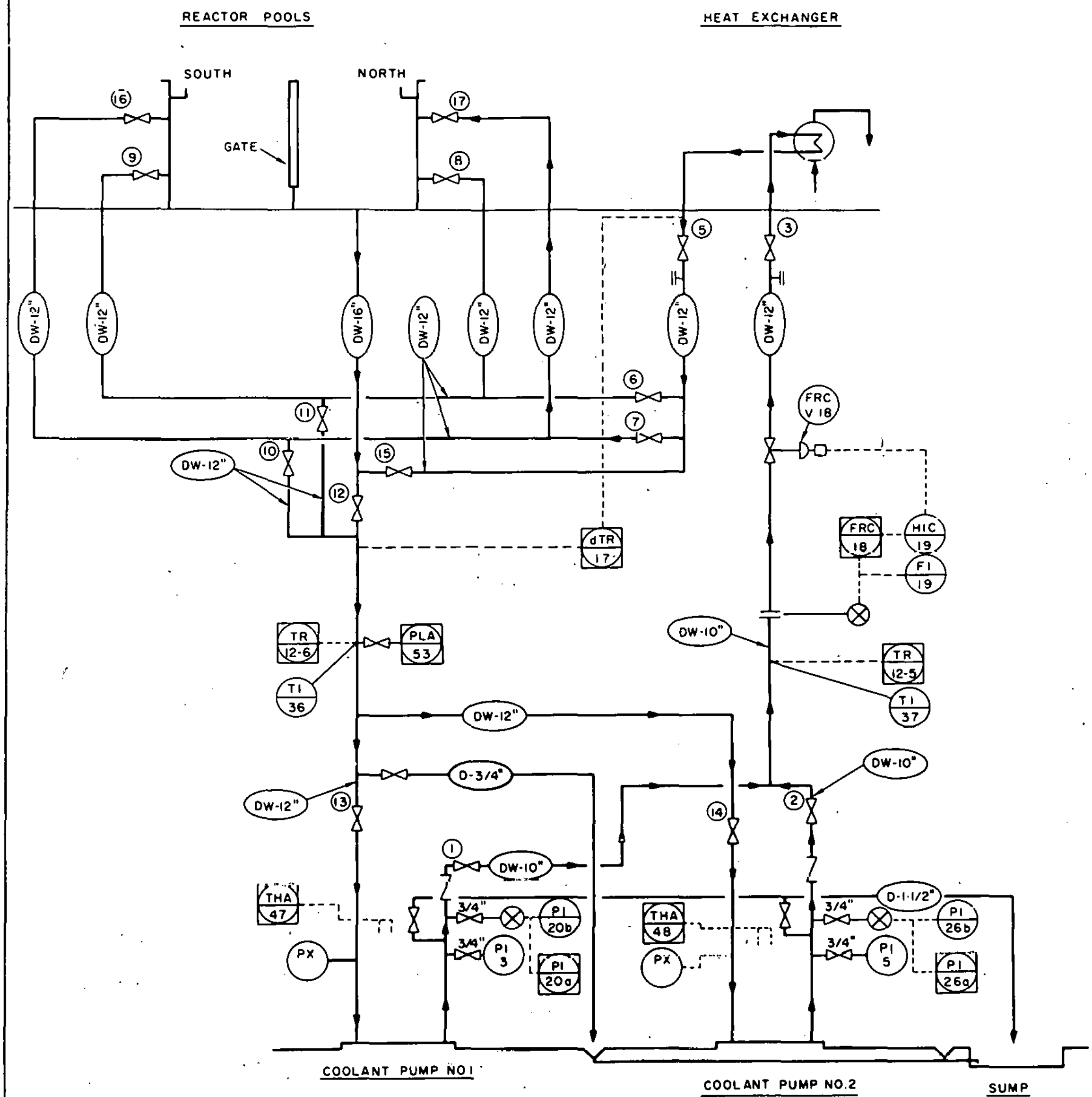


Fig. 10 Process flow sheet (NRTS-61-1995).



ALL PROCESS VALVES CLOSED EXCEPT:
VALVE 8
VALVE 11
VALVE 13
VALVE 14
VALVE 1
VALVE 2
VALVE 3
VALVE 5
VALVE 15

Fig. 11 Process flow configuration — upflow through center nozzle in North tank (NRTS-62-254).



ALL PROCESS VALVES CLOSED EXCEPT:
VALVE 12
VALVE 13
VALVE 14
VALVE 1
VALVE 2
VALVE 3
VALVE 5
VALVE 7
VALVE 17

Fig. 12. Process flow configuration - down-flow through the core, in high nozzle of North pool.

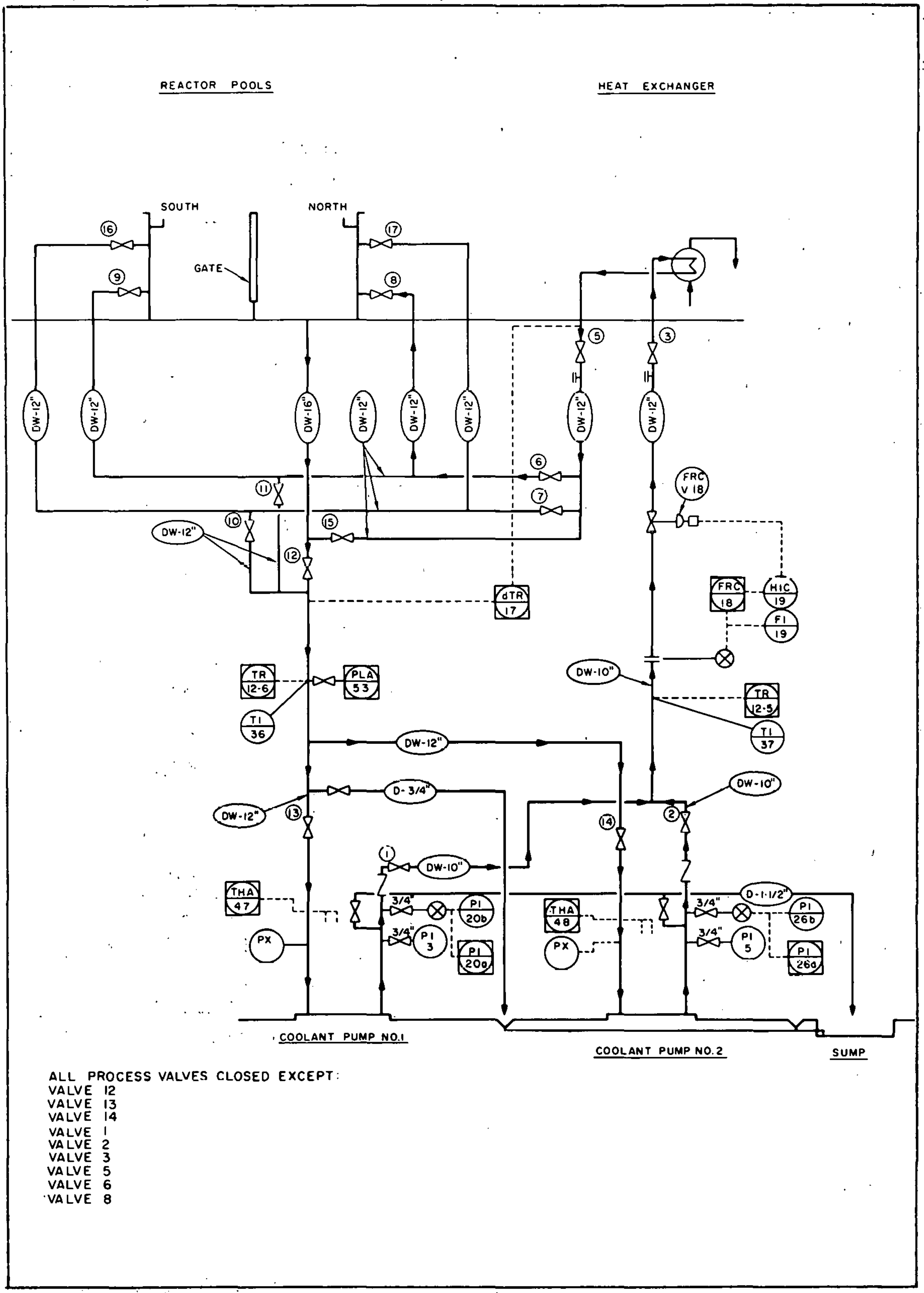


Fig. 13 Process flow configuration - down-flow through the core, in low nozzle of North pool.

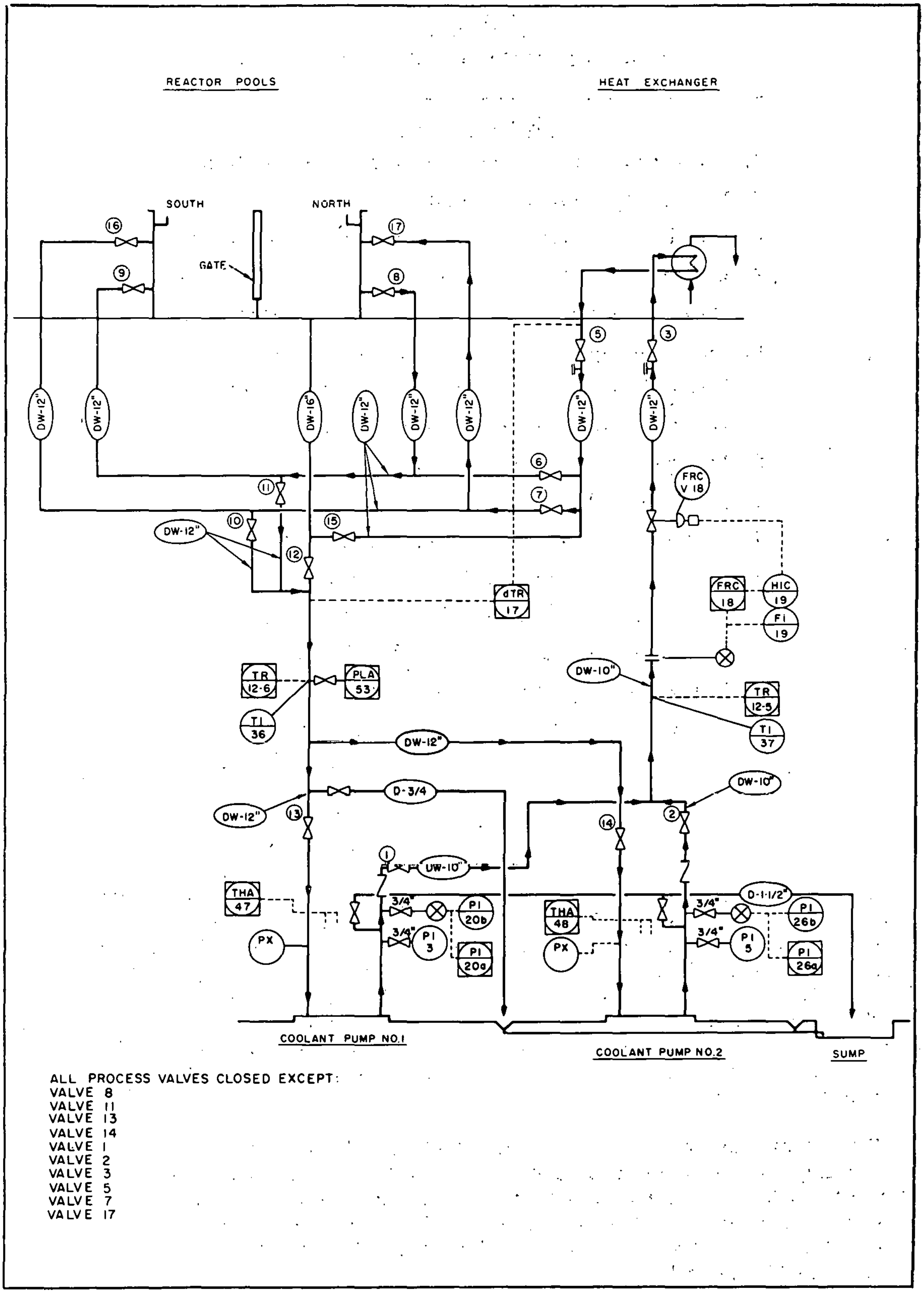


Fig. 14 Process flow configuration - in high nozzle of North pool, out low nozzle of North pool.

TABLE I

COOLANT SYSTEM FLOW PATTERNS

Flow patterns:
1. Up through core, out low nozzle on North pool
2. Up through core, out high nozzle on North pool
3. Up through core, out low nozzle on South pool
4. Up through core, out high nozzle on South pool
5. Down through core, in low nozzle North pool
6. Down through core, in high nozzle North pool
7. Down through core, in low nozzle South pool
8. Down through core, in high nozzle South pool
9. In high nozzle north, out low nozzle north
10. In high nozzle north, out low nozzle south
11. In low nozzle north, out high nozzle north
12. In low nozzle north, out high nozzle south
13. In high nozzle south, out low nozzle south
14. In high nozzle south, out low nozzle north
15. In low nozzle south, out high nozzle south
16. In low nozzle south, out high nozzle north

2. PUMPS

The coolant system includes two 2500-gpm single stage, horizontal split-volute type, centrifugal rigidly mounted pumps. The pumps which were manufactured by Worthington Pump & Machinery Corp. (8LN-14), deliver 2500 gpm at 1775 rpm and 170-ft discharge head. The pumps are equipped with General Electric 125-hp 480-volt drip proof motors. Hydraulic characteristic curves for the pumps are shown in Figure 15. These curves are based on data from factory tests on the pumps. Plant design considerations and the net positive suction head requirements of the pumps will lower the pump capacity for some of the flow configurations.

The pump casing is cast iron ASTM-A 48-56 with 25,000-psi tensile. The casing was subjected before assembly to a hydrostatic test pressure of 250 psi for 1 hr. The one-piece double-suction inpeller is AISI type 304 stainless steel and the shaft is AISI-SAE 4140. The inpeller wear-rings are 17% chrome steel and the casing wear-rings are AISI type 410 stainless steel. The flanges are rated at 125 lb ASA.

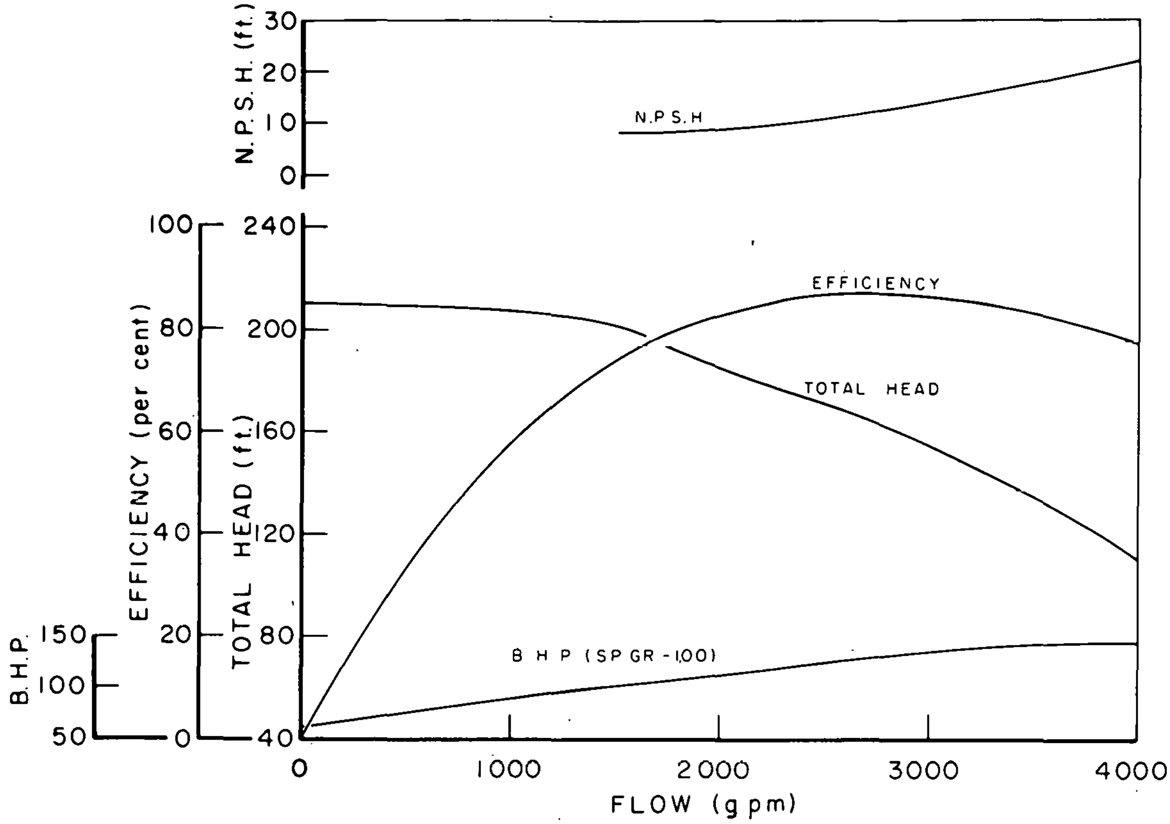


Fig. 15 Primary coolant pump characteristics (NRTS-62-116).

3. HEAT EXCHANGER

The heat exchanger serves to transfer the heat generated in the reactor from the coolant to the waste-cooling water loop for discharge to the Spert lake. The exchanger was designed for a maximum heat load of 4.05 x 10^6 Btu/hr and was manufactured by the Manning & Lewis Engineering Co. of Newark, N. J. It is a vertical, shell and tube, fixed-tube-sheet type, single pass with deionized water on the tube side, and potable water on the shell side. Construction complies with ASME code and T.E.M.A. Class A heat exchanger standards. The exchanger contains a bundle of 320 A-249 type 304 stainless-steel tubes, 3/4-in.-OD x 18 BWG on a 15/16-in. triangular pitch. The design pressure for the tube side is 90 psig at 130°F. There are 18 baffles on a 5-3/4-in. pitch. The heads are ASME flanged and dished type 304 stainless steel. The shell is 3/8-in. carbon steel with a corrosion allowance of 1/16 in. and is designed for 125 psig at 58° to 90°F.

4. CONTROL AND CHECK VALVES

4.1 Flow-control Valve

The coolant flow rate is controlled by a Minneapolis Honeywell S810-11a, 12-in. double-seated, air-to-open, diaphragm-motor valve. The body is cast iron and the trim is type 316 stainless steel. The valve can be operated either automatically or by manual control from both the control center and the reactor building.

4.2 Check Valves

A 10-in. ASA std 150-psi floating-disc check valve is installed in the discharge of each coolant pump to prevent bypass recirculation of the coolant during one pump operation. The valves are mounted vertically. They are type 304 stainless steel and were manufactured by W. G. Rovang & Associates.

4.3 Gate Valves

The various coolant flow patterns are set up with 10 and 12-in., 150-psi wedge-gate valves manufactured by W. G. Rovang & Associates. The seat, gate, body, flange faces, and stem are type 304 stainess steel. The valves are operated by hand wheels from the pump pit.

5. FLOW TUBE

The coolant flow is measured by a 10-in. flow tube installed downstream of the pumps. The tube is a plastic insert Dall flow tube Model DFT-PI manufactured by Builders-Providence, Inc., Providence, R. I. The tube liner and flanges are stainless steel. The design flow is 5000 gpm at 80 psig through the temperature range of 70° to 135°F.

6. PIPING

The coolant loop piping is welded ASTM A-312 grade TP type 304 stainless steel. Pipe 1/2-in. through 1-in. size is schedule 40, 1-1/4-in. through 12-in. size is schedule 10, and 16-in. size is schedule 20. The pipe was fabricated by the Johnson City Metal Fabricating Corp., Johnson City, Tenn.

The design pressure is 125 psig at a temperature range of 40° to 130°F. The welds were heliarced and have 100% penetration.

The fittings used are type 304 stainless-steel butt-welded fittings manufactured in accordance with ASTM A-403-56T.

Flanges 8 in. and larger are type 304 stainless-steel lap joint with forged carbon-steel back-up flanges per ASTM A-181 grade I. Flanges 6 in. and smaller are 150-lb ASA forged stainless-steel welded neck flanges manufactured in accordance with ASTM A-182 grade F 304. Two blind flanges are installed to permit future bypass or paralleling of the existing heat exchanger. The blind flanges are standard ASA 150-lb steel flanges with a 14-gauge type 304 stainless-steel face.

The piping is fixed at the inlets and outlets of the reactor pools, pumps, and the heat exchanger. Elsewhere the piping loops are mounted on spring supports which permit free vertical or horizontal movement of the loops as the system expands or contracts with changes in temperature. In addition, each of the coolant pump suction lines is provided with an expansion bellows to assist in reducing pipe rigidity. The bellows are AISI type 304 stainless steel, Joseph Kopperman and Sons packless corrugated type J-112, with two corrugations and welded ends. They are rated to 50 psig but are subjected essentially only to the hydrostatic head in the reactor pools, approximately 13 psig.

The coolant system is neither insulated nor provided with radiation shielding. Personnel access is not required in the area of the piping at any time that radiation levels may be high. Neither will personnel be permitted in the reactor building, should the pool water be grossly contaminated, until the radiation level has dropped to safe limits.

VI. PROCESS INSTRUMENTATION AND CONTROL

1. GENERAL DESCRIPTION OF INSTRUMENTATION

A wide variety of industrial type instrumentation and control is utilized in the Spert IV process plant. These include pressure, temperature, flow, liquid-level, conductivity, pH, and radiation instrumentation.

The primary process variables are sensed at the reactor building. The signals are transmitted to the reactor building process panel and/or the control center process panel. A block diagram of the plant instrumentation is shown in Figure 16. Because of the 1/2-mile distance between the control center and reactor building, electronic transmission is used for the signals sent to the control center.

Key plant process parameters are monitored on two Panalarm units, one at the reactor building process-control board, the other at the control center process-control board. These units provide warning signals during plant operation.

2. PRESSURE INSTRUMENTS

The pressure instrumentation of the Spert IV reactor process system provides the following indicators and alarms.

2.1 Pump Pressure

There are pressure indicators for the suction and discharge pressure of the primary coolant pumps. These indicators are located at the reactor building and control center process panels. Each system consists of a Honeywell Tel-O-Set PP/I Transmitter and two Marion panel meters. The transmitted signal varies from 4 to 20 milliamperes and is proportional to the pump discharge pressure. Low primary coolant pump suction pressure will activate window 6 of the control center process-panel Panalarm unit.

2.2 Instrument Air Pressure

A Mercoid pressure switch monitors the instrument air. Low air pressure will cause Panalarm window 7 to annunciate at the reactor building process panel.

The remaining pressure instruments are locally mounted gages, serving plant air system, water supplies, and pumps.

3. TEMPERATURE

The process system temperatures are monitored by iron-constantan thermocouples and resistance bulb thermometers.

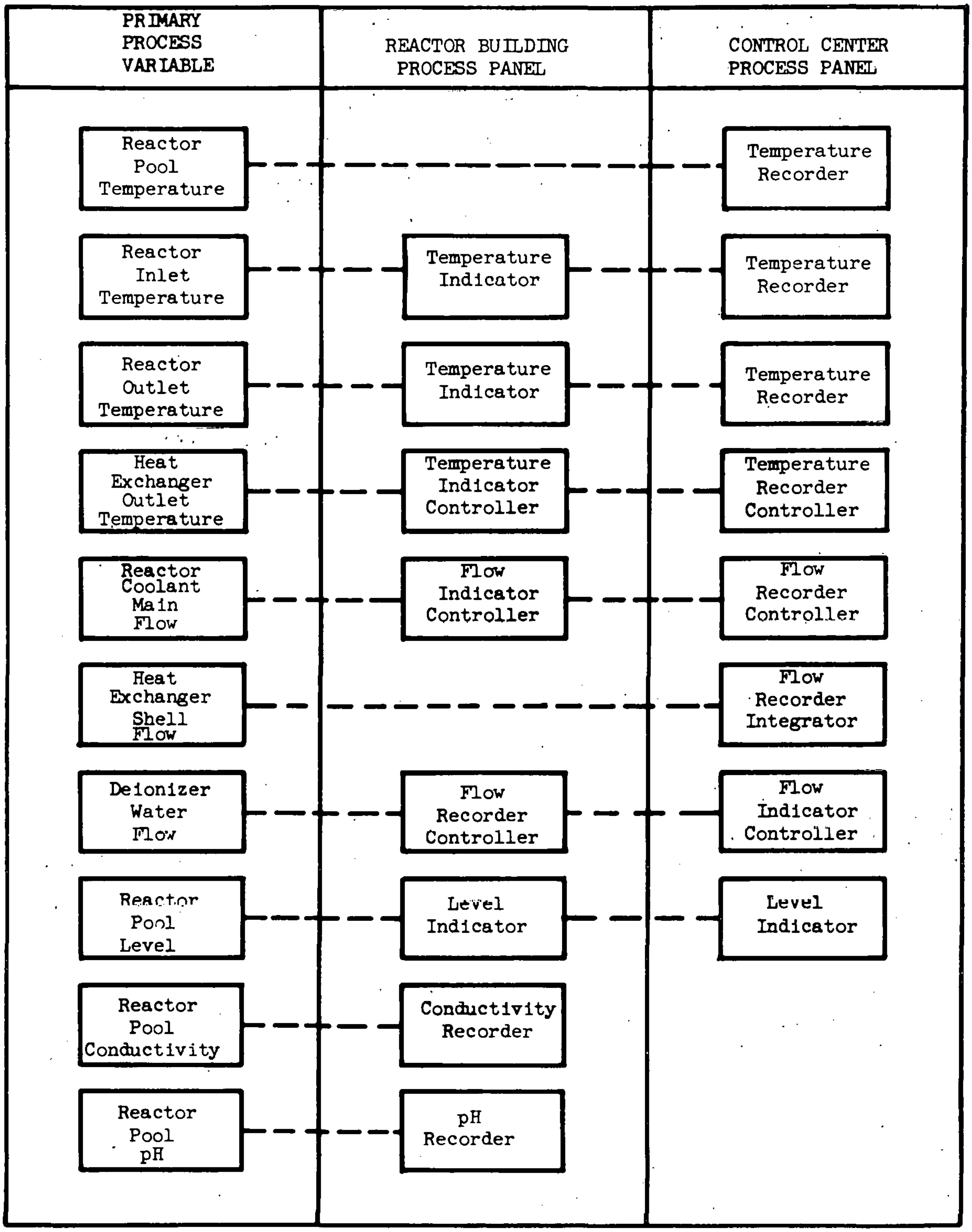


Fig. 16 Plant instrumentation block diagram (NRTS-61-2906).

3.1 Miscellaneous Temperatures

A Minneapolis-Honeywell, Class 15, multipoint temperature recorder is located at the control center process panel. The following temperatures are monitored:

Point Number	Function
1	North pool temperature
2	North pool temperature
3	North pool temperature
4	Heat exchanger coolant discharge
5	Coolant pump discharge
6	Coolant pump suction
7	Outside air intake
8	Basement air
9-12	Spare

The unit is a continuous-balance millivolt type, with a constant voltage unit for standardization.

3.2 Heat Exchanger Differential Temperature

The differential temperature between the coolant pump suction temperature and the heat exchanger outlet temperature is recorded on a Minneapolis-Honeywell single-pen self-balancing D. C. wheatstone bridge circuit with a full scale range of 10°F. The temperature-sensing elements are 500-ohm, high-speed resistance thermometer bulbs built by Minneapolis-Honeywell and are matched and calibrated to within $\pm$ 0.5%.

3.3 Heat Exchanger Temperature Controller

The heat exchanger outlet temperature is sensed by a Minneapolis-Honeywell high-speed resistance thermometer bulb assembly with an accuracy of $\pm$ 0.5°F, located in the exchanger outlet line of the primary coolant. The resistance bulb signal is converted to current by a Honeywell millivolt-to-current transmitter. The temperature signal is indicated on the reactor building process panel and recorded on the temperature recorder-controller located at the control center process panel.

The temperature indicator at the reactor building has a manual loading station so that it is possible to operate the temperature control valve located on the heat exchanger secondary coolant line from the reactor building process panel. The indicating instrument was manufactured by Manning, Maxwell, and Moore and has an output of 1 to 5 milliamperes.

The recorder-controller, located at the control center process panel, is a Leeds and Northrup Model H with a series-60 proportional-control section. This instrument operates the temperature control valve in the heat exchanger secondary coolant line. The temperature control valve is a 4-in., diaphragm-operated, air-to-open device, manufactured by Minneapolis-Honeywell. It is equipped with an electro-pneumatic valve positioner built by Moore Products Co.

3.4 Air Compressor and Water Supply

Palmer Thermometer Inc. furnished three dial-indicator type thermometers. These indicate the temperature of the potable water supply, the air compressor cooling water discharge and the air compressor after cooler discharge. They are 0 to 150°F range instruments with 1°F subdivisions.

3.5 Contaminated Waste Hold-Up Tank

The contaminated waste hold-up tank has a Fenwell temperature control switch to control the tank heaters which prevent freeze-up. The range is 50 to 100°F, with the set point at 50°F.

3.6 Coolant Pump Bearings

Coolant pump bearings are protected by four temperature-alarm switches. They are Fenwall No. 67100 types and actuate Panalarm windows 4 and 5 at the control center process panel.

4. FLOW INSTRUMENTS

4.1 Pool Make-Up from Deionizers

Deionizer water flow to the North and South pools is recorded at the reactor building and indicated at the control center panel. Instrument controls give the option of automatic or manual flow control at the reactor building, and remote automatic flow control from the control center process panel. A selector switch, General Electric, type SB-1, mounted on the reactor building process panel permits flow to the North pool and/or the South pool. The primary sensing element is an orifice plate manufactured by Vickery-Simms Inc. The differential pressure is transmitted by a Robertshaw-Fulton Control Co. transmitter in the form of a 1- to 5-milliampere D. C. signal. This signal drives the recorder-controller in the reactor building process panel which is in parallel with the flow indicator at the control center. Robertshaw-Fulton also built these two instruments. The controller signal is sent to the Microsen electro-pneumatic pressure signal valve positioners which convert the 1- to 5-milliampere signal to a proportional 3- to 15-psi air signal which drives the 1.5-in. Hammer-Dahl flow control valve.

4.2 Main Coolant Flow

The main coolant flow is recorded and controlled from the control center process panel with the option of manual control from a flow indicator on the reactor building process panel. The primary sensing elements consist of a Dall flow tube by B-I-F Industries which sends a differential pressure signal to a Microsen electronic transmitter. The 1 to 5-milliampere signal from the transmitter is sent to a Leeds & Northrup integral Series "60" control unit at the control center process panel. Scale and chart are linear with a range of 0 to 5000 gpm. The control is adjustable over the full scale. The signal also is sent to the flow indicator at the reactor building process panel. The indicator is a Robertshaw Microsen with a manual loading station. The signal from the controller at the control center or from the manual loading station at the reactor building process panel is sent to a Moore model 75 electro-pneumatic positioner which drives the 12-in. Minneapolis-Honeywell Series 800 type 11 double-seated diaphragm control valve. As stated above the flow can be controlled from the reactor building when the manual position switch at the indicator is on. By switching this unit from manual to automatic, control is transferred to the flow controller at the control center.

4.3 Heat Exchanger Secondary Coolant Flow

Coolant flow through the heat exchanger shell is recorded and integrated at the control center process panel. No control is used in this instrument loop. The flow is sensed by a Vickey-Simms orifice assembly installed in the 4-in. line and sized for 100 in. of water differential pressure with 250 gpm of flow. The differential pressure from the orifice is transmitted in the form of a 1-to 5-milliampere D. C. signal to the recorder by means of a Robertshaw Microsen transmitter. Minneapolis-Honeywell provided the recorder for this instrument loop. Mounted within the recorder is a six-digit electronic integrator.

4.4 Deionizer Flow

The flow through the two deionizers is metered by a positive displacement "Rotocycle" made by the Rockwell Manufacturing Co. Alarm on the meter contacts are adjustable from 100 to 100,000 gal. The alarms actuate windows 1 and 2, labeled "Capacity Exceeded," on the reactor building Panalarm unit.

4.5 Pool Make-Up from Storage Tank

During deionizer shutdown, deionized water can be pumped from a storage tank into the reactor pools. This flow is metered with a Brooks rotameter, range 0 to 80 gpm, installed on the fill pump discharge line.

5. LEVEL INSTRUMENTS

5.1 Reactor Pool Level Indicators

The reactor North pool has a Gilbarco level transmitter. The sensing probe is housed in a stilling chamber located in the northwest corner of the main reactor building. A line from the bottom of the stilling chamber is connected to the bottom of the North pool. From the transmitter location, a signal is sent to Minneapolis-Honeywell recorders at the reactor building process panel and control center process panel. The water height may be read to the nearest foot on a 0-to 25-ft scale. The height above the nearest integral number of feet can be read on a 0-to 12-in. scale. A switch permits selection of the foot reading or the inch reading. A green light on the instrument indicates that the foot scale is being used while a red light indicates that the inch scale is being used.

5.2 Deionized Water Storage Level Control

Level control instrumentation on the deionized water storage tank indicates the level, controls it, and actuates a low-level alarm.

The level indicator is a Shaw and Jurs target-type gage. The diaphragm-operated control valve is a Kiely & Mueller 2-in. size. Limit switches mounted on the target gage control the inlet water valve and actuate low-level Panalarm window 3 on the reactor building process panel.

5.3 Contaminated Waste Hold-Up Tank Level Control

The contaminated waste hold-up tank level control is a Minneapolis-Honeywell Tel-O-Set unit which converts the level signal to 4 to 20 milliamperes

D. C. in proportion to the measured level. The signal is sent to two Marion panel meters. One meter is mounted at the bottom of the tank, locally, and the other is on the reactor building process panel. A relay is set to close when the water level reaches 14 ft. This relay actuates Panalarm window 6 at the reactor building panel and Panalarm window 2 at the control center process panel.

5.4 Air Compressor After-Cooler Level Indicator

A 6-in. sight glass water level indicator is provided on the plant air compressor after-cooler.

5.5 Sump Level Control

The sump pump, actuated by high or low water level in the sump, controls the sump level. The switch that controls the sump pump was furnished by Magnetrol Inc. The upper and lower limits are set by two adjustable buoys mounted below the level switch.

5.6 Pool Drain Controls

The North pool can be drained remotely to the building sump by actuating a 2-in. solenoid valve from the reactor building process panel or the control center process panel. The manual control switches were furnished by General Electric, the solenoid actuaters by the Automatic Switch Co., and the split "Y" valve body by Hammer Dahl.

6. CONDUCTIVITY

The conductivity system has three water test locations--- the deionized water supply, the North and the South pools. These three readings are recorded on a Minneapolis-Honeywell multipoint instrument located at the reactor building process panel. The recorder has an adjustable alarm point over the entire range of 0 to 10 micromhos. This alarm point will actuate Panalarm window 4 at the reactor building process panel and Panalarm window 1 at control center process panel. All three conductivity cells were furnished by Industrial Instruments and are temperature-compensated. The two cells measuring the reactor pool conductivity are rated for 1-psi pressure and the cell in the dionized water supply is rated for 100-psi pressure. The recorder is a continuous self-balancing unit with a wheatstone bridge input to match the cell constants furnished.

7. pH MEASUREMENT

The deionized water to the reactor and the water in each of the pools are monitored by the pH measuring system. A continuous sample is passed over the electrodes housed in a stainless-steel flow assembly to insure a representative reading. It is possible, through the use of two switches, to remotely standardize each of the electrode assemblies by operating certain solenoid (Automatic Switch Co.) valves. These switches close the sample valve, drain the flow assembly and fill the assembly with a prepared buffer solution. A multipoint switch allows the reading of the three pH stations, one at a time, through a simple pH amplifier. The pH amplifier has an indicating scale and range of 3 to 10 pH, with an appropriate output to drive a recorder. The unit was built by Beckman Instruments, Inc. A pH recorder, manufactured by Minneapolis-Honeywelll, is synchronized with the switching unit to record one point at a time. It has one adjustable alarm point, which can be set throughout the entire range of the recorder. Panalarm window 5 at the reactor building and Panalarm window 1 at the control center process panel are actuated by this recorder. The selector switches for standardizing of the electrodes are rotary-cam-operated, multistage type SB-1, manufactured by General Electric.

8. RADIATION INSTRUMENTS

There are four systems measuring radiation level-the direct radiation area monitors, a particulate air monitor, a heat exchanger effluent monitor and a sump pump water discharge monitor.

8.1 Area Monitor System

In the reactor building there are five Victoreen area-monitoring sensing elements, two in the basement and three on the first floor. Each element has a self-contained strontium-90 calibration source, which can be exposed by operating a "calibrate" switch located on the respective indicator panel located in the transient instrumentation room in the reactor building. The basic unit is a Victoreen No. 712 and the five plug in stations are No. 715B. Each of the five stations has a 0.1 to 100-r/hr range. A two-channel, two-pen Minneapolis-Honeywell recorder at the control center process panel records the area-monitoring signals as well as the heat exchanger effluent radiation level. A selector switch mounted below the recorder allows the area-monitoring pen to record one station at a time. There is no alarm point on this pen.

8.2 Air Monitor

A Nuclear Measurement Corporation constant air monitor, located in the reactor building, continually samples the air for particulate or gas radiation. The radiation level is recorded on a two-channel, two-pen Minneapolis-Honeywell recorder located at the control center process panel. The other channel is used to monitor the sump pump discharge. If the constant air monitor should alarm due to high radiation, the air conditioning and ventilating system assumes an emergency state. The ventilating fan will stop and the louvers will close to contain any contaminated air in the building. With a key switch at the control center process panel, supervisory personnel may turn on the fan and/or move the louvers in order to exhaust contaminated air from the building.

8.3 Heat Exchanger Effluent Radiation Monitor

The shell side flow of the heat exchanger goes directly to Spert IV lake for disposal. The risk of contamination from this source is very slight. However, the stream is monitored by a submersible Riggs ionization chamber and the signal is indicated and recorded. An alarm point is adjustable over the full range of the recorder. In the event of an alarm, operating personnel could then stop the flow to Spert IV lake. The amplifier and control unit for this radiation system are located in the transient instrumentation room in the reactor building. This unit is a Riggs Nucleonics Corp. Model GA-3, three-decade logarithmic 0.01 to 10-mr/hr instrument with a 1 to 10-millivolt signal output for remote recording. The recording of this signal is accomplished by using the second amplifier and pen section of the area monitor recorder. The alarm point actuates Panalarm window 3 at the control center process panel.

8.4 Waste Water Monitor

The sump pump discharge may go to the leaching pond or be diverted to the hot waste hold-up tank. Hold-up of waste water is automatically controlled through the use of a continuous monitor. The primary element, a Tracerlab Model MW-2 water sampler and scintillation detector (Tracerlab Model MD-5 14), is mounted on the sump pump discharge line. The basic control and amplifier, located in the transient instrumentation room at the reactor building, is a Tracerlab log ratemeter Model MM-1 incorporating a linear amplifier, a log ratemeter alarm and a high-low voltage supply. This instrument has five switch positions for range-changing, thus allowing readings from 2 to 5 decades full scale, The switch must be set at 3 decades to match the recorder scale at the control center process panel. If the radiation level of the effluent is greater than the setpoint, the 6-in. three-way valve, furnished by Minneapolis-Honeywell, will divert the flow from the leaching pond to the hot waste hold-up tank. The valve is installed so that air on the diaphragm causes flow to the leach pond, thus an air failure will cause flow to the hold-up tank. A manual override for this valve has been provided which is mounted on the control center process panel. A solenoid valve opens each time the sump pump operates, thus allowing an effluent sample to be taken. The waste water radiation level is monitored and recorded by the second pen of the air monitor recorder. High radiation will actuate Panalarm window 7 at the control center and window 8 at the reactor building process panel.

9. BOILER INSTRUMENTATION

The boiler unit which serves the heating and ventilating needs for the reactor building has an alarm. This is a standard alarm which functions in the event of a high water level, low water level, or flame failure condition. The alarm signal, labeled "BOILER", is located in Panalarm window 7 at the reactor building and Panalarm window 8 at the control center process panel.

VII. CONTROL SYSTEM DESIGN

1. INTRODUCTION

1.1 General Discussion

This section of the report is devoted to a discussion of the various components of the Spert IV reactor control system with particular emphasis on the functional operation of the items discussed. In order to establish a framework for such a descriptive discussion, consideration is first given to the various requirements which the control system must fulfill.

From a general viewpoint the primary design requirements are that no hazard to personnel shall stem from system operation and that known risks to equipment shall be minimized including those risks demanded by the experimental program. The control system must provide proper manipulation of control units and must furnish information on all operations performed and indications of equipment failures or improper operations. All functions should be performed in such a manner that any component failure which constitutes loss of control shall shut down the system automatically.

These control system requirements which are a consequence of the purpose of the facility and therefore of its mode of operation also must reflect somewhat the philosophy of operation of the facility. The purpose of Spert IV is to provide a facility in which experimental programs can be carried out to develop information on the kinetics of a variety of reactor systems and on the inherent physical mechanisms which affect the neutronic behavior, and thus the safety, of these reactors. The experiments which will be performed include transient power excursions initiated by programmed reactivity perturbations. The Spert IV control system provides two primary means of rapid reactivity addition: ejection of a "transient" rod and fast withdrawal of the control rods, both of which are discussed in more detail below. Control rods in the existing Spert reactors are designed in such a manner that withdrawal of rods removes neutron-absorbing material. In some core designs the rods also include a "fuel follower" so that control rod withdrawal also adds fuel to the core. The transient rod is essentially an inverted control rod of the first type and is used for the initiation of step-wise reactivity perturbations. Raising the transient rod draws neutron-absorbing material into the core and reduces reactivity of the system.

The philosophy of operation of the Spert reactors provides that no nuclear operation of the facilities be conducted with any personnel within approximately one-half mile of the reactor. Thus, the control system design provides for operation of the facility from the control center building, which is approximately one-half mile from the reactor.

The variety of tests to be performed and the short test-time interval for most of the experiments performed led to the selection of a simple control system in which operation is strictly manual with no servo or feedback loops in the control system. Because of the short time scale for the tests, the individual functions required during a transient test, such as ejecting the transient rod, starting data recording and photographic equipment, and scramming control rods, are programmed on a sequence timer and the test is initiated by starting the timer. The reactor operator is always under the direct surveillance of at least one other qualified reactor operator who provides backup and, together with all other persons in the control room, has the authority and responsibility to "scram" the reactor in the event of any unanticipated situation.

Because the action of conventional power level or period scram circuits would in many instances compromise the acquisition of information for which the experiment is conducted, such scram circuits are not used in the control system design. Provision is made, however, for the inclusion of special scram circuits for specific experiments where operator fatigue might become a factor and where the action of the scram circuits would not interfere with the purpose of the experiment. Permanent incorporation of such "scram" circuits in Spert IV would not only in many cases compromise the acquisition of information for which the experiment is conducted, but also would compromise the development of a proper operator attitude. The type of tests performed in the Spert reactors requires that the operator be cognizant of the safety implications of each individual act in the performance of a test. This attitude must carry over beyond the test to all activities such as fuel manipulations and changes in the reactor core, components, and control systems as necessitated by the experimental programs. The development of this attitude can be inhibited severely if an individual believes that he can err and have his error automatically compensated for by an automatic trip circuit. A reliance on protective devices, which frequently must be by-passed or for which the set points must be specified and adjusted prior to each test in order not to interfere with the test, actually would result in a less safe operation. The required attention span of the operator is very brief for most of the experiments performed. Thus, the need for feedback control and safety scram circuits because of the possibility of operator inattention or fatigue is obviated.

2. DESCRIPTION OF CONTROL SYSTEM

2.1 Coupling Magnets

The Spert IV control rods are coupled to the drives by means of electromagnets. To "scram" the reactor, the magnets are de-energized and the control rods are allowed to fall into the core by gravity after a spring-assisted breakaway. The electromagnets used for coupling the rods to the drives can be energized individually so that all rods need not be withdrawn at once. The transient rod operates in the same fashion except that an additional mechanical coupling is provided to insure that the transient rod remains coupled to the drive until the intentional initiation of a power excursion.

The electromagnets of the Spert IV drives are cylindrical in design, with an outer diameter of 5 in. An axial section of the core and armature is of conventional "E-1" appearance. Twelve radial slots are milled partially through the inner and outer shells to impede circulating currents induced upon de-energization, thereby decreasing the release time. Saturation current for the 18-ohm coil is about 0.6 amp, which provides a lifting force of about 900 lb. Normal operating current is expected to be about 0.1 amp, which will lift 300 lb. Release times under operating conditions are expected to be less than 50 msec. The mechanical description of the magnets is contained in

Section VIII and performance curves for the magnets are shown in Figures 37 and 38 of that section.

2.2 Screw Jack

The basic element of a Spert IV control rod drive unit is an acme thread screw jack, the mate of which is a worm wheel driven by a right angle worm. The outside diameter and pitch of the screw jack are 1-1/2 in. and 3/8 in., respectively, resulting in a mechanical efficiency of roughly 35% for the jack, and an over-all efficiency of 15% to 20% for the drive. A screw or worm which is less than 50% efficient is "non-overhauling" or self-locking. This "inefficient" mechanism was chosen for its self-locking feature. The drive is simple, compact, and rugged. The speed ratio of the worm and wheel is 6:1. Thus, 16 revolutions of the input shaft are required for 1 in. of linear motion of the screw jack.

2.3 Mechanical Power

The mechanical power for the drive is obtained from a 1-horsepower, 4-pole, 3-phase induction motor, operating on 208-volt, 60-cycle power. Constancy of speed, overload capacity, ease of control, and low cost are among the characteristics which make induction motors eminently suitable for the rod drives. Ease of reversing and absence of auxiliary starting windings and switches dictated the choice of polyphase over single-phase motors. This motor is mounted integrally with a Graham variable speed transmission, with an output speed range of 0 to 200 rpm. The driving sprocket is identical in number of teeth with the driven sprockets. Both the motor shaft and the transmission output shaft are equipped with spring-set, magnetically released disc brakes to limit coasting. The maximum rate of 200 rpm produces rod motion at the rate of 12-1/2 in. per min.

The transient rod drive is identical to the control rod drives except that it consists of a single screw jack unit, the worm shaft of which is directly coupled to the output shaft of a second Graham transmission for independent operation. No chains and sprockets are necessary for the transient rod.

2.4 Rod Position and Speed Indication

Telesyn self-synchronous motion-transmitting systems are geared to the worm shafts of the transient rod drive and control rod drive number four. These operate Veeder-Root digital counters at the control console which are calibrated to indicate rod drive position in hundredths of inches. The speed ratio controls of the Graham transmissions are operated and monitored by similar Telesyn systems with pilot motors driving Telesyn transmitters at the control relay rack which in turn drive Telesyn receivers on the Graham transmissions and on monitoring Veeder-Root counters at the control console.

2.5 Control System Power Circuit

The NRTS electrical power standard for applications up to 100 horsepower is 480-volt, 60-cycle, 3-phase. Delta-connected ungrounded secondaries are in general use because a double fault is required to disable such a system. However, ungrounded secondaries have the disadvantage of being able to accumulate static charges. Experience with unusual equipment failures at Spert III, for which such charges seem to be the only plausible explanation, led to the choice of a Y-connected secondary with a grounded neutral for the Spert IV facility.

Experience with damage to 480-volt wiring which occurred in connection with a steam leak at Spert III led to the choice of 208-volt Y-connected power for the Spert IV control system because the Spert IV building can be expected to provide a high humidity environment on occasion. Most of the Spert IV control system operates directly from the 120/208-volt power, furnished from a single 6-kva bank of transformers, with small 6-volt and 28-volt transformers being required only for panel indicator lights and annunciator buzzers. System voltages are limited to 120 volts above ground. Full load currents of the drive motors are less than 4 amp each.

The control system 3-phase, 480-volt primary power (Figure 17) is obtained from the main bus of the reactor building motor control center through a 40-amp trip, 100-amp frame, air circuit breaker in cubicle D4. This bus is fed directly from the Spert IV substation through an 800-amp air circuit breaker with 50,000-amp interrupting capacity. The 480-volt input terminals at the relay rack are back-wired for the safety of personnel. Three leads from the input terminals run to a 600-volt barrier type terminal strip with 24 terminals which serves as a junction block for the 3-phase transformer bank. Each transformer is a dry-type, four-winding, 2-kva unit with two 240-volt windings and two 120-volt windings, connected in series and in parallel, respectively, to provide step-down from 480 volts to 120 volts. The three transformers are connected delta-Y to provide 120/208-volt power for control relays and drive motors. The neutral and 3-phase power leads from the transformer secondary are designated ϕ_o, ϕ_{1a}, ϕ_{2a}, and ϕ_{3a}, respectively. The neutral is grounded solidly at the relay rack and at the control console, but is otherwise insulated and carried throughout the system providing the option of a floating neutral in case this should ever be desired.

An induction disc type undervoltage and phase failure relay set to open at about 200 volts monitors the transformer secondary. The control system is energized from the transformer secondary through an NEMA size 1 open-type magnetic contactor designated the main power contactor. Downstream from the contactor, ϕ_{1a}, ϕ_{2a}, and ϕ_{3a}, become ϕ_1, ϕ_2, and ϕ_3, respectively.

As shown in Figure 18, the main power keyswitch on the control console controls the main power contactor, subject to interlocks provided by contacts of a relay-rack door relay, a connector continuity relay, and the phase failure

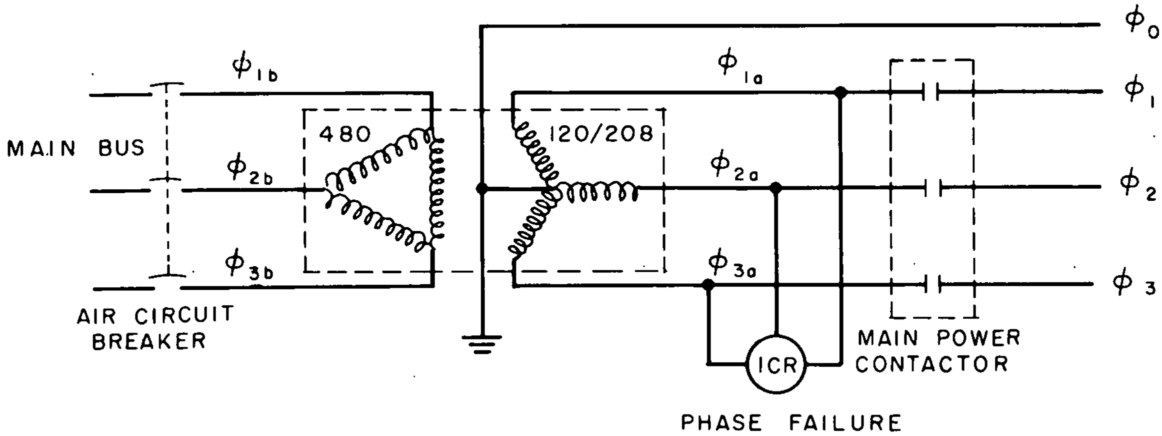


Fig. 17 Control system power circuit (NRTS-62-271).

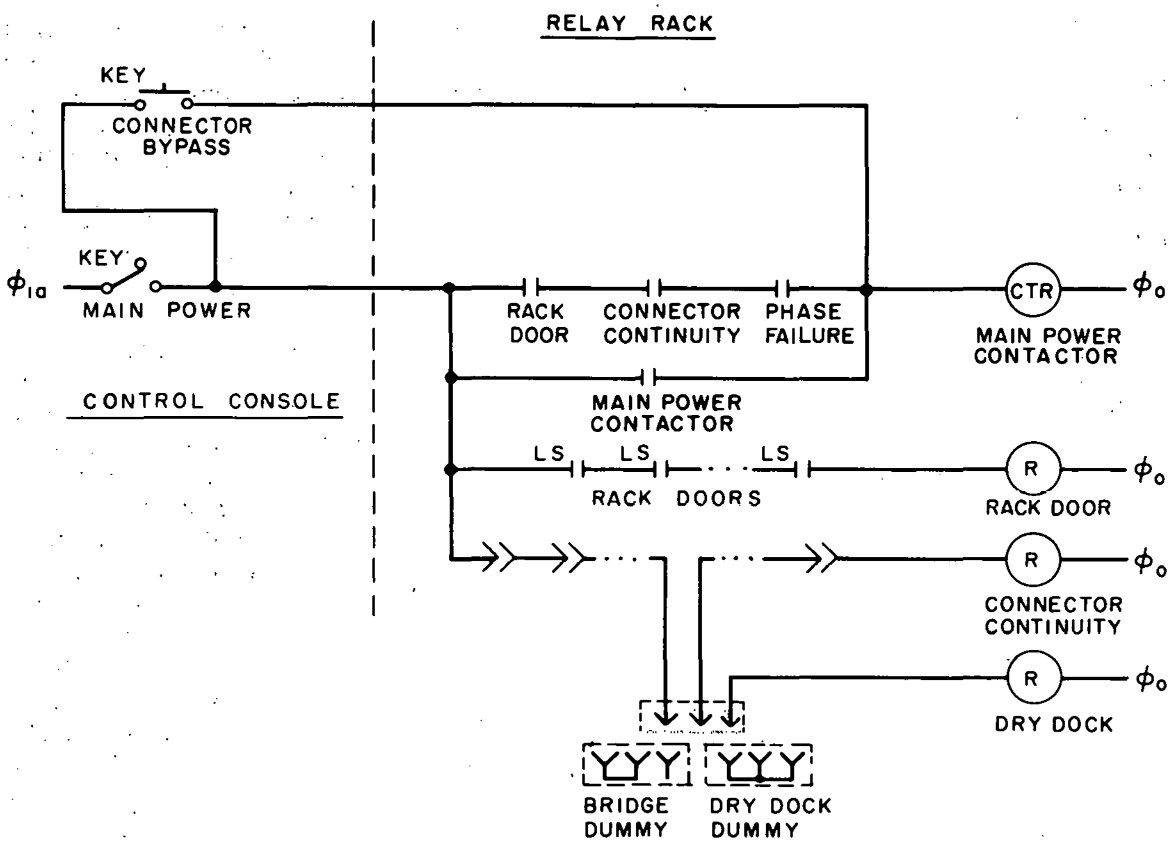


Fig. 18 Control system power control circuit (NRTS-62-258).

relay. The door relay is controlled by a series of switches actuated to close the circuit when the doors of the control system relay racks are closed. This is for safety of personnel working on the equipment in the racks. A red light to indicate "POWER ON" is also placed by each door handle. Similarly, the connector continuity relay is controlled by a circuit which traverses all cable connectors in the control system in such a manner that disengaging or misengaging any connector in the system interrupts the circuit. Those connectors in the system which are mechanically interchangeable, are rendered electrically noninterchangeable by variation in the choice of pins used for this circuit.

Presence of any of these inhibitory conditions prevents turning main power on, but an auxiliary pole of the main power contactor is wired to prevent the loss of main control power by subsequent occurrence of any of these conditions. The console annunciator sounds a buzzer alarm and lights an identifying light if the main power keyswitch is at "ON," and any of these inhibitory conditions occur. A momentary contact keyswitch is provided, designated the "connector bypass switch", to allow an operator to turn main power on despite the interlocks if it becomes necessary to check control circuitry with drive motor cables disconnected.

The dry dock relay shown in Figure 18 is energized by the connector continuity circuit through the dry dock dummy cable receptacles when the drives are in dry dock. This relay is used in connection with the warning light and horn system described in subsection 2.10 beginning on page 43.

2.6 Rod Drive Insert--Withdraw Circuits

Standard NEMA size-0 reversing motor starters are used to control the 1-horsepower, 208-volt, 3-phase induction rod drive motors. Electrically, the two starters comprise four units, designated the control rod insert contactor, control rod withdraw contactor, transient rod insert contactor, and transient rod withdraw contactor (Figure 19). Basic control of these contactors is from

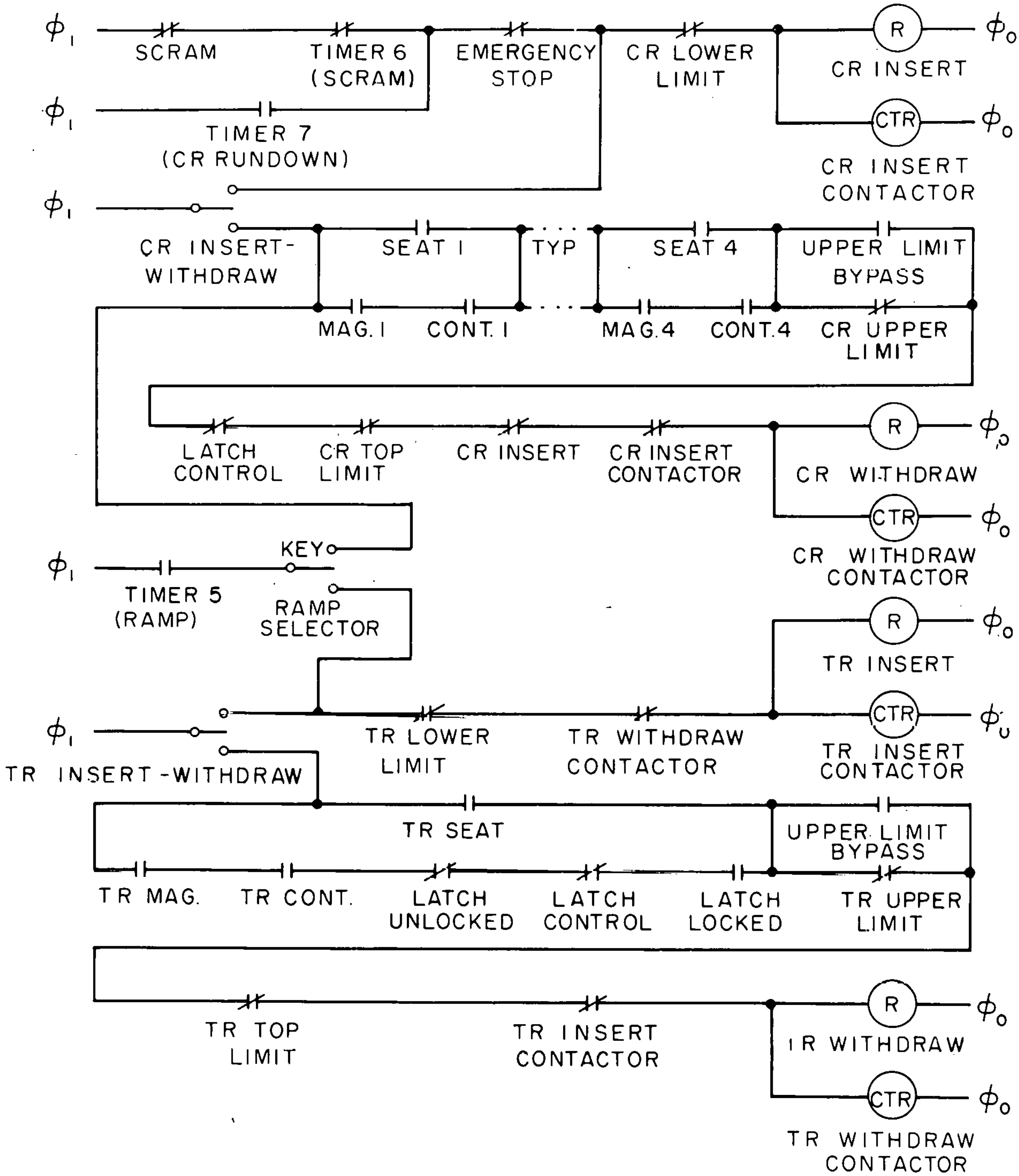


Fig. 19 Control system insert and withdraw circuit (NRTS-62-269).

two pistol grip insert-withdraw switches on the control console. The "OFF" positions and control rod insert position are maintained by detents. The control rod withdraw and the transient rod insert and withdraw positions are spring-returned and must be maintained by the operator. No inhibitions are included in the control rod insert circuit except for the lower limit indication which prevents mechanical damage to the drive if it were to be driven to its lower extreme. The electrical interlock normally included in a reversing starter to prevent simultaneous energizing of both coils has been altered as shown in Figure 19. Each contactor except the control rod insert contactor is inhibited in the usual manner by its opposite member; eg, normally closed auxiliary contacts of the transient rod withdraw contactor inhibit energizing the transient rod insert contactor. Each starter is mechanically interlocked in standard fashion to prevent short circuits should the electrical interlocks malfunction. Each contactor has an auxiliary control relay (eg, control rod insert relay) operating in parallel with it. These are necessary for other control functions to be discussed elsewhere, but the control rod insert relay also serves an additional function here. In case of a malfunction of the control rod withdraw contactor, operation of the control rod insert relay by means of the control rod insert switch increases the certainty of interruption of control current to the withdraw contactor. Then, since the withdraw interlock has not been included in the insert circuit, the insert coil is free to operate, causing the drive motor to insert and, by means of the mechanical interlock, positively opening the withdraw circuit.

Contacts of the scram relay are provided in the control rod insert circuit to initiate automatic rundown of the drives following a scram. For power excursion tests the sequence timer can be used to program either control rod drive insertion (Timer 7 contacts in Figure 19) or a scram to terminate the test. If the test is terminated by such a programmed scram, the automatic drive rundown in inhibited by the Timer 6 contacts (see Figure 19). This feature is provided to prevent electrical noise from the motor drives from possibly reducing the quality of the data being obtained at this time from the various transient instrumentation channels.

Contacts of an "emergency stop" relay which is operated by a spring-return switch on the control console are provided to permit the operator to override an automatic control rod drive insertion and assume manual control of the insertion in the event of an obvious malfunction in the drive mechanism.

Insertion of the transient rod drive is inhibited by the lower limit switch and by an interlock of the withdraw contactor. Because insertion of the transient rod increases reactivity, this withdraw contact interlock remains in its usual form whereas it was altered in the case of the control rod circuit (see Figure 19).

The inhibitions on withdrawal of both transient rod and control rods correspond to features of the mechanical design of the drives. For normal withdrawal of the transient rod, the rod will be held in contact with its drive by its magnet, and, in addition, the mechanical latch will be locked to prevent inadvertent dropping of the rod should the magnet fail. Therefore, contacts of the contact relay, the magnet current control relay, the latch solenoid current control relay, and the latch-locked-position and latch-unlocked-position switch relays of the transient rod are included as interlocks in the transient rod withdrawal circuit as five independent indications that the required conditions exist. The "latch locked" and "latch unlocked" switches were provided to give indication that the latch fingers are either completely closed or completely extended and are not jammed in some intermediate position.

Limit switch relay contacts also are included in both the transient rod and control rod withdraw circuits along with the electrical interlocks with the insert contactors. The "upper limit" switches indicate drive positions corresponding to complete withdrawal of the poison section of the control rods from the reactor core. The "top limit" switches indicate the extreme position of drive withdrawal, which is required in order to clear the bridge for removal of the drive system to dry dock.

The control rod drive is not equipped with mechanical latches. Contact with energized magnets as indicated by contact switch relays and magnet current relays is required for control rod withdrawal. Selective withdrawal of individual control rods is possible since this requirement is by-passed for each rod individually by contacts of its seat switch relay.

Because dropping the transient rod prior to the programmed schedule could cause a premature excursion, the transient rod latch is opened (arming) immediately before the planned drop. As a precaution against accidental dropping of the transient rod during the few seconds between "arm" and "fire", the transient rod latch control circuit, described later, is designed to reclose the latch if any drive motion is attempted while the latch is open. As a second precaution contacts of the latch control relay are included as an inhibition to control rod withdrawal, lessening the likelihood that the transient rod would be dropped accidentally by vibration.

In order to facilitate withdrawal of the drives to their mechanical limits for maintenance purposes, etc., an "upper limit by-pass" relay has been provided which is operated by a maintained contact keyswitch on the control console. This "upper limit by-pass" switch also deactivates the Klaxon horns and flashing red lights which normally operate whenever drives are raised from lower limit. It also prevents the scram which normally would occur should loss of transient rod contact be indicated without the fire relay being energized. The key for this "upper limit by-pass" keyswitch is removed from the switch for all normal operations and is kept under the administrative control of the Nuclear Test Section group leader.

The annunciator on the control console alarms whenever the by-pass switch is in the "ON" position at the time the main power switch is turned on.

Control rod withdrawal and transient rod insertion also can be programmed by means of a sequence timer, for tests in which the transient power excursion is initiated by a "ramp" reactivity addition. This is accomplished by a single timer-operated relay interlocked with a ramp selector keyswitch.

Because reactor control always must be maintained even at the risk of individual component damage, the control-rod-drive motor-overload relays serve only to actuate an annunciator alarm to warn of motor overheat. Thus, in emergencies, the motors may be operated even though the overheat alarm indicates risk of damage to the motors.

2.7 Rod Magnet Control Circuits

A full-wave 3-phase bridge rectifier provides current at about 8 volts for the five rod magnets, as shown in Figure 20. The rectifier is constructed from 6.3-volt, 1.2-amp filament transformers and six 1.2-amp germanium diodes. No filtering is necessary. Current to each magnet can be monitored by a shunted galvanometer and adjusted by a rheostat. If, at a later date, power level and/or period scrams are required, a suitable power amplifier can be inserted in the circuit as shown at "A" in the magnet 1 circuit of Figure 20.

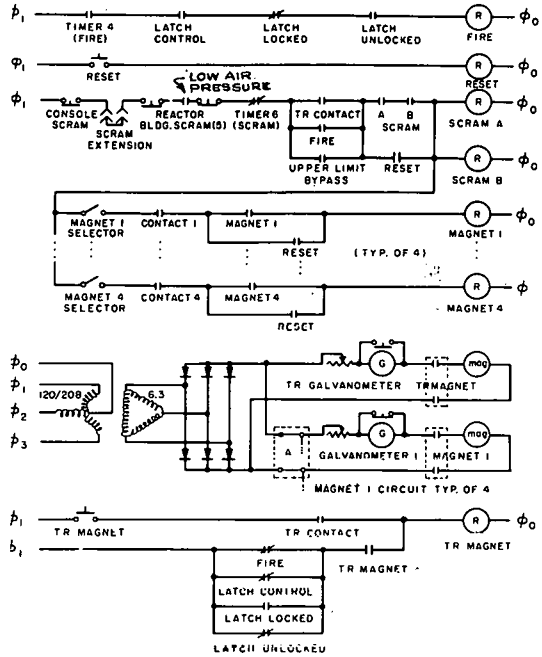


Fig. 20 Control system magnet control circuit (NRTS-62-183).

The scram circuit is similar to an ordinary motor starter circuit with multiple stop-button stations, except that two parallel relays are used and holding contacts of each are in series. Either relay is able to scram the reactor despite malfunction of the other. Manual scram buttons are permanently installed at the control console and at five locations in the reactor building. An extension cord jack is provided at the control console for additional hand-held scram buttons in the console room. Contacts of timer relay 6 operated by the sequence timer provide for programmed scrams. Automatic scram is provided in case of accidental loss of transient rod magnet contact or a reduction of plant air pressure to less than 20 psig. If neither the "upper limit by-pass" relay nor the fire relay are operated, and the transient rod contact relay indicates loss of contact, the scram circuit is opened, de-energizing the control rod magnets.

Control rods are selected for withdrawal by closing appropriate "magnet selector" switches on the control console. Actual energization requires the rod to be in contact with the magnet as indicated by the magnet contact switches and is accomplished by operating the "scram reset" button after magnet selection. Thus, selector switches can retain a given configuration through successive runs of the reactor, but resetting of the magnets requires deliberate action by the operator on each occasion.

The transient rod magnet is energized by a momentary pushbutton on the control console as shown in Figure 20. This magnet cannot be de-energized manually except by turning off control system main power.

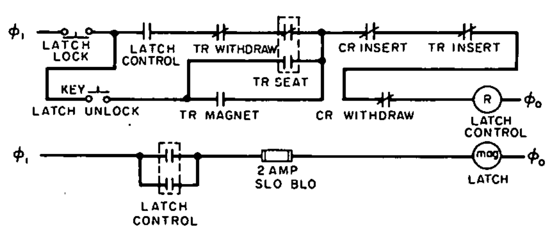


Fig. 21 Control system transient rod latch control circuit (NRTS-62-239).

The transient rod latch is unlocked by a momentary keyswitch on the control console which operates a relay circuit as shown in Figure 21. Energizing of the transient rod magnet current control relay was a necessary condition for transient rod withdrawal, thus, no further operator action is required for transient rod withdrawal. Energizing of the transient rod magnet current control relay also is included as a permissive condition in the latch control circuit. An attempt to insert any of the control rods after the transient rod latch is opened will drop out the latch control relay, reclosing the latch. An attempt to withdraw any of the control rods under these conditions is prevented by the latch control relay. If it is desired to abort the experiment, the latch can be reclosed by the "latch lock" pushbutton. With the rods positioned and the latch unlocked, the reactor is now ready for the initiation of a transient. The "timer start" keyswitch can be operated allowing power excursion to proceed as programmed. Contacts of timer relay 4 (see Figure 20) energize the "FIRE" relay which in turn de-energizes the transient rod magnet current relay, stopping magnet current and dropping the transient rod into the core. Latch control and latch-unlocked indications are included as permissive conditions in both steps of this sequence, so that any malfunction of latch mechanism or circuitry will preclude actual firing of the transient rod.

The latch control relay holding circuit bypasses the magnet current control relay so that the latch will not attempt to reclose when the transient rod is fired. When the rod drops to seat position, the holding circuit is interrupted to reclose the latch. Normally open contacts of the same seat relay are used to allow the latch to be held open with the magnet de-energized in order to withdraw the drive from lower limit position with the rod remaining at "seat."

With the exception of the 40-amp, 480-volt input circuit breaker, and the 2-amp "slow-blow" fuse in the latch magnet circuit, there are no current-interrupting protective devices in the Spert IV control system, because of a general policy that all circuits be designed to remain operable up to the point of self destruction. Alternating current solenoids have very high inrush currents and quickly burn up if held open by a jammed mechanism. Experience with jammed latches on Spert II and Spert III revealed the desirability of protecting latch solenoids in cases where a "fail-safe" configuration is possible, as is the case in the transient rod latch circuit (Figure 21) where a burned fuse merely prevents unlatching the transient rod.

2.8 Sequence Timer Circuits

The Spert IV control system is provided with Multiflex timers, manufactured by Eagle Signal Corp., for sequence programming of experiments. Two basically similar units differing in range capacity by a factor of ten are used together. Very precise timing is available for sequences requiring no more than 30 sec, while sequences requiring up to 5 min also can be programmed but with less precision. Settings can be made with accuracy within 1/4% of full scale.

The basic timer circuit is shown at the top of Figure 22. This is an elaboration of what is referred to by the manufacturer as the "no voltage reset arrangement", protected against automatic restarting. The 30-sec timer and the 300-sec, or 5-min, timer are wired to operate from the same starting keyswitch and stop pushbutton. Energizing the clutch solenoid of each timer by means of the "START" switch engages a clutch and lowers the contact trip bars to ride on a sliding plate. The synchronous motors then drive the sliding plates downward. The contacts close and open as the trip bars drop off the downward-moving plates in accordance with their time settings.

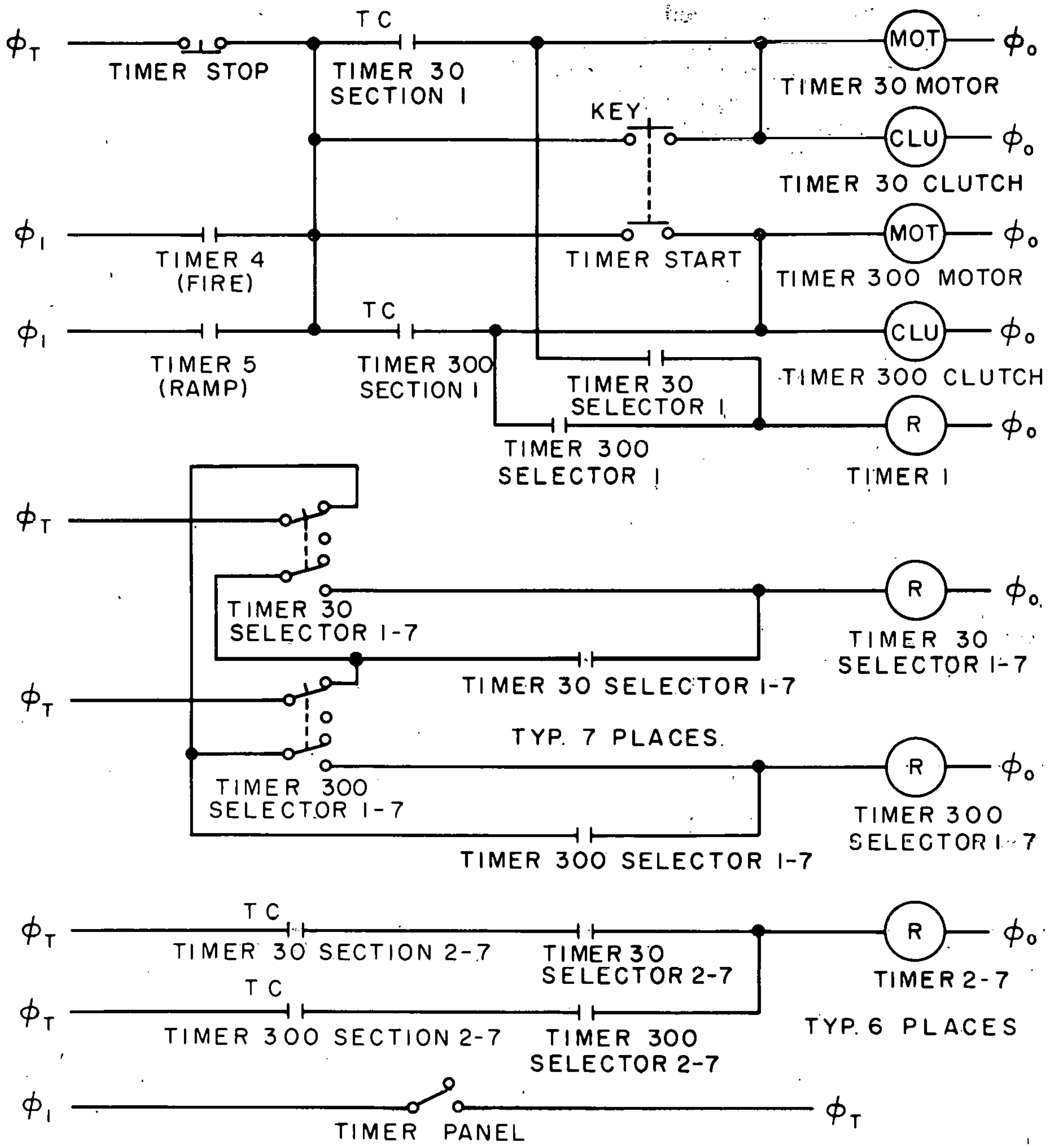


Fig. 22 Control system sequence timer circuit (NRTS-62-238).

De-energizing the clutch solenoid disengages the clutch and raises the trip bars, allowing the sliding plate to reset by spring action to its original position. Section one timer contacts are used as a holding circuit, enabling a momentary switch to be used for starting so that the timer does not repeat its cycle automatically after resetting. Used in this manner, section one closing contact always must operate when the clutch is energized, and section one opening contact determines the length of the timer cycle.

Apparatus is not operated by the timer directly, but by seven timer-controlled relays at the control console and seven other functionally identical relays in the reactor building relay rack. These relays are designated timer 1 relay, timer 2 relay, etc., and are controlled through selector relays by correspondingly numbered contact pairs of the timers. Timer relays 2 through 7 are shown near the bottom of Figure 22. Timer 1 relay is shown with the basic timer circuit because of the internal use of section one contacts. The selector relays, as the figure shows, select whether the 30-sec or 300-sec timer contacts shall control a given timer relay. The timer-selector relays are in holding circuits with momentary selector switches. As is evident from the middle portion of Figure 22, each selector switch in picking up its relay drops out the relay of its opposite member. To drop both of any pair of selector relays out at once, it is necessary to turn off timer panel power. A timer panel switch is provided so that the timers with associated relays and indicator lights may be left inoperative if desired when the control system is being used in non-programmed operations such as core loading. Contacts of timer 4 and timer 5 relays are used to bypass, and thus prevent, accidental function of both the timer panel switch and the timer stop button during programmed transient power excursions, when actuated, by connecting the basic timer circuit directly to phase one power as shown in the upper left portion of Figure 22. Timer 4 relay is permanently assigned to control step transients and timer 5 relay is permanently assigned to control ramp transients.

2.9 Rod Drive Speed Control Circuits

The variable speed transmissions in the Spert IV rod drives are equipped with remotely controlled electrical Shaftrol speed ratio changing mechanisms. As supplied by the manufacturer, these Shaftrol units are driven by a reversible induction stall motor and their action is monitored by remote indicating voltmeters which receive signals from variable-resistance potentiometers mounted with and actuated by the Shaftrol gear trains. This monitor is not sufficiently precise for Spert purposes. The voltmeters and potentiometers have been replaced by Telesyn synchronous-motion transmitters driving Veeder-Root digital counters. To minimize mechanical modification of the Shaftrol units, a Telesyn receiver was substituted for the stall motor in each Shaftrol, and Telesyn transmitters driven by pilot motors were mounted in the relay rack, with additional Telesyn receivers at the control console driving the monitoring digital counters. Stall motors no longer suffice for pilot duty, because the Shaftrol gear trains are likely to be damaged at stall by the additional torque required to drive the digital counters at high speed. Therefore, ordinary shaded pole induction motors are now used for pilot duty and limit switches are installed in the Shaftrols to prevent damage. Reversing is accomplished by switching transmitter Telesyn stator leads, with a delay being incorporated in the pilot motor starting circuits to allow time for the transmitter Telesyn to stabilize before assuming load.

The speed control circuits with their console-mounted operating pushbuttons are shown in Figure 23.

Because the transmission speed ratio is not a linear function of Shaftrol rotation no attempt has been made to choose gear ratios to make the digital counters direct reading. Calibration curves are required to interpret counter readings.

2.10 Rod Drive Warning Lights and Horns

The reactor area has been provided with warning lights and horns which are part of the reactor control system.

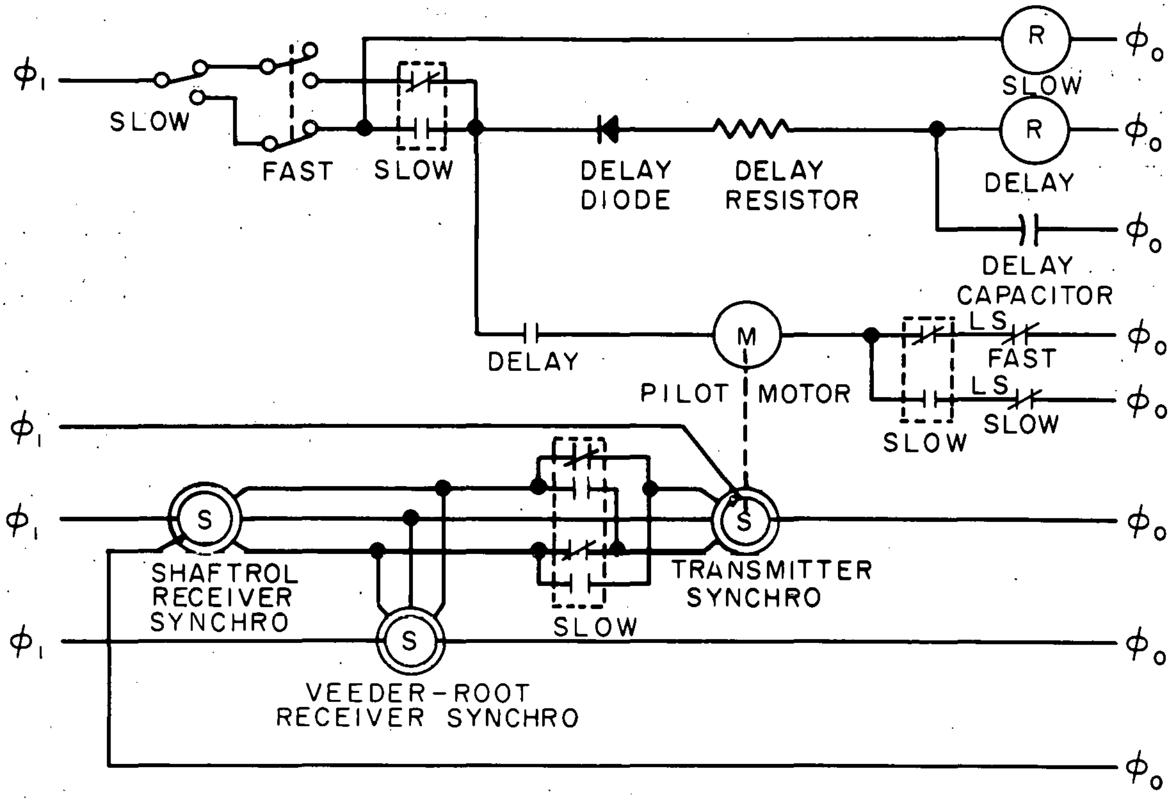


Fig. 23 Control system rod drive speed control circuit (NRTS-62-259).

Whenever control power is on, as it must be to operate the rod drives, if any rod is raised from seat position or if any drive is raised from lower limit position without the use of the upper limit bypass keyswitch, the warning system becomes active. The circuitry is shown in Figure 24.

The horns, one interior and one exterior, are operated with building lighting power by means of the horn relay. The horn relay may be operated by a manual horn button on the control console or by the red warning light contactor. When operated by the red light circuit, the horn is limited to 15 sec of operation by a time delay relay.

The horn relay is one of 18 relays used to control 120-volt reactor building power circuits, and is not an integral part of the reactor control system. The operating coils of these relays have very much higher impedance than the control relays used elsewhere in the system. Because of this, a bleeder resistance shunt is used to prevent spurious operation by capacitive leakage in the long cable leading to the console horn button.

During operation of the reactor, a signal from any seat or lower limit switch, indicating drive motion, energizes the horn relay and the red warning light contactor. Provisions are made such that the warning light contactors are not energized by the console manual horn button, and, as already mentioned, the horn relay operates from the red warning light contactor for only 15 sec. For servicing operations, in which the drives must be operated with personnel at the reactor, the upper limit bypass switch is used to prevent activation of

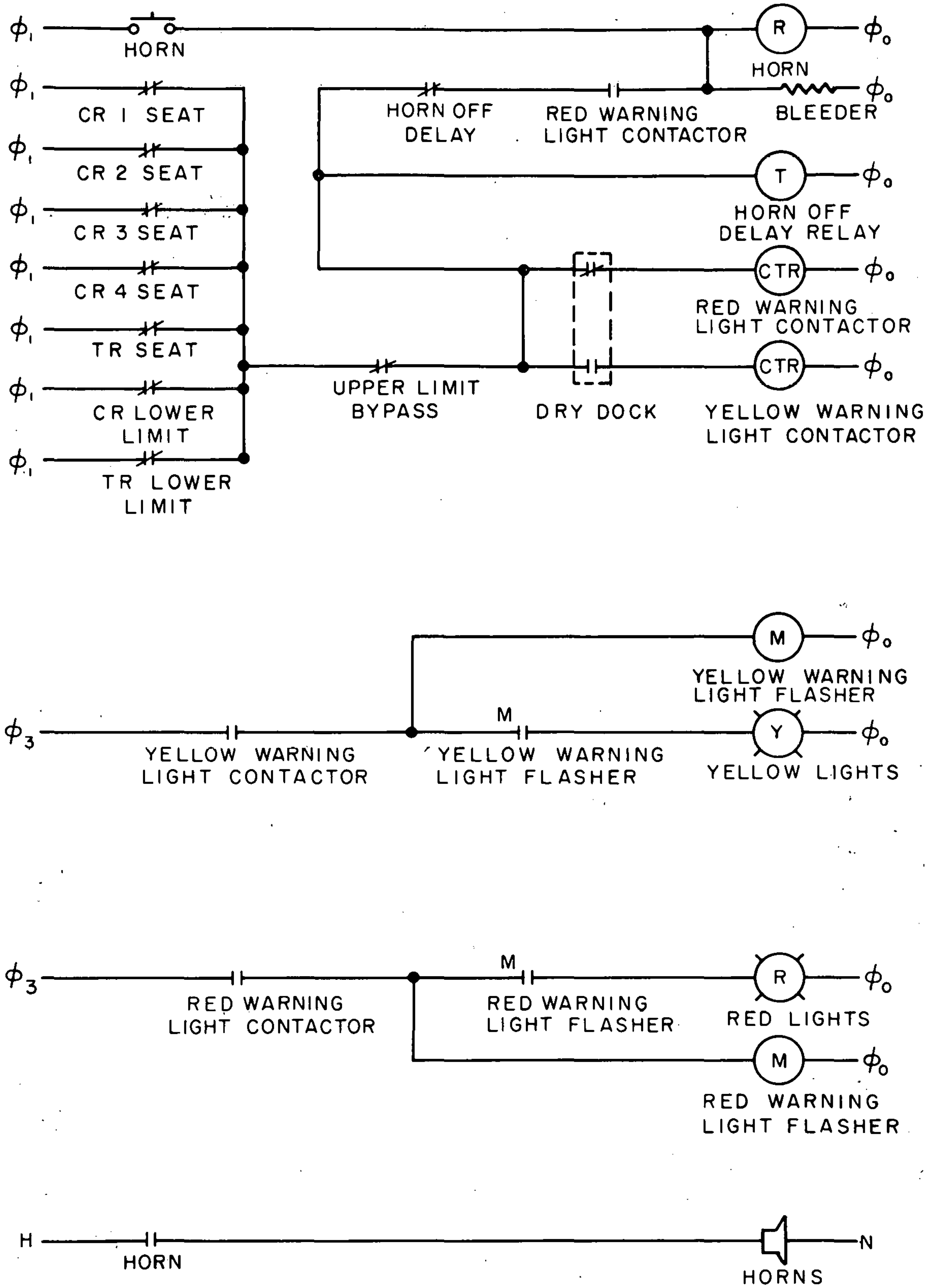


Fig. 24 Control system warning lights and horn circuits (NRTS-62-358).

the red lights and horns. Actuation of the dry dock relay described with Figure 18 causes the seat switch or limit switch signal to activate yellow warning lights rather than the red warning lights and horns when the drives are operated in dry dock.

Phase one power is used for most single-phase requirements in the control system. Because it is desirable not to have 208 volts present in the system wiring except where necessary, phase unbalance is tolerated. This is partially compensated by using phase three power through the warning light contactors to the light flasher circuits. The flashers are Reynolds units having motor-driven cam-operated contacts. Because these contacts may be in closed position when the unit stops, it is not sufficient merely to control the flasher drive motors. The contactors had to be provided to cut off flasher power completely.

2.11 Rod Drive Sensing Switch Circuits

Circuits for magnet contact switches, rod seat switches and shock absorber water level control are shown in Figure 25. Water-filled dashpots are provided to decelerate the rods when scram occurs or when a step transient is initiated. In the event that the reactor is operated with the pool water level below these dashpots, or if rod dropping is checked out in dry dock, auxiliary water is supplied to the dashpots by a small reservoir containing a float switch. This float switch operates a solenoid water valve intended to maintain the level in the reservoir, and a relay which operates an indicating light to warn the control console operator of low water level.

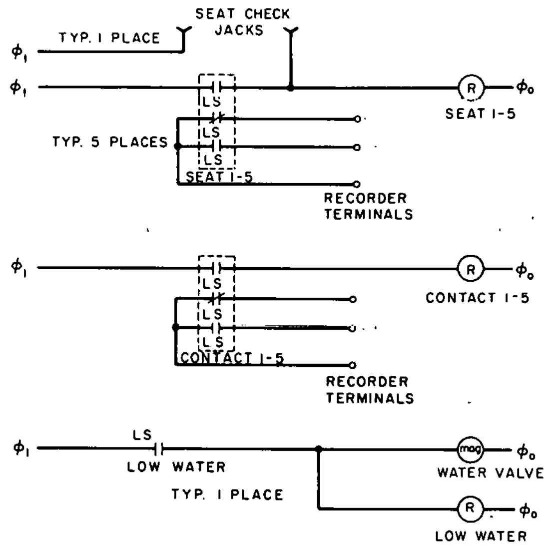


Fig. 25 Control system rod drive sensing switch circuits (NRTS-62-240).

The seat switches operate control relays, which operate control console indicator lights and function in various interlock circuits. Both normally open and normally closed circuits are brought directly from the seat switches for use in the recording of time references on the recording oscillographs. In addition, plug-in jacks at the control console are wired directly to the seat switch relay circuit in such a manner (as shown in Figure 25) that, in event of a power failure, the position of the seat switch on each rod can be checked by an ohmmeter or continuity tester to determine whether the rods are seated in the core. A reading of 900 ohms, the resistance of a seat switch relay coil, indicates the seat switch is open and the rod is not seated. A line resistance reading of about 2 ohms indicates the seat switch is closed.

Magnet contact switch circuits are identical to seat switch circuits except for the continuity-testing feature. Rod drive limit switches (not shown) merely operate control relays, no provision being necessary for the recording of limit switch signals.

2.12 Rod Jam Alarm Circuit

It is conceivable that a rod might possibly stick and not fall to seat position when released for scram or, in the case of the transient rod, for the initiation of a step power excursion. Approximate action in such a case must depend on the particular circumstances at the time. A circuit is provided (as shown in Figure 26) to sound an annunciator alarm if any rod fails to reach seat within 2 sec after being released from its magnet. The rod-jam relay is a thermal relay which picks up after receiving a steady signal for 2 sec. If any rod sticks between magnet and seat so that neither its magnet relay nor its seat relay is energized the rod-jam relay receives a signal. In the case of the transient rod, provision is made that additional indication of the latch being locked and magnet contact existing can forestall a rod-jam signal, which would otherwise occur if the magnet lost its power with the latch locked.

2.13 Control System Electropane Annunciator Plug-in Unit

The control system alarm annunciator is an eight-unit Electropane assembly manufactured by Electro Devices, Inc. The schematic of a typical plug-in unit is shown in Figure 27. The window of each unit contains three lamps colored red, white, and green. Normally, the bulbs are series-connected to line voltage giving a dim light indicating no bulb is burned out. When alarm occurs, all contacts (Figure 27) close, applying full voltage to all bulbs and activating the external audible alarm. Closing the reset button opens the two contact pairs designated reset, silencing the alarm and leaving only the red bulb lit. Clearing the alarm condition then opens the signal contacts and recloses one pair of reset contacts, lighting the green bulb. Closing the reset button again then returns the unit to normal.

The operation of the signal coil can be reversed for use with normally closed external alarm sensing contacts.

2.14 Control System Annunciator Alarm Sensing Circuits

Alarm circuits included in the Spert IV annunciator are shown in Figure 28. The annunciator is powered by control center building power through a separate pole of the control console main power switch, and a second delay relay is incorporated to allow control relays to assume correct

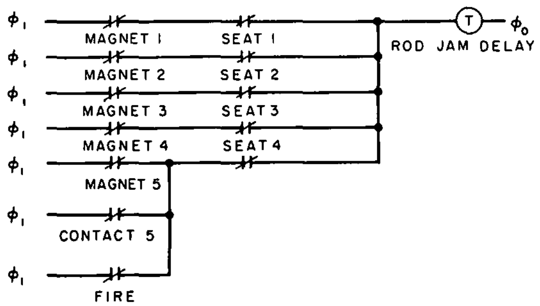


Fig. 26 Control system rod-jam alarm circuit (NRTS-62-677).

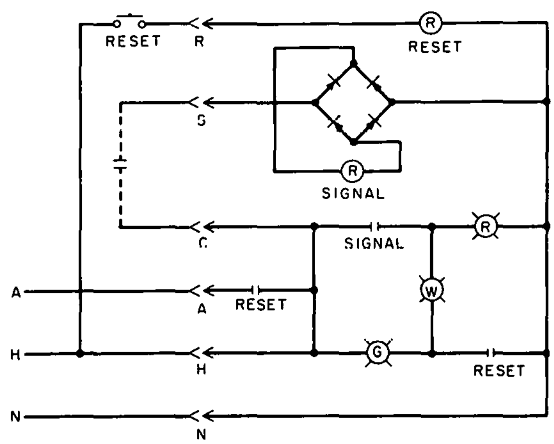


Fig. 27 Control system annunciator plug-in unit circuit (NRTS-62-119).

indications before alarming the annuciator when main power is turned on.

Power failure alarm sounds when low voltage or reversed-phase sequence de-energizes the phase failure relay, closing a pair of contacts.

Connector continuity alarm sounds when the connector continuity relay is de-energized by opening the connector continuity circuit.

The rod-jam alarm sounds when the rod-jam delay relay picks up, after any rod registers neither contact nor seat for more than 2 sec.

When the overheat relays in the drive motor starters open, units four and five alarm.

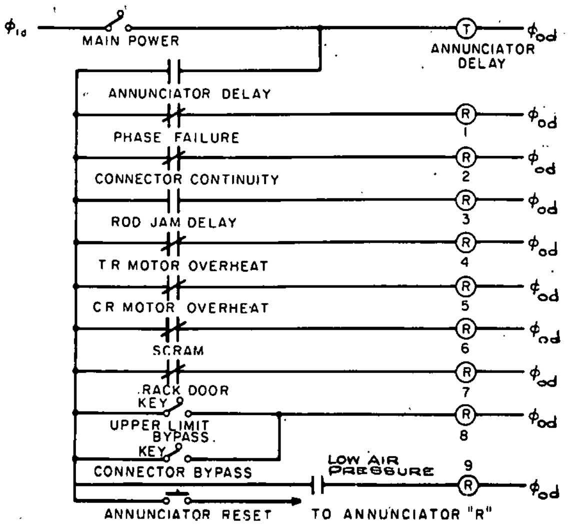


Fig. 28 Control system annunciator alarm circuit (NRTS-62-120).

Scram can be detected by watching seat, contact, or magnet selector indicator lights on the console, but an annunciator unit monitors the scram relay to give immediate unmistakable information in event of its being de-energized for scram.

Opening any relay rack door drops out the rack door relay, causing alarm.

Use of either bypass keyswitch alarms the annunciator, to insure against inadvertent use of these switches, and, in the case of the upper limit bypass switch, to provide alarm when main power is turned on if the upper limit bypass switch has been left on from previous operation.

3. CONTROL CONSOLE

The Spert IV control console is built up from eight standard prefabricated 22-in. metal sections and two 45° "pie" sections. The two pie sections are inserted between the third and fourth and the fifth and sixth rectangular sections, so that the over-all appearance is roughly that of a quadrant of a circle. The operator is seated before the two center sections which contain intercom control switches and the switches and indicating lights which, with the sequence timer panel in section six, constitute the controls of the reactor. The remaining panels are occupied by television monitors, oscillographs, and nuclear instrumentation. Figure 29 is a photograph of the complete reactor control console.

Twelve 19-conductor control cables and one No. 2 AWG, stranded, neoprene-coveted, bonding cable are laid from the control console to the reactor building relay rack for reactor control. Sixty RG8A/V coaxial cables and a number of 19-conductor cables are used for instrumentation, process control, and intercommunication.

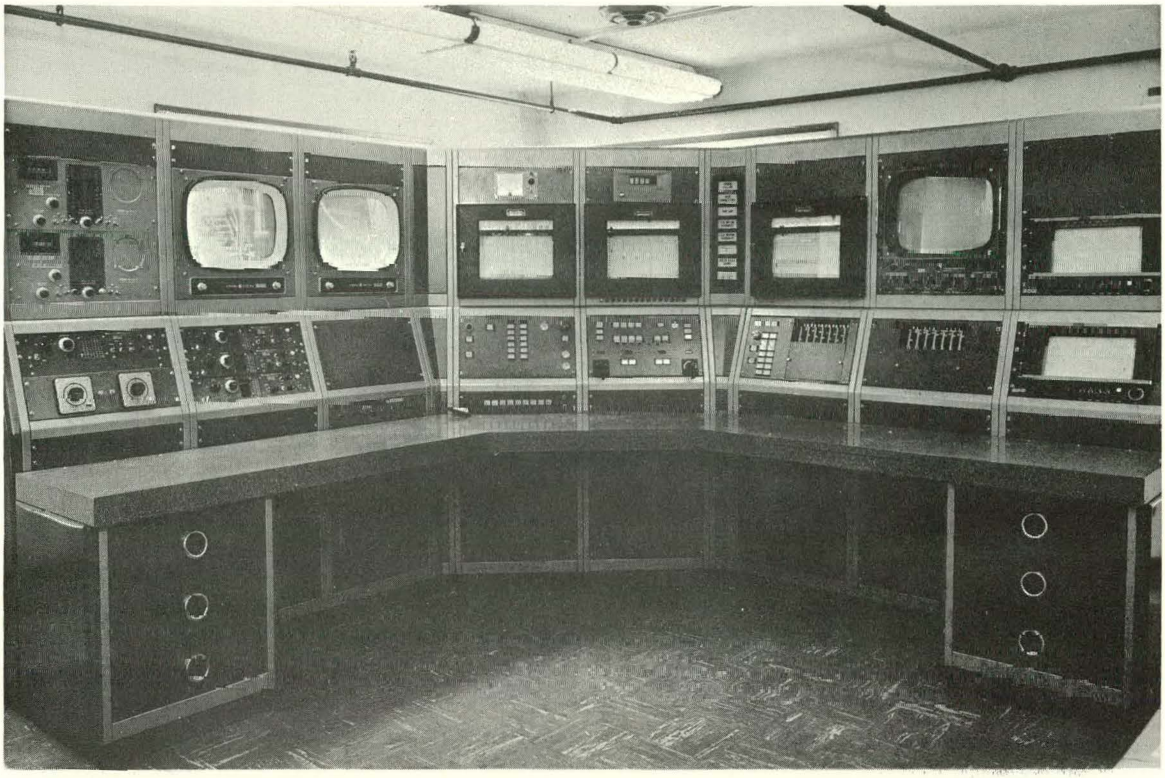


Fig. 29 Photograph of reactor control console (NRTS-62-48).

The reactor control panels are occupied by illuminated push buttons which serve for operation and indication for most of the controls. Two pistol grip switches, control drive insertion and withdrawal, and six locking key-switches control critical functions such as main power, sequence timer, transient rod latch, and interlock bypasses. A red mushroom head push button and a black flush-mounted push button are for scram and manual evacuation horn, respectively. Fixed indicator lights, identical in appearance to the illuminated push buttons, monitor the contact, seat, and limit switches.

Two indicator light colors are used: red and white. Fixed indicators use only white and indicate by illuminating at "ON" or "OFF". Limit switches, contact and seat switches, and the ramp selector keyswitch are monitored in this manner. Momentary push buttons used for drive speed control are continuously illuminated in white only, merely for the sake of appearance and ease of identification.

The annunciator reset momentary push button is left unilluminated except when the annunciator alarms, at which time it is illuminated with a flashing white light.

The scram reset button (also momentary) is normally lighted white. When scram occurs it changes to red. It is located just beneath the mushroom head scram button.

The transient rod latch-relocking button is adjacent to the latch-unlocking keyswitch and is lighted white for identification, changing to red when the latch is unlocked.

Two sequence timer units are used, having ranges of 30 sec and 300 sec, respectively. Each timer has six sections with fully adjustable "ON-OFF" times and a seventh which is not adjustable but is capable of controlling equipment desired to run coincidentally with the timer itself. The timer panel has its own power "ON" switch, a maintained-contact push button, which allows the timers to be left inoperative if desired. This switch is continuously illuminated white when turned on, and the timer stop button adjacent to it is illuminated simultaneously, also white. Each section of the timer is provided with a selector push button which is illuminated white when selected and turns red when it becomes active during a run.

The selector switches are momentary push buttons which energize relay holding circuits. Once picked up, a selector relay can be dropped out only by picking up its opposite member or by turning off timer panel power. That is to say, for example, if timer 30 section two is picked up, the selection can be changed to timer 300 section two by actuating the corresponding selector switch, but to drop out both sections it is necessary to turn off timer power.

The timer panel is provided with a stop button so that a program sequence can be stopped if circumstances require preventing the initiation of a transient. However, if a program has proceeded far enough that a transient has been initiated, timer relays bypass both the timer "STOP" button and the timer power switch so that impulsive or accidental action cannot then interfere with programmed data recording and scram.

Both timers operate from the single timer start keyswitch, but corresponding sections of both timers cannot be used in the same run. However, different sections of the two timers can be used in the same run. This arrangement was chosen so that the various sections could be permanently assigned to certain functions without eliminating the choice of the 30-sec or the 300-sec range for any given function. For example, section four of both timers is assigned to timing the firing of the transient rod. The selector switches determine which timer actually is used in a given run, but regardless of which timer is chosen to fire the transient, section six of either timer still may be selected to terminate the run with a programmed scram.

Magnet selector switches are of the maintained-contact variety so that they can be left unchanged from run to run. These switches are normally unilluminated, but are lighted red when actuated for magnet selection and then turn white when the magnets are actually energized by the operation of the scram reset button. If any magnet loses contact with the rod for any reason, it is de-energized automatically, and its selector switch reverts to red.

Twelve auxiliary relays are provided in the reactor building relay rack which are operated by 12 maintained-contact "ON-OFF" switches on the control console. These switches, normally unlighted, are white when in "ON" position. The switches and relays are used for miscellaneous functions, such as remote control of high voltage to ion chambers, and remote control of 120-volt and 480-volt reactor building power circuits.

A patch panel is provided in the relay rack at which any combination of circuits may be interconnected, involving the auxiliary relays, the timer relays, the remotely controlled building power receptacles, and the instrument room control cable. Eighteen 120-volt relays and one 480-volt relay are provided in the reactor building for controlling the evacuation horns, seventeen 120-volt power receptacles, and one 480-volt, 3-phase power receptacle, respectively. All of these may be controlled from the patch panel.

Four 4-digit Veeder-Root counters are mounted above the insert-withdraw switches and the drive speed control switches on the control console. These indicate withdrawal positions and Graham transmission speed settings of the control rod and transient rod drives. The position indicators read directly in hundredths of inches. The speed indicators read in arbitrary numbers which must be referred to a calibration chart to give actual speed in inches per minute. These could not be made direct-reading by mere choice of gear ratios because the relationship between the drive speed and the speed control shaft angle is not linear.

VIII. CONTROL AND TRANSIENT ROD DRIVE UNITS

1. GENERAL

The control and transient rod drives (Figure 30) are mounted on a drive base plate which is positioned on the control bridge. Since the surface of the bridge is not machined, shims are used for initially setting the base plate level and square and dowels are used for repositioning the plate after removal. Hanging from the underside of the base plate is a structure containing mounting pads for the other drive components. The total drive assembly consisting of the drive units, base plate, and lower structure are removable from the reactor as one unit and can be placed in the "dry dock" for maintenance of any drive unit component.

The drive units were designed to permit reasonably easy changing of control rod location and length of permissible rod travel. An arbitrary core length of 36 in. was selected for design. A design travel of 45 in. was selected for the 36-in. core to permit ejection of the transient rod poison sufficiently below the core to preclude any reactivity effects attributable to transient rod "bounces" following ejection. The control rod drives were made of parts identical to the transient rod drives for ease in design fabrication and maintenance.

It must be recognized that these initial drive units may be grossly modified or replaced with a new design for later testing in Spert IV using other core designs.

Basically, a rod drive unit (Figure 31) consists of an inverted hollow shaft screw jack driven by a variable-speed transmission. Rod speeds are available in a continuous range of 0 to 12 in./min. A Telesyn generator geared to the transmission output drives a receiver and digital readout at the reactor console in the control center area. Rod drive position is indicated to the nearest 0.01 in.

One variable-speed transmission drives the four control rod drives through mechanical connections. The transient rod drive operates as an independent unit.

Control rod pickup and scram actuation are accomplished by means of an electromagnet suitable for underwater service. An adjustable spring "kicker" provides additional acceleration in the initial portion of the control rod drop. The transient rod magnet is equipped with a mechanical latch to prevent inadvertent dropping of the transient rod.

The shock absorbers are integral with the drive units and are suitable for operation either above or below the water level. All control wiring and power leads that move with the magnet are brought up through the hollow screw and the movement is compensated by "koil-cords" mounted in the screw housing.

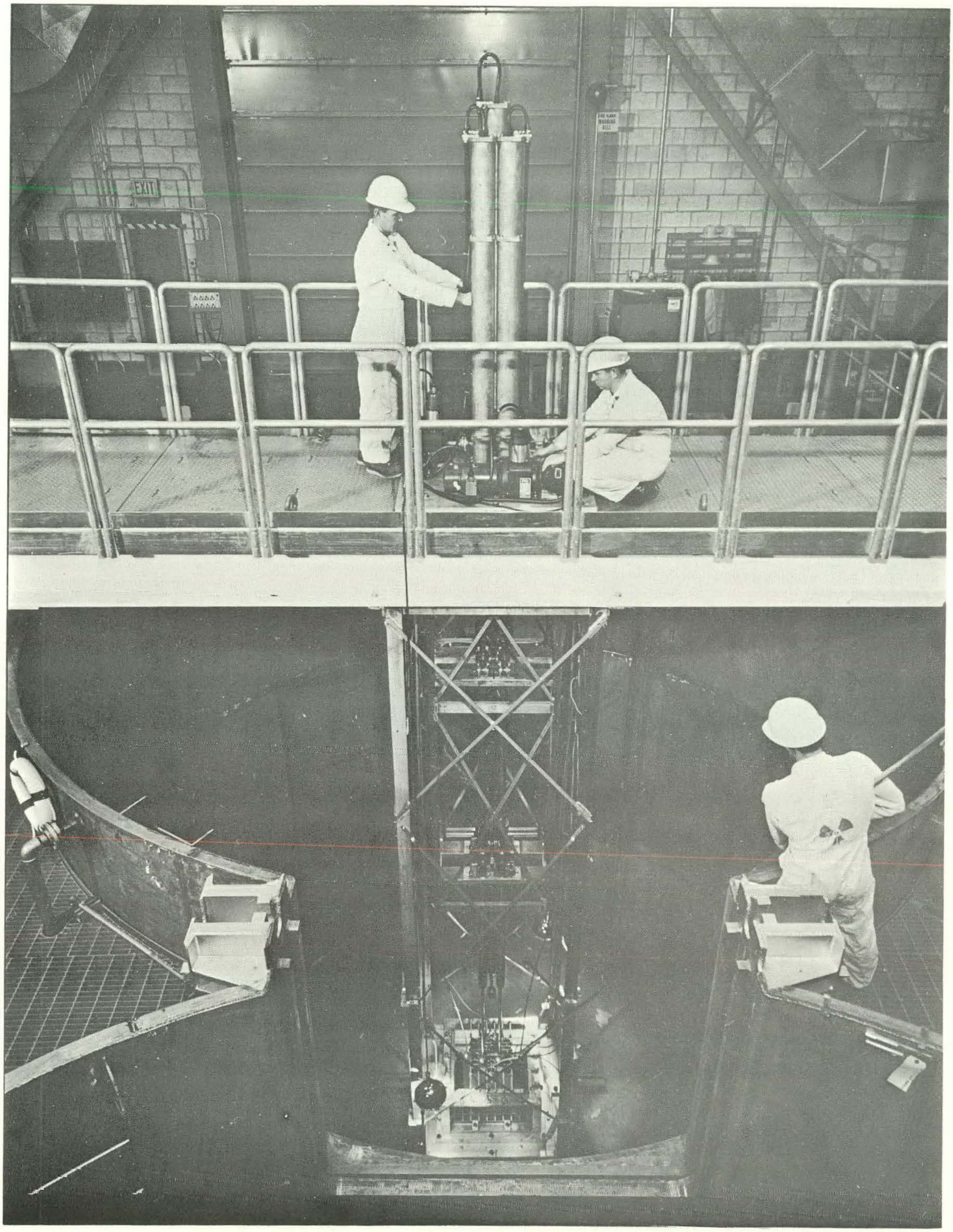


Fig. 30 Photograph of drive system (NRTS-61-6900).

2. DETAILED DESCRIPTION OF THE DRIVE COMPONENTS

2.1 Rod Drive Base Plate

The rod drive base plate consists of a 48-in.-long x 48-in.-wide x 1-1/2-in.-thick 6061-T6 aluminum plate. Three slots 3 in. long x 7/8 in. wide are provided along two opposite edges to permit moving the plate parallel to the control bridge for precise lineup over the core centerline. The base plate is predrilled for two control rod configurations, the 5 x 5 and the 3 x 5 configurations. Stainless-steel "helicoil" inserts are used in the tapped holes for mounting drive components to prevent undue wear on the tapped holes that are used frequently.

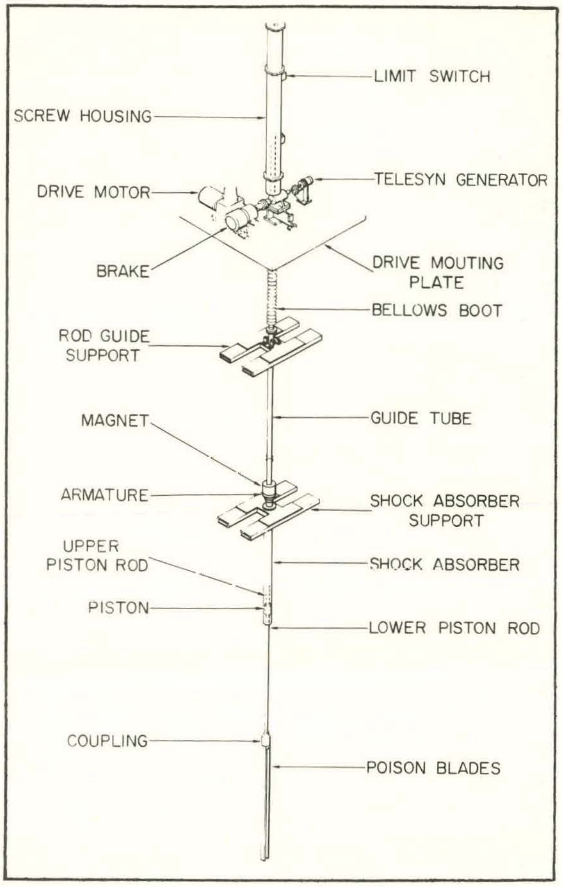


Fig. 31 Control rod drive schematic (NRTS-62-270).

2.2 Drive Unit Lower Guide Structure

The drive unit lower guide structure (Figure 32) is a rectangular framework 36 in. square by 130 in. long fabricated from 1-1/2-in. 6061-T6 aluminum angle. This support is bolted and doweled to the underside of the rod-drive base plate and is used to support the rod roller guides and the shock absorbers.

The rod roller guides, shown in Figure 33, consist of five sets of 6061-T6 aluminum V-notch rollers set in nylon sleeve bearings and mounted on an H-shaped channel frame 60 in. from the top of the support structure. The channel frame is predrilled for the two control rod configurations mentioned in Section 2.1. The V-notch rollers restrict the lateral movement of the square drive rods and also prevent rotation of the drive rods.

The predrilled shock absorber mount at the bottom of the frame also consists of an H-shaped support member fabricated from 6061-T6 aluminum channel. The control rod "seat" switches and the shock absorber mounts are located on this frame. Tubing for water supply to the shock absorber liquid-level control and electrical leads for the seat and level control switch actuation are mounted on the inside face of the guide structure angles.

2.3 Drives

Because of the reliable operation and reproducibility of speeds experienced with the Graham transmissions used in Spert I, a 1-hp model 190 MW5 Graham variable-speed transmission was selected for the Spert IV drives. Figure 34 shows a photograph of the drive units. This transmission has an output speed range of 0 to 200 rpm and is controlled from the reactor console, located in the control center building, by remote electrical control. The transmission is equipped with both a spring-set magnetic release motor brake and a Stearns

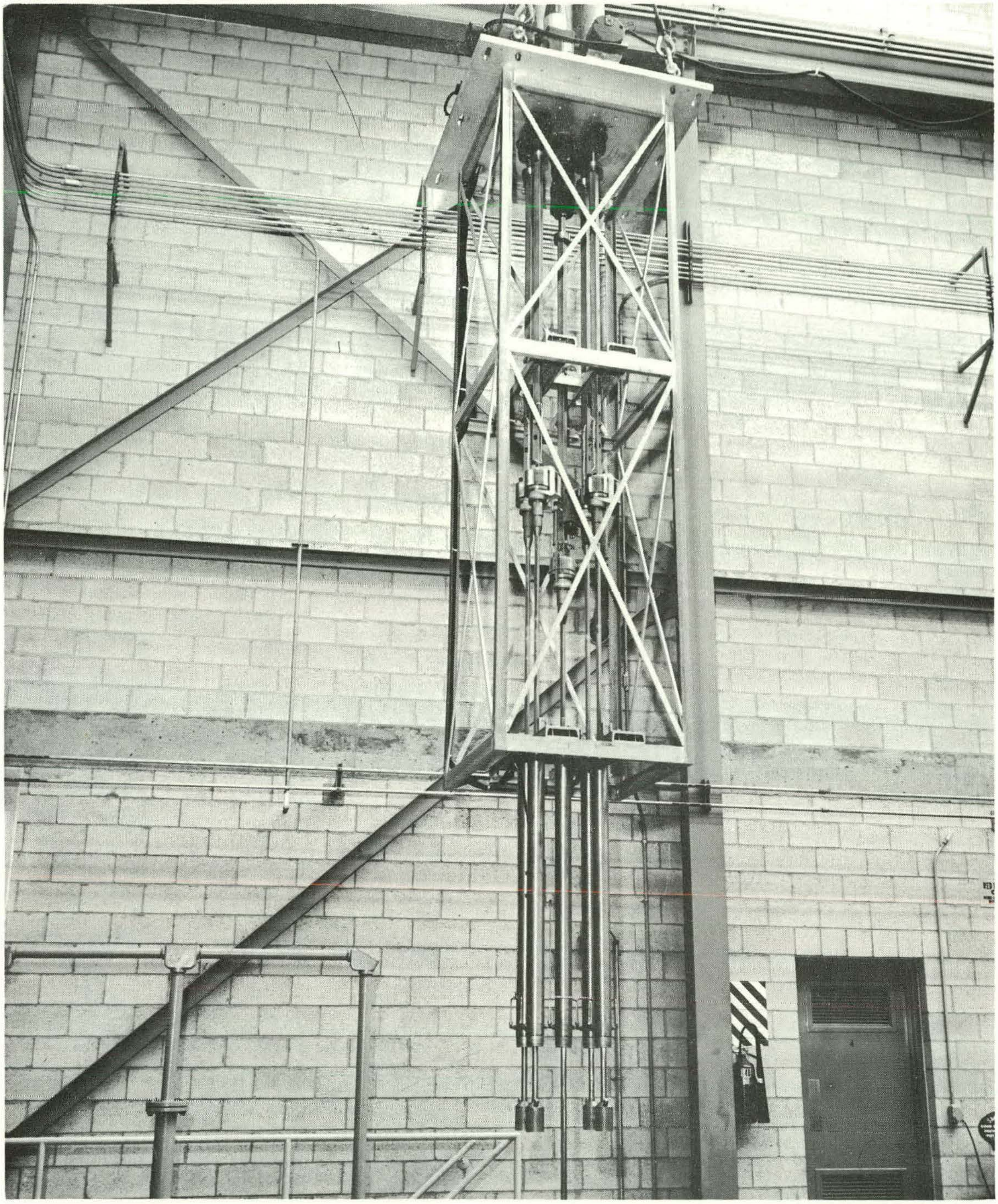


Fig. 32 Photograph of drive unit lower guide structure (NRTS-62-53).

size No. 56 spring-set magnetic release output shaft brake. This one unit drives all four sets of control rods. A 1.6:1 speed reducer on the output shaft drives a Telesyn transmitter which is coupled to a receiver at the Spert IV

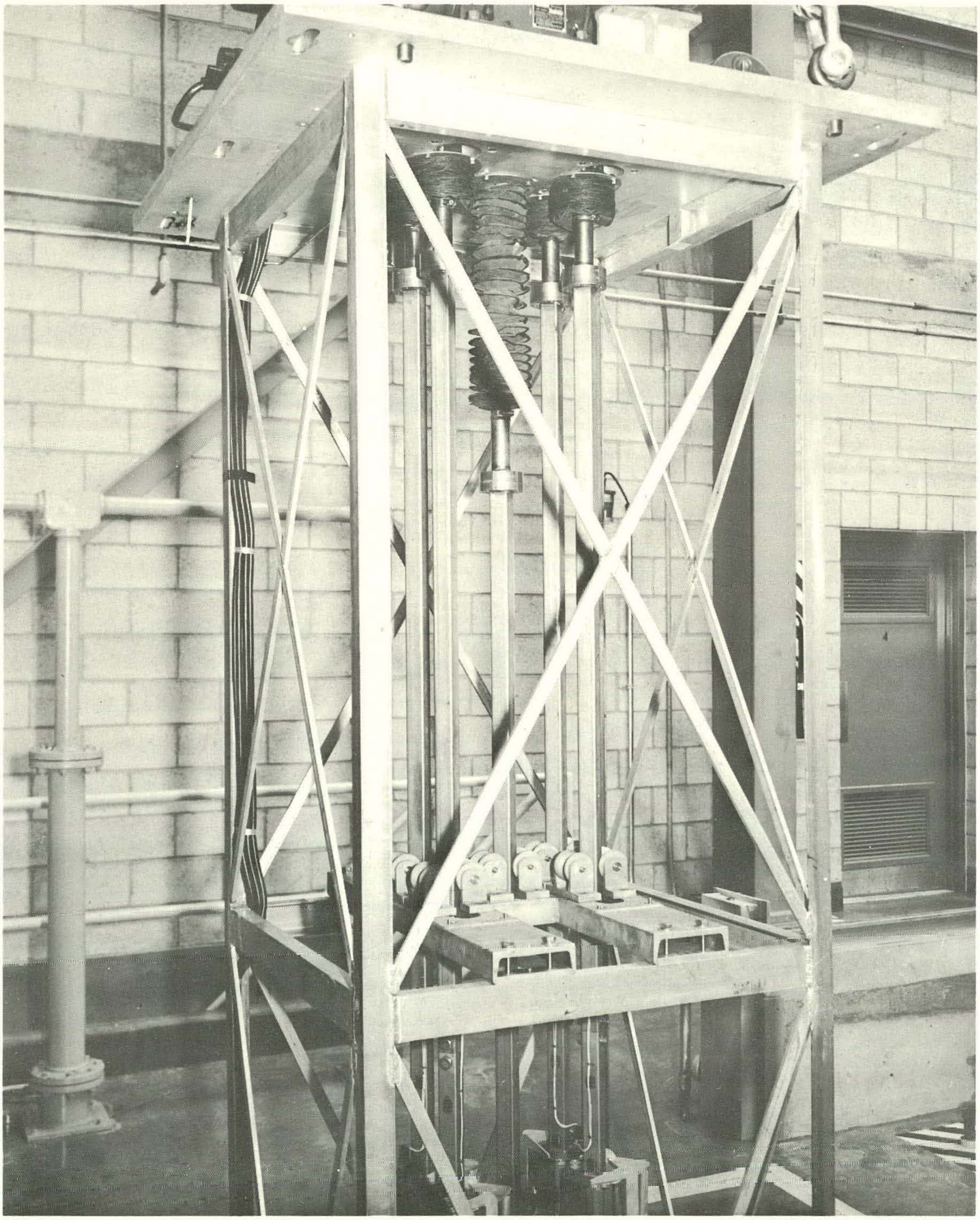


Fig. 33 Photograph of drive unit roller guides (NRTS-62-55).

control-center control room. The receiver drives a digital readout of rod position to the nearest 0.01 in.

An identical unit is used for the transient rod drive. Although the theoretical power required for the present speeds and control rod weights is less than

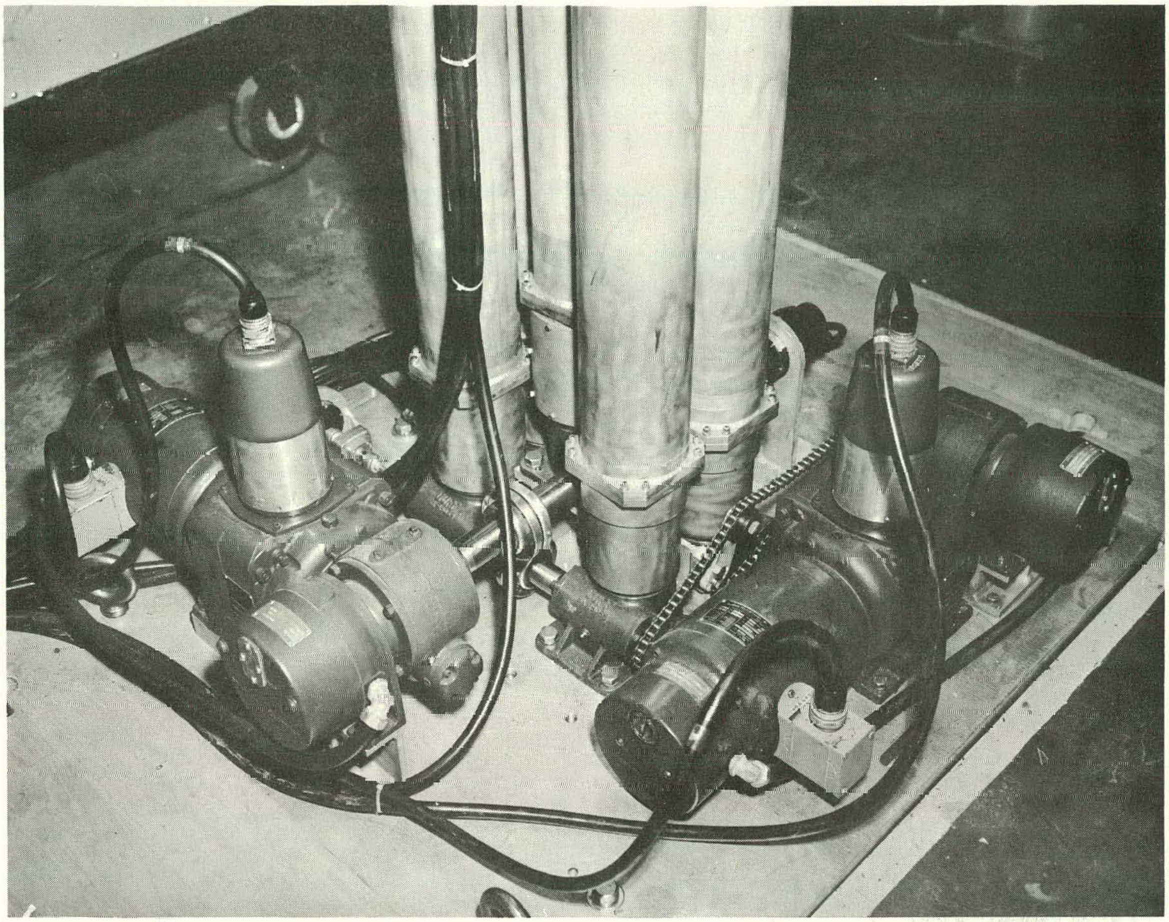


Fig. 34 Photograph of drive unit motors and transmission (NRTS-61-6824).

1/6 hp, the larger (1 hp) size was selected to permit an exchange of gearheads in the transmission to permit higher rates of withdrawal if the future experimental need arises and also to permit the use of a bank of heavier control rods such as the fuel follower type. The use of both a motor brake and an output shaft brake is required for a conical disc friction transmission of any size to prevent overrun.

2.4 Screw Jack

In order to obtain a versatile drive unit that could be easily modified and/or maintained, and have a low over-all height, a commercial inverted worm gear jack was selected as the control rod drive. Also, to enclose the electrical leads from the magnets and limit switches, a 3/4-in.-diameter through hole was required in the lead screw. A five-ton-rated commercial jack with a 1.500-in.-diameter lead screw satisfied these requirements. The 6:1 ratio jack has a bronze 24-tooth worm gear, a hardened-steel worm and an AISI-1015 cold-drawn seamless-steel-tubing 54-in. travel leadscrew with a 0.375-pitch modified square thread. The 24-tooth bronze worm gear is mounted with tapered roller bearings.

Since the five-ton rating for each jack has lifting capabilities approximately 100 times the requirements for any present or future planned control rod banks, no stress calculations were performed in the design of this unit.

The AISI-1015 leadscrew is protected below the control rod drive base plate by an accordian-pleat rubber boot fastened to the underside of the mount base plate and to the stainless-steel lower flange of the leadscrew. This boot also prevents the screw jack lubrication from contaminating the reactor pool water.

Fastened to the top of each jack unit is a 5-1/2-in.-diameter x 46-3/4-in.-long AISI type 304 stainless-steel protective tube for the lead screw. This tube replaces the standard 2-in.-diameter protective tube supplied with the unit. Enclosed within the protective housing is a "koil-cord" which controls the slack in the electrical leads coming from the magnet and limit switches at the lower end of the drive system.

Fastened to the leadscrew within the housing are two nylon adjustable actuating bushings for the upper and lower limit switches. Access for adjustment of the actuators is provided by gasketed cover plates on the tube. These limit switches control the travel of the control rod drives for each particular core. The switches are externally mounted on the protective tube in waterproof enclosures. All electrical leads from a particular drive are brought out of the protective tube through a multiple pin connector which is potted with high temperature polyurethane for waterproofing. The input shaft for each jack is a through shaft used for joining a series of jacks together. The control rod jacks receive power from a single driver.

The in-line control rod jacks are connected together with Falk No. 3F couplings. The two parallel lines of control rod drive jacks are joined together with a No. 40 single pitch roller chain, thus, effectively tying all of the control rod drives together as a single unit.

One output shaft on a control rod drive jack and the output shaft on the transient rod drive are each connected to a Metron 10 A 1.6 R speed reducer with a speed ratio of 1.6:1. The output of the speed increaser drives a No. 5 HG Telesyn generator for providing the drive position indication signal to the control center Telesyn receiver. The Telesyn generator rotates 360° for 0.10 in. drive travel.

Figure 35 is a photograph of the assembly of the control rod and transient rod drive motor, screw jacks, and leadscrew protective housing.

2.5 Magnets

The control rod drive unit magnets (Figure 36) are designed for operation either submerged in water or in air. The magnets are fastened with two 1/2-13 NC x 1-1/2 in.-long stainless-steel stripper bolts at the lower end of the 60-in.-long, 1-1/2-in. x 0.065-in.-wall, type 304 stainless-steel drive guide tube that is flanged to the lower end of the jack leadscrew.

The outer magnet housing is a 5-in.-OD x 4.218-in.-ID x 2-11/16-in.-long Armco iron cylinder with twelve 1/32-in.-wide x 1/4-in.-deep radial slots machined evenly spaced on the inner surface of the cylinder. The slots are used to induce a more rapid decay of the magnetic flux. The inner magnet core consists of a 2.500-in.-diameter by 0.625-in.-ID x 2-11/16-in.-long Armco iron cylinder with twelve 1/32-in.-wide x 15/16-in.-deep slots machined radially in the external surface. The inner surface of the inner magnet core is lined with a 0.625-in.-diameter x 0.047-in.-wall AISI type 304 stainless-steel tubing.

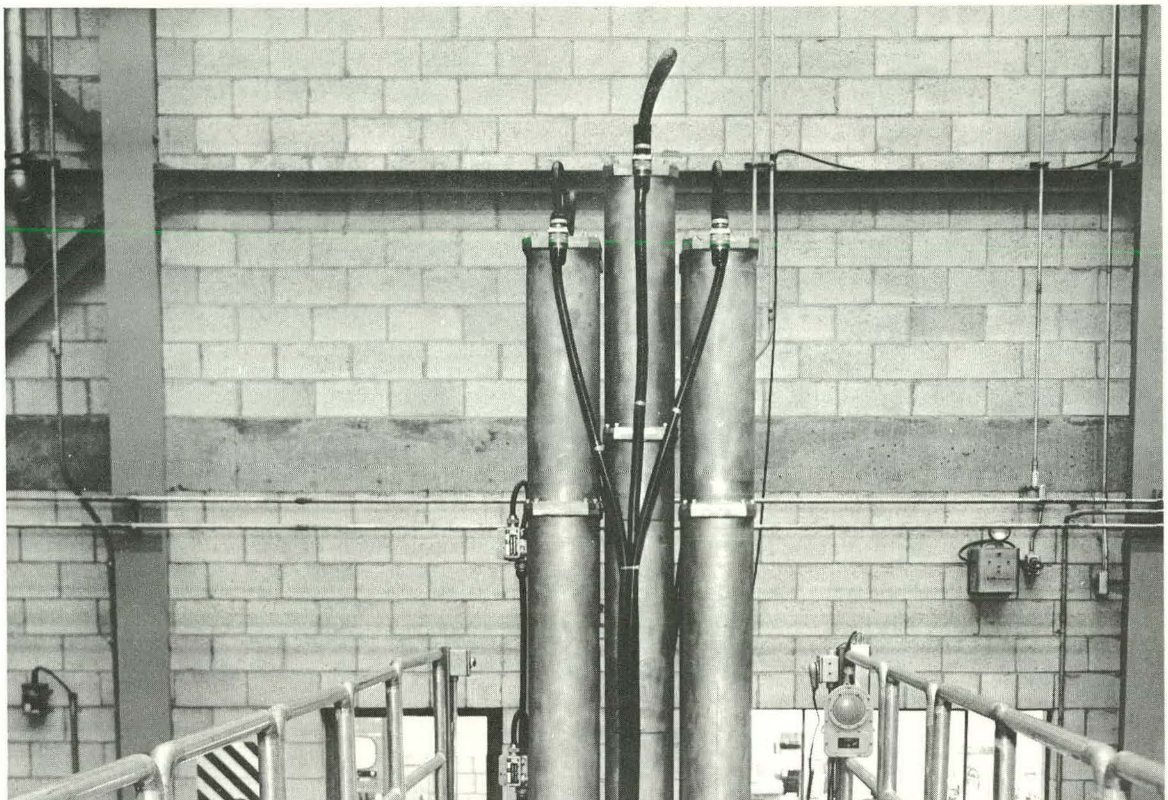


Fig. 35 Photograph of drive unit screw jacks (NRTS-62-52).

The magnetic circuit is completed at the upper end of the magnet housings with a 4.250-in.-OD x 2.500-in.-ID x 51/62-in.-thick Armco iron plate.

The lower face of the magnet is covered with a 0.078-in.-thick x 4.218-in.-OD x 0.625-in.-ID AISI type 304 stainless-steel plate seal welded to the outer housing and the inner housing liner. The upper face of the magnet is covered with a 5-in.-OD x 5/8-in.-ID x 11/16-in.-thick AISI type 304 stainless-steel disc seal welded to the outer and inner magnet housings. A 1-1/2-in.-square x 0.065-in.-wall x 11-3/8-in.-long spring housing is welded to the upper cover plate.

The coil consists of a 1-9/16-in.-wide x 4-3/16-in.-OD x 2-21/32-in.-ID phenolic bobbin filled with No. 22 AWG wire. The last wrap is No. 20 AWG braided wire. The resistance of the coil is 18 ohms. It is wrapped with glass tape over which insulating varnish is applied. A multiple pin connector potted with high temperature polyurethane is used to bring the electrical leads from the magnet.

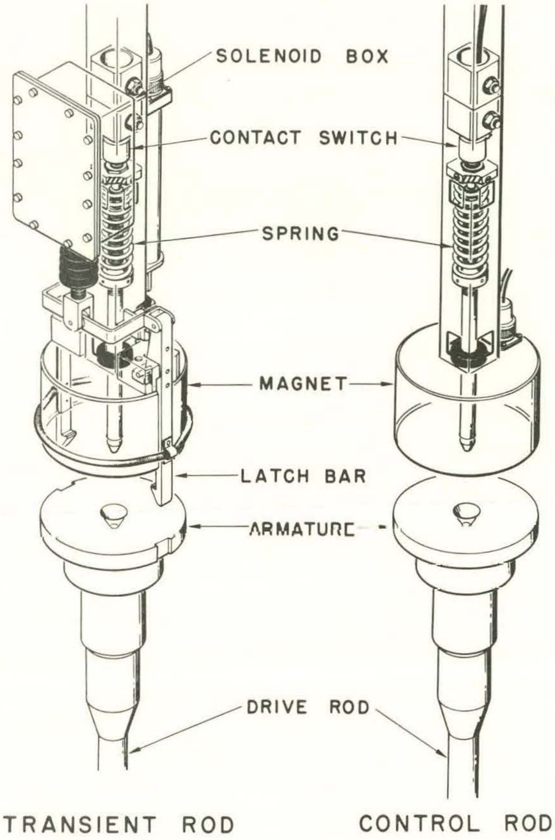


Fig. 36 Control rod drive magnet assembly (NRTS-61-2870).

A 1/2-in.-diameter by 11-1/4-in.-long bullet-nosed plunger is located inside the hollow core of the magnet and extends into the magnet extension tube. The bullet nose on the plunger fits a corresponding depression in the armature face to assure proper lateral line-up of the magnet and the armature. The plunger is driven by a 20-lb/in., 0.135-in.-wire-diameter, 1.260-in.-OD, 16-coil, 17-7 PH spring contained within the extension tube. The upper end of the plunger is threaded to adjust the spring preload to obtain the desired thrust. A "contact" limit switch is actuated by the plunger when the magnet has been driven to within 0.010 in. of the armature.

The transient rod magnet (Figure 36) is identical to the control rod magnets except a mechanical solenoid-operated latch has been added to prevent dropping of the transient rod if the magnetic circuit should be broken inadvertently. The latch is normally in the locked position and requires power to unlatch. Each of the two latch bars are beveled on the end to permit latching to the armature plate without retracting the latch. In normal operation there is no load on the latch bars and 1/8-in. clearance exists between the latch hook and the armature plate.

The 9-lb-at-1-in.-throw pull-type solenoid is mounted in a waterproof enclosure welded to the magnet extension tube. The solenoid plunger is sealed with a neoprene bellows where it passes through the watertight enclosure. A simple toggle mechanism transforms the vertical movement of the solenoid plunger to the horizontal motion to actuate the latch. A detent in the underface of the armature provides a seat for the latch tooth when the magnet is de-energized. These teeth prevent the latch from being retracted whenever there is a load on the latch. A garter spring over the latch arms and the magnet assure that the latch bars always assume a latched position when not actuated by the solenoid. Miniature limit switches mounted on the latch indicate the position of the latch as "latched" or "unlatched".

All parts of the magnets that are not stainless steel are plated with 0.002-in. nickel.

The armature plates for the control rods consist of 5-in.-OD x 3/4-in.-thick Armco iron. The top surface is plated with 0.010-in. nickel to serve as a magnetic flux shim and the remainder plated with 0.002-in. nickel. A 0.993-in.-diameter x 60° tapered hole provides the detent for the magnet push rod. Each armature is mounted on a stainless-steel "mono-ball" bearing to permit a maximum of 1° of armature tilt. The freedom to tilt assures full face contact of the magnet and armature during operation.

Performance of the magnets as determined by test is shown in Figures 37 and 38.

2.6 Shock Absorbers

The shock absorbers for the control rod and transient rod drives are identical. Figure 39 is a photograph of the shock absorber assembly and Figure 40 is a pictorial view of an individual unit. The function of the transient rod and the control rods during "scram" conditions is such that it is essential that the rods be inserted into the core as rapidly as possible. This in turn

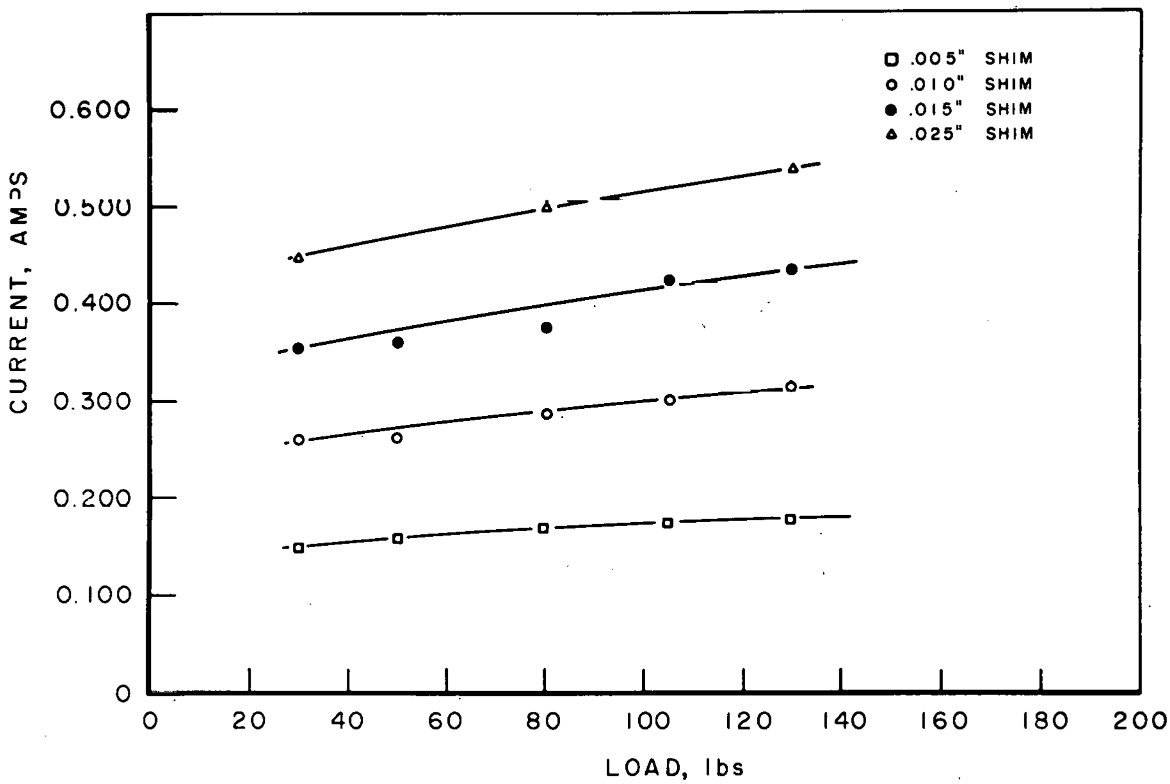


Fig. 37 Control rod drive magnet load performance curve (NRTS-62-62).

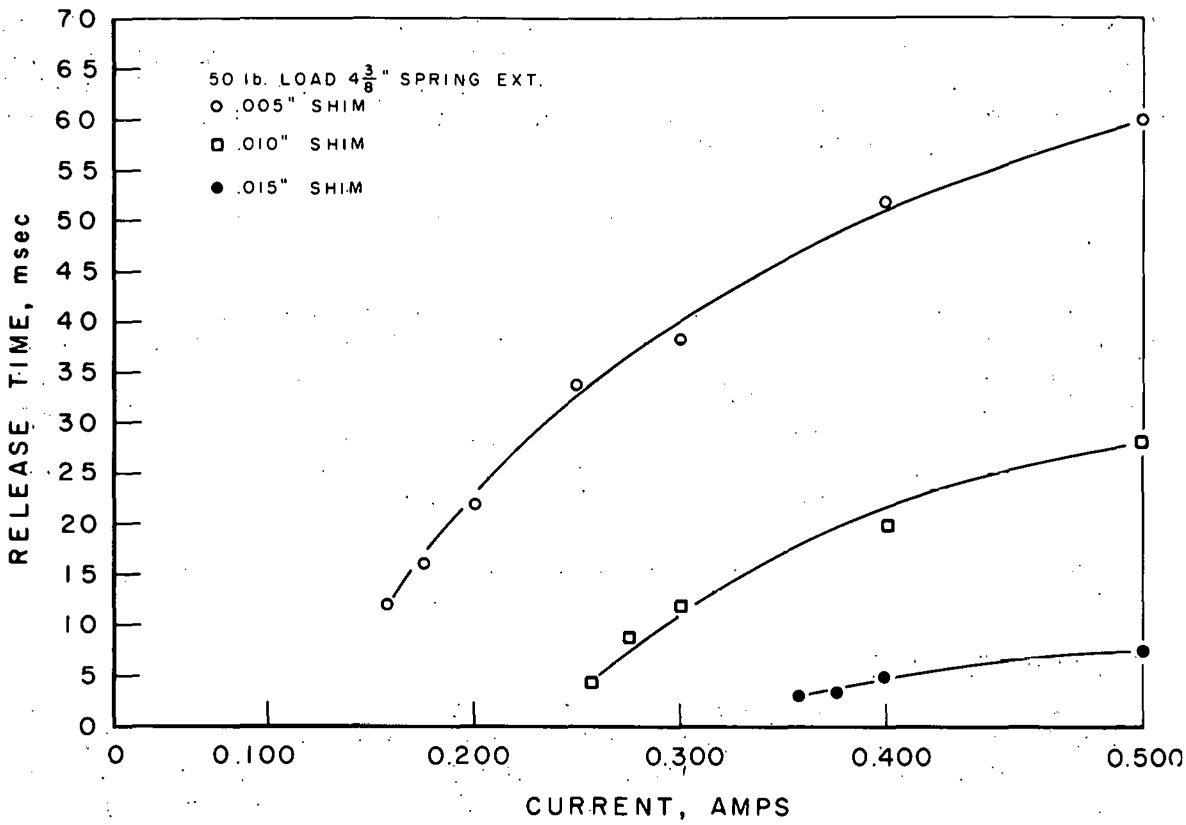


Fig. 38 Control rod drive magnet release time curve (NRTS-62-117).

introduces a requirement that the rods be decelerated as smoothly as possible and come to rest without "bounce-back" at a predetermined position such that the magnets may be recontacted and "seat" switches may be actuated verifying that the rods have indeed been inserted.

Since the experimental program may dictate as much as 18 ft variation in the head of water in the reactor pool, and the first core is designed for blade-type rods incapable mechanically of use as a compression column, it followed that the shock absorbers had to be located below the magnet assembly and above the core in a region where the environment would be either water or air and the stresses on the rods would be in tension.

The shock absorbers were therefore designed as individual in-line hydraulic rams enclosed in a water reservoir. The water level in each reservoir is maintained by one common level control. A piston attached to the drive rod passes through a cylinder in which controlled bleed holes have been drilled. As the piston passes each row of holes, these holes are cut off producing essentially a "step-linear" deceleration. The piston comes to rest on a spring which permits overrun of the drive units to compress the spring and bring all magnet faces to contact elevation. The shock absorbers are mounted on the drive unit frame and are an integral part of the drive unit assembly.

The 1.844-in.-diameter 6061-T6 aluminum piston is connected to the 1.000-in. AISI type 304 stainless-steel drive rods with 3/4-16 NF threads. The screwed connections are pinned with a No. 2 taper pin. The upper drive

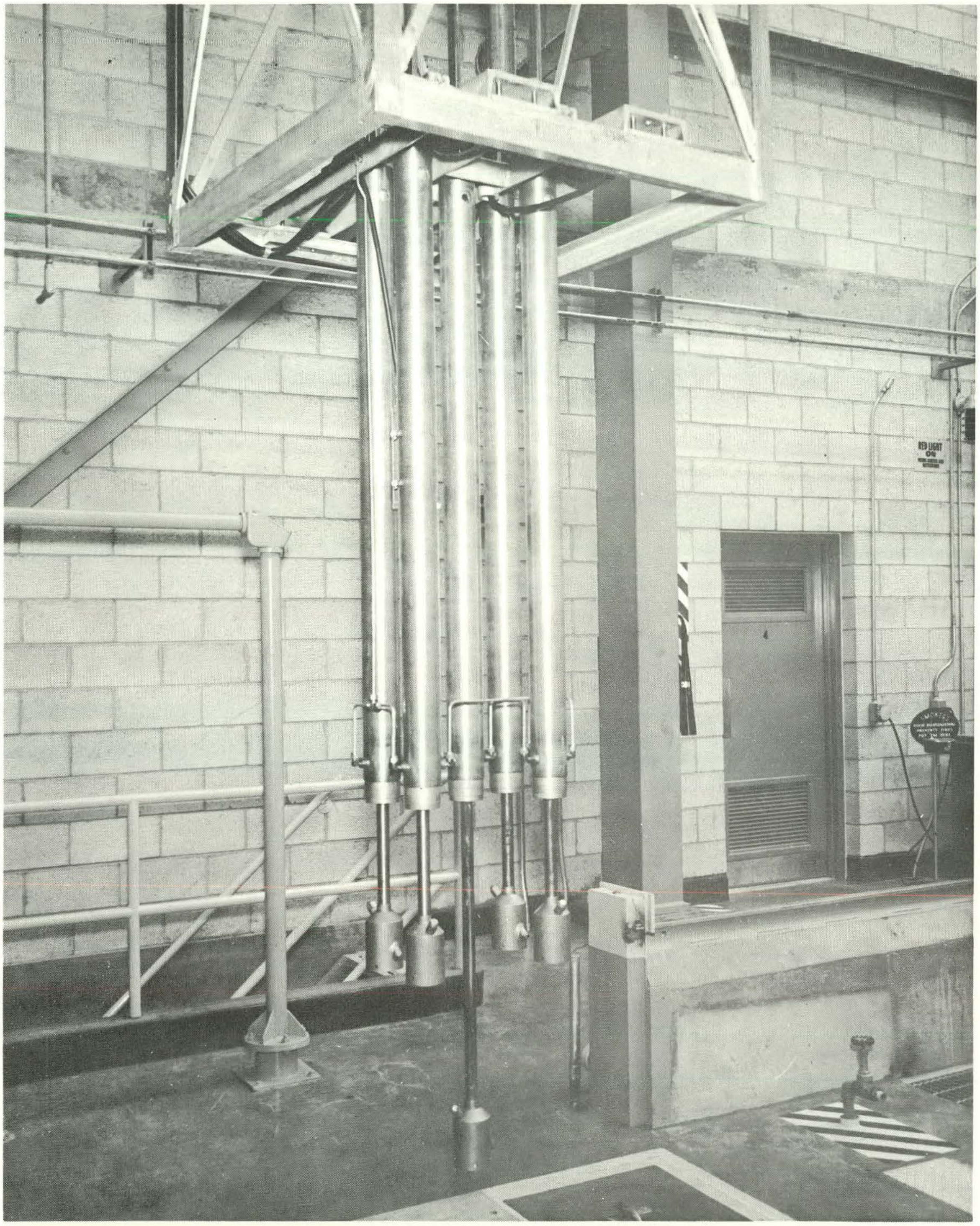


Fig. 39 Photograph of control rod shock absorber (NRTS-62-50).

rod connects to the 17-4 PH pilot holding the monoball bearing for the magnet armature with a 3/4-10 NC thread and is pinned in place. The tapered pilot is used to center the drive rod in the shock absorber in the down position and also serves as the actuator for the seat switch. The lower drive rod connects to the control rod couples with a 3/4-10 NC thread and is pinned in place.

The piston travels in a 51-3/8-in.-long x 2.25-in.-OD x 0.188-in.-wall AISI type 304 stainless-steel inner tubing. This inner tube is drilled with five groups of four equally spaced holes 3/4 in. diameter by 7 in. long and one group of three equally spaced holes 1/4 in. diameter by 2 in. long as well as seven rows of 5/32-in.-diameter bleed holes on 1/2-in. centers. The 0.250-in.-wire-diameter, 1.75-in.-OD, 8-coil 17-7 PH stainless-steel spring is held in by an AISI 304 stainless-steel seal housing which screws into the inner tube with a 2-7/8-in.-OD x 2-in.-ID x 1/16-in.-thick service sheet gasket between the seal housing and the outer tube. Inserted in the seal housing is a 1.378-in.-OD x 1.002-in.-ID x 1-3/8-in.-long nylon bushing which guides the lower drive rod. A National Oil seal fits against the bushing and is held in by an AISI 304 stainless-steel seal retainer which is attached to the seal housing with four 1/4-20 NC x 3/4-in.-long socket head cap screws.

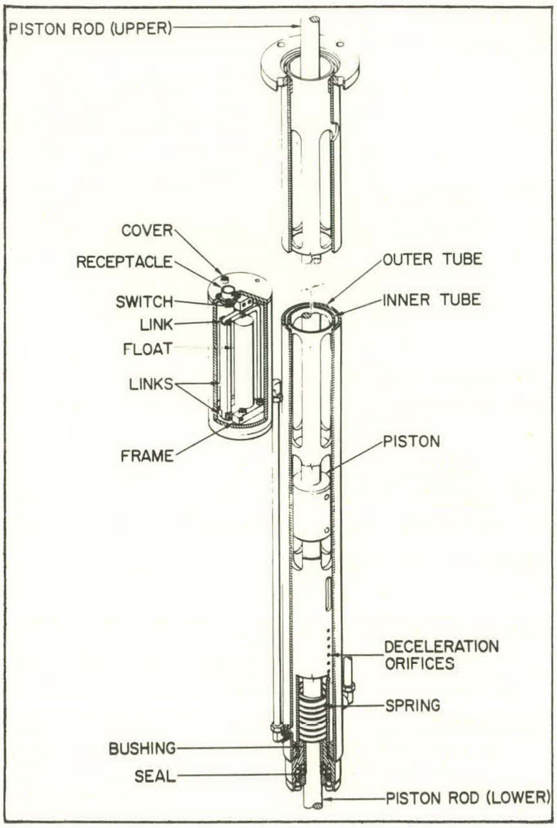


Fig. 40 Control rod shock absorber pictorial (NRTS-62-272).

The inner tube is contained in a 51-1/16-in.-long x 3-in.-OD x 0.120-in.-wall AISI type 304 stainless-steel outer tubing. This outer tube is attached to the shock absorber mount with four 3/8 - 16 NC - 2 x 3/8-in.-long cap screws. Attached to one of the outer tubes is the common 6061-T6 aluminum level controller. Through mechanical linkages, a float actuates a microswitch which opens a solenoid and allows water to enter the five shocks and fill the reservoirs to a height of approximately 27 in.

IX. AUXILIARY EQUIPMENT

1. WATER TREATMENT SYSTEM

1.1 Introduction

Water for the Spert site is supplied by two deep-well pumps, one at the control center designated as No. 1 Well Pump House and one 500 yds southeast of the control center designated as No. 2 Well Pump House. The No. 1 well has a 20-stage, 10-in. submersible pump having a capacity of 400 gpm at 500 ft of head and the No. 2 well has a 15-stage, 10-in. line shaft pump having a capacity of 550 gpm at 545 ft of head.

Either or both pumps operate intermittently on storage tank level controls to supply two interconnected storage tanks having a total capacity of 75,000 gal. Low-level alarms on the tanks are connected with the ADT system at the fire department and are set to alarm when the water level in the tank drops to 5-1/2 ft.

Water is distributed to the various reactor areas by two parallel booster pumps which, in conjunction with a pressure control valve, maintain 70-psig pressure on the Spert distribution line. Under normal operating conditions only one booster pump is in service; but whenever supply pressure at Spert IV drops below 40 psig the second booster pump will be put in operation manually.

Well water is supplied to Spert IV at 50 psig. The well water is used for equipment cooling, heat exchanger cooling water, emergency fill in the North pool, utility purposes, and water treatment equipment supply.

The water treating equipment for the Spert IV facility is located at the reactor building in room 6. The system consists of a water softener, two parallel demineralizer units, and associated piping, controls, and alarms.

Table II presents a typical analysis of Spert well water.

1.2 Softener

The softener shown in Figure 41 consists of a resin and brine tank, totalizing meter, and associated piping. When supplied with 12 gpm of water with an initial hardness of 9 grains/gal as $CaCO_3$, the softener will deliver water free of hardness as shown by a Bautron and Baudet test.

The soft water is used for boiler fill and make-up water, plant hot water service, and for regeneration of the deionizers.

Table III gives the engineering and operating data for the softener.

The water hardness is determined frequently to indicate when regeneration is required.

1.3 Demineralizers

There are two separate demineralizer units each with a maximum flow rate of 50 gpm and a minimum capacity of 15,000 gals of demineralized water

TABLE II

ANALYSIS OF SPERT WELL WATER

	ppm
Ca	39
Mg	14
Fe	0.04
F	0.1
Mn	0.01
Na	8.8
K	27
B	0.05
SiO_2	26
NO_3	1.2
HCO_3	158 ppm as $CaCO_3$
Cl	16
SO_4	7
Dissolved solids	205
Hardness	147
pH	8.2
Specific conductance at 25°C	332 micromho/cm

with a specific conductance of 2 micromho/cm or less. Figure 42 is a photograph of the demineralizer installation.

The two demineralizer units consist of two vertical tanks lined with 100-mil plastesal, anion and cation resin beds, two 55-gal polyethylene chemical regenerate tanks and electric mixers, conductivity indicators and recorders, totalizer meters, and associated piping, controls, and alarms.

Water is supplied directly to the demineralizers from the Spert distribution piping and, after removal of ionic impurities by the resin beds, the demineralized water flows to the North or South pools or to a 6000-gal PVC-lined storage tank. The flow pattern to the reactor pools is determined by a selector switch located on the reactor building process panel. A conductivity recorder alarms at the process and Spert control panels on high conductivity which indicates a need for regeneration.

Table IV gives the engineering and operating data for the demineralizers.

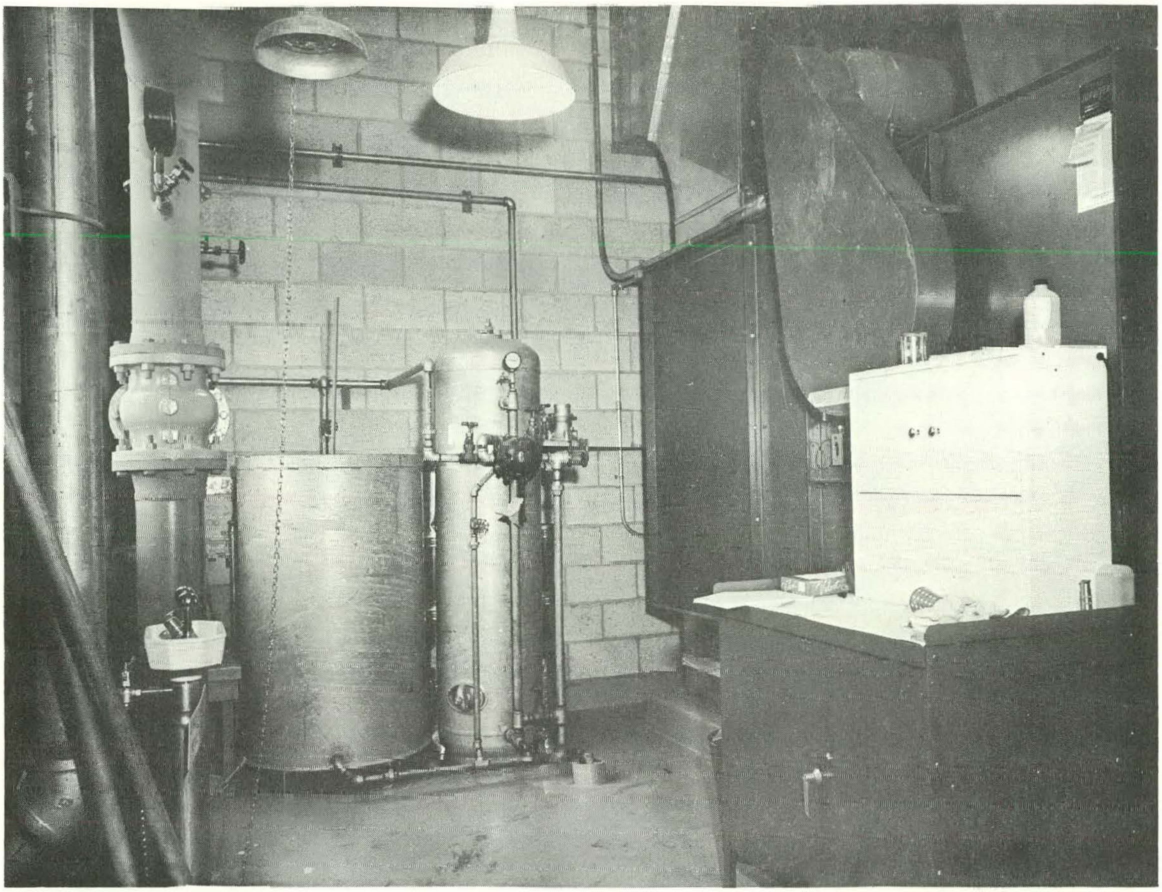


Fig. 41 Photograph of water softening unit (NRTS-61-7097).

TABLE III

SOFTENER ENGINEERING AND OPERATING DATA

Resin Tank	
Shell OD (in.)	18
Shell height (in.)	60
Design pressure (psig)	125
Material	galvanized steel
Resin	
Type	high capacity polystyrene
Quantity (ft^3)	4
Bed depth (in.)	24
Regenerate	NaCl
Gals/regeneration	10,000
Maximum flow rate (gpm)	20

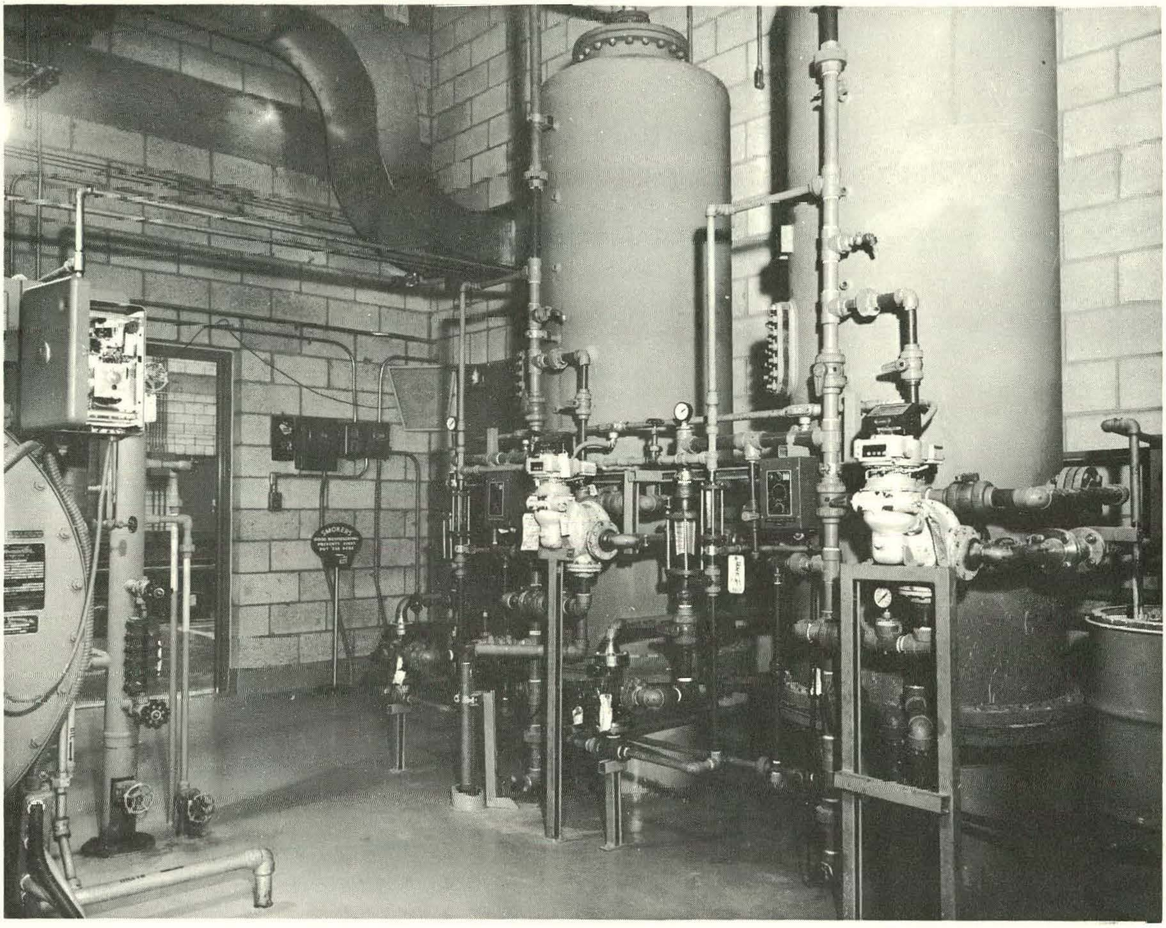


Fig. 42 Photograph of plant demineralizer unit (NRTS-61-7088).

1.4 Clean-up System

The clean-up system is comprised of a stainless-steel 50-gpm centrifugal pump and stainless-steel piping which permits suction to be taken on either pool, and discharging to the deionizers or the sump for disposal.

The purpose of the clean-up loop is to reclaim low-mineral-content water which is unsatisfactory for reactor use. This low-mineral-content or off-standard deionized water can be returned to a high purity state at little cost due to the increased capacity of the deionizers which is realized by using off-standard deionized water as supply for the deionizers.

Because of the possibility that the reactor water can contain fission products the following conditions must be met before the system can be used for reclamation purposes:

> (1). A representative reactor water sample will be taken. If the gross fission products are less than 3 x 10^{-5} microcurie/cc the clean-up system can be used.

TABLE IV

DEMINERALIZER ENGINEERING AND OPERATING DATA

Resin Tank	
Shell OD (in.)	36
Height (ft)	8
Design pressure (psi)	100
Piping	
Caustic	black iron with brass valves
Acid	PVC (schedule 80)
Demineralized water	PVC (schedule 80)
Cation Resin	
Type	high capacity sulfonated polystyrene
Quantity (ft^3)	72.5
Depth of bed (in.)	21
Anion Resin	
Type	quaterary amine-Type I-strongly basic
Quantity (ft^3)	23
Depth of bed (in.)	36
Regenerate	
Acid	H_2SO_4 86° Baume
Base	flake caustic
Gals/regeneration/unit	15,000
Specific conductance at 25°C	less than 2 micromho/cm
Maximum flow rate/unit (gpm)	50
Maximum pressure drop (psi)	10

(2). Double block and bleeder valves at the branch supplying the deionizers, and the two softener-deionizer cross ties will be removed.

(3). Install double block and bleeder valves at the clean-up pump discharge.

The preceding procedures isolate the clean-up system from the potable water system and will prevent contamination of the potable water.

Upon completion of the clean-up cycle the preceding procedures will be repeated in reverse.

2. COMPRESSED AIR SYSTEM

Instrument and utility air is supplied by the plant air system.

The plant air compressor is a vertical-type, heavy duty, 7- x 5-in., single-stage, double-acting, water-cooled, carbon ring air compressor rated at 59.5 SCFM at 5000 ft above sea level, and discharging at 125 psig.

The compressor discharges to the after-cooler and a cyclone entrainment separator to an air receiver having a capacity of 110 cu ft. The air receiver serves to balance the air system pressure and to prevent an excessive loss in pressure during periods of heavy demand.

From the receiver the air goes to an after-filter which removes particles larger than 3 to 4 microns.

At this point the air system is divided into two branches: instrument and utility air.

Utility air is reduced to 60 psig for general plant use such as air bibs, gate seal between North and South pool, and with a further reduction to 8 psig for use in regeneration of the deionizers.

The instrument air branch feeds to a dual-tower air dryer. One tower is in operation at all times while the other tower is being reactivated. A timing device controls pneumatic shifting, electrical heating, and cooling of the towers. Indicating lights, thermometers, and pressure gages are provided to indicate which tower is in operation.

Each tower air dryer is rated to handle 30.5 SCFM, in continuous service, while receiving air at 125 psig, 85°F and 100% relative humidity. The outlet dew point is less than -30°F.

After leaving the air dryer the air is reduced to 30 psig for instrument use.

An interlock is installed between the utility and instrument air lines which shuts off utility air in the event the air system pressure drops below 60 psig. This prevents loss of, or a delay in losing, the instrument air.

A low air pressure alarm alarms and indicates on the reactor building process panel. The alarm point is set at 27 psig.

In the event of complete loss of air pressure all air-controlled process valves are automatically closed and the ventilating system will shift to emergency conditions. A description of the emergency air system is included in Section IX-4, "Heating and Ventilatng System".

3. WASTE DISPOSAL

The four liquid wastes which have to be disposed of at Spert IV during operations are chemical wastes, waste cooling water, contaminated waste, and sanitary waste.

3.1 Chemical Wastes

Chemical wastes are produced during regeneration of the deionizers. The chemical waste is directed to the leaching pond by gravity flow. The drain line is a 6-in. Saran-lined steel pipe equipped with a heating coil where earth cover is only 3 ft, to prevent freeze-up during extremely cold weather.

3.2 Waste Cooling Water

Waste cooling water is raw water that has passed through the heat exchanger. This water originates from the Spert distribution piping and is routed to the heat exchanger at 40 psig. After passing through the exchanger the water is directed to the Spert IV lake.

The waste cooling water effluent line of the heat exchanger is provided with a radiation monitor and alarm system to detect any leak of water bearing radioactive isotopes from the tube side (reactor cooling water) to the shell side (exchanger cooling water) of the heat exchanger.

The detection unit indicates in the reactor building and records at the control center. High radiation alarms will sound at the reactor building and at the control center (see radiation instrumentation Section VI-8).

The effluent high activity alarm set point will be set at 0.8 mr/hr above background.

Procedures to be followed upon a high radiation level in the heat exchanger effluent will be dictated by the level of activity and the state of the reactor.

3.3 Contaminated Waste Water

Contaminated (radioactive) waste water from whatever source is intended to be drained or flushed into the building sump for disposal.

The contaminated waste system is composed of a sump pump, continuously monitoring loop, sampling station, hold-up tank, leach pond and all piping, controls, and alarms necessary for operation.

The sump pump is a vertical, turbine-type pump with a capacity of 300 gpm at 70 ft of head. The pump operates automatically on level control and discharges via a 6-in. line to the leaching pond or contaminated waste hold-up tank. The pump can be controlled from the Spert control center.

A 3-in. loop is provided in the sump pump discharge line for purposes of monitoring and sampling. Flow is accomplished in the loop by the use of a restricting orifice. A monitor in the loop working in conjunction with alarms, indicators, and controllers automatically operates a 3-way valve at the contaminated waste hold-up tank. High radiation alarms sound at the reactor building and Spert control center. The radiation level is recorded at the control center and is indicated at the reactor building.

Contaminated waste water with a radioactive isotope content greater than 50 counts/min above background would be diverted automatically to the contaminated waste hold-up tank for decay before disposal to the leaching pond.

The contaminated waste hold-up tank is 26 ft in diameter and 16 ft high, with a storage capacity of 61,000 gal. The contaminated waste hold-up tank is provided with local and remote level indicators, heating coils to prevent freeze-up during cold weather, high-level alarms that sound at the reactor building Panalarm and at the control center process Panalarm when the level reaches 14 ft and a valve to permit draining to the leaching pond when the contaminated liquid waste has decayed to "RCG" as defined in National Bureau of Standards Handbook No. 69.

3.4 Sanitary Waste

The sanitary waste disposal system consists of a septic tank, leaching pit, and piping required to connect the above with the change room facilities.

The septic tank is 8 ft below grade and located 100 ft south of the reactor building. The tank is a watertight precast concrete cylinder with a capacity of 800 gal below the inlet piping. The tank is equipped with two manhole covers with lifting bails and inlet and outlet baffles.

The leaching pit is 60 ft south of the septic tank. The pit is constructed of standard 8-in. pumice blocks laid on a 10-in. reinforced concrete footing. The pit is 8 ft in diameter and extends 8 ft below and 5 ft above inlet piping. The sides and bottom of the tank are surrounded by 18in. of minus-1-1/2-in. rock and gravel. The cover is a poured concrete reinforced slab and provided with a manhole and cover with lifting bail. The top of the pit is 1 ft below grade.

The piping which connects change room facilities with the disposal system is 4-in. concrete pipe with cement-caulked heel and spigot joints.

4. HEATING AND VENTILATING SYSTEM

The major heating and ventilating equipment is located in Room 6 at the Spert IV reactor building.

The heating system is comprised of a package boiler complete with controls, fuel oil storage tank and booster pump, boiler make-up pump and condensate receiver tank, chemical addition tank and pump, steam distribution piping and all interconnected piping to enable the various components to function as a unit. The ventilating system components consist of an air tempering and distribution system, and automatic temperature and ventilation controls.

Heat is supplied at 13 psig by an Orr and Sembower 50-hp, 1,670,000-Btu/hr gross output, three-pass firetube boiler which is equipped with an Orr and Sembower mechanical pressure-atomizing fuel oil burner assembly designed for use with No. 2 fuel oil and a capacity of 15 gal/hr.

The boiler controls consist of a pressure control, a type 29 RF5 "Fireye" control, and a "Magnetrol" level control.

The boiler operates between 9 psig and 13 psig. These limits are controlled by the pressure controller. When the steam pressure in the boiler drum drops to 9 psig the pressure control relay closes and provides power for the "Fireye" control which programs the operating sequence of the blower and/or burner motors, ignition and fuel valves. The "Fireye" also provides a purge period before and after each firing period and automatically shuts the boiler down if a flame failure occurs. The control is reset manually after a flame failure.

The "Magnetrol" controls the on-off operation of the make-up pump and in conjunction with the "Fireye" shuts off the oil supply if high or low water conditions exist. The "Fireye" recycles automatically after the high or lower water condition is remedied. The boiler must be manually blown down on high water conditions. An underground 30-in. x 6-ft vitrified clay pipe with a wooden cover and filled with large aggregate serves as a blowdown pit for the boiler. The pit is located 20 ft northwest of the reactor building.

When the boiler malfunctions (ie, flame failure, high or low water) an alarm is sounded at the Spert IV building and Spert IV central control room. A flashing light in the Panalarm at both of the above locations indicates boiler malfunction.

Boiler trim is provided for a 15-psig system and includes a water column, steam pressure gage, boiler and gage glass blowdown valves, and a safety valve.

The water column is a 3-stage unit with try cocks, gage glass, drain and shut-off valves.

The safety valve is a Kunkle ASME standard relief valve with a relief pressure of 15 psig and a capacity of 28 lb of steam/hr.

A 3000-gal fuel oil storage tank is located 5 ft below the surface and 30 ft north of the reactor building. A 1-1/2-gpm fuel oil recirculating pump located in the reactor building basement takes a suction on the tank and delivers fuel oil at 15 psig to an oil reservoir located in Room 6. The reservoir serves as a pressurized oil supply for the burner rotary gear fuel pump. The recirculating pump operates simultaneously with the burner fuel pump.

The condensate receiver located in Room 6 at the reactor building is 24 in. in diameter and has a capacity of 100 gal. The tank shell is 3/16 in. thick and is vented to the atmosphere. The interior and exterior of the receiver is galvanized.

The receiver is supplied by condensate from the steam heating coils under normal conditions. If the water level in the receiver drops below 8 in. a McDonnell Miller series No. 21, float-operated make-up valve opens and allows softened water to flow into the receiver until the level is 10 in.

The make-up is an Aurora Pump Model D4, single-stage, centrifugal pump with a capacity of 7.9 gpm at 15 psig. The make-up pump takes suction from the condensate receiver and discharges to the boiler. The pump is controlled by the "Magentrol" level controller on the boiler.

The boiler water treatment system is a Milton Roy package system consisting of a reciprocating, positive-displacement, controlled-volume, chemical addition pump with a maximum capacity of 1.2 gal/hr, and a 50-gal platform chemical mixing and storage tank complete with cover and sight glass. Trisodium phosphate and sodium sulfite solutions are stored in the tank for injection into the boiler water system. The chemical addition pump operates simultaneously with the make-up pump and discharges to the condensate receiver.

The boiler water is tested daily for phosphate, sulfite alkilinity, and hardness. The phosphate concentration is maintained at 40 to 60 ppm and the sulfite concentration at 4 to 8 ppm. The boiler is given a blowdown as necessary to keep total alkilinity below 400 ppm.

The air tempering unit is a Trane Torrivent Model 224-1. The unit incorporates large coil face "non-freeze" Model NS heating coils, combination filter and mixing box with top and bottom opposed porportional mixing dampers, throw-away medium capacity filters, adjustable V-belt drive, belt guard and a 15-hp 480-volt 60-cycle 3$\emptyset$ motor. The two ventilating fans are 24-in. double-inlet forward-curved mutliblade-type centrifugal fans driven by a common shaft at 703 rpm. The tip speed is approximately 4400 ft/min. The unit is mounted on vibration insulators.

The capacity of the unit is such as to permit heating 22,350 cfm of air at 5000-ft elevation from -20°F to +60°F when using 10-psig saturated steam. The unit provides approximately 6 air change/hr in each room of the reactor building.

A galvanized-steel ductwork system complete with dampers, air-operated damper control motors, and heating coils distributes and porportions warm air to three zones in the reactor building. Zones 1 and 3 are the east and west building wings respectively and zone 2 is the reactor room and the basement.

The following results are attained by the controls in the ventilating system.

(1) Temperature Controls. - Comfort conditions are maintained in the building wings and the reactor room by pneumatic thermostats which in turn control the steam flow to the zone heating coils.

Discharge air from the ventilating fans is controlled at 60°F by controlling the ratio of recirculated air to fresh air in the system and steam flow to heating coil in the "Torrivent" tempering unit.

Ventilating fans are shut down automatically if the temperature of the heating coil in the "Torrivent" tempering unit is less than 42°F, unless this controller is over-ridden at the Spert IV control room process panel. This will prevent freeze-up of the heating coil.

(2) Ventilation Controls. - There are two patterns of air flow attainable in the reactor building known as normal and emergency conditions.

Normal conditions exist when fresh air is mixed with recirculated air from the basement and distributed to the various rooms in the building. A slight positive pressure is maintained in the reactor room by a static pressure regulator which in turn controls exhaust damper 3 located in the reactor room high bay. Dust infiltration, and consequently particulate contamination of the reactor water, is thereby minimized. A positive pressure greater than that in the reactor room is maintained in the building wings by preset air volume to the reactor room and building wings. The normal air flow pattern is then from the wing annexes to the reactor room and basement to cold air return.

The emergency ventilation is provided for the purpose of purging the building of air-borne activity if meteorlogical conditions are satisfactory.

The following is the sequence of events for changing the ventilating system from normal to emergency conditions:

Upon a signal of high radiation from the air particulate monitoring system, "CAM", the ventilating fans will shut down automatically, and a high radiation alarm will signal at the Spert IV instrument room and Spert IV control room at the control center. This will prevent spread of air activity within the building. At this time, if meteorlogical conditions are favorable, a key switch at Spert IV central control room may be turned to emergency conditions. This switch will de-energize FCV-106 which with other controls shuts off steam flow to the coil supplying heat to the main reactor building duct, and vents air on damper motor operators so that 100%-fresh air is taken in from the outside and discharged to the reactor building wings and basement. The air is purged to the atmosphere through damper 4 located in the southeast corner of the reactor building high bay. The heating system will be taxed during prolonged emergency ventilating conditions to maintain comfort conditions in the building so a lowering of building temperature is to be expected.

5. TRANSIENT INSTRUMENT ROOM AIR CONDITIONER

The air conditioner serves to remove heat load generated by the transient instruments located in room 4.

The air conditioner is a wall-mounted Carrier Model 63 E5 with a heat removal capacity of 54,000 Btu/hr as defined by ARI Standard 210-58. The evaporator is composed of a hermetically sealed compressor and motor. The evaporator is driven by a 1/2-hp, 440-volt, 3ϕ, 60-cycle motor and the condenser fan is driven by a 1/4-hp motor. Proportional dampers permit 0% to 100% of room or outside air to be admitted to the intake. Safety devices include a built-in overload relay on the compressor, and a pressure relief device.

The on-off operation of the air conditioner is controlled by a thermostat to maintain 72°F temperature in the transient instrument room.

6. DRY DOCK

A dry dock is installed for the purpose of safely performing any repairs or adjustments on the control rod drives, seat switches, magnets, latches and shock absorbers.

The dry dock is composed of three levels: (a) Drive support plate level on the operating floor, (b) Intermediate level at 8 ft 2 in. above pool bottoms, (c) Lower level at the pool bottom level.

The drive support plate rests on a 6- x 6-in. oak framework and the control rod drive extensions are 2 ft above the pool floor level when fully extended.

The intermediate platform is constructed of 8-in. channel and is supported by four 2-1/2-in. schedule-40 pipe columns. The intermediate platform is complete with railing, bar grating, and a controlled access gate. The space below the platform and between the support columns is caged on three sides with chain link fencing and the fourth side with a folding controlled access gate.

The access gates are provided with a solenoid-operated latch and a micro-switch to indicate gate position. The latches are controlled and the gate positions indicated at the Spert IV control room. Yellow flashing lights at the dry dock cages indicate that a control rod is withdrawn from its seat.

The warning lights and cage-access gate arrangement are provided to protect personnel in the reactor area from being injured by the control rod drives during repair, adjustment or operation in the dry dock.

7. BUILDING CRANE

The crane is a 12-ton double-girder, single-trolley unit made by Craneveyor Corp. The span of the girders is 42 ft 8 in. and the hook lift is 58 ft.

The bridge, trolley, and hoist are driven by 2-speed, double-wound motors with magnetic brakes.

The hoist has two 6-ton hooks connected by a 3-ft (center of hook to center of hook) load bar and a 12-ton hook at the center of the load bar. The double 6-ton hooks provide better rotation control of hoist loads.

Multispeed hoist, bridge, and trolley push button stations plus a jog (up and down), and emergency stop push button stations are stacked to form a single-unit crane-control pendant. Two pendants are provided; one with 45 ft of cable for reactor building use and one with 10 ft of cable for use at the Spert IV control center control room during remote crane operation.

The bridge and trolley are provided with automatic end stops and the hoist is equipped with automatic upper and lower limit switches. The emergency stop button shuts off all power to the crane and spring-loaded brakes prevent any motion when the crane power is off. The jog push button raises or lowers the hoist in 1/8-in. increments.

The crane component speeds are listed in Table V.

TABLE V

CRANE COMPONENT SPEEDS

	Low	High
Hoist	5.1 fpm	15.3 fpm
Bridge	18.2 fpm	54.6 fpm
Trolley	18.9 fpm	56.7 fpm

8. RADIATION MONITORING SYSTEMS

There are five separate and distinct radiation monitoring systems installed at the reactor building: constant air monitor (air particulate); remote area monitors; sump pump discharge monitor; waste cooling water monitor; and portal monitors.

8.1 Constant Air Monitor

The constant air monitor (CAM) is Nuclear Measurement's Model AM-2A and is composed of a detector, count rate meters and recorders, compressor with interchangeable flow orifices, filter holder and disposable filters, and alarm circuitry.

The detector indiscriminately measures β, and γ radiation collected on the filter (and some direct radiation) and in conjunction with the count rate meter and recorders, the radiation level in the building is indicated and recorded at the reactor building and control room. The range of the count rate meter and recorder is 50 to 50,000 counts per minute.

The flow through the compressor is 1 cfm and the filters remove particles larger than 0.3 micron.

The alarm circuitry is composed of an alert and alarm circuit and controls operation of the vent fan.

Alert conditions are signified by a constant ringing bell for 30 sec in conjunction with a flashing yellow light on the CAM.

Alarm conditions are signified by an intermittent ringing bell and flashing red light on the CAM.

The vent fan is shut off automatically by high radiation alarm from the CAM and will return to normal on alert or low radiation conditions.

8.2 Remote Area Monitors

The remote area monitoring system consists of one basic control unit, five sensing elements, recorder, and inter-connecting signal cable between the various components.

The basic control unit is Victoreen's Model 712 and consists of five indicating meters with a range of 0.1 to 100 r/hr. The indicating meters have a calibration potentiometer, high radiation alarm set point and light and reset control.

The elements are sensitive to γ radiation only and are capable of operating in fields of 0.1 to 100 r/hr. Each sensing unit is completely water proofed and includes a self-contained, remotely operated calibration source.

The basic control unit is located in Room 4 (instrumentation bunker) at the reactor building and the five sensing elements are distributed throughout the building as noted below.

(a) West wall of Room 5.

(b) East wall of Room 5 at stairway to the basement.

(c) On the bridge over the reactor pool.

(d) East wall of basement.

(e) West wall of basement at the south end.

The remote area radiation monitor recorder is located at Spert IV in the control center.

8.3 Sump Pump Discharge Monitor

The sump pump discharge monitoring system consists of a sampler, detector, amplified rate meter, recorder and alarms.

The sampler is Tracerlab Model MW-2 and consists of a 3-in. stainless-steel flange pipe tee with 2 in. of lead shielding around the run of the tee. The sampler is installed in the sump pump discharge monitoring loop described in "Contaminated Waste Water" (Section IX-3-3.3).

The detector is a watertight γ scintillation unit secured by a leak proof, shock-resisting collar in the sampler. The detector has a 1-1/2-in.-diameter by 1-in.-thick sodium iodide, thallium-activated crystal to function with a photo-multiplier tube and a μ-metal shield. Maximum pressure on the detector is 30 psig.

The preamplifier, rate meter, and indicating panels are located in Room 4 at the reactor building. The rate meter is a five-decade log rate meter. The rate meter has a self-contained regulated high voltage supply to permit calibration of the meter. The panel meter has five scales indicating counting rates directly in counts per minute with four decades full scale on the meter and one scale indicating high voltage. A suppression switch permits the selection of 10, 100, 1,000, 10,000, and 100,000 counts per minute suppression for greater accuracy in reading.

The activity level of the sump pump discharge is indicated and recorded at Spert IV control-center control room. The reactor is provided with a high radiation set point which energizes the Panalarm unit and both the control-center control room and the reactor building process panels.

8.4 Waste Cooling Water Monitor

The waste cooling water monitoring system consists of a basic control unit, detector, and recorder.

The basic control unit is a Riggs Nucleonics Model GA-3A and is located in Room 4 at the reactor building. The count rate meter and scale are three-decade log units with a range of 0.01 to 10 mr/hr.

The detector is an ionization chamber type for detecting γ radiation in a water stream at a level of 0.01 to 10 mr/hr. The detection chamber is housed in a 4- x 10-in. stainless-steel water-proof housing.

High radiation level is indicated in Room 4 and indicated and recorded at the Spert IV control center process panel.

8.5 Portal Monitors

There are two portal monitors installed at the Spert IV reactor building and each is composed of sensing elements and a console. One monitor is provided at the north entrance door and the other at the change room entrance door from Room 5 (reactor room).

Each monitor consists of eleven β, γ, geiger tubes located at the top, sides, and bottom of a door frame. The monitors have a sensitivity of 0.15 microcurie of γ and 0.45 microcurie of β on body surfaces.

The console is wall-mounted and houses eight meters. Seven meters for the top and sides of the door frames and one for the four counters below the door frame grating. Each meter has an individual high level alarm set point, zero and high voltage adjustment. The console sounds a local alarm when any of the preset alarm levels have been reached by any of the channels.

9. WARNING HORNS AND LIGHTS

Six warning lights and two warning horns are installed at the reactor building.

The warning lights are Crouse Hinds No. VGR 216 bracket-type weatherproof fixtures with 150-watt lamps, red globes, and guards. The lights are located as follows:

(a) East wall of Room 5 (operating floor).

(b) West wall of Room 5 (operating floor).

(c) East wall of Room 9 (basement).

(d) West wall of Room 9 (basement).

(e) Outside north entrance.

(f) Outside south entrance.

The lights are connected to a flasher unit located in Room 7 (process control room). The unit is Reynolds Electric Co. Model LBDS-4 with a flashing speed of 60 flashes/min.

One warning horn is installed on the south wall of the building adjacent to the roll-up door, and the other on light pole B 30 yds northwest of the building.

The interior horn is a Faraday No. 133 L and the exterior horn is Faraday No. 136, which is a weather-proof type horn. The two horns are connected in parallel and controlled through the relay panel.

The horns will sound and the lights will flash whenever a control rod is withdrawn from its seat switch or whenever the drives are withdrawn above the lower limit switch.

In addition the horn can be sounded by a push button located on the console at the Spert IV control center.

X. PLANT CASUALTY EVALUATION

The experimental programs for the Spert IV plant will include several types of tests with a variety of reactor cores. Some experiments will have a planned high probability of damage to the plant or equipment and, because of the exploratory nature of the program, there is an attendant non-negligible probability that this probing of reactor behavior may result in more extensive mechanical damage than anticipated. For both of these situations, the hazards report[1]* has treated the principal problems of protection of personnel engaged in the experiments and the protection of all other persons, including the public at large. The cases considered therein were treated in such a fashion that the results, which were found acceptable even under the extreme assumptions postulated, were independent of the particular form of the incident. That is, it was not an essential part of the argument that details of the postulated incidents be specified with regard to the consequences or likelihood of the failure of a particular component. While this maximum accident analysis covered the primary area of concern in hazards evaluation, it is also of importance to examine the possible consequences of various types of failures in the plant. Accordingly, this section discusses the consequences of failures in the principal plant systems. Since the presentation of system failures encompasses the failures of individual components, these generally will not be singled out for further examination, but in a few special cases individual component failures will be discussed as a part of the respective system analyses.

1. AIR SYSTEM (See Section IX)

One of the plant systems which can affect plant safety is the air system. The air system is used primarily to provide low pressure air for air-actuated instrumentation and diaphragm-operated valves. These valves include the primary flow control valves, the three-way valve on the sump pump discharge line and the dampers on the ventilating system. Two modes of gross system failure are of importance: loss of air pressure and air system overpressure. They are discussed below.

1.1 Loss of Air Pressure

Should air pressure be lost because of compressor failure, air line failure, or for any reason, the primary coolant system flow control valves will close. These are spring-loaded diaphragm-operated valves. The result of these valves closing would be a loss of coolant flow to the reactor core. The consequences of the loss of flow would be dependent upon the condition of the core under test at that time. For the initial core and the presently contemplated mode of operation, estimates [2] have been made that indicate that no hazard due to meltdown of fuel would exist. Loss of air pressure to 27 psig or below will alarm the "low air pressure" annunciator on the control center process panel. Should this occur while the reactor is running, the reactor automatically will be scrammed by the low air pressure scram interlock and procedures will call for manual shutdown of the pumps.

* Numbers in brackets refer to entries in Section XVI, References.

Deliberate operator action is required before operation of either the reactor or the pumps can be resumed after the air supply is re-established. Failure to shut off the pumps will cause the pumps to run in a "shut-in" condition which will, after a period of time, cause overheating of the pumps. Should the air loss be momentary and the pumps not be shut off, flow would be re-established at the original set point but the reactor would have been scrammed automatically prior to the re-establishment of flow.

The three-way valve on the sump pump discharge automatically will divert the sump water to the contaminated storage tank. Since the loss of air may be associated with other effects of planned experiments or accidents, the sump water may contain contamination and this action will prevent water from going directly to the leaching pond during an air failure.

The North (reactor pool) tank air-operated drain valve will close automatically on air failure to retain water in the pool.

The air-operated valve on the deionizer discharge line to the reactor pools will close, thus preventing the addition of water during air loss. In case of true emergency 150 gpm of raw water is available to the tanks. The valve on this line is manually operated from a location near the North pool. The availability of this valve hand wheel would be dependent upon the radiation level at the time.

Air pressure loss automatically would switch the ventiliation louvres to "emergency" conditions. This entails using 100%-fresh air through the steam heating coils. During severely cold weather, damage to the steam coils by freezing could occur. This can be prevented by turning off the blowers manually from control center or the reactor building.

Loss of air to the reactor pool sealing gate would permit leakage of water between the pools. Since the bottom of the gate is only 5 ft below the maximum water level, and since core designs allow cores to extend to within 6 ft of the top of the pool, no fuel would be uncovered. Thus the leakage would result at worst in high radiation levels because of the decrease in shielding thickness.

1.2 Air Overpressure

Air overpressure of significant magnitude to cause rupture of any part of the air system would result in air pressure loss with the consequences noted above.

1.3 Component Failure

The failure of individual components will lead to one of the two situations discussed above and the possible effects of such occurrences are encompassed by the above general discussion as indicated by the following examples of the considered incidents:

(a) Compressor failure would lead to low air pressure throughout the system and eventually to complete loss of air supply.

(b) Failure of valves, pipe, fittings, etc. would result in loss of the air system if the leak were in excess of the capabilities of the compressor.

(c) Simultaneous loss of pressure control on the compressor and failure of all relief valves to function would result in overpressure and rupture which would be an air loss to the system.

2. LOSS OF ELECTRIC POWER

A second system which must be analyzed from the standpoint and consequences of failure is the electric power distribution system. The two major distribution nets in this system include the reactor control power net and the regular building power net.

2.1 Reactor Control Power (See Section VII)

Control power can be lost either through loss of the Spert IV substation, by loss of the 480-volt transformer and/or building power breaker, or by the loss of individual components in the control system.

If control power is lost for any of these reasons, the control rods will scram automatically by magnet release. Upon return of the power, automatic rundown of the control rod drives will occur. Positive reactor operator action is required before the control rod magnets can be re-energized and the reactor operated.

In the event that control power is lost and building power is maintained during an upflow experiment, procedures will require an immediate manual shutdown of the coolant pumps. During the short period of time following reactor scram and the pump shutdown, the control rods are not mechanically restrained from moving. This does not present a safety hazard for the follwing reasons:

(a) The weight of the control rods is approximately 60 lb and the exposed cross section area of the rods is 1.25 sq. in. requiring a core ΔP in excess of 48 psi to cause movement of the rods. This ΔP is in excess of pump capacity and in excess of the core support capabilities.

(b) If the design of the initial control rods were to be changed, the above statement might not be valid. However, because of the presence of the rod drives, it still would be impossible for the rods to be withdrawn by flow to a position higher than their initial position before "scram".

2.2 Loss of Building Power (See Section II)

Other power failures which do not result in loss of reactor control power at the reactor building would have no effect on the reactor control system. However, loss of building power will cause loss of the reactor instrumentation power supplies and amplifiers at which time operating procedures require that the reactor be scrammed immediately. In order to determine that the reactor is safely shutdown re-entry into the building will be made using portable neutron instruments since the normal neutron-counting circuits would be inactive during power outage. Building power loss also would stop all auxiliary equipment except the deionizers. Manual reset of the motor starters at the reactor building is required for re-establishing coolant flow following a power failure. Since the reactor must be in a shut-down condition prior to re-entry into the reactor building, flow cannot be re-established while the control rod drives are raised.

Building power low voltage will cause the constant air monitor to alarm which in turn switches the plant ventilating system to "emergency" conditions. With low voltage all radiation monitoring devices will alarm. Under these conditions procedures require that the building be evacuated immediately, if occupied. Re-entry will be made according to standard procedures with Health Physicist escort and instrumentation.

Power loss will negate all Health Physics monitoring instruments and therefore procedures call for the building to be evacuated immediately.

Battery-powered emergency lights automatically provide sufficient illumination in all hazardous areas of the reactor building if the regular power fails.

3. LOSS OF POOL WATER

Loss of water from the reactor pools must be considered in terms of possible radiation hazards and damage to equipment. The events which could lead to a loss of water are ruptures of the glass ports in the North pool, in the pool walls, in coolant lines, or in the 2-in. drain lines in the North pool.

If any of these events occurred, under circumstances leading to the loss of as much as one-half the capacity of either pool, the pump area would be completely flooded to a height of 4 ft causing possible damage to all motors except the sump pump motor. The sump pump motor is located 10 ft above the bottom of the pump area.

If the volume of one pool were lost, the water in the basement would reach a level of 1-1/2 ft above the bottom of the pools; if the volume of both pools were lost, the water would reach a level of 4-1/2 ft above the bottom of the pools. Even in the latter case, the sump pump motor would not be submerged and use of the sump pump to reduce the water level would be permitted.

If both tanks were ruptured and the 4-1/2-ft level of water were achieved in the basement, there would be no net buoyant force on the tanks because the water would not drain below the 4-1/2-ft level of either tank. However, if the water from one tank were lost during a period when the other tank was empty and valved off, the water displaced by the tank (approximately 411 cu ft) would weigh approximately 29,390 lb. The weight of the tank without appurtenances or piping is approximately 27,800 lb resulting in a net buoyant force of 1590 lb. Since each tank is fastened to the foundation by means of the equipment hold-down bolts and is further weighted with piping and other equipment, the net buoyant force is not sufficient to cause any movement or property damage. A flooding alarm is located in the pump pit and will actuate a Panalarm unit in the control center if the water level in the pump pit reaches 1 ft in depth. This alarm will give warning of any flooding problem.

Another problem presented by loss of coolant would be core exposure. This condition would exist if the window port 5 ft from the pool bottom broke, if the North pool ruptured below this 5-ft window port, or if the coolant line through which upward flow is directed ruptured at the North pool. The present core height is 7 ft; in the event the 7-ft or 9-ft window broke, the core would still be covered with water. Loss of core moderator-coolant poses no special hazard or probability of extensive damage to either the building or the plant equipment. However, it could entail special working conditions to effect repair and cleanup, depending on the water contamination, if any, and the condition of the specific core installed in the reactor.

If the 2-in. drain pipe in the North pool leading to the sump or cleanup pump broke, approximately 48 hr would be required to drain the pool. Since the security guard inspects the plant twice in each 8-hr shift, even if this occurred on a non-working shift, not much flooding would occur before its discovery. Removal of the water from such flooding would be accomplished from the control center sump pump control. No significant plant damage would occur from flooding.

The greatest personnel hazard exists if the glass viewing port should break while personnel are directly in front of the window as they would be while setting up photographic equipment. Since the windows are glass, breakage could occur accidentally. Calculations show that in an idealized situation the water level in the North tank would drop to the 5-ft level in approximately 1 min. The windows are protected by slide-gate valves operated from the main floor. Manual operation of the slide gate from full open to full closed requires slightly over 1 min as determined by actual test. It thus does not appear feasible to attempt closing the slide gate when and if a window is broken with people in the building and a hot core in the reactor tank. In these conditions the building will be evacuated immediately and re-entry made only after the condition of the plant has been determined. To reduce hazard, the windows are protected by the gate valves at all times except during the set-up of an experiment or during a nuclear test. During a nuclear test no personnel will be present in the reactor building and re-entry will be made in the normal manner with a Health Physicist escort. The viewing ports are currently being equipped with mechanical gate actuators which will close the gates in less than 10 sec. These actuators can be used for opening or closing the gates from either the control center or the reactor building. No use will be made of the viewing ports until the actuator installation is complete.

4. LOSS OF HEATING AND VENTILATING

Since the heat loss from the building will be slow, and auxiliary portable heating units are available from the Central Facilities stores, no hazard of post-incident damage should occur as a result of the failure of the heating and ventilating system. Explosion of the boiler probably would damage both the ventilating fans and the deionizing equipment located in the same room. Since this equipment is located in a wing building, damage to the main reactor building would not occur.

5. FIRE

The only major source of fire hazard is from electrical equipment fires or malfunction of the boiler "fire-eye" resulting in a fire box explosion.

Fire protection is furnished by local appropriate fire extinguishers and the AEC fire department located 6.6 mi distant. An ADT alarm system is used to manually report fires in the Spert area. An automatic ADT alarm is actuated by a heat sensing device in the main reactor building ceiling. Since the boiler is located in a separate room in the wing building and is separated from the main building by a non-combustible 12-in. pumice-block wall, fires originating in the boiler room would be effectively delayed from spreading to the rest of the reactor building.

Protection from fire in the main building structure, which is erected of 12-in. pumice block with bare structural steel, is accomplished by limiting the storage of combustibles in the main reactor building by strict administrative control.

6. EARTHQUAKES

AEC design criteria for construction at the NRTS specifies that the NRTS is in zone 2 (moderate damage) in an anticipated damage zone scale ranging from 0 (resulting in no damage) to 3 (resulting in severe damage). Although minor damage is possible at the NRTS to zone 2 buildings, the August, 1959, earthquake north of West Yellowstone, Montana, caused no known physical damage at the NRTS even though the intensity of this earthquake was rated at "VII" on the modified Mercallis scale.

Some disturbance of instrumentation might be expected and possible fluctuations in water levels are likely. However none of these would result in any serious plant problems.

XI. PLANT DRAWINGS AND PHOTOGRAPHS

The reactor and reactor facilities as they normally would appear to a person entering any of the reactor entrance doors or at normal stopping places for a plant tour of inspection are shown by the photographs in this section. The location of all reproducible drawings pertaining to the Spert IV facility also is specified.

1. PLANT PHOTOGRAPHS

Figure 43 - General view of the north and west exterior of the reactor building.

Figure 44 - View of the core structure and pools taken from the building crane T. V. location.

Figure 45 - View toward North pool from first landing on stairway to basement.

Figure 46 - View directly down the basement stairway taken from first landing.

Figure 47 - View directly down the basement stairway taken from dry dock landing.

Figure 48 - View from dry dock landing looking south.

Figure 49 - View from dry dock landing looking at North tank glass viewing windows.

Figure 50 - View from pool floor level looking into pump pit.

Figure 51 - View of pumps and piping taken from pool floor level landing.

Figure 52 - View of demineralizers as seen from outside door.

Figure 53 - View of demineralizers and regeneration equipment as seen from just inside the utility room exterior door.

Figure 54 - View of water softeners and ventilating system plenum taken from just inside the utility room exterior door.

Figure 55 - View of boiler and ventilating system plenum as seen from the utility room exterior door.

Figure 56-57 - General view of reactor building main floor as seen from north personnel entrance door.

Figure 58-59 - General view of reactor building main floor as seen from south personnel entrance door.

Figure 60 - General view of reactor building main floor as seen from change room door.

Figure 61 - View of reactor pools from utility room door, main floor level.

Figure 62 - View of overhead crane and roof as seen from south personnel entrance door at main floor level.

Figure 63 - View of overhead crane and roof as seen from mechanical assembly room at main floor level.

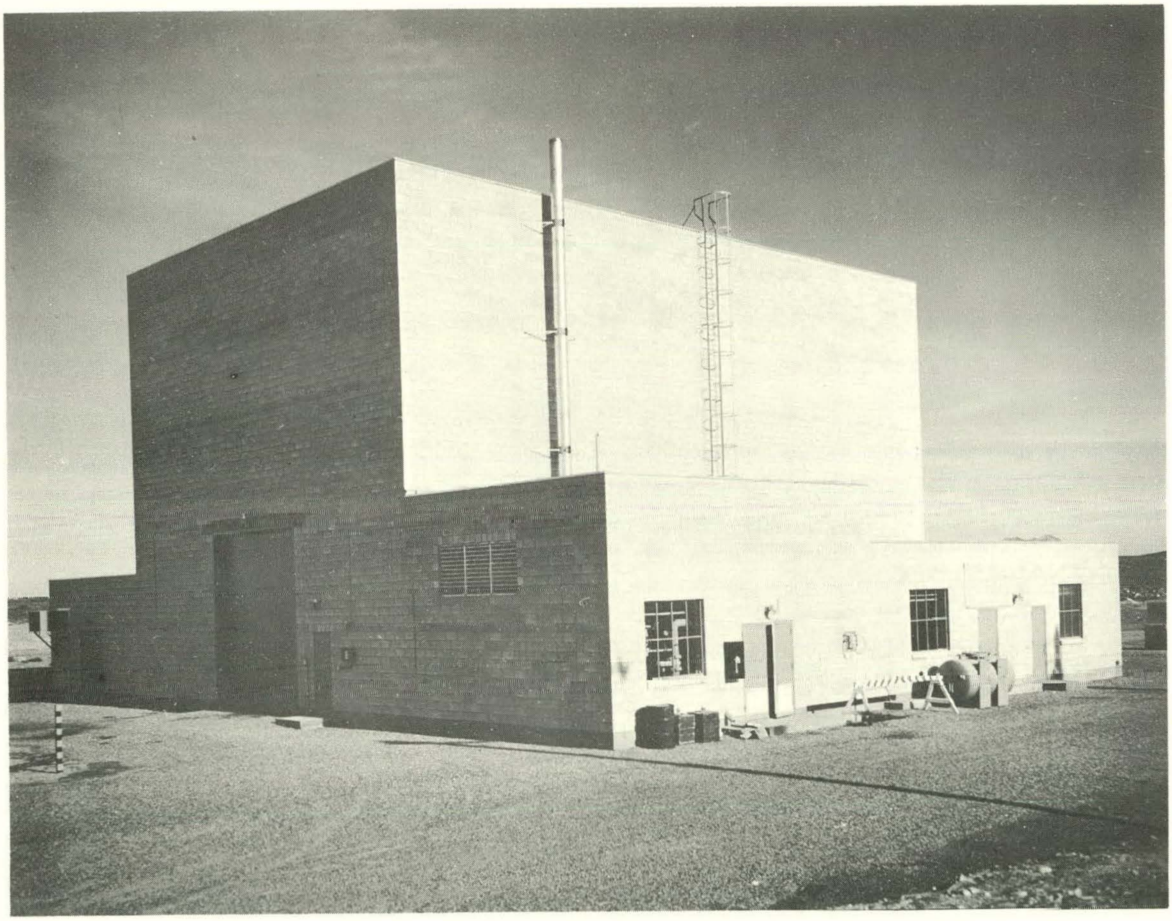


Fig. 43 Photograph of north and west exterior of reactor building (NRTS-61-6381).

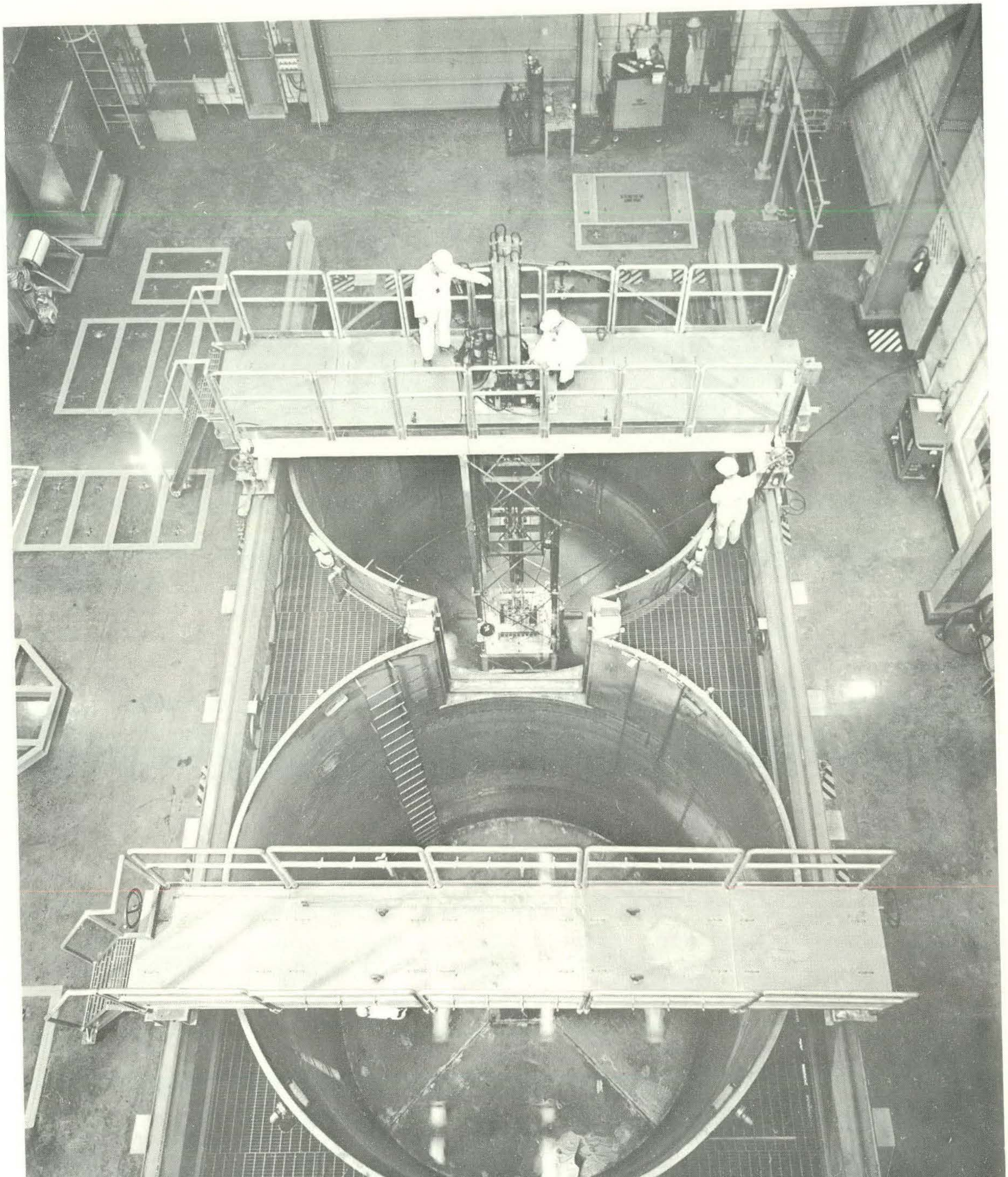


Fig. 44 Photograph of pools and core structure (NRTS-61-6898).

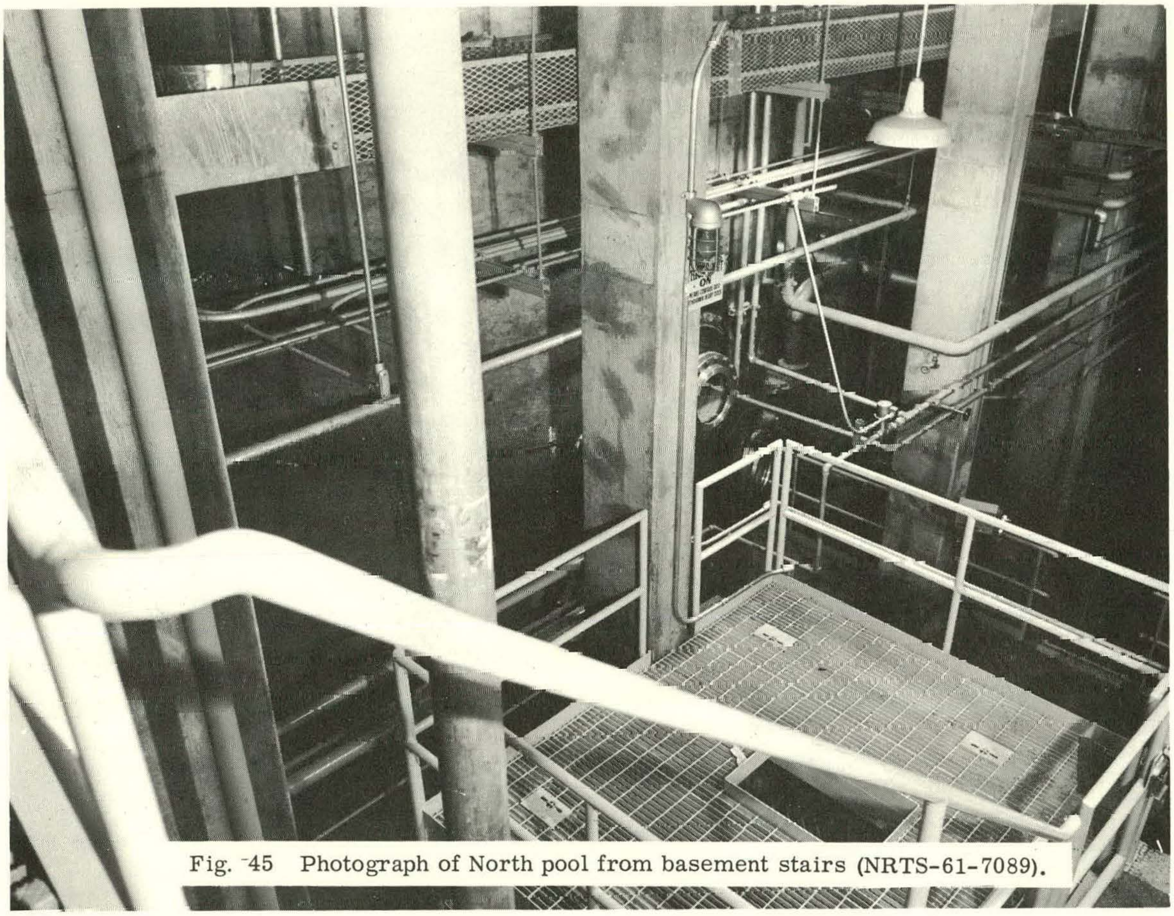


Fig. 45 Photograph of North pool from basement stairs (NRTS-61-7089).

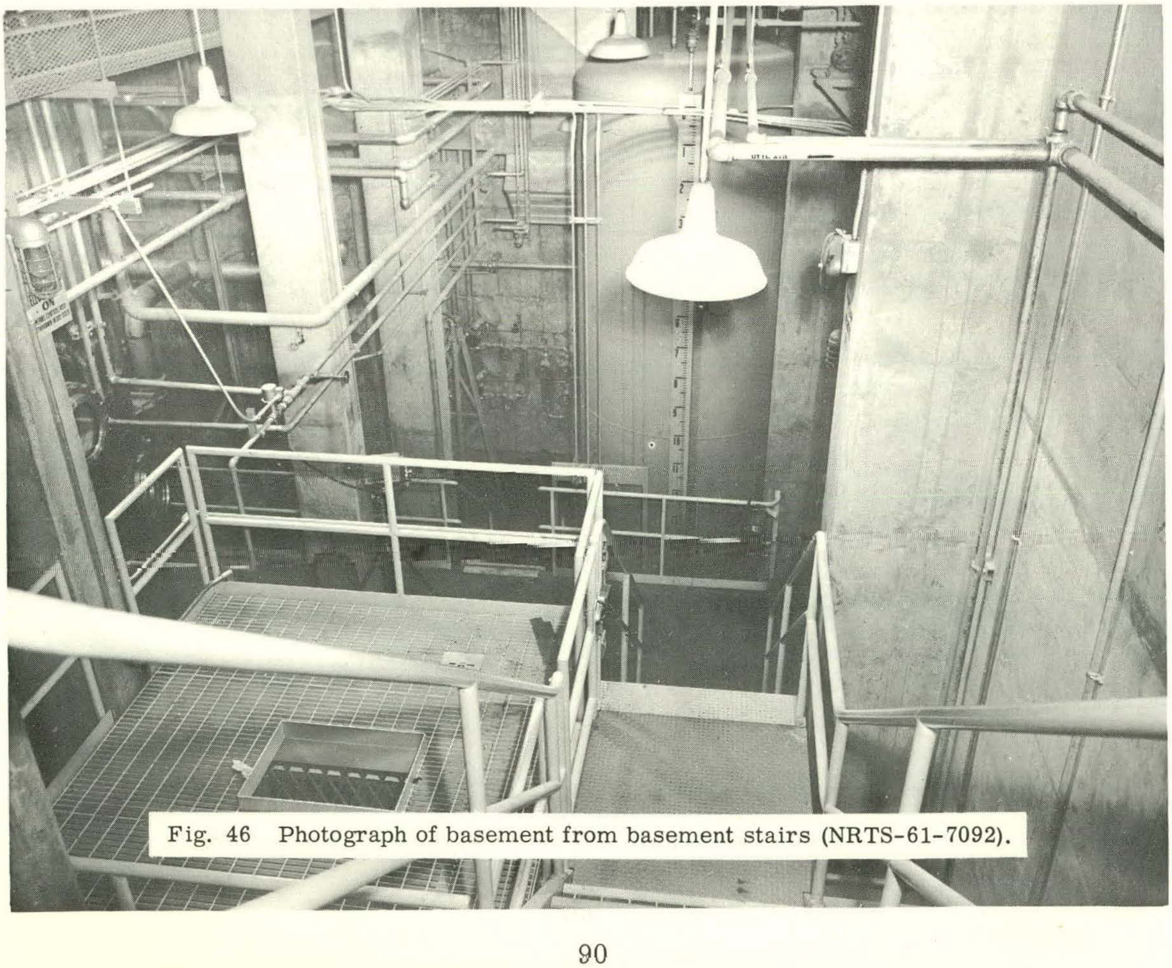


Fig. 46 Photograph of basement from basement stairs (NRTS-61-7092).

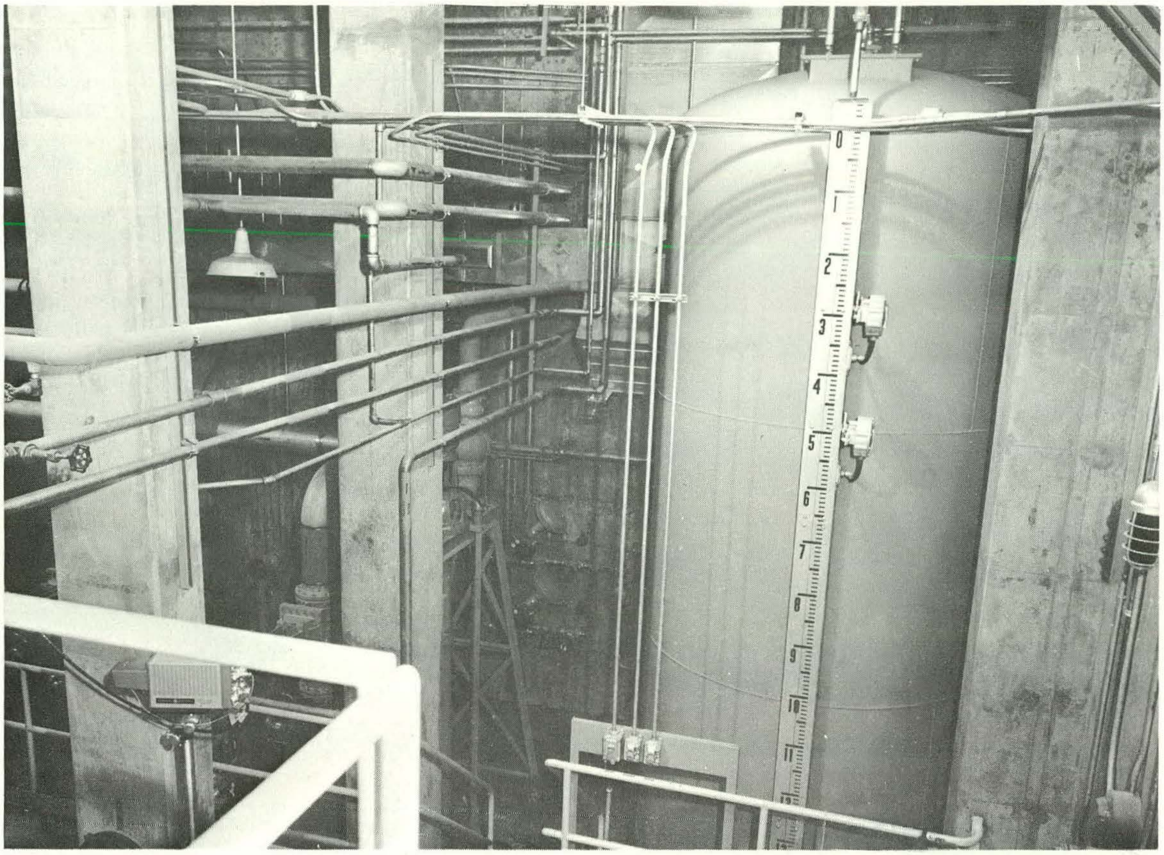


Fig. 47 Photograph of basement from dry-dock landing looking down stairway (NRTS-61-7090).

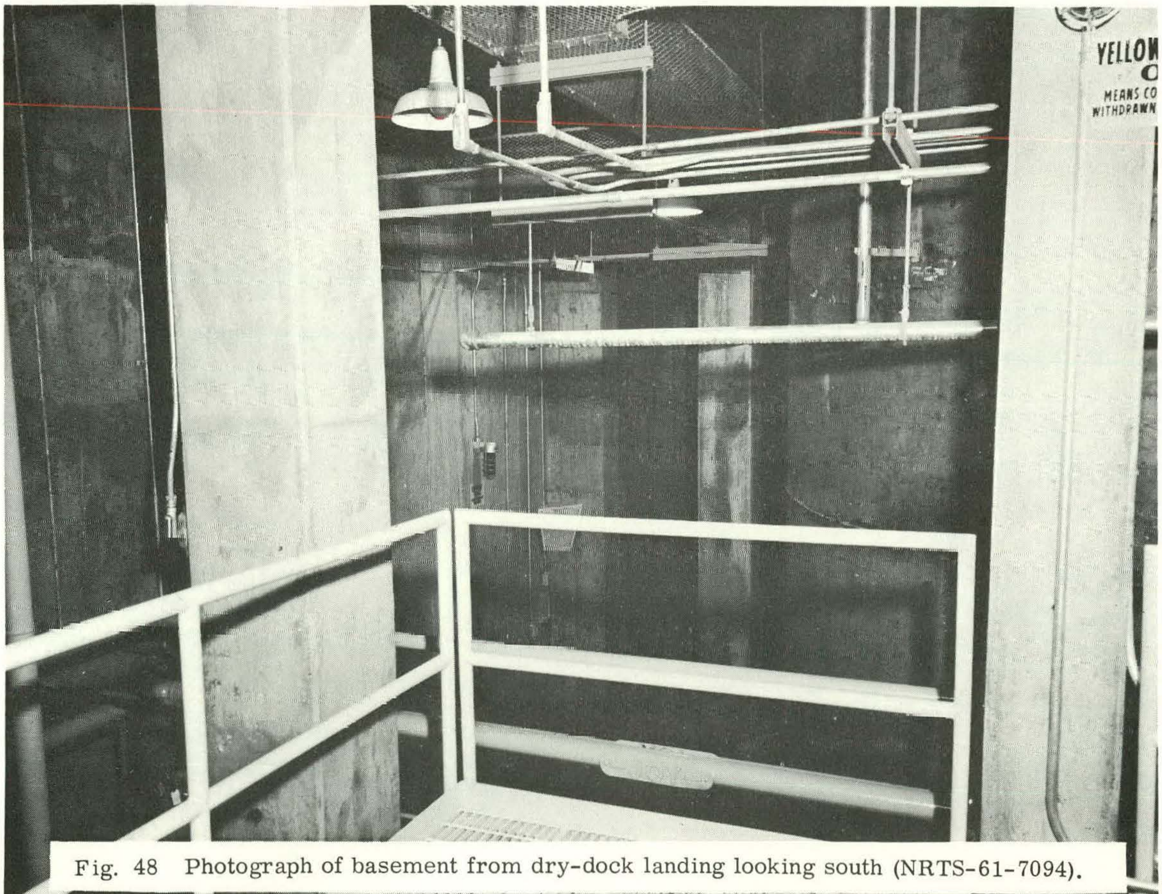


Fig. 48 Photograph of basement from dry-dock landing looking south (NRTS-61-7094).

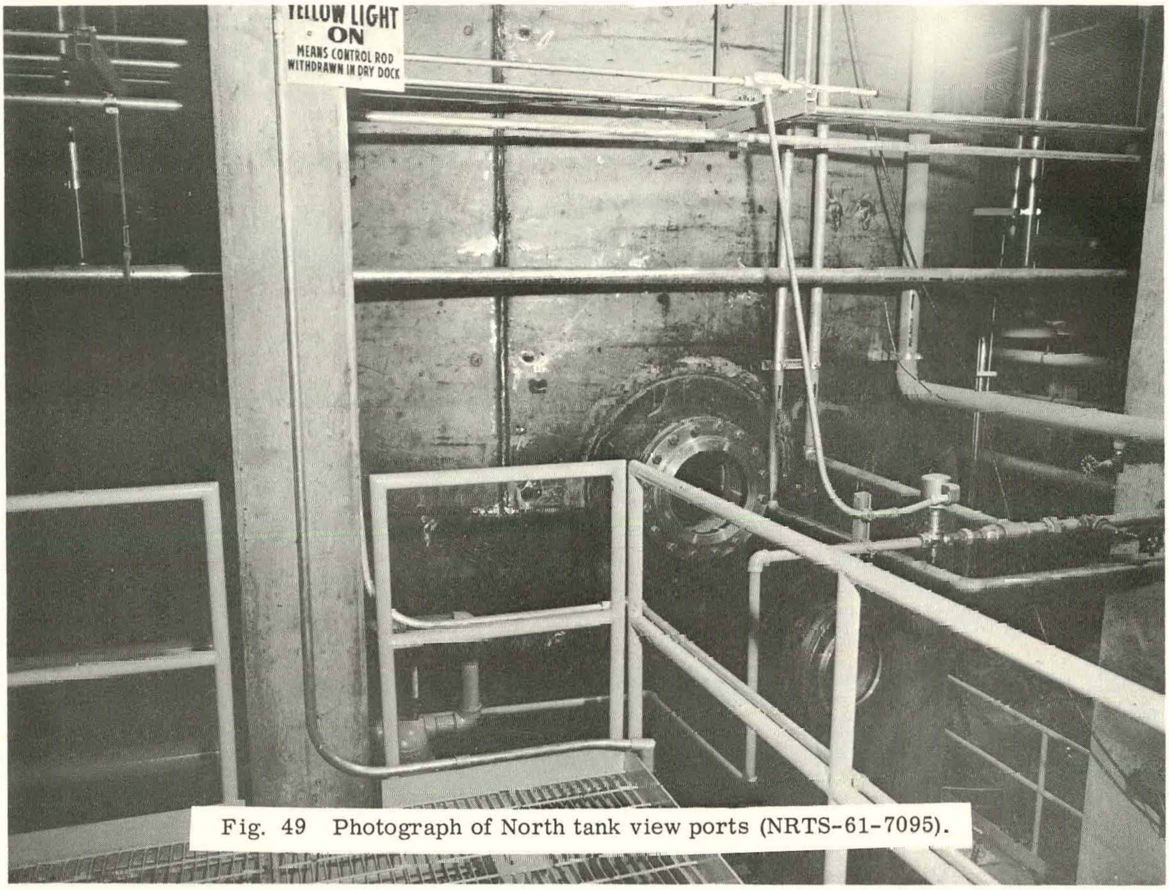


Fig. 49 Photograph of North tank view ports (NRTS-61-7095).

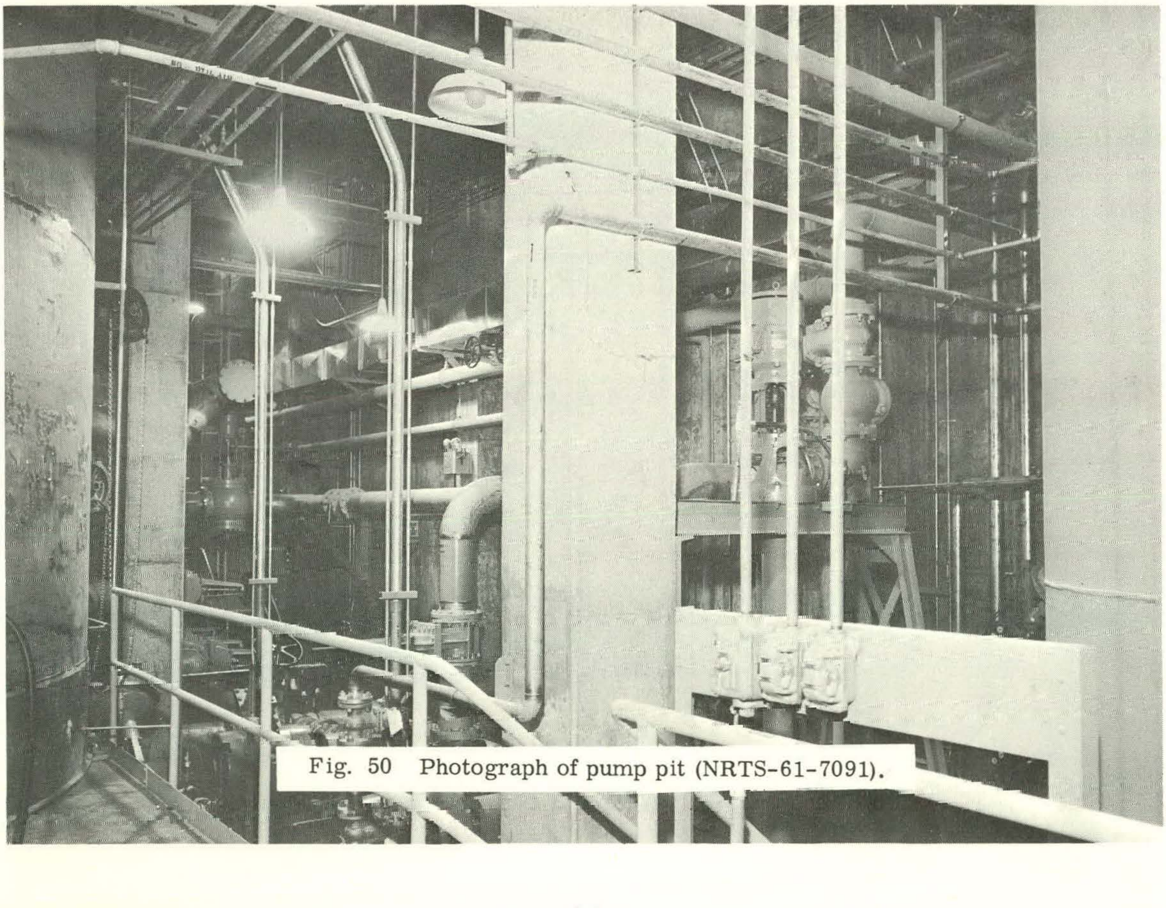


Fig. 50 Photograph of pump pit (NRTS-61-7091).

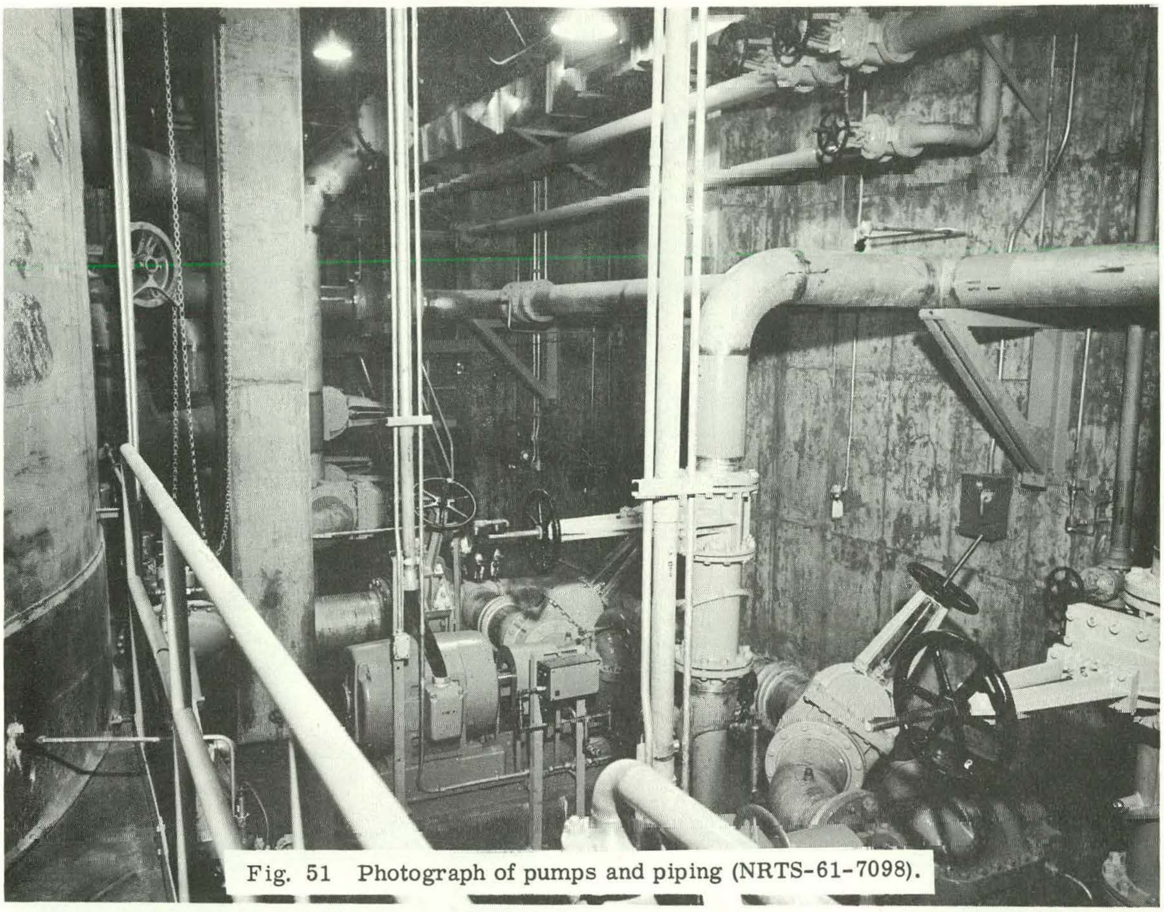


Fig. 51 Photograph of pumps and piping (NRTS-61-7098).

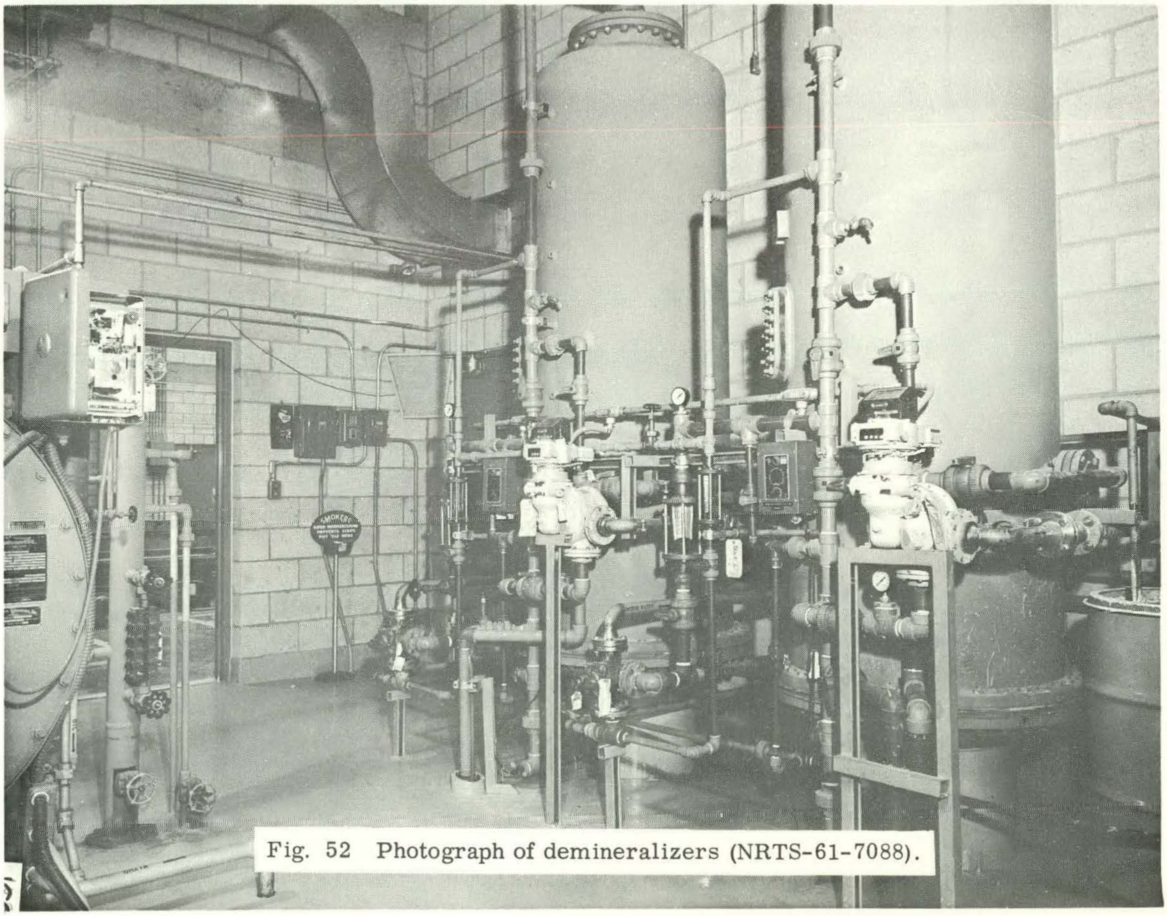


Fig. 52 Photograph of demineralizers (NRTS-61-7088).

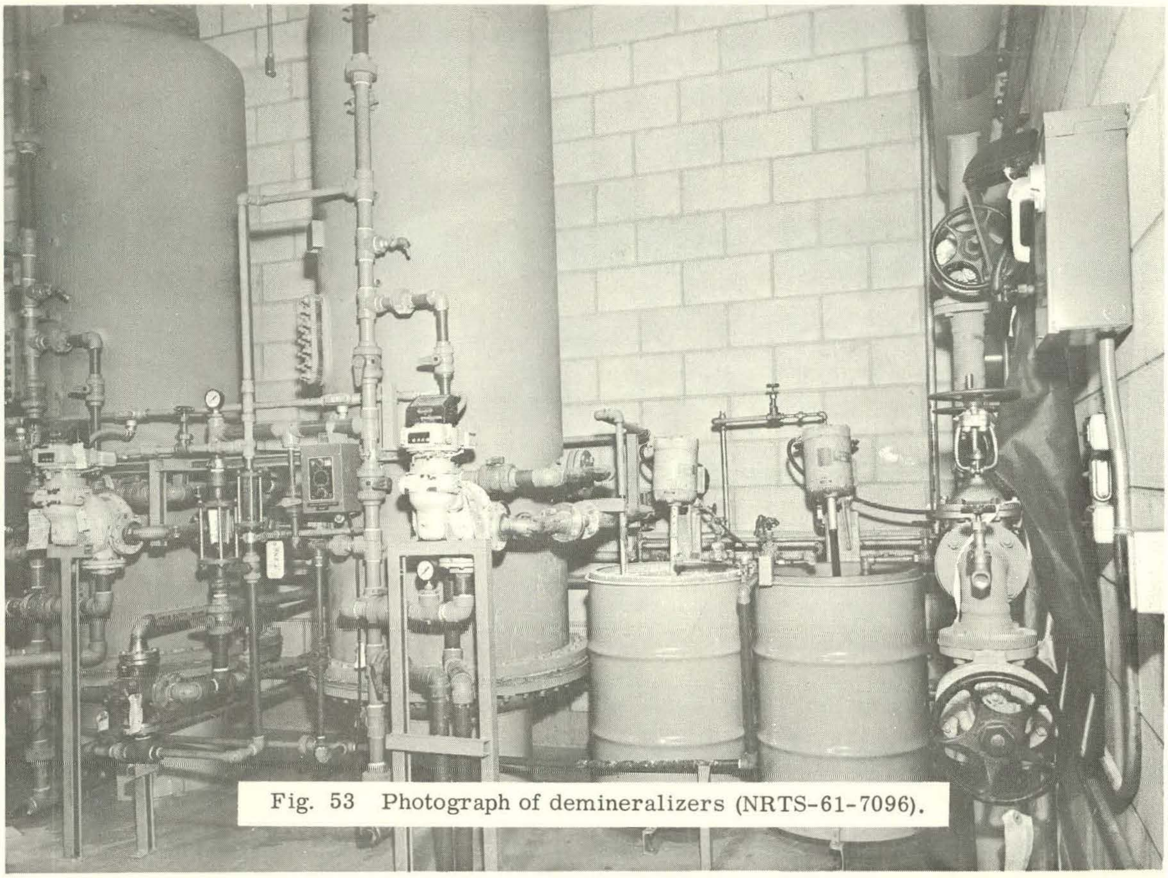


Fig. 53 Photograph of demineralizers (NRTS-61-7096).

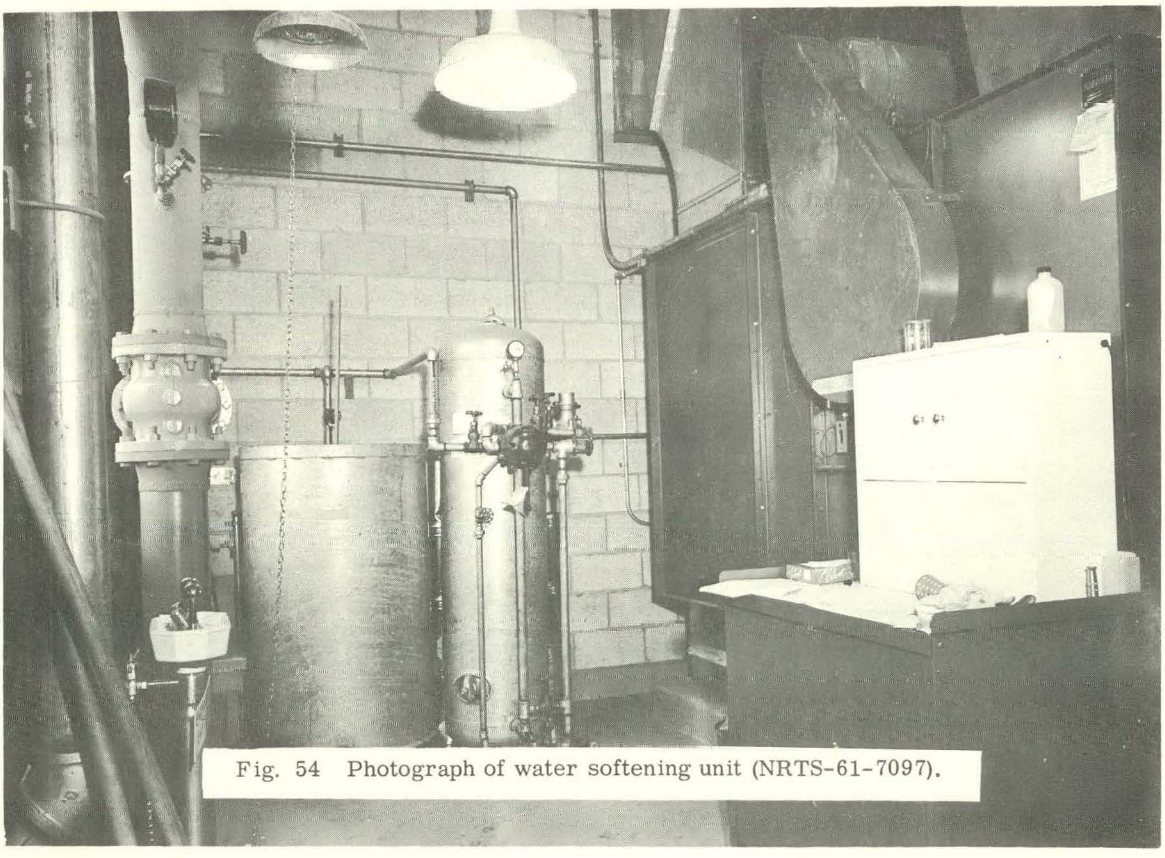


Fig. 54 Photograph of water softening unit (NRTS-61-7097).

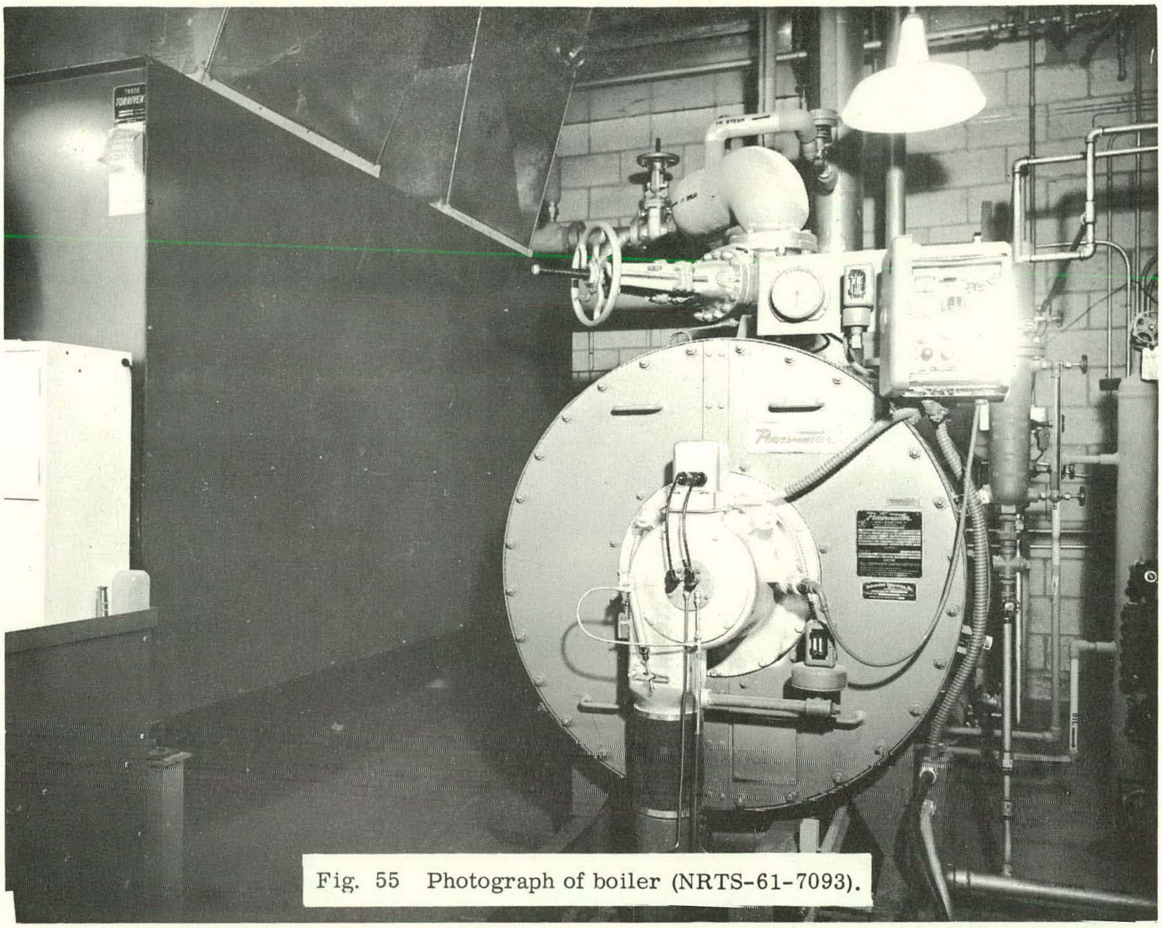


Fig. 55 Photograph of boiler (NRTS-61-7093).

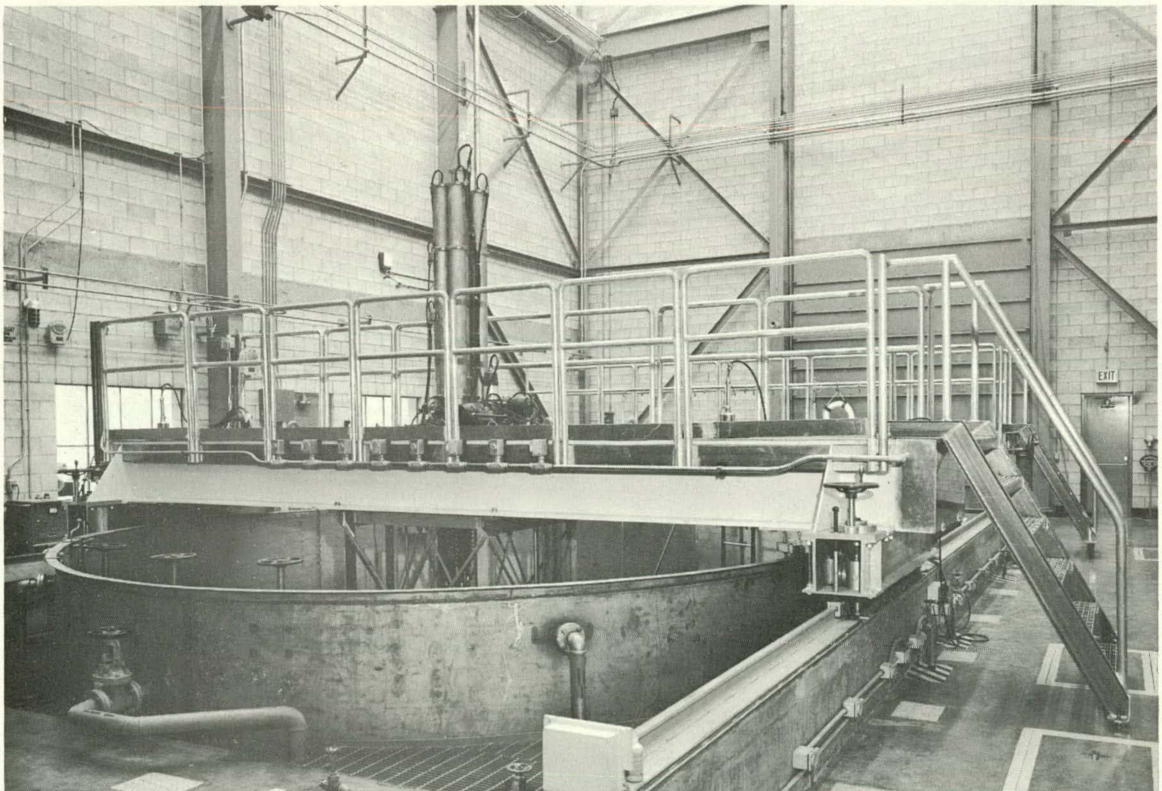


Fig. 56 Photograph of reactor building main floor from north personnel entrance door (NRTS-61-7196).

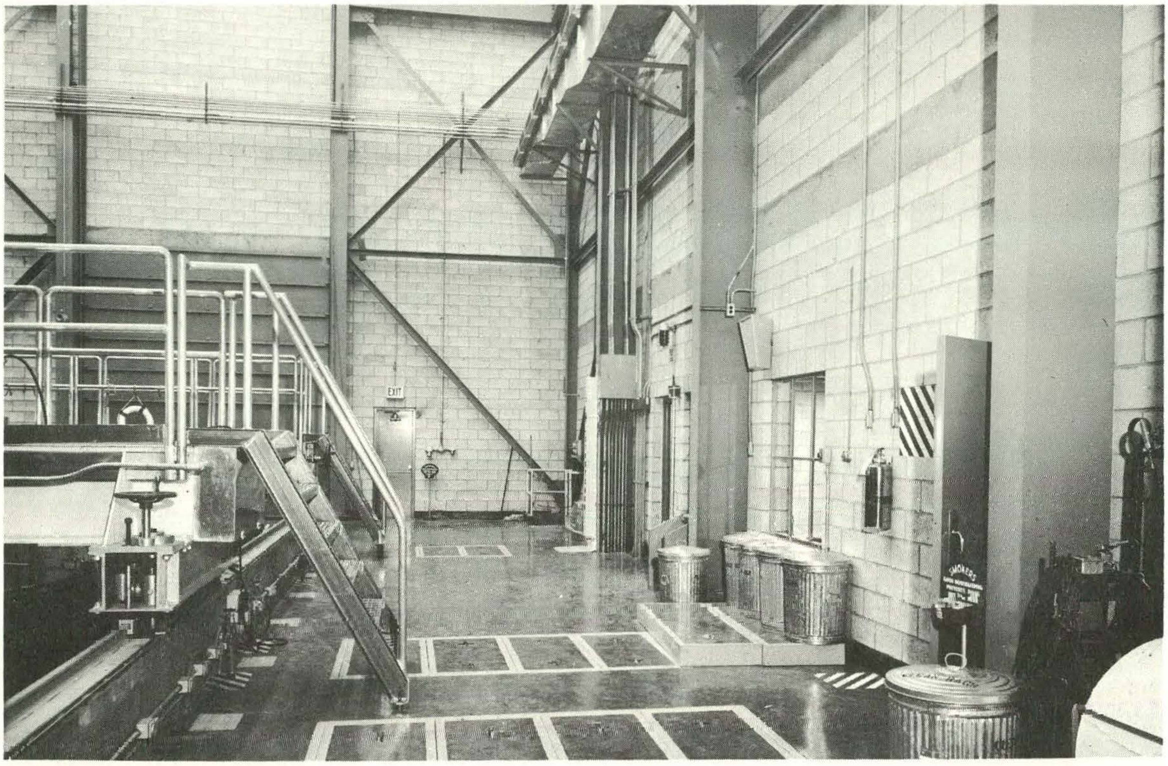


Fig. 57 Photograph of reactor building main floor from north personnel entrance door (NRTS-61-7198).

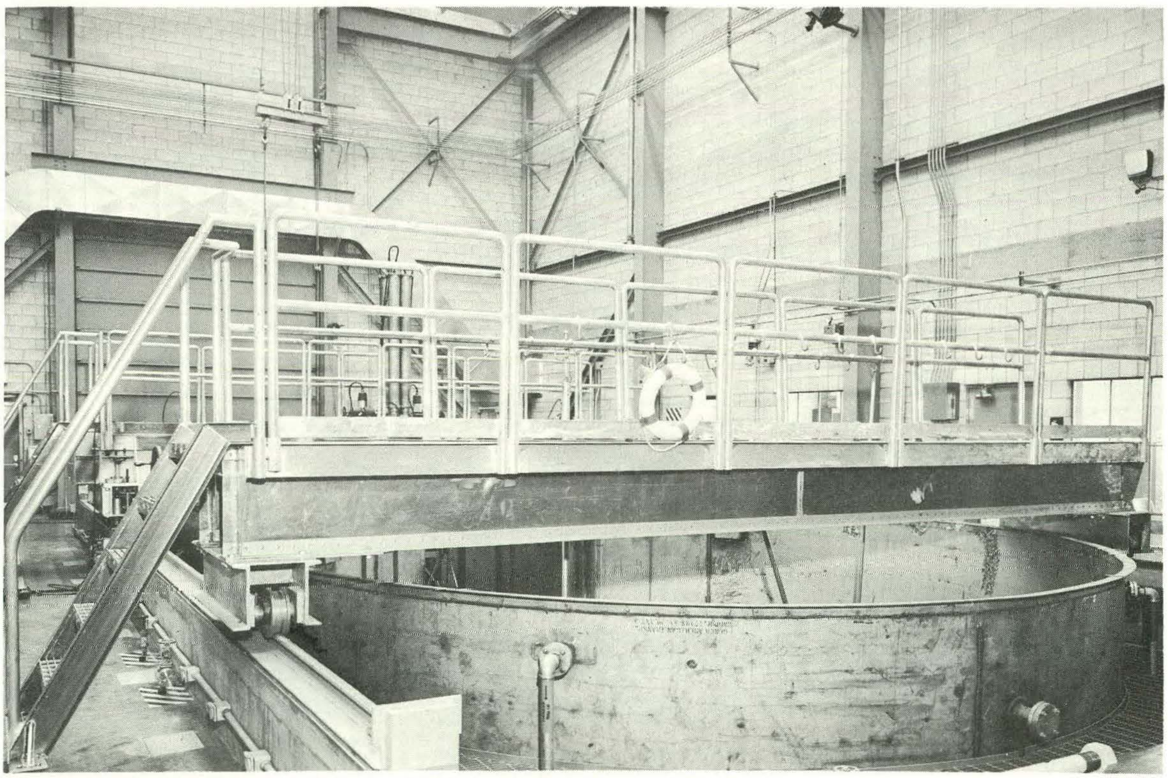


Fig. 58 Photograph of reactor building main floor from south personnel entrance door (NRTS-61-7195).

Fig. 59 Photograph of reactor building main floor from south personnel entrance door (NRTS-61-7194).

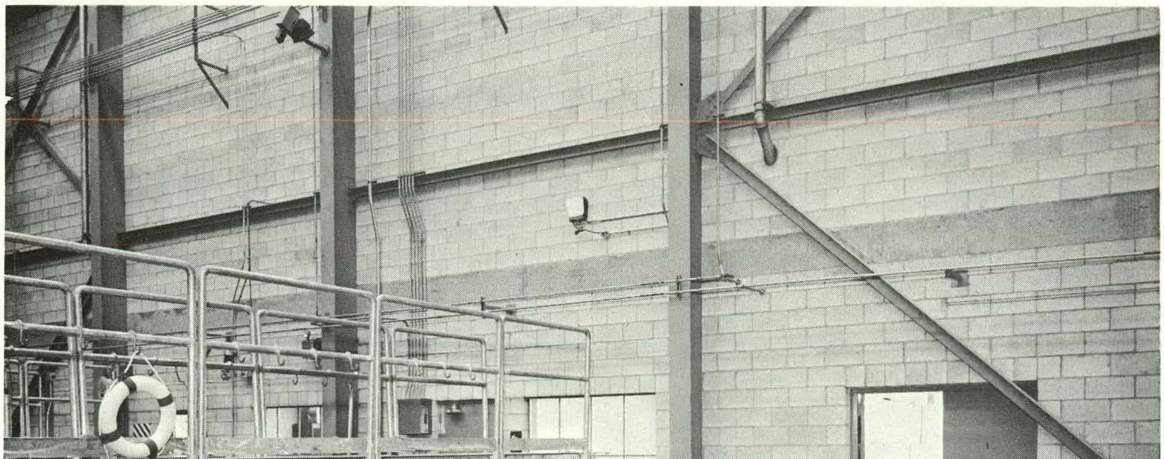


Fig. 60 Photograph of reactor building main floor from change room door (NRTS-61-7199).

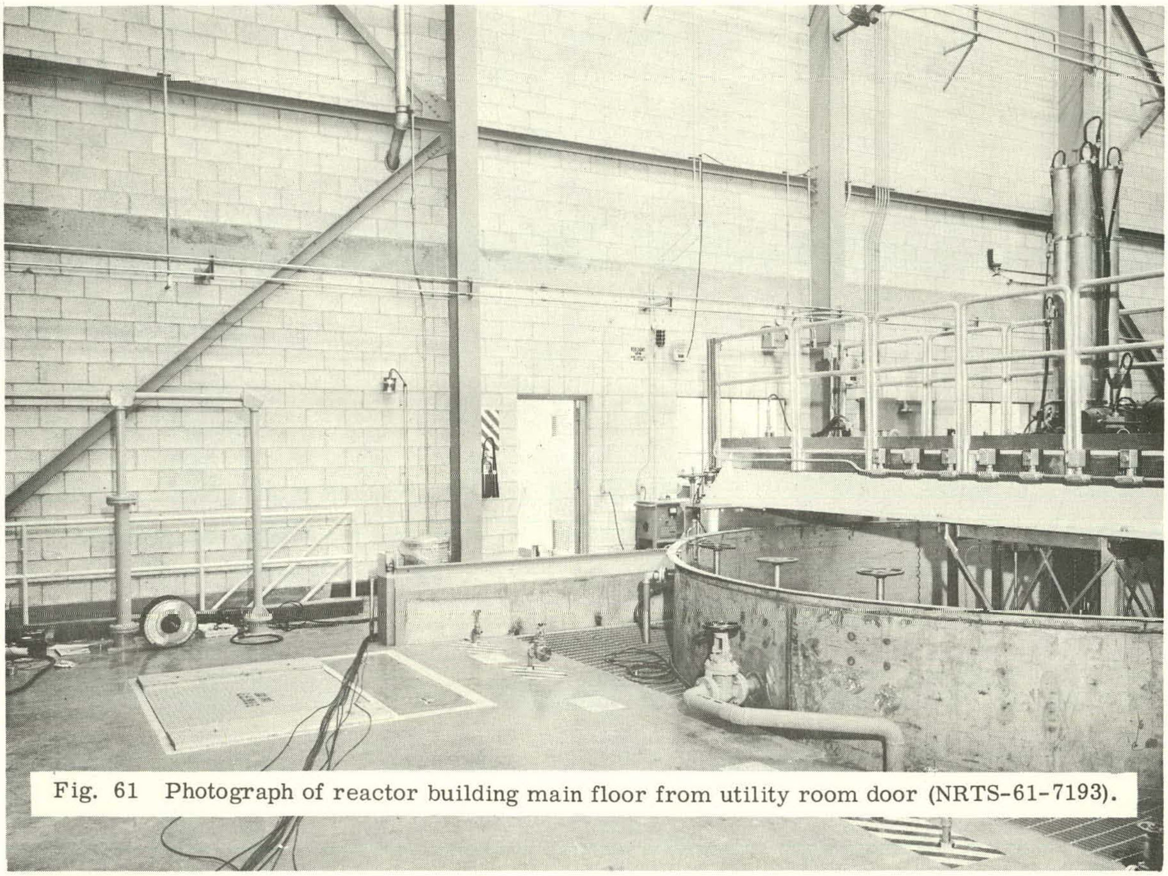


Fig. 61 Photograph of reactor building main floor from utility room door (NRTS-61-7193).

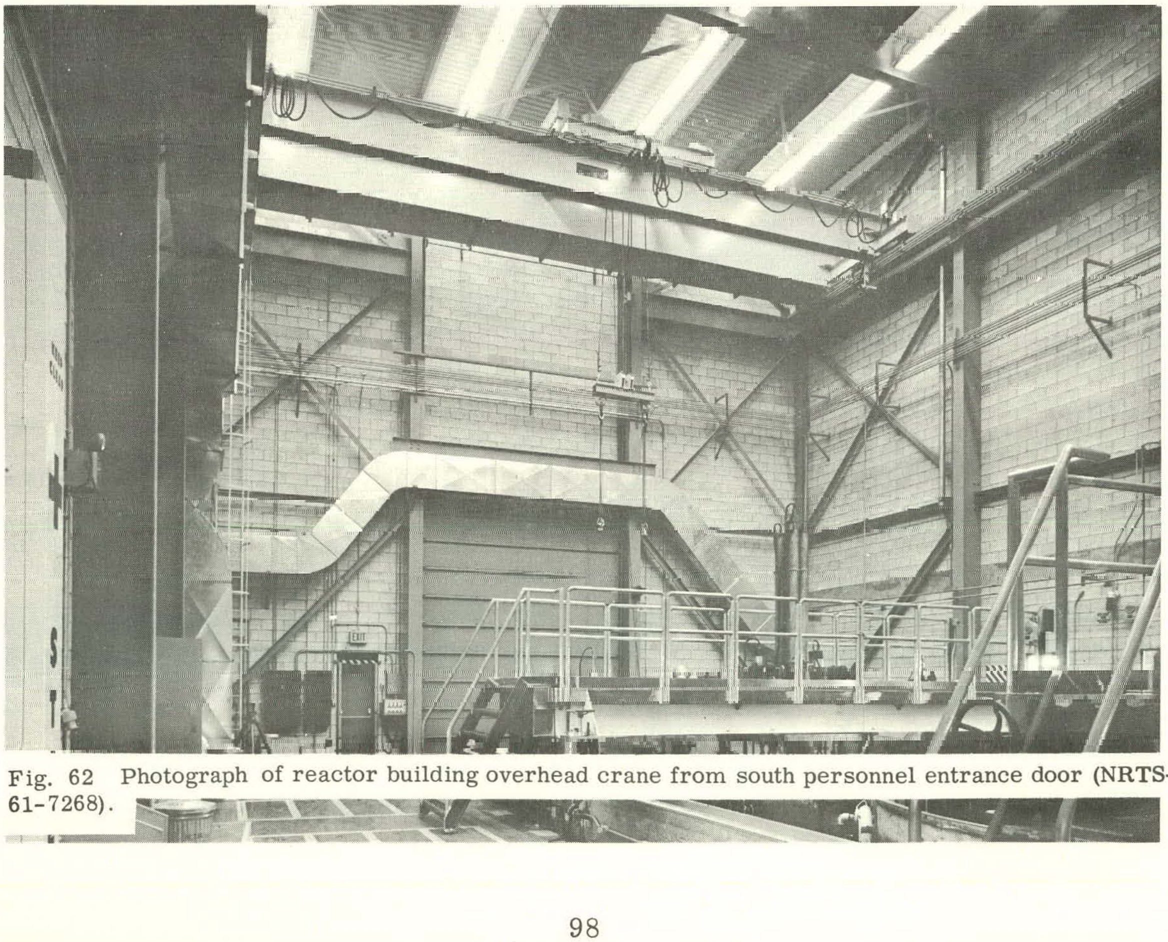


Fig. 62 Photograph of reactor building overhead crane from south personnel entrance door (NRTS-61-7268).

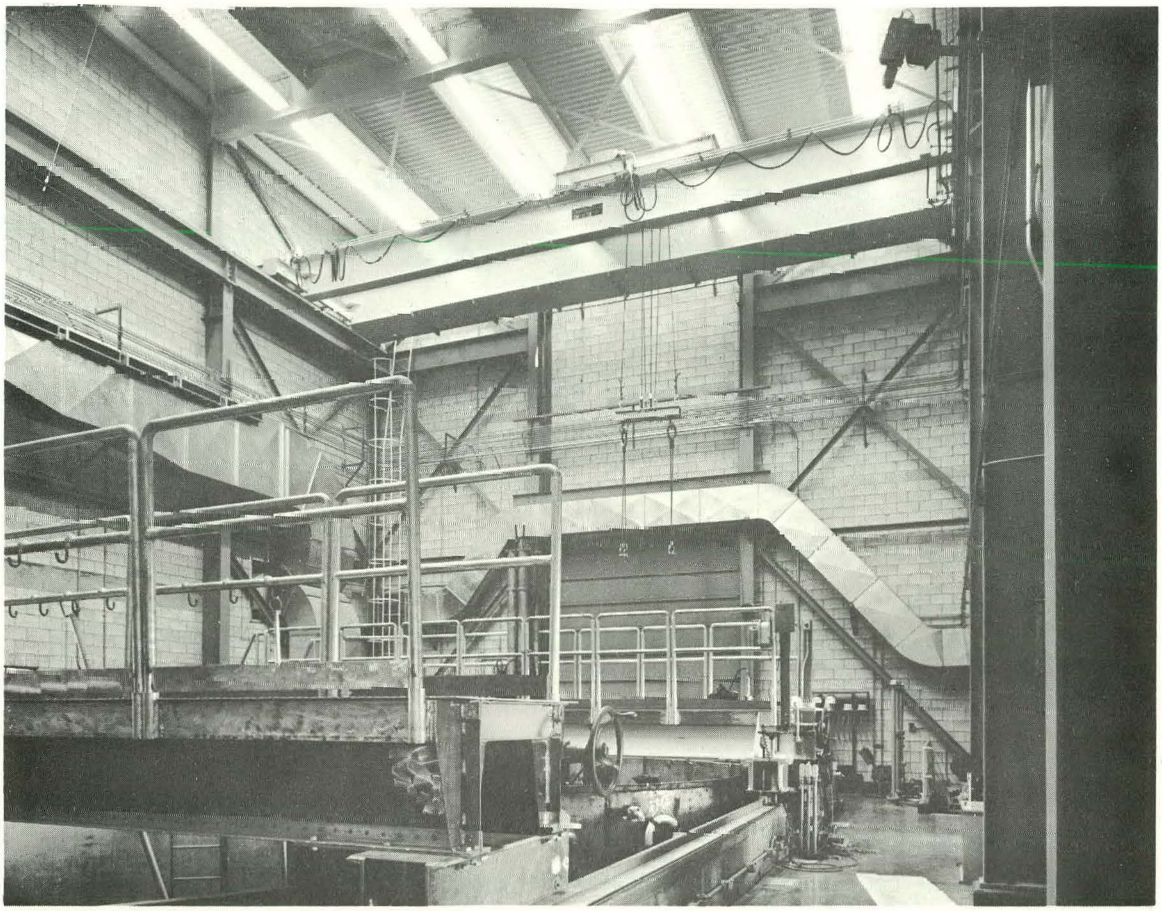


Fig. 63 Photograph of reactor building overhead crane from mechanical assembly room (NRTS-61-7267).

2. PLANT DRAWINGS

Reproducible copies of all drawings furnished by the B. D. Bohna and Co. are kept in the Idaho Falls drafting section of Phillips Petroleum's division engineering branch. Prints of these drawings are available at the Spert control center and in the reactor building.

Original drawings of Phillips Petroleum drawings pertaining to Spert IV, ie, control rod drives, core structure, fuel, handling tool, etc., are retained at the Central Facilities offices of the Spert design section of Phillips Petroleum's division engineering branch. Reproducibles of these originals are also retained 5 miles from Central Facilities at the MTR design and drafting section of division engineering to preclude loss of originals should fire or other disaster endanger either the MTR or Central Facilities.

Prints of the Phillips Petroleum Spert IV drawings are available at both the reactor building and at the Spert control center.

Equipment drawings and other vendor literature are retained at the Spert control center for a master file and working copies are kept in the office of the reactor building.

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XII. APPENDIX A
DESIGN DATA SUMMARY

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DESIGN DATA SUMMARY

1. <u>Tanks, Pools, and Vessels</u>

1.1 <u>Reactor Pool and Storage Pool</u>

Construction: welded, rolled, type 304 stainless-steel plate

Inside diameter: 20 ft

Height: 25 ft

Wall thickness, top 23 ft: 5/16 in.

Wall thickness, bottom 2 ft: 5/8 in.

Bottom thickness: 1/2 in.

Design pressure

Below gate: 25-ft hydrostatic load plus 50-psi static surcharge

Above gate: 6-ft hydrostatic load plus 10-psi static surcharge

1.2 <u>Deionized Water Storage Tank</u>

Construction: carbon steel, PVC-lined, ASME, dished and flanged heads

Capacity: 6000 gal

Outside diameter: 7 ft, 6 in.

Over-all length: 18 ft, 2 in.

Wall thickness: 3/16 in.

1.3 <u>Contaminated Waste Hold-up Tank</u>

Design

Standard: American Water Works Assoc. Standard D100-59

Seismic loading: Zone 2

Snow load: 30 lb/ft^2

Wind load: 30 lb/ft^2

Construction: welded carbon steel

Inside diameter: 26 ft

Height, straight section: 16 ft, 8 in.

Capacity: ~ 61,000 gal

Tank heater capacity: 15 kw

1.4 <u>Air Receiver</u>

Design: ASME

Design pressure: 125 psig

Design temperature: 450°F (max)

Dimensions

Outside diameter: 3 ft, 6 in.

Over-all height: 10 ft

Wall thickness: 5/16 in.

Volume: 110 ft^3

1.5 Air Dryers

Design: ASME

Number: 2

Design pressure: 165 psi

Design temperature (max): 500°F

Shell thickness: 0.322 in.

Head

Type: round and flat

Thickness: 0.75 in.

Capacity/tower: 30 SCFM

2. Pumps

2.1 Primary Coolant

Type: single stage, double suction, horizontal, split volute type centrifugal pump

Number: 2

Capacity: ea 2500 gpm at 170-ft hd

Motors

Rating: 125 hp

Type: 480 volt, 3Ø, 60 cycle

rpm: 1800

2.2 Sump Pump

Type: vertical, single stage, turbine

Number: 1

Capacity: 300 gpm at 70-ft hd

Motors

Rating: 75 hp

Type: 480 volt, 3Ø, 60 cycle

rpm: 3450

2.3 Clean-up Pump

Type: single stage, horizontal, centrifugal

Number: 1

Capacity: 50 gpm at 140-ft hd

Motor

Rating: 5 hp

Type: 480 volt, 3ϕ, 60 cycle

rpm: 3450

2.4 Fill Pump

Type: single stage, horizontal, centrifugal

Number: 1

Capacity: 50 gpm at 50-ft hd

Motor

Rating: 1-1/2 hp

Type: 480 volt, 3ϕ, 60 cycle

rpm: 3450

2.5 Condensate Pump

Type: single stage, horizontal, centrifugal

Number: 1

Capacity: 7.9 gpm at 15-ft hd

Motor

Rating: 1/4 hp

Type: 115 volt, 1ϕ, 60 cycle

rpm: 1750

2.6 Chemical Addition Pump

Type: reciprocating, positive displacement, controlled volume

Number: 1

Capacity: 1.2 gpm (max) at 600 psig (max)

Motor

Rating: 1/3 hp

Type: 115 volt, 1ϕ, 60 cycle

rpm: 1725

2.7 Fuel Oil Booster Pump

Type: rotary, positive displacement

Number: 1

Capacity: 1-1/2 gpm

Motor

Rating: 1/4 hp

Type: 120 volt, 1ϕ,60 cycle

rpm: 1200

3. Piping

3.1 Deionized Water

Type: seamless and welded as per ASTM A-120

Wall thickness

1/2 in. to 1 in.: schedule 40

1-1/4 in. to 12 in.: schedule 10

16 in.: schedule 20

Material: stainless steel type 304

3.2 Raw water, Fuel Oil, Air, Acid, Steam and Condensate

Type: standard weight black iron and galvanized as per ASTM A-120

3.3 Acid Waste

Type: glazed, vitrified clay with bell and spigot ends, and Saran-lined steel

3.4 Sanitary Waste

Kind of piping: 4-in. concrete with cemented bell and spigot joints

4. Auxiliary Equipment

4.1 Water Treatment System

Softener

Type: polystyrene resin

Capacity/cycle: 10,000 gal

Flow rate (max): 20 gpm

Deionizer

Number: 2

Type: mixed bed

Max. flow rate/unit: 50 gpm

Effluent water conductivity: $<2 \mu$mho/cm

4.2 Compressed Air System

Compressor

Type: stationary, single stage, double acting, water cooled

Bore: 7 in.

Stroke: 5 in.

Piston ring assembly: carbon

Capacity: 59.5 SCFM of 125-psig air at 5000 ft and 100°F

4.3 Heating and Ventilating System

Boiler

Horsepower: 50

Btu/hr gross output: 167 mbh

EDR steam radiation gross: 6,970 sq ft

Steam output at 212°F: 1,725 lb/hr

Heating surface fireside: 252 ft^2

Heating surface water side: 272 ft^2

Passes: 3

Type burner: pressure atomizing fuel oil burner

Draft: mechanical forced draft

Oil consumption: 15 gal/hr No. 2 fuel oil

Guaranteed fuel to steam efficiency: 80%

XIII. APPENDIX B

REACTOR COMPONENTS (TYPE D CORE)

XIII. APPENDIX B
REACTOR COMPONENTS (TYPE "D" CORE)

The initial core in the Spert IV facility is an aluminum "loose leaf" plate-type core and will be used initially for the study of the instability phenomena first noted in the Spert I facility but which could not be fully investigated due to the limited facilities available in Spert I. A description of the mechanical components of this initial core is listed below:

1. Fuel Assemblies

The fuel designed for the initial core in Spert IV has a nominal height of 24 in. Fuel assemblies are the Spert type "D" assemblies which are 3- x 3-in.-square and have removable plates (Figure 64). Basically, the fuel assembly consists of a square-end box, a 2.996-in.-square aluminum retaining can with a 0.060-in.-thick wall, two grooved side plates, 12 fuel plates and a lifting bail. The lifting bail and square end box further serve to hold the plates in the assembly. Complete disassembly of the fuel is accomplished by removing 12 machine screws. Removal of four machine screws releases the lifting bail and the fuel plates. The square-end box is machined from commercial 6061-T6 aluminum square tubing and the can is commercial square 6061-T6 aluminum tubing purchased in accordance with Engineering Specifications SPT 1012 contained in Appendix "C".

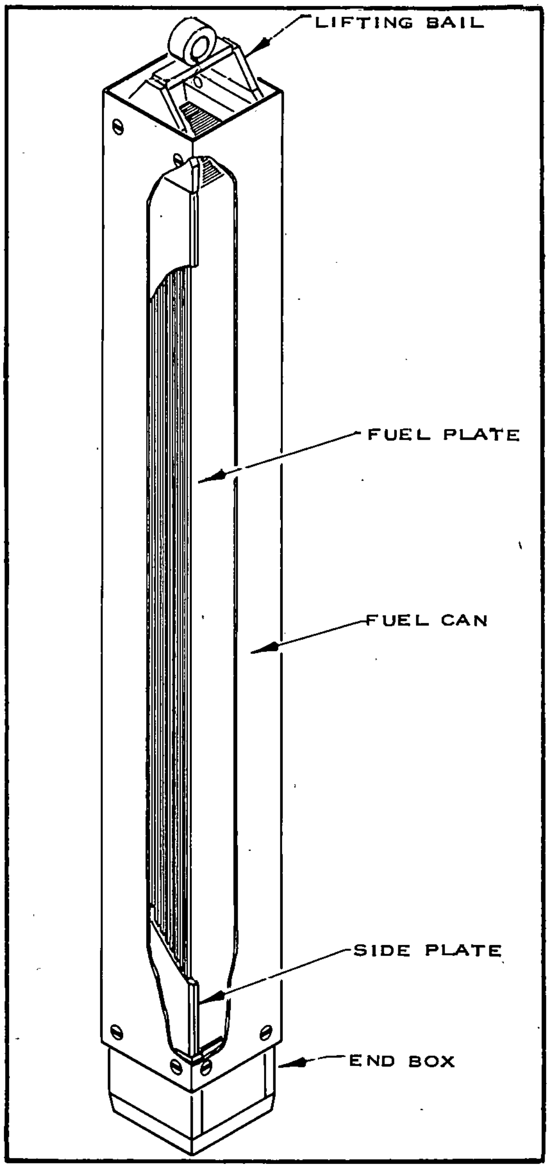

Fig. 64 Spert type D fuel assembly (NRTS-61-1991).

The fuel in each plate consists of 14 g of U-235 alloyed with aluminum melting stock to produce a core 0.020 in. thick x 2.45 in. wide x 24 in. long. The fuel plate core is clad with 6061 aluminum to produce a fuel plate 2.704 in. wide x 25-1/8 in. long x 0.060 in. thick. The water channel spacing between plates is nominally 0.179 in. Since the fuel spacing can be changed readily by removal of fuel plates or fabrication of new side plates, the water channels may be varied as experimental conditions dictate. Fuel specifications are shown in Appendix "C".

The initial hydraulic studies of the Spert type D fuel assembly [3] revealed the need for a modification to the upper lifting bail to help flatten the flow distribution through the center plate section of the assembly. This modification was made and hydraulic tests were conducted on the 18-plate, type D fuel assembly to determine the channel flow distribution for both up-flow and down-flow conditions. The tests were run in the ETR Hydraulic Test Facility which was described previously (3). The channel flow distributions were obtained by inserting two tubes in each channel with one tube measuring the static pressure near the bottom of the channel while the other tube measured the static pressure near the top of the channel. The channel frictional pressure drop was correlated to the flow by means of the Fanning friction correlation. In addition, the over-all fuel assembly pressure drop was measured.

A plot of over-all pressure drop vs flow is shown in Figure 65 for both up-flow and down-flow at a water temperature of approximately 85°F. The relationship may be expressed by

$$GPM = 74.7 \Delta P^{0.549} \quad [4]$$

where the pressure drop is in psi. The calculated relationship for up-flow also is shown for comparison purposes.

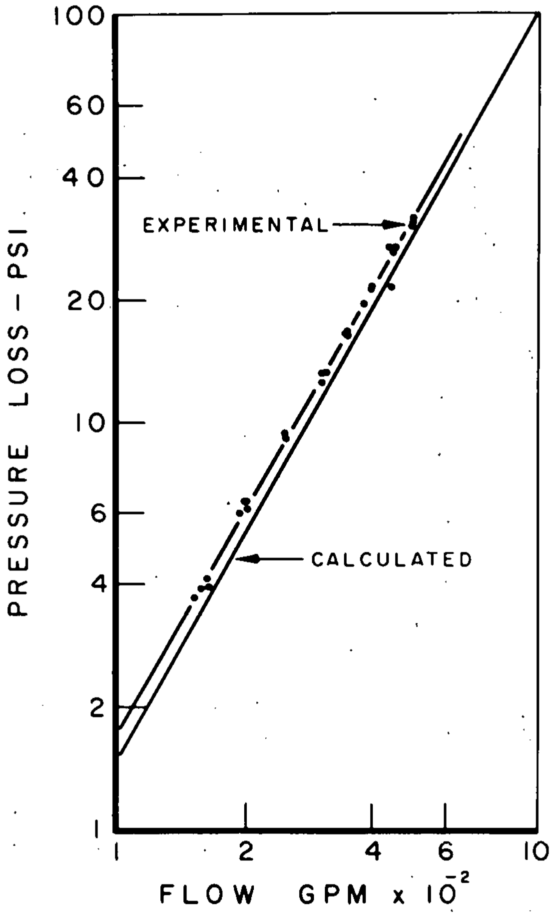

Fig. 65 Pressure drop vs flow for 18-plate assembly (NRTS-60-6106).

The calculation deviates from the experimental correlation by approximately 10%. The over-all pressure drop with the modified lifting bail was not appreciably different from the pressure drop with the original lifting bail.

In Figures 66 and 67 the channel flow distribution is shown as a function of over-all flow rate and flow direction. The channel flow was measured for 11 of the 19 channels. The channel flow for the remaining eight channels was determined by assuming symmetrical flow distribution. For down-flow, Figure 66, the distribution is relatively flat except for a flow depression in the outer two channels. For upflow, Figure 67, the flow distribution is relatively flat except for the increased flow in the outside channels. The flow direction dependence in the outside channel flow rate is probably due to entrance effects.

The actual flow in any given channel may deviate from that shown in the figures since it was assumed that each channel spacing was the average width between plates (0.092 in.) rather than the maximum (0.105 in.) or the minimum (0.079 in.). However, motion pictures taken of the fuel plates during previous flow studies showed that all the plates exhibited random vibration in their support slots.

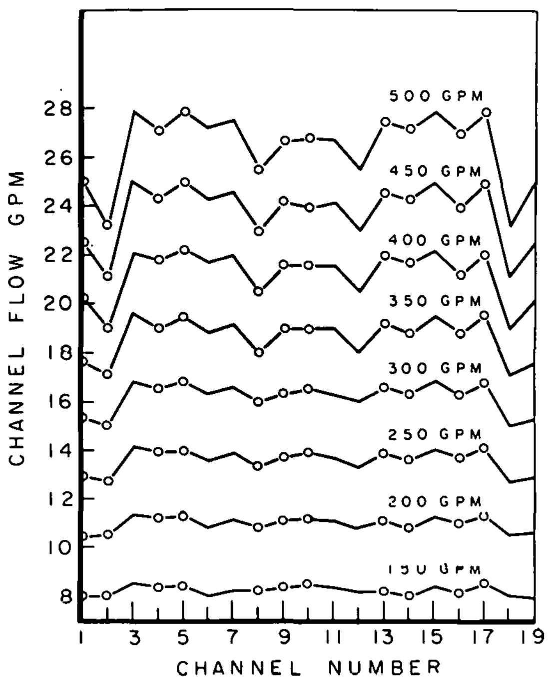

Fig. 66 Channel flow distribution for 18-plate assembly downflow (NRTS-60-6104).

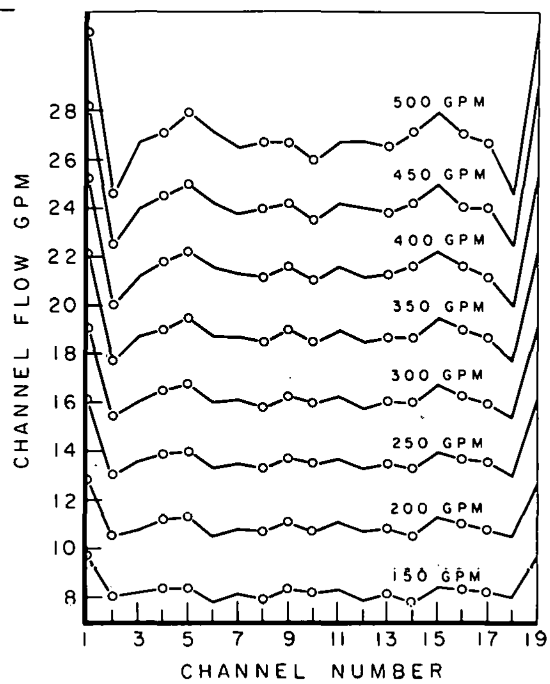

Fig. 67 Channel flow distribution for 18-plate assembly upflow (NRTS-60-6105).

For up-flow, the most erratic channel flow distribution occurred at the highest flow rate measured, 500 gpm. At this flow rate, the maximum deviation from the average was 16% while the standard deviation from the average was 1.4%. The maximum deviation occurred in the outside channels.

For down-flow, the maximum deviation and standard deviation from the average channel flow were 12% and 1.3%, respectively, for an assembly flow rate of 500 gpm. In general, the maximum deviation and the standard deviation decreased with decreased assembly flow rates.

Hydraulic tests were conducted on a 12-plate type D assembly in the ETR Hydraulic Test Loop. The outside channels of the side plates were one-half the width of the inside channels. The over-all pressure drop-flow relationship is shown in Figure 68 for a temperature of 85°F. Additional tests are planned to obtain the channel flow distribution.

2. Control and Transient Rods

A control rod for the D core consists of two flat blades secured to a yoke by three 5/16 - 18 NC x 7/8-in. flat-head socket-cap screws per blade which are staked at assembly to prevent loosening of the screws and consequent separation of the blades from the yoke.

An integral part of the yoke is a spring-loaded pin and support pad which rests on the control rod fuel assembly lifting bail. This feature accomplishes positive mating of the control rod yoke and drive rod coupling when the drives are at their lower limit, and consequent safe coupling of the drives and the control rods can be readily made.

Each blade of a control rod is made from two flat sections; one 6061-T6 Al and the other "Binal"* with 7% boron, which are tapered on one end and riveted at the tapers by four 1/4-in. 100° flat-head aluminum rivets to form a blade 5/16 in. thick, 2 in. wide, and 58-1/4 in. long. The upper section of the control rod blade is the "Binal" section and is 32-7/8 in. long. The lower section is the aluminum follower and serves as both a guide and flux peak suppressor as the control rod blades are withdrawn.

The four sets of control rods operate in a guide slot that replaces a portion of the fuel in the standard type D fuel assembly. The end boxes of the control rod are equipped with lower blade guides which mate with the blade guides in the fuel assembly. The lower blade guides have been perforated with 3/4-in. drilled holes to allow easy escape of water from the guides when the rods are dropped.

The transient rod for initiating excursions by the sudden addition of reactivity is identical to the control rods except as noted.

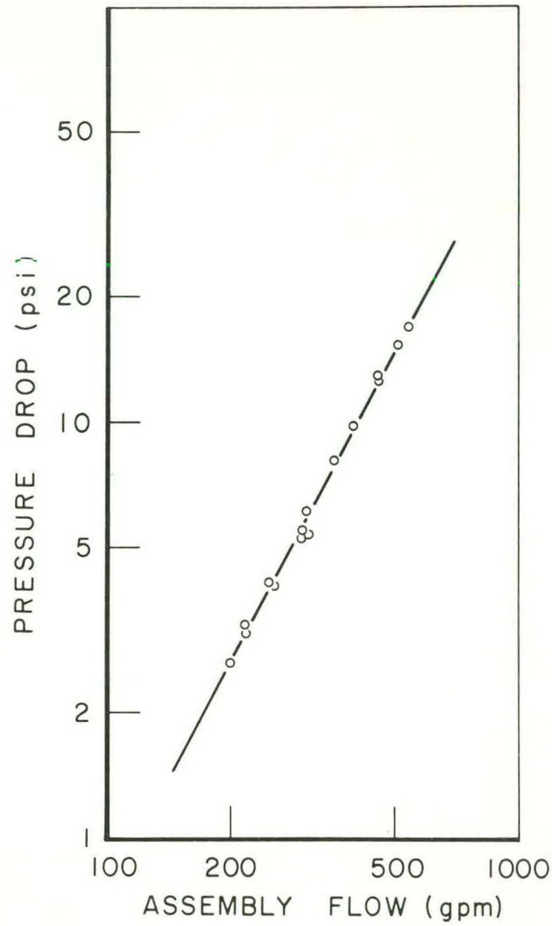

Fig. 68 Pressure drop vs flow for 12-plate assembly (NRTS-62-257).

1. The over-all length is 60-1/4 in.
2. The lower section is the "Binal"
3. The "Binal" section is 25-3/8 in. long.
4. The screw hole pattern for securing the blades to the yoke is the inverse of the control rod screw hole pattern.

Because of the position of the "Binal" section in the transient rod the rod is raised to decrease reactivity and dropped to add reactivity.

Figure 69 illustrates the relationship of control rod, yoke, guides, side plates and end boxes to each other.

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* Trade name for the Sintercast Corp. Aluminum-Boron powder-metallurgy processed material.

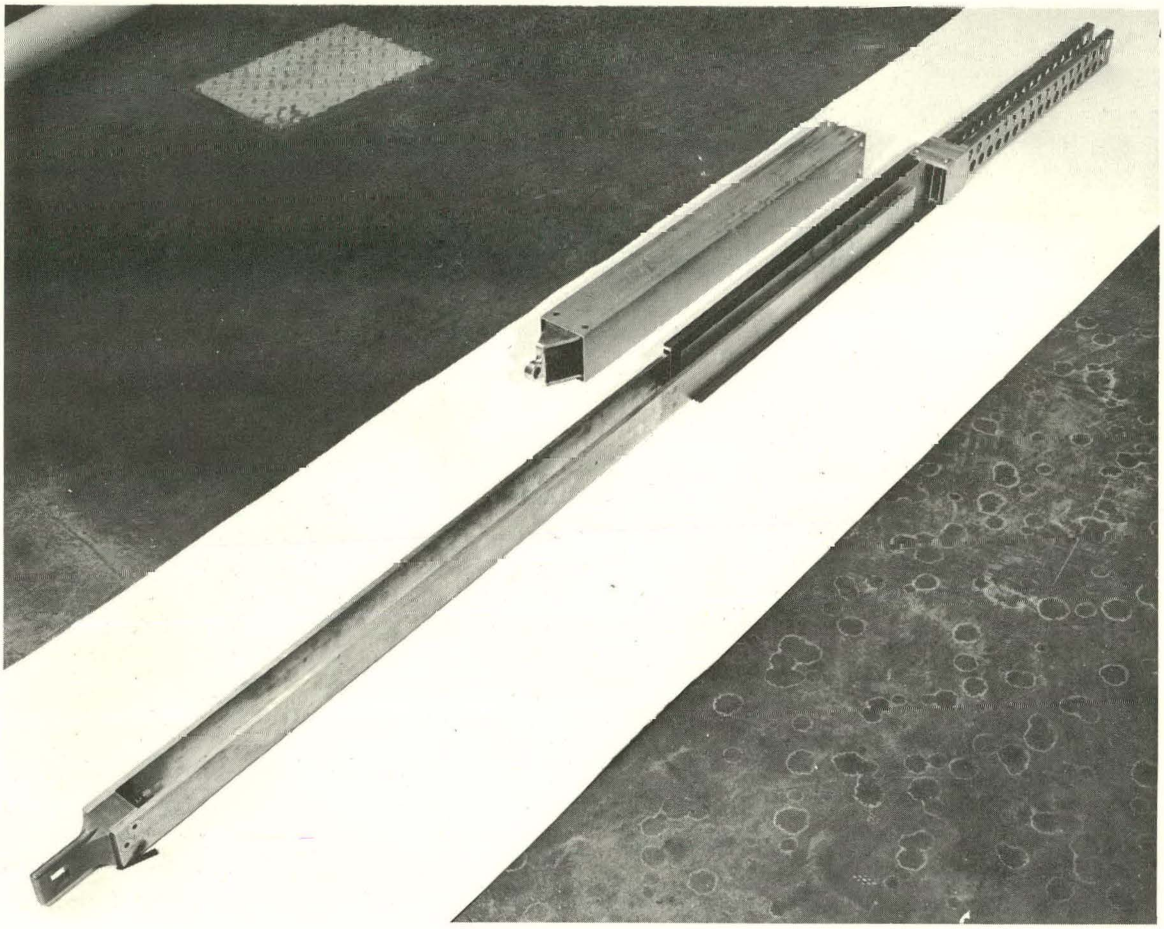

Fig. 69 Photograph of control rod assembly (NRTS-61-7269).

3. Core Support Structure

The core structure, shown in Figure 70, consisting of the grid support plate, lower grid assembly, flow skirt and flow skirt weldment is supported from the reactor control bridge by the core support weldment. Cross bracing on the core support weldment is removable with tools from the working bridge, thus providing more accessible working space around the reactor core. Lateral movement of the core support weldment is prevented by the sway-stand weldment mounted to the H-section on the reactor pool floor. The sway-stand weldment is not physically connected to the core support weldment, thus allowing for vertical movement of the core structure due to thermal expansion.

3.1 Lower Grid. The lower grid consists of 6061-T6 aluminum plate 4 in. thick by 40-1/4 in. square. A 27.5-in.-square section has been removed from the center of the plate and replaced with an interlocking 6061-T6 aluminum egg crate as shown in Figure 71. The egg crate is fastened with twelve 0.250- x 5/8-in.-long stainless-steel dowels and bolted with 144 No. 10 - 24 NC x 5/8-in.-long stainless-steel cap screws to the grid plate. The egg crate unit divides the square section into a 9 x 9 lattice of 3- by 3-in.-cells. The lower grid assembly bolts to a grid support plate by nine 1-in. stainless-steel captive screws. The grid support plate is fastened to the core support weldment by

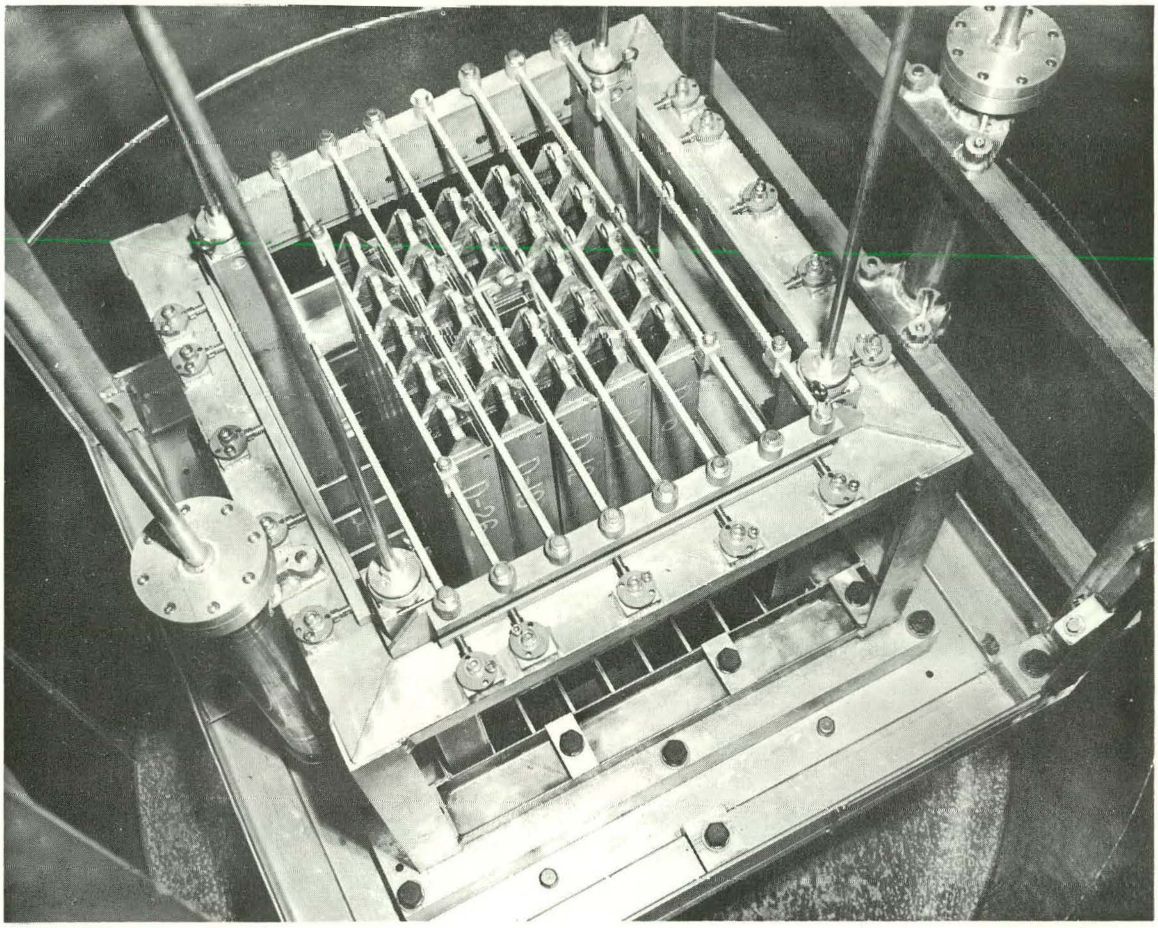

Fig. 70 Photograph of reactor core structure (NRTS-62-146).

eight 1-in. stainless-steel captive screws, thereby connecting the lower grid to the reactor control bridge. The grid support plate serves as the mounting surface for a flow transition weldment which connects through an expansion bellows to the 16-in. flow nozzle at the bottom of the reactor pool.

3.2 Flow Skirt. The flow skirt is a removable type 6061-T6 aluminum open-ended 27 3/4-in.-square box, 5/16 in. thick and 23 1/16 in. high. A photograph of the flow sheet is shown in Figure 72. The flow skirt is fastened to the flow skirt weldment by 72 No. 10 24 NC x 3/8-in.-long stainless-steel cap screws. Installation and removal of the flow skirt can be accomplished by removing the flow skirt weldment from the lower grid assembly.

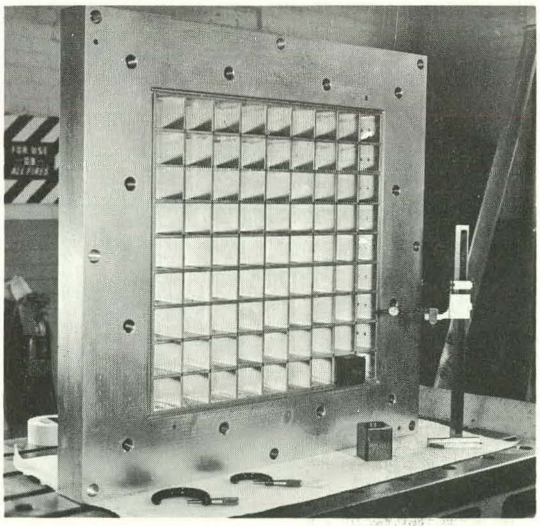

Fig. 71 Photograph of reactor core structure lower grid assembly (NRTS-61-1161).

3.3 Flow Skirt Weldment. The flow skirt weldment bolts to the lower grid assembly with twelve 1-in. stainless-steel captive screws and contains the fuel assembly hold-down mechanisms and fuel assembly location cams. Figure 73 is a photograph of the core structure showing the flow skirt weldment.

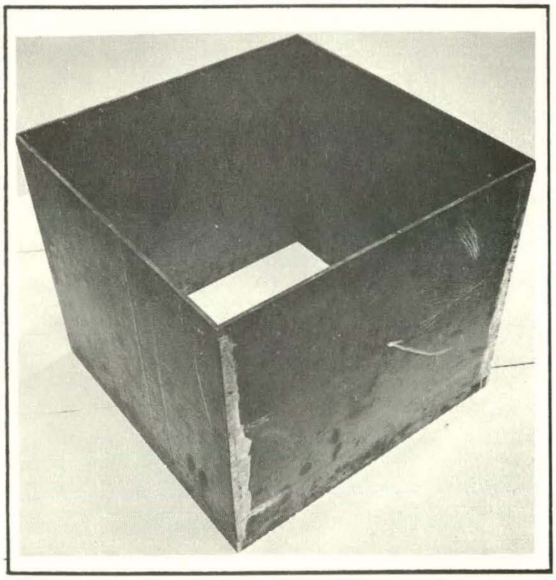

Fig. 72 Photograph of reactor core structure flow skirt (NRTS-62-184).

The fuel assembly hold-downs consist of 3-1/4-in.-high by 3/8-in.-wide horizontal 6061-T6 aluminum bars bolted to the top of the flow skirt weldment with 1/2 - 13 NC stainless-steel captive screws. These bars restrain the fuel assemblies in a vertical position between the lower grid assembly and the top surface of the flow skirt weldment. Sliding bar clamps on the hold-down bars restrict the lateral movement of the fuel assemblies.

The fuel assembly location cams, consisting of 5 cams per side for a total of 20 cams, position the fuel assemblies laterally for full core loading.

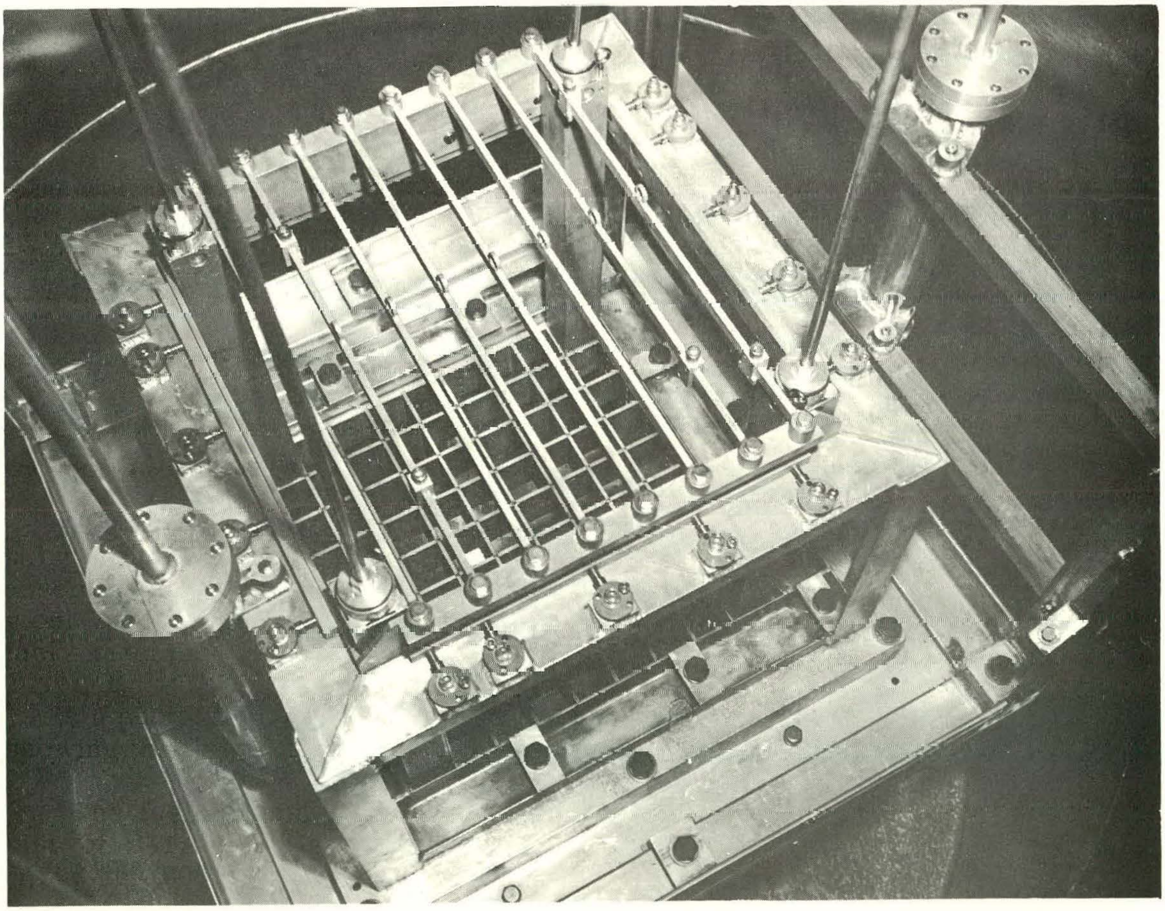

Fig. 73 Photograph of reactor core structure flow skirt weldment (NRTS-62-145).

XIV. APPENDIX C

SPECIFICATIONS

SPECIFICATIONS FOR SPERT TYPE "D" FUEL CANS

Engineering Specification SPT-1012
May 3, 1960

SPECIFICATIONS FOR SPERT TYPE "D" FUEL CANS

A. SCOPE

This specification covers the material, dimensions, and shipping of material for the SPERT type "D" fuel cans.

B. MATERIAL

Material shall be 6061-T6 drawn aluminum square tubing.

C. DIMENSIONS

1. The tubing shall be 2.996 $^{+.002}_{-.008}$ inches square outside dimensions by .062 $\pm$ .006 wall thickness.

2. Tubing shall be furnished in 5-foot lengths or multiples of 5-foot lengths.

3. The outside corner radius shall be 1/16-inch maximum.

4. Tubing shall be straight within .030 inch in 5 feet.

5. Tubing shall be square within 0.015 inch T.I.R. at any cross section.

D. SHIPPING

Tubing shall be packaged in such a manner as to protect the dimensional tolerances of the tubing during shipping.

Specifications SPT-1025
Phillips Petroleum Company
Atomic Energy Division
April 14, 1961

SPERT TYPE "D" FUEL PLATES

INDEX

SPERT TYPE "D" FUEL PLATES

A. Scope

These specifications cover the requirements for fabrication, inspection, cleaning, and packaging for shipment, SPERT fuel plates, Type D for use in the SPERT reactor facilities at the United States Atomic Energy Commission's National Reactor Testing Station near Idaho Falls, Idaho.

B. Drawings

The following Phillips Petroleum Company drawing is incorporated in and made a part of these specifications:

SPT-D-7025, Rev. 3	SPERT Type D Fuel Plate, Assembly and Details

C. General Description and Requirements

1. The SPERT Type D fuel plates are individual rectangular plates having the dimensions shown on the drawing. Each fuel plate consists of a uranium-aluminum alloy core enveloped and centered in an aluminum cladding on all surfaces. The cladding shall be metallurgically bonded to the alloy core.
2. The seller is required to supply two thousand two hundred and fifty (2250) SPERT Type D fuel plates.
3. Each fuel plate supplied shall have an identifying letter and number. The identification shall consist of the letter "D" followed by a number. The numbers shall run serially starting with the number 600. The size and location of the identification is shown on the drawing, and shall be of sufficient depth to produce a sharp, clear impression.
4. A chemical assay for total uranium content shall be made on samples taken from each uranium-aluminum alloy ingot. The seller will assay a minimum of two (2) samples from each ingot. The seller will also provide the buyer with two (2) samples properly identified from each ingot. The seller shall supply the buyer with

a report containing the serial number of each ingot, location from which samples were taken from each ingot, and the results of analysis of each sample taken. The homogeneity of the uranium in the uranium-aluminum alloy ingots shall be determined by the chemical analyses of the samples taken from the ingots. The variation in the uranium content of the ingots shall be maintained within tolerances such that the fuel content of the fuel plates as specified in other sections of this specification will be met. Following fabrication of all fuel plates, the seller shall supply the buyer with a report identifying each fuel plate with the ingot from which fabricated.

5. Radiographs shall be made of at least 10 per cent of the fuel plates and furnished to the buyer. The plates to be radiographed shall be selected at random from the finished plates.

6. Destructive tests shall be conducted on at least five (5) fuel plates selected at random. The tests shall consist of sectioning the fuel plates as specified by the buyer and preparing at least three (3) photomicrographs of each plate. Two prints of each photomicrograph shall be supplied to the buyer.

D. Materials of Fabrication

1. The following materials which meet the indicated specification shall be used when applicable:

Fuel Plate Core	The uranium-aluminum alloy core shall consist of uranium enriched to a minimum of 93 per cent in the isotope U-235 alloyed with aluminum melting stock ASTM Specification B24-46, Grade 9930A.
Fuel Plate Cladding	Fabricated from aluminum alloy 6061, ASTM Specification B209-57T Alloy GS11A, Aluminum alloy 6061.

2. Subject to the provisions of the contract the procedure outlined in this paragraph 2 shall be used for determining U-235 content of each plate submitted by the seller.

These data will be used to determine acceptance or rejection with regard to the fuel loading of each plate and also will determine the B quantity of the material balance of Section F.

Each fuel plate shall contain nominally 14.0 ± 0.2 gram U-235. Compliance with fuel content specifications will be determined by the buyer from chemical analyses of the ingot samples supplied by the seller and/or by non-destructive assay of each completed fuel plate by the buyer. Where non-destructive assay by the buyer is employed, procedures and techniques will be used by the buyer such that the measurement uncertainty will not exceed .1 per cent. Fuel plates for which the non-destructive assay content differs from 14.0 grams in excess of 0.3 grams may be returned, at the discretion of the buyer, to the seller. Upon request of the seller and prior to time of assay of the first item by the buyer under this contract, measurement procedures, techniques and criteria for determining compliance with the total U-235 specification per item by the buyer will be explained and demonstrated to the seller. Failure to agree upon a particular measurement, set of measurements, or the final material balance, shall be a dispute concerning a question of fact within the meaning of the article of the attached contract entitled DISPUTES. However, nothing contained herein concerning accountability shall excuse the Contractor from proceeding with the contract.

E. Financial Responsibility

1. Pursuant to the terms of the contract, the seller will be held financially responsible for all U-235 provided in highly enriched form by the buyer through Atomic Energy Commission channels to accomplish the purpose of this contract. This responsibility will be defined in terms of a final material balance constructed from:

   A = The total quantity U-235 provided the seller as determined by the supplier (AEC) or as mutually agreed upon by the supplier and the seller.

   B = The total quantity U-235 as measured by the buyer by chemical assay or ingot samples and/or non-destructive assay of accepted fuel plates and samples.

C = The total U-235 measured by the buyer or his agent through sampling and weight measurements of the U_3O_8 resulting from scrap processing by the seller.

D = The total U-235 as determined by the buyer from sampling and assay of unused enriched uranium metal to be returned to the buyer upon completion of the contract.

X = Total U-235 unaccounted for.

2. A charge will be made to the seller against the total contract price of all accepted plates for material unaccounted for as defined by X = A-B-C-D. The charge schedule which takes into account the measurement uncertainties associated with the quantities entering into the material balance is as follows:

   a. For X equal to 200 grams or less, no charge will be assessed.

   b. For X in excess of 100 grams, the financial charge will be calculated by the formula: Charge ($) = 1000 + (X-200) 17.

3. If subsequent to delivery to the buyer's designated carrier and before measurements can be made by the buyer or his agent there occurs physical loss of U_3O_8 or excess U-235, the quantities "C" and "D" used in implementing the financial responsibility formula will be as determined by the seller, provided,

   a. such determinations are reasonable and consistent with process technology and both the seller's and buyer's measured quantities and estimated losses,

   b. the values, along with a statement and supporting data relative to the method (s) of their determination, shall be delivered to the buyer or his agent at the time the material is physically transferred to the buyer,

   c. such loss has not resulted from defective or improper shipping containers or packaging furnished by the seller under the contract.

If any of the above three conditions are not met, then the seller shall be financially responsible for such loss in accordance with definition "C" and "D" of the financial formula set forth above. The buyer may, at his option, waive its right to measure any part or all of U-235 in U_3O_8 or excess U-235 returned in lieu thereof accept the seller's measurements thereon for the purpose of implementing the above financial formula.

4. The seller may review all measurement procedures, techniques, and calculations made by the buyer prior to assay of the first item under this contract. The seller's representative may witness any or all determinations made by the buyer relative to items under this contract, provided notification is given at the time of shipment from the seller's plant.

5. Redetermination of contents of particular items will be made by the buyer upon written request from the seller, provided such requests are made prior to the buyer placing the items in use or 60 days after shipment, whichever occurs first. Failure to agree upon a particular measurement, set of measurements, or the final material balance shall be a dispute concerning a question of fact within the meaning of the article of the attached contract entitled DISPUTES. However, nothing contained herein concerning accountability shall excuse the Contractor from proceeding with the contract.

6. In addition to the financial responsibility of the seller for U-235 accountability as described in Section E of this specification, the seller will be held financially responsible for maintaining the purity and enrichment level of the uranium caused to be supplied to the seller by the buyer. This responsibility shall extend throughout the entire period that the uranium is in the custody of the seller, his agents, or subcontractors.

   For degrading the enrichment of the uranium, whatever its form, returned to the buyer, or its designees, a charge of twelve dollars ($12) per kilogram U-235 per one-tenth (.1) percentage point degradation from the initial enrichment, as determined by the buyer or his agent, will be charged the seller. In addition, the seller will be charged for costs incurred by the buyer or its designees for any processing changes resulting from foreign material in quantities in excess of specification values, or original content values when not specified, introduced into the uranium by the seller or its subcontractors.

All costs incurred by the buyer or his agent resulting from degradation and contamination of uranium as described in this section (E. 6) shall be deducted from the contract price or paid the buyer by the seller.

F. Fabrication

1. The fuel plate dimensional deviation will be held within the limits shown on the drawings listed in Section B of this specification.

2. Each fuel-bearing plate shall be X-ray photographed or fluoroscoped prior to final shearing to establish the outline of the alloy core material.

3. Uranium shall be uniformly distributed throughout the fuel plate cores. The fuel density in grams of U-235 per unit area over an area of one square centimeter within any plate shall not deviate from the average fuel density for a plate by more than ± 5 per cent, except in the area within 2 inches of the end of the fuel plate. In this area a deviation of 7 per cent will be permitted to account for concentrations of fuel (dogbone) frequently encountered in these areas. The average fuel density for a plate is defined as the total U-235 contained in that plate, divided by the total area of the fuel core. There shall be no detectable non-uniformity in the radiographs. Detection of non-uniformity shall be cause for rejection.

4. Control of total U-235 content shall be exercised on the fuel plates. The U-235 content of each plate shall be nominally 14.0 ± 0.2 grams.

5. Each fuel plate shall be rolled by a combination of first, hot rolling, and second, cold rolling. The final reduction of the fuel plate thickness shall be accomplished by cold rolling and shall not be less than 15 per cent nor more than 20 per cent.

6. A blister test shall be conducted on each fuel plate at 1000°F for one hour. The

plates submitted by the seller shall be free of blisters or any blemish which indicates a separation of the clad from the fuel core.

G. Cleaning

1. The seller shall take all precautions necessary to maintain a high standard of cleanliness during fabrication to insure that no foreign materials or corrosion products are present in the finished units.

2. All oil and grease shall be removed by use of a satisfactory degreasing agent, and the surface rinsed with water and dried.

3. Surfaces of individual fuel plates shall be free of uranium, dirt, scum, scale, grease, pencil marks, and other foreign materials. Fuel plates shall be entirely free of blisters and unreasonable scratches or gouges.

H. Deviations from Specifications

1. Notwithstanding other provisions of these specifications, the buyer at his option, and in writing, may waive seller's minor deviations from requirements of the specifications and drawings where the failure to meet any specific requirement either alone or in combination with other such failures will not, in the opinion of the buyer, significantly reduce the efficiency of performance of the fuel plates. Acceptance of a fuel plate by the buyer with one or more such deviations from the specifications shall not be construed to mean the buyer approves or will approve similar deviations in plates not yet delivered under the contract. If such deviations allowed by the buyer under this provision result in less costs to the seller than would have been incurred had all the requirements of the specifications and drawings been fully met, then the contract price shall be adjusted downward by an amount corresponding to such reduced costs and the contract modified in writing accordingly.

I. Packaging and Shipping

1. The buyer will provide suitable shipping boxes for the fuel plates. The seller is responsible for loading the finished plates in the shipping boxes in a clean, dry condition, free of grease and other extraneous materials.

2. The seller shall take all necessary precautions during packing to protect the plates from damage during shipment.

3. Shipment will be f. o. b. fabrication site.

J. Testing, Inspection, and Data Records

1. A certified copy of inspection and test records covering all items in Section J. 1. shall be supplied to the inspector designated by the buyer. A duplicate copy shall be included in the shipping containers with the fuel plates. The seller shall allow the buyer's inspector free access to the portions of the plant engaged in fabrication of the fuel plates at all reasonable times. The seller shall provide the buyer's inspector all assistance necessary to perform tests and inspections he desires.

a. The seller shall inspect all fuel plates to determine conformity to drawings listed in Section B. The seller will not ship any fuel plate deviating from the specifications, drawings, dimensions, or tolerances without the buyer's approval.

b. The seller shall supply the buyer with inspection data, process control data, manufacturing data and other information specified in other sections of these specifications. In addition, the following shall be supplied:

(1) Radiographs of 10 per cent of the fuel plates selected at random. Radiographs shall be identified by the serial number of the plates. The buyer may specify exposure conditions to insure usable film for interpretation.

(2) Results of seller's analyses of samples from each ingot.

(3) The serial number and U-235 content of each fuel plate with the corresponding serial number of the ingot from which the fuel plate was fabricated.

(4) Three (3) microphotographs from each of the five (5) fuel plates sectioned for destructive tests.

(5) Two (2) five gram alloy samples from each ingot.

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XV. APPENDIX D
ENGINEERING CALCULATIONS

(M. K. Shane)

XV. APPENDIX D
ENGINEERING CALCULATIONS

The calculations used in the design of the control-rod-drive mounting plate, control rod guide tubes, shock-absorber support, shock absorber and guide-roller frame weldment, core support weldment, core components, and viewports are described below.

1. Control-Rod-Drive Mounting Plate

The control-rod-drive mounting plate supports all sub-assemblies that provide motion to the control rods. This plate is supported by the control bridge beams along two edges in the manner of a simply supported beam with some restraint because of the hold-down bolts.

An analysis of all load combinations is quite involved, for these loads include static and dynamic values which vary over wide ranges according to the condition being considered. However, the most severe stresses occur in the drive mounting plate when the drive assembly is being moved from its position in the dry dock and placed on the control bridge beams. This work is done with the overhead crane, and an assumed shock load equal to three times the static value is included in the calculations.

The 4-ft-square by 1-1/2-in.-thick plate is assumed to be cut into thirds. The center third supports the entire weight of the worm-gear screw-jack units, screw housings, guide tubes and magnets (870 lb); plus one drive motor (200 lb); plus one-third of the weight of the shock absorber frame weldments, shock absorbers, control rod blades, couplings, etc (1/3 of 600 = 200 lb).

Assuming this piece to be a simply supported beam and the loads to be concentrated at their centers of gravity, the reactions at the supports are found to be:

$$R_1 = \frac{1}{36}[35(100) + 18(870) + 3(200) + 1(100)] = 552 \text{ lb}$$

$$R_2 = 100 + 870 + 200 + 100 - R_1 = 718 \text{ lb}$$

The maximum moment, M, is located at the point of application of the 870-lb load.

$$M = 18(552) - 17(100) = 8230 \text{ lb-in.}$$

Because of clearance holes in the plate, the effective beam width, b, is reduced from 16 in. to 4 in. Solving for the section modulus, Z:

$$Z = \frac{bd^2}{6} = \frac{4(1\ 1/2)^2}{6} = 1\text{-}1/2 \text{ in.}^3$$

where d = depth of beam

The tensile stress, S, for static conditions is then:

$$S = \frac{M}{Z} = \frac{8230}{1\ 1/2} = 5487 \text{ psi}$$

and the assumed shock load equal to three times the static load gives a maximum stress of:

$$S_{(max)} = 5487(3) = 16{,}461 \text{ psi}$$

Since 6061-T6 aluminum has a yield strength of 35,000 psi, this affords a factor of safety of roughly two for the drive mounting plate.

2. Control Rod Guide Tubes

The control rod guide tubes are 1-1/2-in.-square stainless-steel tubes with 0.065-in.-thick walls. Sections are joined by means of flanged joints and/or pinned connections.

When the control rods are being raised these tubes must be able to support the weight of the control rod magnets, magnet armature, etc. down to and including the control rod blades (77 lb total/tube). When the control rods are being lowered these tubes must be able to compress the springs in the bottom of the shock absorbers until satisfactory magnet contact is achieved on all magnets.

The only questionable stresses involved when raising the load are the bearing stresses, S_B, at the pinned joints. This calculation for a control rod guide tube with a 1/2-in.-diameter pin through it is:

$$S_B = \frac{P}{A_B} = \frac{77}{2[1/2 \times 0.065]} = 1185 \text{ psi}$$

where P = 77-lb load

A_B = projected bearing area (in.^{2})

When the guide tube is transmitting the force necessary to compress the shock absorber spring, there is danger of failure as a column. The maximum force, P, that this spring is able to deliver is:

$$P = \frac{GFd^4}{8ND^3} = \frac{11{,}000 \times 1 \times \overline{.250}^4}{8 \times 6\,1/2 \times (1\ 1/2)^3} = 245 \text{ lb}$$

where G = torsional modulus for 17-7PH stainless steel

F = deflection (in.)

d = wire diameter (in.)

N = number of active coils

D = mean diameter of coils (in.)

Using the Euler formula for failure of a long column by elastic instability, it is found that the guide tube maximum allowable load, Q, is as follows:

$$Q = A \left[\frac{18{,}000}{1 + \frac{1}{18{,}000} \left(\frac{L}{r}\right)^2} \right] = 0.3516 \left[\frac{18{,}000}{1 + \frac{1}{18{,}000} (122)^2} \right] = 3464 \text{ lb}$$

where A = guide tube cross sectional area (in.^2)

L = guide tube length (in.)

r = guide tube radius of gyration (in.)

The control rod guide tubes therefore were found to be amply safe, and the physical dimensions necessary for guiding the rods are the limiting design parameter.

3. Shock Absorber Support

The shock absorbers are supported by a weldment that is in turn supported by the shock absorber support framework. This shock absorber support weldment is fabricated from 6-in. aluminum channel and reinforced in the area of the shock absorber clearance holes.

This support framework must be strong enough to support the weight of the shock absorbers when filled with water (55 x 5 = 275 lb) while simultaneously resisting the force applied by the drive units when compressing the shock absorber springs (245 x 5 = 1225 lb). The force exerted on the shock absorber when decelerating the falling mass of the control rods, control rod couplings, etc., up to and including the magnet armature (57 x 2 x 5 = 570 lb) is less than the force necessary to compress the shock absorber springs and is not included in the calculation.

An examination of the physical dimensions of the support framework and the placement of the loads shows that the most highly stressed portion of this weldment is the inside flange of either 3-ft-long channel in the vicinity of the shock absorber clearance holes.

The section modulus, Z, of the channel at this location is reduced from the book value because of the removal of material for the clearance holes and bolt holes. The area, A_R, and moment of inertia, I_R, of the removed section are:

$$A_R = bh = 4 \times 0.200 = 0.800 \text{ in.}^2$$

$$I_R = \frac{bh^3}{12} = \frac{4 \times (0.200)^3}{12} = 0.00267 \text{ in.}^4$$

where b = length of removed portion of web (in.)

h = web thickness (in.)

The location of the principal axis of the cross section, $\overline{X}$, changes also and is calculated as follows:

$$\overline{X} = \frac{A_B X_1 - A_R(X_R)}{A_B - A_R} = \frac{2.40(0.51) - 0.80(0.10)}{2.40 - 0.80} = 0.715 \text{ in.}$$

where A_B = book value for area of channel

X_1 = book value for distance from principal axis to face of channel

X_R = distance from centroid of removed area to face of channel

The moment of inertia, $\overline{I}$, of the modified cross section is:

$$\overline{I} = (I_B + A_B d_1^2) - (I_R + A_R d_2^2)$$

$$= [0.69 + 2.40(0.715 - 0.51)^2] - [0.00267 + 0.80(0.715 - 0.100)^2]$$

$$= 0.486 \text{ in.}^4$$

where I_B = book value of moment of inertia of section of channel

d_1 and d_2 = distance principal axis of section is moved to correspond to new location

then, the new section modulus, $\overline{Z}$, is:

$$\overline{Z} = \frac{\overline{I}}{c_w} = \frac{0.486}{(1.92 - 0.715)} = 0.388 \text{ in.}^3$$

where c = distance from principal axis to the most remote point of the section

The calculation for the maximum stress assumes that the loads (55 + 245 = 300 lb/shock absorber) are concentrated at the center of the shock absorbers. The inside flange of either 3-ft-long channel then has three 150-lb concentrated loads (300/2 = 150 lb), 3 in. apart, and the distance between supports is 34 in. The reactions at the beam supports, R_1 and R_2, are:

$$R_1 = R_2 = \frac{3(150)}{2} = 225 \text{ lb}$$

The maximum bending moment, M, is at the center of the beam and is:

$$M = 17(225) - 3(150) = 3375 \text{ lb-in.}$$

Neglecting the fact that the weak section is actually 3 in. from the center of the beam, a conservative approach, the maximum bending stress, $S_{(max)}$, in the flange is:

$$S_{(max)} = \frac{M}{Z} = \frac{3375}{(0.388/2)} = 17{,}400 \text{ psi}$$

where $Z = 1/2$ of $\bar{Z}$

The value for stress provides a factor of safety of two based on yield for the most highly stressed portion of the shock absorber support weldment.

4. Shock Absorber and Guide Roller Frame

The shock absorber and guide roller frame weldment is fastened under the control rod drive mounting plate with eight, 3/4-10NC, bolts. These bolts, and the members of the frame weldment, must be capable of supporting the dead weight of the frame weldment (133 lb) plus the maximum shock absorber loads as calculated in the preceding section (5 x 300 = 1500 lb).

The tensile stress, S_t, in the bolts is equal to the total force divided by the total area of the bolts at the root diameter of the threads. The area of one bolt at the root diameter of the thread is:

$$A = \frac{\pi}{4} [0.750-2\ (0.6495\ p)]^2 = 0.302 \text{ in.}^2$$

where p = pitch = $1/N$

N = threads/inch

$$\text{therefore } S_t = \frac{1500 + 133}{8\ (0.302)} = 676 \text{ psi}$$

This stress is quite low and affords a factor of safety for stress concentration at the thread root and for shock loads.

The stresses in the members of the frame weldment were calculated in a similar manner ($S = P/A$) and found to be less than 600 psi. The size of the members was not reduced, however, because it was felt that a reduction of mass would introduce problems in vibration.

5. Core Support Weldment

The core support weldment is fastened under the control bridge beams with eight 1-8NC bolts. These bolts, and the members of the support weldment, must resist the forces applied by the weight of the weldment (610 lb), the core support plate (480 lb), the grid assembly (400 lb), the flow skirt assembly (170 lb), and a possible complete loading of fuel cans (960 lb).

The tensile stress in the bolts is calculated as in the preceding section. The area of one bolt at the root diameter of the thread is:

$$A = \frac{\pi}{4} \left[1.000-2 \ (0.6495 \ p)\right]^2 = 0.551 \text{ in.}^2$$

$$\text{and } S_t = \frac{610 + 480 + 400 + 170 + 960}{8 \ (0.551)} = 594 \text{ psi}$$

Again, the tensile stress in the bolts is quite low and an ample factor of safety is provided for stress concentration at the thread roots and for shock loads.

The bottom of this core support weldment is prevented from swaying by a sway stand. To prevent the possibility of thermal expansion causing a force to be exerted on the core support weldment by the sway stand, a gap of 1 in. was allowed between these parts. Assuming a temperature rise, dT, of 130°F and a full tank of water, the change of length, dl, of the submerged parts is:

$$dl = La_{a1} \ (dT) = 300 \ (1.31 \times 10^{-5}) \ 130 = 0.51 \text{ in.}$$

where a_{a1} = 1.31×10^{-5} in./in./°F = coefficient of linear expansion for 6061-T6 aluminum

L = original length of submerged parts

It is seen, therefore, that the 1-in. gap will not close due to thermal expansion of the submerged parts.

This change of temperature of the water will cause a relative movement between the core centerline and the control rod blades. The core support framework is constructed from 6061-T6 aluminum, while the submerged control rod drive parts are constructed from type 304 stainless steel. The difference in the coefficient of linear expansion for these materials will cause the control rods to raise with respect to the core centerline as follows:

$$(dl_{al} - dl_{sst}) = L' \ dt \ (a_{al} - a_{sst})$$

$$= 213 \ (130) \ (1.31 - 0.96) \ (10^{-5})$$

$$= 0.097 \text{ in.}$$

where L' = length of stainless steel in common with length of aluminum

a_{sst} = 0.96×10^{-5} in./in./°F

The transition section is attached to the bottom of the core support plate, and these parts transmit the change in length of the core support weldment and other submerged parts to the expansion joint. The change in length that this expansion joint would have to compensate for with a change of temperature of 130°F is as follows:

$$dl = 130\ [280\ (1.31 \times 10^{-5}) + 9\ (0.96 \times 10^{-5})] = 0.49 \text{ in.}$$

As the expansion joint provided has a rated movement of plus or minus 2 in. no stress is to be expected from thermal expansion of the parts.

6. Core Components

In the analysis of the core components, the strength of the grid is of primary concern. This grid is fabricated from aluminum plates that are notched to enable them to fit together in a manner similar to the construction of an egg crate. However, the depth of this notch is shallow on one set of plates and deep on those that are at 90 degrees. On those plates with the deep notch, the section is further reduced by a clearance hole for a screw. For this reason, it is assumed that the only plates capable of supporting a load are those having the shallow notch. The section modulus, Z, of one of these plates is calculated as follows:

$$Z = \frac{bh^2}{6} = \frac{0.300\ (4 \div 1\ 1/4)^2}{6} = 0.378 \text{ in.}^3$$

where b = plate width (in.)

h = plate height less notch less tapped hole.

As these plates are evenly spaced every 3 in., the loads and unit pressures imposed on a 3-in.-wide strip (1 1/2 in. on each side of the plate) is used in the calculations. The weight of nine fuel cans (9 x 11.779 = 106 lb) is assumed to be uniformly distributed along the plate, and a unit prssure, p, is added to this load. This pressure, p, is numerically equal to the pressure drop across the core with "downflow" conditions. The reaction R at either end of the plate is:

$$R = 1/2\ [\,106 + pA\,] = 1/2\ [\,106 + p\ (3 \times 27)\,]$$

where A = area = 3L

L = length of 3-in. strip

The maximum moment, M, is:

$$M = 1/2\ RL = 1/2\ [\,1/2\ (106 + 81_p)]\ 27 = \frac{27}{4}\ (106 + 81_p)$$

Substitution of the above values in the equation for maximum tensile stress and solving for p gives the value of the maximum allowable pressure drop across the core for "downflow". A safety factor of two for the maximum allowable stress is used in the following calculation:

$$s = \frac{M}{Z} = 17{,}500 \text{ psi}$$

$$\frac{27}{4} \frac{(106 + 81_p)}{(0.378)} = 17{,}500$$

$$p = 10.9 \text{ psi}$$

For "upflow" conditions, the hold-down structure would be analyzed in a manner similar to the above calculation except that the weights of the fuel cans act in the opposite direction. No calculation is included for the existing hold-down bars, however, for these bars and associated hardware were designed for static operating conditions.

7. View Ports

View ports are provided near the bottom of the North tank. The lowest view port is approximately 20 ft below the top of the tank, and the assumption is made that the view port is a circular plate with edges fixed and a uniform load over the entire surface. The maximum radial stress, S_r, at the edge of the plate is:

$$S_r = \frac{3W}{4\pi t^2} = \frac{3\ p\ D^2}{16\ t^2}$$

where W = total applied load = $\frac{p\pi D^2}{4}$ (lb)

t = thickness of plate (in.)

p = pressure (psi)

D = diameter of plate (in.)

The manufacturer's data on the viewport material show the ultimate stress to be equal to 15,000/25,000 psi. Using the lowest pubulished figure for stress and inserting the known physical dimensions of the view port, the equation is solved for pressure, p, as follows:

$$p_{ult} = \frac{16\ t^2\ Sr}{3\ D^2} = \frac{16\ (1)\ (15{,}000)}{3\ (16)^2} = 312 \text{ psi}$$

The static pressure due to a depth of 20 ft is:

$$p = 0.433\ (20) = 8.66 \text{ psi}$$

The view ports therefore have a factor of safety of over 30 to compensate for shock loads and possible stress concentrations due to scratches, uneven gasket surface, etc.

XVI. REFERENCES

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