80
12/170

IDO-17036
November 1965

SPERT III REACTOR FACILITY:
E-CORE REVISION

MASTER

J. Dugone, Editor

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

PHILLIPS
PETROLEUM
COMPANY

ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

PRINTED IN USA. PRICE \$4.00. AVAILABLE FROM THE CLEARINGHOUSE FOR FEDERAL
SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

IDO-17036
AEC Research and Development Report
Reactor Technology
TID-4500 (44th Ed.)
Issued: November 1965

**SPERT III REACTOR FACILITY:
E-CORE REVISION**

J. Dugone
EDITOR

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

**PHILLIPS
PETROLEUM
COMPANY**

Atomic Energy Division

Contract AT(10-1)-205

Idaho Operations Office

U. S. ATOMIC ENERGY COMMISSION

ABSTRACT

The Special Power Excursion Reactor Test III (Spert III) reactor is a pressurized-water, nuclear-research reactor which has been constructed to provide a facility for conducting reactor kinetic behavior and safety investigations. The investigations are designed to provide information for the advancement of pressurized-water and boiling-water reactor technology and safety. The facility has been designed for operation up to pressures of 2500 psig, temperatures of 650°F, and flows of 20,000 gpm and incorporates essential features typical of pressurized-water and boiling-water reactors. This report is an updating of the original facility report [1], and describes the engineering features of the reactor and supporting process equipment with the E-core and other modifications that have been made to date (1965).

CONTENTS

ABSTRACT	ii
I. INTRODUCTION	1
II. BRIEF DESCRIPTION OF THE FACILITY	3
1. GENERAL AREA LAYOUT	3
2. REACTOR VESSEL AND COOLANT SYSTEM	4
3. REACTOR CONTROLS AND CONTROL ROD DRIVES	5
4. AUXILIARY EQUIPMENT	5
III. PLANT SITE AND BUILDINGS	6
1. SPERT SITE	6
2. CONTROL-CENTER AREA	6
3. SPERT III AREA	6
4. SPERT III REACTOR BUILDING	7
4.1 General	7
4.2 Main Reactor Building	9
4.3 Wing Building	9
4.4 Spert III Control Room -- Control-Center Building	9
IV. DETAILED DESCRIPTION OF REACTOR COMPONENTS	10
1. REACTOR VESSEL	10
1.1 Design Philosophy	10
1.2 Design Conditions	10
1.3 General Description of Reactor Vessel	11
1.4 Vessel Support	13
1.5 Shop Hydrostatic Test	14
1.6 Cleaning Procedure	14
1.7 Insulation	15
1.8 Radiation Shielding	15
1.9 Thermowells	15
1,10 Experimental Thermocouples	16
2. REACTOR CORE COMPONENTS	17
2.1 Fuel Assemblies	17
2.2 Control Rods	19
2.3 Transient Rod	20
2.4 Core Filler Pieces	20
2.5 Thermal Shields	22

2.6	Internal Support Structure	23
V.	COOLANT AND PRESSURIZATION SYSTEMS	26
1.	PRIMARY-COOLANT SYSTEM	26
1.1	General Description	26
1.2	Pressurizer Vessel	27
1.3	Makeup Pump	29
1.4	Primary-Coolant Pumps	31
1.5	Heat Exchangers	33
1.6	Primary-System Control and Check Valves	34
1.7	Flow Tubes	36
1.8	Piping	36
1.9	Expansion	38
1.10	Insulation	38
2.	SECONDARY-COOLANT SYSTEM	39
2.1	General Description	39
2.2	Pumps	39
2.3	Piping and Valves	40
VI.	PROCESS INSTRUMENTATION AND CONTROL	41
1.	GENERAL DESCRIPTION	41
2.	PRESSURE	43
3.	TEMPERATURE	43
4.	LEVEL	44
5.	FLOW	45
6.	CONDUCTIVITY	46
7.	INTERLOCK SYSTEM	46
VII.	SPERT III CONTROL AND TRANSIENT ROD DRIVES	47
1.	MECHANICAL DESIGN CRITERIA	47
2.	DESIGN SAFETY CRITERIA	48
3.	DESCRIPTION OF DRIVE UNITS	49
3.1	Design Bases	49
3.2	Description of Drive Unit	50
4.	CONTROL AND TRANSIENT ROD OPERATION	60
4.1	General	60
4.2	Rod Withdrawal	60

4.3	Rod Insertion	61
4.4	Safety Features	61
VIII.	CONTROL SYSTEM DESIGN	64
1.	INTRODUCTION	64
2.	DESIGN PHILOSOPHY	64
3.	DESCRIPTION OF CONTROL SYSTEM	65
3.1	Drive Motors and Power Requirements	65
3.2	Motor Power	65
3.3	Motor Speed Selector	67
3.4	Power Control	67
3.5	Control Rod Control Switches	70
3.6	Control and Transient Rod Relay System	71
3.7	Transient Relay Systems	76
3.8	Scram, Ramp, and Run-down Circuits	77
3.9	Sequence Timer Circuits	81
3.10	Control Console	82
3.11	Rod Position	82
3.12	Alarm and Warning System	83
3.13	Rod Drive Air System	86
IX.	AUXILIARY EQUIPMENT	89
1.	WATER TREATMENT SYSTEM	89
1.1	Introduction	89
1.2	Softener	89
1.3	Demineralizer	89
1.4	Storage Tank	90
1.5	Chemical Addition	91
1.6	Analysis of Reactor Water	91
2.	AUXILIARY-COOLANT SYSTEM	91
2.1	Cooling Tower	92
2.2	Circulating Pump	92
2.3	Piping	92
3.	COMPRESSED AIR SYSTEM	92
3.1	Service Air System	92
3.2	Instrument Air System	92
4.	WASTE DISPOSAL	93
4.1	General	93
4.2	Reactor Building Sump	93
4.3	Hot Waste Storage Tank	93
4.4	Leaching Pond	93
4.5	Auxiliary Leaching Pond	94

5. HOT STORAGE FACILITIES	94
5.1 Dry Storage Holes	94
5.2 Hot Storage Well	94
6. AUXILIARY REACTOR SHUTDOWN SYSTEM	94
X. REFERENCES	96
APPENDIX A -- DESIGN DATA SUMMARY	97
1. GENERAL REACTOR DESIGN DATA	99
2. REACTOR COMPONENT DESIGN DATA	99
3. THERMAL DESIGN DATA -- 64-ASSEMBLY CORE	102
4. PRIMARY-COOLANT SYSTEM DESIGN DATA	102
5. SECONDARY-COOLANT SYSTEM	104
6. AUXILIARY EQUIPMENT	104
APPENDIX B -- ENGINEERING CALCULATIONS	107
1. HYDRAULICS	109
1.1 Core Area	109
1.2 Heat Exchangers	109
2. HEAT TRANSFER	110
2.1 Core Area Heat Transfer Coefficients	110
2.2 Engineering Hot Channel Factor	111
2.3 Heat Exchangers	115

FIGURES

1. Plan of Spert site	3
2. Process flow diagram	4
3. General layout of Spert III area	7
4. Plan of reactor building	8
5. Vertical view of reactor building	8
6. Reactor-vessel assembly	11
7. E-core cross section	17
8. 25-rod fuel assembly	18

9. Control rod assembly	19
10. Transient rod assembly	20
11. 1F and 2F filler pieces	21
12. 3F and 4F filler pieces	21
13. Plan view of thermal-shield instrumentation	22
14. Lower grid flow skirt and control rod guides	24
15. Upper grid structure	25
16. Sketch of pressurizer pressure vessel	28
17. Pressurizer water level -- indicated versus true level	30
18. Hydraulic characteristics for primary pumps	32
19. Motor characteristics for primary pumps	32
20. Primary pump heating data for 5000-gpm flow per pump	33
21. General features of the heat exchanger	34
22. Hydraulic characteristics of secondary pumps	40
23. Simplified block diagram of process instrumentation	42
24. Drive-unit assembly	50
25. Drive-unit installation	51
26. Gear-box section	53
27. Drive-screw section	53
28. Control -rod air-cylinder section	56
29. Transient-rod air-cylinder section	57
30. Shock absorber unit	58
31. Parts of shock absorber unit	58
32. Drive -rod-seal assembly	59
33. Drive motor power system	66
34. Motor speed selector system	68
35. Power control system	69
36. Miscellaneous relays	71

37. Control and transient rod relay system	73
38. Transient rod relay system.	75
39. Scram circuits	78
40. Ramp and run-down circuits	80
41. Sequence timer circuits	81
42. Control console	83
43. Alarm circuits	84
44. Flow diagram of the control rod air system.	87
45. Top air pressure required as a function of vessel pressure	87
46. Auxiliary shutdown system	95
47. Amount of boric acid required to compensate for reactivity	95
48. Time required to poison the Spert III reactor system	95
B-1. Pressure drop through 25-rod fuel assembly	109
B-2. Pressure drop through control rod and 16-rod fuel assembly	109
B-3. Pressure drop at tube side of heat exchanger	110
B-4. Pressure drop at shell side of heat exchanger	110
B-5. Heat transfer coefficient.	111
B-6. Flow requirements and outlet temperatures as a function of inlet temperature -- heat load of 1 MW	117
B-7. Flow requirements and outlet temperatures as a function of inlet temperature -- heat load of 10 MW.	118
B-8. Flow requirements and outlet temperatures as a function of inlet temperature -- heat load of 30 MW.	118

TABLES

I. Thermowell Locations and Replacements.	16
II. Pressure Instruments	43
III. Temperature Instruments	44
IV. Level Instruments	45

V. Flow Instruments	46
VI. Spert III Control Rod Drive Safety Backups	62
VII. Engineering and Operating Data for Softener	90
VIII. Chemical Analysis of Spert Well Water	90
IX. Engineering and Operation Summary -- Demineralizer	91
X. Chemical Analysis of Primary Coolant Water	91
XI. Estimated Hot Channel Factors -- Type "E" Core	115
XII. Calculated Heat Exchanger Performance Data (Extrapolated from A. O. Smith Curves).	116

SPERT III REACTOR FACILITY E-CORE REVISION

I. INTRODUCTION

The Special Power Excursion Reactor Tests (Spert) Project, operated by Phillips Petroleum Company, was established as part of the U. S. Atomic Energy Commission's reactor safety program in 1954 and is directed toward experimental and theoretical investigations of the kinetic behavior and safety of nuclear reactors. The Spert III reactor facility described herein has been constructed as a part of this safety program to fulfill the need for a facility in which to conduct reactor behavior and safety studies under operating conditions typical of pressurized-water and boiling-water reactors.

General objectives in the Spert III facility design were: (a) to provide a facility in which reactor power excursion tests could be performed and experimental information gathered on the kinetic behavior of the reactor, (b) to incorporate in the design a complete reactor and coolant system typical of existing and proposed pressurized water power reactors to permit an investigation of safety problems common to this class of reactors, and (c) to incorporate sufficient flexibility in the overall design to permit studies on several core designs.

Sustained reactor power operation was not a primary objective in the design of the facility. However, power operation for a limited time has been incorporated in the design to permit limited investigations at conditions which duplicate, as nearly as economically feasible, those nuclear and hydraulic conditions normally found in pressurized-water-reactor systems. Power operation is also required to obtain experimental information on the effect of high initial power on reactor behavior and on the transient response and hydraulic stability of the overall coolant system.

The major components of the facility include a reactor vessel, a pressurizing vessel, and two primary-coolant loops including pumps and heat exchangers. The reactor vessel and primary-coolant piping system are designed for operation at pressures up to 2500 psig and temperatures up to 650°F. Coolant flow rates up to 20,000 gpm through the reactor core are available. The heat removal capacity of the two evaporative heat exchangers is 60 MW, limited by the water supply to periods of about 30 minutes. Supporting auxiliary equipment necessary for operation of a reactor of this type has also been included.

The reactor and major plant equipment are remotely operable, with the controls located in a control-center building approximately 1/2 mile from the reactor. The reactor shielding has been limited to that which will permit work on the shutdown reactor to proceed following short time power operation.

The conceptual design of the Spert III facility reflects the recommendations and suggestions of many individuals and laboratories interested in reactor-kinetics problems and who provided advisory consultation during the initial phases of the Spert program. The Spert III E-core safety analysis is presented in another report [2].

The conceptual design of the facility was prepared by Phillips Petroleum Company in 1955. Engineering design and inspection were completed by the Stearns-Roger Manufacturing Company under contract to the USAEC. The architect-engineering work performed by the Stearns-Roger Manufacturing Company did not include the design of the fuel, the core support structure, the control rods and control rod drives, and the reactor control system. These items were designed and supplied by Phillips Petroleum Company. Construction was accomplished by a lump-sum contract with Paul Hardeman, Inc. as the prime contractor. Construction was completed and the facility accepted for operation by Phillips Petroleum Company in October 1958.

This report describes the Spert III Reactor Facility with the E-core and other modifications that have occurred since the construction of the facility. Earlier publications [1, 3, 4,] have been based on preliminary design and as-built information and are not necessarily representative of the facility with its modifications to date (1965). The report 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 process equipment. For those readers interested only in gross features of the facility, a brief general description is presented in Section II, and a design data summary of the reactor and major plant equipment is tabulated in Appendix A. Detailed design and engineering information on the facility are presented in the remaining sections.

II. BRIEF DESCRIPTION OF THE FACILITY

For orientation purposes, a brief description of the general features of the facility is presented in this section. For those readers interested, particulars of the reactor or equipment are discussed in succeeding sections of this report.

1. GENERAL AREA LAYOUT

The Spert III reactor and major plant equipment are provided with remote control instrumentation and equipment permitting operation from an isolated control center. The building housing the reactor is located about 1/2 mile from the control center as shown in Figure 1.

Fig. 1 Plan of Spert site.

2. REACTOR VESSEL AND COOLANT SYSTEM

Insofar as is economically feasible, the facility incorporates the nuclear and hydraulic features of conventional pressurized-water and boiling-water reactors. The reactor vessel and coolant system are designed for a maximum operating pressure and temperature of 2500 psig and 650°F. The coolant system is comprised of two coolant loops as shown in Figure 2. Each of the primary loops consists of two canned-rotor pumps operating in parallel, a heat exchanger, two flow tubes, and flow-control and check valves. The total flow capability of both loops is 20,000 gpm, and the heat removal capacity is 60 MW. Coolant enters the reactor vessel at the bottom, flows upward through the reactor core, reverses direction, and flows downward through the thermal shields leaving the vessel near the bottom.

Fig. 2 Process flow diagram.

The primary-coolant pumps are canned-rotor pumps, each capable of delivering 5000 gpm against a head of 328 ft of water. The pumps require the circulation of cooling water through coils surrounding the motor stator. Heating of the primary coolant to 650°F may be accomplished in about 14 hr, independent of nuclear energy input, by utilizing the energy transferred to the coolant by the pumps.

The coolant flow is controlled by remotely operated gate valves. The rate of flow is metered by either an 8-inch or a 16-inch flow tube to achieve an accurate measurement depending on the rate. The primary piping system was built with spun cast AISI Type 304L stainless steel. The piping was sized to

permit water velocities up to 30 ft/sec in order to reduce the pipe sizes and the attendant heat losses. On the basis of Spert experience, it can be concluded that centrifugally cast stainless steel pipe has given adequate service under cyclic operation and probably is equal to high-pressure stainless steel piping fabricated by other methods [5].

The reactor vessel is a multilayer-type vessel, and the control rod drives are mounted on the bolted top head. Thermal shields comprised of four concentric rings of stainless steel with a combined metal thickness of 5-3/4 inches are provided to minimize the vessel-shell stresses during operation. Because of the remote operation of the facility, the only shielding provided is a 6-inch-thick lead wall (one thickness of lead brick) around the reactor vessel to permit work around the reactor in a shutdown condition. Primary-system pressure is maintained by a pressurizing vessel in which steam pressure is generated by electric immersion heaters.

3. REACTOR CONTROLS AND CONTROL ROD DRIVES

Because of the nature of the experiments performed in Spert III, no automatic period-scam circuit is provided. Provisions are made, however, for the inclusion of special scam circuits, such as the differential temperature (ΔT) and loss-of-flow scams. Also, various interlocks are built into the control system that help prevent a series of unwise operations that could mechanically damage the core or drive units or that could initiate a premature excursion.

The control rod drives are air driven with a mechanically positioned lead screw controlling the position and rate of movement during withdrawal. Reactor scam is accomplished by rapidly exhausting the air on one side of the air piston and allowing the compressed air to drive the piston to a seat.

4. AUXILIARY EQUIPMENT

All equipment necessary to provide services for an integrated reactor plant has been furnished. Raw water and electric power are furnished from the control-center area.

III. PLANT SITE AND BUILDINGS

1. SPERT SITE

The Spert site is located within the boundaries of the NRTS approximately 50 miles west of Idaho Falls, Idaho. The site was selected to permit transient testing to be carried out safely.

A general plan of the Spert site has been previously shown in Figure 1. The nucleus of Spert operations is the control center. The reactor sites have been arranged in a semicircle lying approximately 1/2 mile from the control center and at a nominal 1/2 mile from each other. The reactor areas are spaced at approximately 60° increments to give four reactor sites in the 180° arc. Spert III lies approximately east of the control center.

2. CONTROL CENTER AREA

The control-center area forms the nerve center of the Spert operations. Within the fenced area are the control-center building and raw water storage and distribution equipment for the Spert site. The control-center building for the proposed Power Burst Facility will also be located in this area. The control-center building houses the reactor controls and instrumentation for all of the Spert reactors, administration offices, a dark room, and instrument- and mechanical-work areas.

Water for the Spert site is supplied from two wells located near the control-center area. Well No. 1 is 653 ft deep and well No. 2 is 1217 ft deep. A 400-gpm deep well pump on well No. 1 and a 550-gpm well pump on well No. 2 supply water to the two ground-level storage tanks. A total capacity of 75,000 gallons of ground-level storage is available. An automatic level control maintains the tank levels by intermittent operation of the pumps.

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

3. SPERT III AREA

The Spert III reactor is located approximately 1/2 mile east of the control center. A general layout of the area is shown in Figure 3.

A substation located outside and to the west of the main building provides electrical power for the area. The Spert III substation consists of a 2000-kVA, 3-phase, 13.8-kV-480-V transformer system for operation of the four primary-coolant pumps and a 500-kVA, 3-phase, 13.8-kV-480-V transformer system for all other uses.

Fig. 3 General layout of Spert III area.

Raw water is pumped from the control-center area through a 4-inch pipe, and is supplied to the building at about 70-psig pressure. Equipment located within the Spert III area in the immediate vicinity of the reactor building includes a 10,000-gal deionized water storage tank north of the building and an induced-draft, single-cell cooling tower southeast of the building. In addition, a 3000-gal fuel oil storage tank and an 8000-gal hot-storage tank for radioactive liquid waste are buried to the west and east of the building, respectively.

Nonradioactive liquid waste and/or radioactive liquid waste whose activity level is within AEC disposal tolerances are dumped directly into the ground in two open leach pits. One leach pit, located about 100 ft north of the building, is provided for the disposal of chemical waste from the water treating equipment, while a second leach pit located 450 ft southeast of the building is utilized for normal process waste. Both pits are isolated by fences.

4. SPERT III REACTOR BUILDING

4.1 General

The reactor building consists of a main reactor building housing the reactor and coolant systems and a wing structure housing auxiliary equipment, electrical switchgear, process controls, instrumentation, etc. A reactor building plan and section are shown in Figures 4 and 5, respectively.

Fig. 4 Plan of reactor building.

Fig. 5 Vertical view of reactor building.

4.2 Main Reactor Building

The main reactor building is a 40- by 80- by 30-ft-high, steel-girdered, pumice-block structure. A 10-ton crane spans the 40-ft width of this area and serves the entire length of the building. The reactor vessel is located below floor level in a 16- by 18- by 27-ft-deep pit centered 20 ft from the south wall. A 27- by 12-ft-deep process-equipment pit extends from the reactor pit to the building north wall and is separated from the reactor pit by a 3-ft-thick concrete wall. The pressurizer, primary-coolant pumps, secondary-coolant pumps, makeup pump, air compressors, and process piping are located in the process equipment pit. A 4- by 4-ft opening in the ground-level floor at the northeast corner of the building serves as a dry dock for the vessel head. A stainless steel lined, radioactive-fuel storage canal 8 by 6 by 16 ft deep is located in the floor at the southeast corner of the building. Additional storage facilities for radioactive materials include eight 6-inch-diameter, dry-storage holes located near the east building wall. Each hole is 16 ft deep and supplied with a lead plug 10 inches thick. Seven 11-ft-deep, 6-inch-diameter carbon steel pipes buried in the floor to the west of the reactor pit provide locations for ionization chambers.

4.3 Wing Building

On the west side of the main reactor building is a 20-ft-wide by 12-ft-high wing structure extending the full length of the building. At the north end of the wing is a utility room housing the water treatment equipment and building heating plant. Adjacent to this is the reactor building control room housing the process controls and instrumentation and switchgear associated with reactor process equipment and control system. The air-conditioned room at the south end of the wing is allotted for transient instrumentation. It is separated from the control room by an office and washroom.

4.4 Spert III Control Room -- Control-Center Building

The Spert III control room is located in the control-center building. This air-conditioned room contains the process instrument panel and nuclear console which provide remote control operation of the major process equipment and reactor from the control center.

IV. DETAILED DESCRIPTION OF REACTOR COMPONENTS

1. REACTOR VESSEL

1.1 Design Philosophy

The reactor vessel contains the nuclear components which will be used for various experimental reactor safety tests. These tests will be used to evaluate the effects of varying the operational parameters (flow, temperature, and pressure) of the core. Incidental to and as a direct consequence of these tests, the reactor vessel will be subjected to stresses induced by shock loading, gamma-heat generation, thermal cycling, and pressure cycling.

If the ASME Boiler and Pressure Vessel Code, 1956 Edition, which was the latest edition at the time of design, had been used wholly as a guide for the design of the vessel, the required wall thickness would have posed serious problems in fabrication. In addition, the heavy wall thickness would have imposed undesirable operating conditions upon the system, particularly in respect to internal heat generation caused by gamma radiation and to heating and cooling rates incident to bringing the vessel in and out of service. The code formulates rules only for the design of equipment for steady-state operation, indefinite lifetime, and for pressures up to 3000 psi maximum. Above 3000 psig, the code rules concerning risk to personnel and property are established. However, these conditions do not apply for the Spert III facility because this facility is located in a remote and sparsely populated area. Personnel are to operate the facility remotely, at infrequent intervals, and only for short periods of time.

Section VIII of the ASME Vessel Code was followed in determining head and flange thicknesses, except that a safety factor of 4 and welding efficiency of 95 percent were used in accordance with accepted practice on multilayer vessels. The shell, which is subject to internal heat generation, is designed for 750°F, while the main vessel flanges, heads, and other parts not subjected to internal heat generation are designed for a temperature of 700°F. The stresses in the pressure parts were examined for the effect of internal heat generation, pressure surges, and temperature shocks. These cyclic stresses are well below the yield point of the material. The material used in the layered shell has a minimum tensile strength of 105,000 psi and a minimum yield strength of 75,000 psi. The allowable code design stress for this material is 22,800 psi at 750°F.

1.2 Design Conditions

The reactor vessel is designed for the following conditions:

Design pressure	2500 psig
Design temperature	700 °F
Bulk-coolant temperatures	70-650 °F
Steady-state operating pressure	2500 psig

Estimated maximum transient pressure	3500 psig
Steady-state reactor power	60,000 kW

The number of normal temperature cycles, based on 10,000-hr operating life, is estimated at less than 1000 cycles.

Due to the limited quantity of secondary cooling water, operating time is limited to approximately 30 minutes at maximum design power; and at lower power levels, the operating time increases in direct proportion to the decrease in power.

1.3 General Description of Reactor Vessel

The reactor pressure vessel is illustrated in Figure 6. As indicated in the figure, it is an all-welded, multi-layer vessel. It is built up from concentric layers of relatively thin steel plate, progressively wrapped, tightened, and welded together around an inner pressure-tight cylinder. The vessel consists of a flanged full-opening top head, a cylindrical shell, a bottom hemispherical head, and a bottom tee. The ID of the vessel is 48 inches, overall length is 23 ft 9 inches, and the dry weight is approximately 42 tons. All inside surfaces in contact with the primary coolant water are clad with Type 304L stainless steel or are lined with a manually applied Type 309 Cb stainless steel metal weld.

1.31 Top Head. The top head serves to mount the control rod drives and to maintain proper alignment between the core and the control rods. It is a bolted full-opening type, which permits frequent removal and suitable accessibility to the reactor internal structure. Twenty-eight 4-inch-diameter bolts fasten the top head to the body of the reactor vessel.

The head assembly consists of a hemispherical forging and a forged bolting flange per A. O. Smith (AOS) specification 5002 having a minimum tensile strength of 85,000 psi and a minimum yield point of 51,000 psi. The

Fig. 6 Reactor-vessel assembly.

inner surfaces are clad with a 1/4-inch-thick Type 309 Cb stainless steel weld deposit to give the required corrosion protection. The average head thickness is 3-1/2 inches.

The attachments to the head consist of five 3-inch control rod nozzles and four 6-inch flanged access nozzles. Attachment of these nozzles is accomplished by welding into the head forging. The nozzles are forged neck halves per AOS specification 5002, lined with 1/8-inch-thick, Type 304L stainless steel, rolled-in liners.

1.32 Cylindrical Shell. The vessel shell is constructed in accordance with standard A. O. Smith Company multilayer construction and consists of a 1/2-inch inner pressure shell, wrapped with eleven 1/4-inch-thick load-bearing layers for a total wall thickness of 3-1/4 inch.

The inner pressure shell is fabricated from four 3/4-inch-thick single-clad plate halves, rolled and welded to form a cylinder. Plate material is ASTM A-225, Grade B, clad with 1/8 inch Type 304L stainless steel. After welding, the OD was machined to a surface finish of 125 rms and a final wall thickness of 1/2 inch.

The load-bearing plates, carbon steel per AOS specification 1146 A, were wrapped around the inner shell in successive layers. The longitudinal weld seams of each layer were contour ground prior to wrapping the succeeding layers, and advanced 75° from the preceding layer seam to ensure a tight fit of the layers.

Located near the bottom of the shell are six equally spaced 8-inch coolant-outlet nozzles. Each nozzle consists of three parts: (a) a forged neck half which penetrates the vessel wall, (b) a transition piece welded to the neck half, and (c) a thermal shield welded inside the nozzle.

The forged neck halves are 1-1/2-inch-thick single-clad plate, ASTM A-225, Grade B, FBQ backing plate, clad with ASTM A-264, Grade 3, Type 304L 1/8-inch nominal stainless steel and welded into the vessel shell. The shell area around the nozzles is step-wrapped with five additional layers of 1/4-inch plate for reinforcement.

Welded on the end of each nozzle is a 2-inch section of 8-inch Schedule 160, ASTM A-312, Type 304L stainless steel pipe. This section is provided for making the transition from the forged carbon steel neck half to the Type 304L stainless steel primary-coolant piping. A 1/8-inch-thick, Type 304L stainless steel thermal shield is welded inside each nozzle to reduce the temperature shock at the transition weld area.

Two feet below the vessel bolting flange are six 4-inch flanged access ports and a 4-inch nozzle. The six access ports are of the same construction as the top head access ports, while the nozzle is of the same construction as the coolant-outlet nozzles.

Two iron-constantan thermocouples in thermowells, located just above the core and spaced 180° apart, penetrate the vessel shell for measuring the temperature of the primary coolant leaving the core.

Sixteen iron-constantan thermocouples are embedded in the vessel wall at varying depths and extending the length of the core area. These thermocouples were installed to furnish information regarding the temperature distribution in the vessel wall.

1.33 Bottom Head Assembly. The bottom head assembly consists of a multiple plate spherical sector, a bottom head forging, a thermal shield support ring, and a forged flange for bolting the bottom tee to the head assembly.

The multiple plate spherical sector, 3-1/4 inches thick, is made up of four pie-shaped segments formed and welded together into a hemisphere. The inner layer is 1/2-inch-thick clad plate; the balance of the head thickness is composed of 1/4-inch-thick plates conforming to AOS specification 1146 A.

The head forging and flange are forged steel, ASTM A-105, Grade II, modified with a 1/4-inch-thick Type 309 Cb stainless steel weld deposit on the inside surfaces.

1.34 Bottom Tee. The flanged tee bolts to the reactor vessel bottom head and serves as an entrance for the primary-coolant water from the two primary-piping loops. This tee is flanged to permit removal for inspection inasmuch as the thermal stresses and shock will be greatest in this area.

The tee is of forged construction, 54 inches long, 18-7/8 inch ID with a 3-inch wall thickness. The material is ASTM A-105, Grade II modified with a 1/4-inch-thick Type 309 Cb stainless steel weld deposit on the inside surfaces.

Two 3-inch access ports and a 2-inch access port are welded into the bottom side of the tee. The nozzles are forged necks, per A. O. Smith Company Specification 5002, clad with a 1/8-inch-thick, Type 304L stainless steel rolled-in liner.

1.35 Top Head Closure. Top head closure of the vessel is effected by two concentric gaskets between the mating faces of the top head flange and the vessel bolting flange. The gaskets are spiral wound Flexitallic, Type 304L with Canadian-asbestos filler, each with a 3/4-inch face. A 1/4-inch vent, from the annulus between the gaskets, is brought out between the bolt studs for monitoring intergasket leakage.

1.36 Bolting Assembly. The top head is bolted to the vessel by means of twenty-eight 4-inch nominal-diameter, high alloy carbon steel stud bolts (SAE-4140), tapped into the vessel flange and extending through the top head flange. The top head is retained by special cap nuts. The bolt and cap nuts are center-drilled to permit the insertion of depth micrometers for pretensioning of the bolts.

1.4 Vessel Support

The vessel support is designed so that movement in both the vertical and horizontal directions, due to thermal expansion of the vessel, is free and

unrestrained. The support, or cradle, consists of six 1-1/2-inch-thick curved-plate segments bearing on the underside of the bottom head. The supporting segments, properly gusseted and reinforced around the outside with a horizontal plate, are welded to a circular horizontal base plate which is bolted to an I-beam framework at an elevation approximately 11 ft above the floor of the reactor pit.

Four equally placed, vertical sway braces are provided at the top of the vessel to prevent rocking of the vessel in operation. The sway braces are spring-loaded supports anchored to the reactor pit wall which bear on the circumference of the vessel bolting flange.

1.5 Shop Hydrostatic Test

A hydrostatic pressure test was conducted with the vessel in the horizontal position at one and one-half times the design pressure. Prior to the test, 78 strain gauges were applied to the outside surfaces of the vessel, and 30 gauges were applied to the bottom tee forging. Wherever possible, each strain gauge had a gauge located in a similar stress location so their readings could be compared. After application of the gauges, several pressure cycles were conducted to stabilize the gauges and to check gaskets for proper seating. A water column installed on the suction side of the test pump was used to detect any unexpected yielding of the vessel during these initial pressure cycles. The hydrostatic test was conducted by increasing vessel internal pressure in steps up to 3750 psig and then dropping the pressure in the same manner. The final pressure was held for 8 hr at 3750 psig.

Strain gauge readings were taken at each pressure step. From these readings, an evaluation was made of the strains which existed at the design pressure of 2500 psig and the test pressure of 3750 psig. With few exceptions, strain gauge readings, for similar locations, were comparable for all pressure steps. This indicated reliability of gauges and uniformity of stresses. The maximum stress recorded for the vessel was 24,100 psi at 3750 psig and 16,000 psi at 2500 psig. The maximum stress recorded for the tee was 30,000 psi for 3750 psig and 20,160 psi for 2500 psig. The overall length and circumference of the vessel were measured at zero psig and, again, at the full-test pressure of 3750 psig. The vessel grew 1/8 inch in circumference and 1/16 inch in length while under pressure.

1.6 Cleaning Procedure

The interior surface finish of the vessel was specified to be 125 rms or better in order to eliminate crevices or pockets where residue or radioactive material could lodge. All grinding, polishing, and buffing required to obtain such a finish was done in such a manner so as to prevent contamination of the stainless steel surfaces.

All interior surfaces of the vessel which are in contact with the primary coolant water must be free of oil, grease, and other foreign materials. The surfaces were first cleaned with a detergent solution (150 lb of trisodium phosphate to 500 gal of 140°F water) to remove oil, grease, and other lubricants. The surfaces were then pickled with a 125°F solution of 10 percent by volume of nitric acid (70 percent) and 2 percent by volume of hydrofluoric acid (60 percent). After draining and rinsing, a copper sulphate test was made for the detection of metallic iron on the stainless steel surfaces. Upon satisfactory

completion of the copper sulfate tests, the vessel was then rinsed and dried, and all openings were sealed for shipping. The final cleaning procedure was done in the field after complete assembly and testing of the system.

1.7 Insulation

The outside of the vessel shell, bottom head, and the bottom tee are covered with 4 inches of foamglass insulation to reduce system heat losses. The insulation, in block form, is attached to the vessel by means of metal bands and wire mesh. Foamglass was used not only for its insulating properties, but also because it is relatively unaffected by radiation.

1.8 Radiation Shielding

The cylindrical portion of the reactor vessel is surrounded by a 6-inch-thick lead shield to provide biological shielding for operating and maintenance personnel after shutdown. The shield extends from the bottom head seam of the vessel up to the underside of the vessel bolting flange.

The shield consists of a single course of stacked 2- by 6- by 6-inch lead bricks, curved on all sides to eliminate beam holes and to ensure proper fit of one brick against the other. The bricks are fabricated of corroding lead, ASTM B-29, with a maximum silver content of 0.0035 percent. Special lead brick bridging segments are used where the primary-coolant piping and access nozzles penetrate the shield. Lead wool is compacted into gaps surrounding the nozzles and piping.

An air gap exists between the shield and the vessel insulation to provide cooling air for the inside surfaces of the shield. The temperature is not expected to rise sufficiently to cause sagging of the shield; but in the event this should happen, fans will be installed to pull cooling air through the gap.

An opening, approximately 3 by 3 ft, has been installed in the shield opposite the core to facilitate access to the thermocouples and strain gauges attached to the vessel wall. The opening is reinforced with structural steel to permit removal of bricks from the shield without having to unstack the entire shield. Support for the shield is provided by an extension of the vessel I-beam support structure.

1.9 Thermowells

As originally constructed, the Spert III primary coolant system contained eighteen thermowells. Eleven were commercially supplied wells with 9-inch immersion lengths, two were commercially supplied wells with 6-inch immersion lengths, and the remaining five were field-fabricated with 8-inch immersion lengths. During a mechanical disassembly of the Spert III core structure, the immersion section of a field-fabricated thermowell was found in the bottom of the reactor vessel. An analysis [6] of this thermowell failure indicated that the failure resulted from fatigue. The fatigue was probably due to harmonic excitation of the thermowell caused by the primary pump impeller frequency. Consequently, replacement thermowells were installed in a total of 13 locations, as shown in Table I. Eleven of these were commercial wells having a 5-inch immersion length, 3/4-inch tip diameter, 1-inch base diameter, 1-inch NPT mounting thread, and a 1/2-inch internal thread. The remaining two were commercial wells having a 6-inch immersion length and other dimensions identical to the well described previously.

TABLE I

THERMOWELL LOCATIONS AND REPLACEMENTS

Location	Original	Replacement
East Reactor Outlet	Commercial 9-inch	Commercial 5-inch
West Reactor Outlet	Commercial 9-inch	Commercial 5-inch
Pressurizer Vessel	Commercial 9-inch	None
East Heat Exchanger Inlet	Commercial 9-inch	Commercial 5-inch
West Heat Exchanger Inlet	Commercial 9-inch	Commercial 5-inch
East Primary Pump Suction	Commercial 9-inch	Commercial 5-inch
West Primary Pump Suction	Commercial 9-inch	Commercial 5-inch
East 16-inch Flow Tube	Commercial 9-inch	Commercial 6-inch
West 16-inch Flow Tube	Commercial 9-inch	Commercial 6-inch
East Reactor Inlet	Commercial 9-inch	Commercial 5-inch
West Reactor Inlet	Commercial 9-inch	Commercial 5-inch
East 8-inch Flow Tube	Commercial 6-inch	None
West 8-inch Flow Tube	Commercial 6-inch	None
Torus	Field-Fabricated	Commercial 5-inch
East Heat Exchanger Outlet	Field-Fabricated	Commercial 5-inch
West Heat Exchanger Outlet	Field-Fabricated	Commercial 5-inch
East Primary Pump Discharge	Field-Fabricated	Removed and Plugged
West Primary Pump Discharge	Field-Fabricated	Removed and Plugged

The two 8-inch flow tube wells were not replaced since they were originally of the commercial type with a 6-inch immersion length. The pressurizer vessel thermowell was not replaced since it was not subject to coolant flow. The two primary-pump discharge wells never had been used and were plugged.

1.10 Experimental Thermocouples

Seven stainless steel sheathed, chromel-alumel thermocouples connected in parallel are located four feet below the core to measure the temperature of the primary coolant entering the core. The core exit temperature is measured with four similar thermocouples. These four thermocouples are attached to the outside edge of the upper grid and extend into the annulus between the core and reactor vessel.

2. REACTOR CORE COMPONENTS

The core volume is that part of the reactor internal structure that is confined by the upper and lower grids and the core skirt. Vessel size and geometrical considerations limited the actual available volume for fuel assemblies to sixty-eight (68) 3- by 3-inch-square sections. Since the core skirt is cylindrical, the space between the square fuel assembly positions and the cylindrical core skirt must be occupied by appropriately shaped filler pieces.

2.1 Fuel Assemblies

The oxide core selected for investigation in the Spert III reactor is composed of cylindrical stainless steel fuel rods, supported in stainless steel assemblies. A cross section of the core is shown in Figure 7. As may be noted from the figure, location of a cruciform poison (transient) rod in the geometric center of the reactor necessitates fuel assemblies of two sizes in order to maintain symmetry of the lattice. The larger of the two assemblies is designated as the 25-rod fuel assembly and the smaller designated as the 16-rod fuel assembly.

Fig. 7 E-core cross section (64 assembly loading).

2.11 25-Rod Fuel Assembly. Figure 8 is a pictorial cross section of the 25-rod fuel assembly. It is a stainless steel unit consisting of fuel rods, fuel can, two intermediate grids, upper and lower fuel grids, and upper and lower end-box adaptors.

Fig. 8 25-rod fuel assembly.

The fuel rods are 40.8 inches long, excluding end plugs, by 0.466 inch in outside diameter with a wall thickness of 0.020 inch. The materials of the tubing and end plugs are Types 348 and 347 stainless steel, respectively. The fuel is in the form of uranium dioxide pellets, 4.8 weight percent enriched, 0.420 inch in diameter. Each of the fuel rods contain 38.5 grams of U-235 with an active fuel length of 38.3 inches which leaves a 2.5-inch expansion space at the top of the fuel rod. In this expansion space, a compression spring is positioned to keep the fuel pellets in place. The gas in the fuel rods is 95 percent helium to improve the heat transfer across the gas gap.

The 25 fuel rods are spaced on a 0.585-inch-square pitch in the fuel can by upper and lower fuel grids and two intermediate grids. The upper and lower fuel grids are designed so that 9 rods are removable. The intermediate grids are spaced 13 inches and 26-1/2 inches above the lower fuel grid to minimize vibration and thermal bowing of the fuel rods. A total area of 120 square inches in the form of slots has been removed from each stainless steel fuel can. These slots are so arranged that by proper orientation of the fuel assemblies, the slots in adjacent assemblies can be either aligned or blocked. The alignment of the slots allows core cross flow.

Upper and lower stainless steel end boxes are attached to the fuel can by welding. The function of the end-box attachments is to adapt the fuel assembly to the supporting grids and hold the element in the reactor core. The lower end box is equipped with a Bellville spring which permits thermal expansion and eliminates compressive stresses when the upper grid is seated. A key is attached to the lower end box which fits into a slot in the lower grid ensuring proper positioning and orientation of the element. The overall dimensions of the fuel assembly are 2.975 inches wide by 2.975 inches thick by 52.750 inches long. The average heat transfer surface, based on the active fuel length, is 1402 in.² per assembly, and the nonmoderator-to-moderator ratio is 1.03 for a 3-by 3-inch cell.

2.12 16-Rod Fuel Assembly. The 16-rod fuel assembly is similar to the 25-rod fuel assembly. The fuel rods and the fuel rod pitch are the same as used in the 25-rod fuel assembly. The assembly contains 16 fuel rods, 4 of which are removable. The overall dimensions of the fuel assembly are 2.476 inches wide by 2.476 inches thick by 52.750 inches long. The average heat transfer surface is 897 in.² per assembly, and the nonmoderator-to-moderator ratio is 0.942 for the assembly.

2.2 Control Rods

There are eight control rods located in the core, two of which are provided in each quadrant. The two rods are joined by a yoke and driven by a single drive mechanism. The rods are the fuel-poison type, containing fuel in the lower section and a poison (neutron-absorber) material in the upper section. A control rod assembly is shown in Figure 9.

Fig. 9 Control rod assembly.

The fuel section consists of 16 fuel rods containing a total of 616 g of U-235. The fuel rods are the same as those used in the fuel assemblies. They are spaced on a 0.585-inch-square pitch to give a flow area of 2.80 in.². The average heat transfer area is 897 in.² per control rod, and the overall dimensions of the fuel section are 2.496 inches wide by 2.496 inches thick by 45-41/64 inches long. The 16 fuel rods are brazed into position to prevent rotation.

The poison section is a square box constructed of 0.186-inch-thick Type 18-8 stainless steel plate containing 1.35 weight percent boron-10. The overall dimensions of this section are 2.496 inches square by 46 inches long.

Flux suppressors are provided in the region between the end of the poison box and the top of the fuel rods. There are 12 suppressors in each control rod assembly. Six of the suppressors are 2.184- by 1- by 0.030-inch plates while the other six are 2.340- by 2-23/32- by 0.030-inch plates. These suppressors are made of the same type material (boron-10 stainless steel alloy) as the poison section. Eight rubbing pads of Haynes No. 40 alloy, located four each at the junction of the fuel and poison sections and on the lower end box, position and guide the control rod through the housing.

2.3 Transient Rod

The transient rod is a cruciform-shaped rod with blades 5-1/8 inches wide by 3/16 inch thick. The upper section, which is normally in the core, is 56 inches long and constructed of 18-8 stainless steel. The lower section, which is the absorber section and normally extends below the core, is 38 inches long and constructed of 1.35 weight percent boron-10 stainless steel. The two sections are welded together and machined smooth. Haynes No. 40 alloy bushing pads, 4-5/8 in. square by 0.09 inch thick, guide and position each blade in the transient rod guide. The yoke section, constructed of 347 stainless steel, is pinned to the upper section of the rod and forms the coupling between the transient rod and transient rod drive. See Figure 10 for the transient rod assembly.

Fig. 10 Transient rod assembly.

2.4 Core Filler Pieces

Experimental considerations necessitated a core structure in which the number of fuel assemblies could be varied to meet the versatile operating

conditions set forth in the Spert III program. Dummy spacer assemblies of the same overall dimensions as the 25-rod fuel assemblies are therefore required to fill the lattice positions unoccupied by fuel assemblies in order to maintain hydraulic equilibrium for various core sizes. These assemblies are designated as Type 4F filler pieces.

Due to the square geometry of the fuel assemblies, it is possible only to approximate the cylindrical geometry of the core skirt in assembling a core. Spacer assemblies, designated as Type 1F, 2F, and 3F filler pieces, are provided to complete the cylindrical geometry of the core skirt and, thus, maintain hydraulic balance through the core.

The heat generation in all the spacer assemblies is relatively small and, therefore, requires a low coolant flow rate. This is accomplished by orificing the end boxes.

The filler pieces are of similar construction except for geometry. The units consist of three main components, an upper end box, filler box, and a lower end box, each of 18-8 stainless steel. The flow-restricting orifices are located at the entrance and exits of the filler box in the lower and upper end boxes. In all filler pieces except the 1F, relief holes are drilled through the upper end box above the orifice plate to allow coolant flowing on the outside of the spacer to exit via the end box and out through the upper grid. These holes are sized to maintain flow equilibrium with the adjacent fuel assemblies.

The filler box is constructed to the size and shape required. The upper and lower end boxes are then welded to the filler box, and the unit is machined to the final dimensions. A Bellville spring and key are attached to the lower end box to complete the assembly. See Figures 11 and 12 for a pictorial assembly of the four filler piece types.

Fig. 11 1F and 2F filler pieces.

Fig. 12 3F and 4F filler pieces.

2.5 Thermal Shields

Due to the high operating pressure of the Spert III reactor, the pressure vessel was made as small as possible to reduce the vessel cost. However, gamma-heat generation within the vessel walls becomes a more serious problem as the vessel walls approach the core. Thermal shields were provided to attenuate the gamma radiation and protect the vessel wall. The thermal shields were designed as four concentric shells with a water annulus between each shell and are positioned between the core skirt and the vessel wall. The annuli were selected to approximate as nearly as possible a uniform temperature rise through the thermal shields. Iron-constantan thermocouples sheathed in 304 stainless steel are embedded in the thermal shields to measure the equilibrium gamma-heat generation during power operation of the reactor. Also, pitot tubes are installed in flow annuli between the shields to measure the coolant distribution through the thermal shield region. The thermal shield layout depicting the position of the attached thermocouples and pitot tubes is shown in Figure 13.

Fig. 13 Plan view of thermal-shield instrumentation.

The thermal shields, constructed of centrifugally cast Type 304L stainless steel, are labeled inner thermal shield, intermediate thermal shield, reactor thermal shield, and outer thermal shield. The reactor thermal shield extends vertically from approximately the top of the core to the base of the vessel where it is secured to a support ring which is welded to the vessel wall. The entire core structure, the core skirt, the inner, intermediate, and outer thermal shields are suspended from the reactor thermal shield. The inner, intermediate, and outer thermal shields extend vertically for approximately the active length of the core.

Coolant flow is proportioned to annuli between the core skirt, thermal shields, and the vessel wall by aid of a flow control scoop and spaced flow relief ports in the support thermal shield. The flow control scoop is mounted to the top of the support thermal shield.

2.6 Internal Support Structure

2.61 Reactor Core Flow Skirt. The reactor core flow skirt is a Type 304L stainless steel cylinder, 32 inch ID by 33.25 inch OD by 54 inches long suspended from the reactor thermal shield by six equally spaced ears located between the flow scoop and the top grid. Thus, in addition to supporting the core skirt, the ears provide a passageway for cooling water to the inner thermal shield annuli. A lower grid support is mounted inside the flow skirt to permit the testing of cores with fuel length up to 45 inches.

Design conditions for the flow skirt included a 5200-lb loading of fuel and grid weight in addition to the force exerted by the springs on the fuel assemblies. This force is estimated to be a total of about 2000 lb. During flow, there is essentially no load on the flow skirt or the lower grid support at a pressure drop through the core of ≈ 9 psi.

The thickness of the flow skirt was based primarily on the gamma heating expected and the cooling water available to maintain the thermal stresses within reasonable limits. The maximum differential pressure across the flow skirt is approximately 50 psi at full-flow conditions.

2.62 Lower Grid. The lower grid consists of a forged AISI Type 304L plate 3 inches thick and 31.970 inches in diameter. Sixty-four 2.505-inch diameter through holes are provided for the end boxes of the 25-rod fuel assemblies and core fillers. Fifty-six of these holes have four keyways to permit alignment of the fuel in any of four directions. The remaining eight holes have only one keyway. These holes are used for odd-shaped core filler pieces that cannot be oriented in more than one position. Eight holes 2.130 inches in diameter are provided for the four 16-rod fuel assemblies and the four outside core filler pieces. The holes for the four 16-rod assemblies also have four keyways for fuel alignment, and the four filler-piece holes have only one keyway. Eight 2.596-inch-square through holes are provided for passage of the control rods. Recessed mounting holes for the control rod guide tubes are provided around the control rod holes on both the top and bottom surfaces.

The center of the grid contains a cruciform through hole 5.835 inches long in each direction and 0.470 inch wide. The ends of the slots are also the center of 1-inch-diameter, water-coolant holes. Recessed mounting holes for the upper transient rod guide tube are provided on the lower surface for the transient rod lower guide tubes. Seventy-six 1-inch holes, eight 9/16-inch holes, and four 7/16-inch-diameter through holes are provided to feed cooling water between the fuel assemblies.

The total design load on the lower grid consists of 5000 lb of fuel and accessories and 2000 lb of spring force from the fuel assemblies for a total of 7000 lb when the upper grid is in place. This load lessens with upflow and becomes essentially negligible with pressure drops across the core of 9 psi or more. Assembly of the lower grid, flow skirt, and guide tubes is shown in Figure 14.

2.63 Guide Tubes. The control rods and the transient rod are guided in special tubes to prevent damage to adjacent fuel assemblies and to help control the velocity of coolant past the control rod fuel and the adjacent fuel assemblies. These guide tubes are made of upper and lower sections. The upper sections

Fig. 14 Lower grid flow skirt and control rod guides.

are located in the core area extending from the recessed mounting in the lower grid into the upper grid. The lower sections attach to the bottom face of the lower grid and extend below the core. All lower guide tubes are fastened to at least two other guide tubes by means of angle brackets. The angle brackets serve also as straightening vanes for the incoming coolant water. Stops, consisting of strips riveted to each inside surface of the lower guide tubes, are provided in the bottom end of the lower guide tubes to allow the rods to rest in position if uncoupled from the drive rods. The guide tubes are made of Zircaloy-2.

2.64 Upper Grid. The upper grid structure retains and/or positions all of the elements in the core region. It is machined from cast equivalent of Type 304L stainless steel which was selected for its corrosion resistance and welding characteristics. The grid is 7 inches thick and 42 inches OD.

Five types of holes are provided in this grid in addition to square through holes for the control rod and transient rod guide tubes. For convenience, these holes have been numbered Types I through V. The Type I holes, of which there are 40, are for positioning and holding the upper ends of assemblies in "non-removable" locations. "Nonremovable locations" is a term applied to those locations where fuel assemblies cannot be removed without first removing the upper grid. The 13 Type II holes permit the insertion and removal of the 25-rod fuel assemblies without removing the grid and are designated as "removable positions". The upper ends of the fuel assemblies are held and positioned by retainers which fit in the hole and are, in turn, held in place by lugs on rotational locking collars mounted in adjacent holes. The six Type III holes are for positioning and holding the upper end box of the two 16-rod fuel assemblies in the nonremovable locations, and four periphery spacer assemblies with Type 16 rod end boxes. The 11 Type IV holes are for positioning and holding the upper end box of 25-rod fuel assemblies in nonremovable locations and, in addition, are used to mount the locking collars for holding the retainers in adjacent removable positions. The two Type V holes permit the removal and insertion of the 16-rod assemblies without removing the grid.

Grooves are milled in the under side of the grid on the center line of each fuel assembly row to permit steam and/or gases to be vented from the upper end of the core. Forty-eight equally spaced 1/2-inch-diameter holes are provided on a 34-1/2-inch-diameter circle to provide cooling in the heavy section of the grid. Also provided on a 34-1/2-inch-diameter circle are 48 equally spaced lapped holes for mounting instrumentation brackets. Figure 15 is a photograph of the upper grid structure.

Fig. 15 Upper grid structure.

V. COOLANT AND PRESSURIZATION SYSTEMS

A schematic flow diagram of the Spert III coolant systems has been shown previously in Figure 2. Pressurized water in the primary-coolant system serves as the heat-transfer medium to remove heat generated in the reactor core and reject it through heat exchangers to the secondary system. Design heat removal capacity of the primary system is 60 MW, and the design maximum-operating pressure, temperature, and flow are 2500 psi, 650°F, and 20,000 gpm, respectively. The system is maintained under pressure by an electrically heated pressurizer vessel. The secondary-coolant system supplies water to the secondary side of the heat exchangers for the generation of steam which is released to the atmosphere. Limited deionized water storage capacity restricts the maximum operating time of the coolant systems to 30 min at the design heat removal rate of 60 MW. Detailed descriptions of the coolant systems and their component parts follow.

1. PRIMARY-COOLANT SYSTEM

1.1 General Description

The primary-coolant system consists of the pressurizer vessel and two primary-coolant loops, each of which contains two pumps, a heat exchanger, the necessary equipment for flow and temperature control, and a number of drain and vent valves for draining and venting the various components of the system. Primary coolant leaves the reactor through six radial 8-inch nozzles near the base of the reactor vessel. The 8-inch nozzles discharge into a 12-inch header ring which surrounds the reactor and, in turn, discharges into the two 16-inch primary-coolant loops. The coolant in each loop then flows to a heat exchanger where the flow is divided, part passing through a 10-inch line to the exchanger with the balance bypassed through a 16-inch line around the exchanger. Division of the flow in each loop is accomplished by means of two control valves, one in the heat exchanger inlet line and one in the bypass line. The heat exchanger line and the bypass line are rejoined in the process pit beyond the exchanger, and the coolant flow continues on to the primary pumps. Two 5000-gpm pumps are installed in parallel in each loop. Coolant flows through the pumps and on through either of two flow measurement and control stations installed in parallel in each loop. In order to obtain a better accuracy for widely variable flow rates, one station contains an 8-inch flow measurement tube and control valve and the other contains a 16-inch tube and valve. Capacities are 200 to 2200 gpm and 2000 to 11,000 gpm for the 8-inch and 16-inch tubes, respectively. After leaving the flow valves, the coolant flows to a flanged tee at the base of the reactor vessel where the two loops rejoin and enter the reactor to complete the circuit.

Pressure in the reactor vessel and coolant loop is maintained and controlled by a pressurizer system. This system consists of a pressurizer vessel connected to the reactor outlet piping by a 4-inch static water leg. The pressure is regulated by automatically controlled immersion heaters which maintain a constant pressure steam dome in the pressurizer vessel. A pressure control valve plus five safety relief valves serve to protect the primary system from excess pressure.

The primary-coolant system is initially filled with deionized water by means of the west secondary-coolant pump. After initial filling, makeup water is supplied by means of a 5-gpm high pressure makeup pump. The excess water during heat-up is discharged from a connection in the reactor inlet piping through a control valve to the atmosphere. The addition of makeup water and blowdown of excess water are controlled from the pressurizer liquid level controller in order to maintain a near constant volume of water in the primary system.

The reactor and pressurizer vessels are mounted in fixed positions on structural steel. All other equipment and piping are mounted on spring supports and bearing plates to allow movement as the system expands and contracts with heating and cooling. Specifications for all piping and equipment in the primary system required design for operation at 2500 psi pressure and 650°F temperature.

1.2 Pressurizer Vessel

The pressurizer is used to produce and transmit a pressure greater than the primary-system saturation pressure to the primary system, thus preventing boiling of the coolant. During normal operation, the electrical heaters in the pressurizer are submerged in water. By heating the water, a steam dome is created in the upper half of the vessel which serves as a cushion to absorb pressure surges in the primary system.

The original pressurizer vessel in the Spert III system failed on October 26, 1961. Subsequent investigations [7] led to the conclusion that because of an error in the indication of the existing level-control instrumentation on the vessel, it was possible for one or more of the electric heaters to operate above the steam water interface thus causing superheating of the steam and producing overtemperature in the upper half of the vessel. The failure mechanism was third-stage creep at elevated temperature.

A new pressurizer vessel has been installed in the system. The pressurizer was reinstrumented to eliminate the deficiencies in the original design. A description of the pressurizer vessel and instrumentation follows. The instrumentation is also described in Section VI.

The pressurizer is a 2-ft 9-inch-ID by 15-ft 2-11/16-inch- (overall) high vessel of ASTM A-212, Grade B, FBX, with nickel-clad (ASTM B-162) in accordance with ASTM A-265. The fittings are of the same material as the vessel. The vessel was designed for 2750 psig and 700°F service. It was fabricated by the Chicago Bridge and Iron Company at Birmingham, Alabama. A sketch of the vessel is shown in Figure 16.

The vessel walls are 3.15 inches thick including the 0.100-inch nickel cladding thickness. The vessel top is a 2:1 axis-ratio elliptical head with a minimum thickness of 2.9 inches including the 0.100-inch cladding and contains a bolted flange access port of 16 inches ID. The bottom head is a 2:1 axis-ratio elliptical dished head. Minimum thickness of the dished head is 2.9 inches including the 0.100-inch clad thickness. Vessel length between the top and bottom head weld lines is 11 ft 5-7/8 inches. Total vessel weight is approximately 13 tons. Horizontal movement is prevented by two sway braces connecting to the vessel near the top.

Fig. 16 Sketch of pressurizer pressure vessel.

zation. The bottom nozzle connects to a 4-inch line which is normally open during operation through a 4-inch piston-operated valve and connects to the reactor outlet piping.

The top head of the pressurizer contains a four-inch nozzle. For operation of Spert III as a boiling water reactor, this nozzle can be used to connect the pressurizer to the reactor through a 4-inch line equipped with a blind flange. Operation of Spert III as a pressurized water reactor requires this pressurizer-reactor connection be closed. Also connected to the 4-inch nozzle and pipe arrangement are the pressurizer safety relief valves, rupture disc, and a 2-inch line to the pressure control valve. There are two safety relief valves in parallel which are set to operate at 2850 psi. A rupture disc is located upstream of each safety relief valve and is set to relieve at 2680 psi at 700°F.

The liquid level control system is attached to the pressurizer vessel by three 1-inch nozzles. This liquid level control system consists of two units. One measures the overall vessel level with a range of 12 ft while the second measures the level in the top half of the vessel with a range of 6 ft. Essentially, all control functions are on the 6-ft range with the exception of the two lower heaters which are interlocked with the 12-ft range system.

Sixteen 480-volt, stainless steel sheathed, hairpin-type electrical heaters are installed horizontally in the lower half of the pressurizer. The heaters are mounted through 3-inch, 2500-psi flanges. Each heater has a capacity of 12 kW giving a total heating capacity of 192 kW. The heaters are controlled by the pressurizer pressure-control system and are wired for step control so that the upper six heaters turn off automatically at a pressure 25 psi below the control point; four heaters below the upper six shut off at 15 psi below the control point; and the next four lower heaters shut off at 10 psi below the control point. The two heaters at the bottom of the vessel shut off on the control point. The pressure is maintained at the control point by the reverse of the above procedure, ie, two heaters turn on if the pressure drops 5 psi below the control point, then four additional heaters turn on if the pressure continues to drop to 15 psi below.

A four-inch nozzle is installed in the bottom head of the pressurizer and is used to connect the pressurizer vessel to the primary system. This nozzle is provided with a thermal shield to reduce any thermal shocks to the lower head during pressurization or depressurization.

The three 1-inch nozzles are located 1 ft 2-1/2 inches, 6 ft 10 inches, and 13 ft 7/8 inch above the bottom head and connect to the liquid level control system. The distance between the upper two nozzles is 6 ft 2-7/8 inches and the upper and lower nozzles, 11 ft 10-3/8 inches. A differential pressure cell is connected between the upper two nozzles. The signal from a differential pressure cell drives the liquid level recorder-controller. This controls the blowdown valve when the water level is 2.5 ft above the uppermost heater, controls the makeup pump when the water level is 1.5 ft above the uppermost heater, and shuts off the upper two banks of heaters when the level is below 1 ft (3 ft above the uppermost heater). The signal sent by the differential pressure cell is the true water level if the pressurizer fluid is at room temperature. However, at elevated temperatures, the recorded pressurizer water level is in error because of the density difference between the reference leg and the pressurizer fluid. A correction can be made from the chart shown in Figure 17. This figure also shows that at a level reading of 1.5 ft for the 6-ft range instrument, the true water level will be within 3 inches of the reading at all temperatures. This, coupled with the previously mentioned controls, should prevent operation of the heaters unless they are submerged. Another differential pressure cell is located between the upper and lower nozzles and is used to observe the water level when the water level is beyond the range of the control cell. This signal is also recorded and is used to cut out the lower two banks of heaters when the water level drops to 4.25 ft above the bottom of the level indicator. Also, the top half of the pressurizer is equipped with a glass sight gauge for a visual level indication.

The vessel is equipped with seven thermowells located at 1-ft 8-inch increments along the axis of the vessel. These thermowells contain thermocouples that are recorded. A high temperature (675°F) from any thermocouple will cut off the pressurizer heaters and provide an alarm at the control center and reactor building. The vessel also contains two sets of embedded thermocouples. These are located approximately 2 ft and 12 ft 8 inches above the bottom plate. Each set consists of three thermocouples on a 3-inch triangular pitch. One thermocouple is within 7/8 inch of the inside surface, another within 1-1/2 inches, and the last within 2-1/2 inches. The temperatures of these thermocouples are recorded and used to determine the thermal gradients in the vessel wall during plant upsets or other thermal shock conditions.

The volume of water in the pressurizer is approximately 410 gallons, and the steam dome volume is approximately 28 ft³. The pressurizer is insulated with 3 inches of 85 percent magnesia insulation. Total calculated heat loss from the pressurizer, including losses through the uninsulated heater flanges and other uninsulated surfaces as well as the insulated surface, is about 50,000 Btu/hr or 15 kW when operating at 668°F. Based upon calculations supplied by the vendor, the pressurizer heating and cooling rates must be limited to 135 °F/hr to minimize thermal stresses in the vessel. The capacity of the electric heaters (192 kW) is such that they give about this heat-up rate. The calculated time to raise the pressurizer temperature from 70 to 668°F is 6-1/4 hr for an average heating rate of 96 °F/hr. When cooling and depressurizing the system, the pressurizer pressure is bled down in small increments over a sufficient time period so that a 100 °F/hr cooling rate is not exceeded.

1.3 Makeup Pump

An inverted, vertical reciprocating pump with a capacity of 5 gpm, at a pressure of 2500 psi, is installed to provide makeup water for blowdown,

Fig. 17 Pressurizer water level -- indicated versus true level.

volume shrinkage, and leakage in the primary system. The pump takes suction from the 12,000-gal deionized water storage tank and discharges into the primary system in the inlet piping at the bottom of the reactor.

The pump was manufactured by the Aldrich Pump Company and is a Model Triplex Direct Flow, with a 7/8-inch-diameter plunger and 2-1/2-inch stroke. It is powered by a 10-hp, Type "CE", explosionproof General Electric motor.

The liquid end of the pump is mostly 304 stainless steel. This includes the solid forging cylinders, stuffing box barrels, and cylinder throats. The plunger gland is 303 stainless steel; the valve seats and plug are 440 C stainless steel; and the plunger is K Monel. The crankshaft is die-forged, precision-ground,

high-grade carbon steel with tapered roller bearings. The connecting rods are drop forged with renewable steel-backed, babbitt-lined, precision crank-pin bearings. The makeup pump has a crankshaft-driven, lube-oil pump to provide positive lubrication to all bearings with the exception of the main bearings which are splash lubricated.

1.4 Primary-Coolant Pumps

Two 500-horsepower, canned-rotor, single-stage centrifugal pumps are installed in each of the two primary-coolant loops. The pumps were manufactured by the Byron Jackson Division of the Borg-Warner Corp. and are designed to deliver 5000 gpm at a temperature of 650°F with a suction pressure of 2500 psi and a discharge pressure of 2585 psi, maximum.

The pump case and impeller are Type 304 stainless steel. The pump volutes are welded into the primary-piping loop to make the pump an integral part of the system. The motor assembly contains the rotor and all bearings and includes a heat-barrier member at the bottom end through which the rotor shaft extends to support and drive the pump impeller. All wetted parts, with the exception of the bearings and the motor liners, are of suitable stainless steels. The bearings consist of two self-aligning sleeve bearings and a pivot-shoe-type thrust bearing employing graphitar inserts against a nitrided surface. The motor-winding cavity is filled with inert gas at approximately one atmosphere pressure. An inconel-stator-liner tube, hermetically seal-welded at either end, isolates the windings from the water-filled cavities. The motor rotor is also protected with an inconel jacket or "can", hermetically seal-welded at each end, which isolates the copper bars and steel laminations within the rotor.

In use, high pressure primary-system water solidly fills the motor rotor and bearing cavities serving as bearing lubricant and also to remove motor heat to a heat exchanger constructed externally to the tubular motor frame. Communication of this water with the water in the pump cavity is permitted only through the close restriction around the shaft immediately above the impeller hub. The thrust-bearing disk is utilized as a pump to circulate the high pressure water within the rotor and bearing cavity. Flow is upward around the rotor, returning from the top of the motor through heat-exchanger tubes to the bottom end immediately above the heat-barrier member and entering through the hollow portion of the rotor shaft to the disk for recirculation. Fresh water from a closed-loop, atmospheric-pressure, cooling-tower system at an 80°F maximum inlet temperature is circulated at a rate of at least 50 gpm through an external cooling jacket around the heat exchanger tubes to remove the motor heat. The cooling rate is sufficient to maintain a water temperature of about 150°F within the motor and bearing cavities even when the pumped liquid temperature is as high as 650°F.

Each pump is mounted on four spring supports which are in turn mounted on a ball-bearing plate support. This arrangement permits the pumps to move freely in any direction as the primary loops expand or contract with heating and cooling. Sway braces are also provided to minimize vibration and whiplash action of the piping. Hydraulic characteristic curves for the pumps are shown in Figure 18, and motor-characteristic curves are shown in Figure 19. These curves are based on data from factory tests on the pumps.

Fig. 18 Hydraulic characteristics for primary pumps.

Fig. 19 Motor characteristics for primary pumps.

The net energy input to the pumps is used to bring the primary system up to the desired operating temperature before nuclear experiments in the reactor are begun. The heat input varies both with pumping rate and primary-system temperature. Curves showing pump power input, heat loss to the motor cooling water, and the net heat input to the primary system versus system temperature for one pump operating at 5000 gpm are shown in Figure 20. In actual practice, it takes about 14 hr to heat the primary system from 70 to 650°F with pump heat at a total flow rate of 20,000 gpm.

Fig. 20 Primary pump heating data for 5000-gpm flow per pump.

1.5 Heat Exchangers

The two heat exchangers, one in each primary loop, serve to transfer the heat generated in the reactor from the primary to the secondary-coolant loops for discharge to the atmosphere. The exchangers were designed for a maximum heat load of 30-MW each and were manufactured by the A. O. Smith Corporation of Milwaukee, Wisconsin. Figure 21 shows the general features of the exchangers. They are horizontally mounted shell and U-tube type with removable shell. The primary coolant flows inside the tubes, and deionized or raw water is used in the shell side as the secondary coolant. The channels or water boxes are made of carbon steel with Type 304L stainless steel cladding. Each exchanger contains a bundle of 367 Type 304L stainless steel U-tubes, 5/8 inch OD by 16-BWG on a 13/16-inch triangular pitch, which provide a heat-transfer area of 1940 ft². Design pressure for the tube side is 2500 psi at 668°F. The shell and shell-side piping are constructed of carbon steel and are designed for a maximum pressure of 75 psi.

Fig. 31 General features of the heat exchanger.

When the primary system is in operation, only part of the primary coolant flows through the exchanger which then rejoins the bypassed stream in a downstream mixing chamber. Division of flow between each exchanger and its bypass piping is accomplished by two motor-operated valves which apportion the flows to maintain the temperature of the mixed streams at the desired reactor inlet temperature.

The exchangers can be operated as evaporators with the steam generated on the secondary side discharged to the atmosphere or as liquid-to-liquid exchangers with the secondary water discharged to the leach pond. Utilization of the exchangers as evaporators restricts the minimum temperature of the primary coolant, leaving the reactor and entering the heat exchangers, to about 270°F. With an inlet temperature of 270°F, the maximum heat removal is about 1 MW per exchanger. With reactor outlet temperatures of 400°F or greater, the exchangers are designed for a maximum capacity of 30 MW each. Deionized water is used as the secondary coolant when the exchangers are operated as steam generators. Deionized water storage tank capacity (12,000 gal) is sufficient to permit evaporative operation at a total power of 60 MW (30 MW per exchanger) for a period of about 30 minutes.

For operation at primary-coolant temperatures below 270°F, the exchangers may be used as liquid-to-liquid exchangers with either deionized or raw water used as the secondary coolant. Maximum heat-removal capacity when operating in liquid-to-liquid service is limited by the flow capacities of the water supply systems to about 5.5 MW per exchanger when using deionized water and 4.4 MW per exchanger with raw water. Storage capacities of the two water systems limit operating time at capacity to about 20 min when using deionized water and 3-1/2 hours when using raw water.

1.6 Primary-System Control and Check Valves

1.61 Heat-Exchanger Temperature Control Valves. Flow through the heat exchanger tube bundles and the 16-inch heat exchanger bypass lines is divided and controlled by motor-operated gate valves. Each loop contains a 10-inch valve in the heat-exchanger inlet line and a 16-inch valve in the bypass line. The valves may be operated automatically or manually from the control center, from the reactor building instrument panel, or with hand wheels on the motor operators.

The valves were manufactured by the Cooper Alloy Corp. [a], Hillside, New Jersey. Originally the materials of the valves were as follows: The valve bodies, gates, and bonnets were made of 304L cast stainless steel, with stellite wearing surfaces on the gate and body seat rings, and in the back seat ring in the bonnet. The stem was made of 304L stainless steel, chrome plated for hardness. The stuffing-box bonnet bushing, lantern ring, and upper gland were made of Haynes Stellite No. 6. There were five rows of mica-impregnated, square asbestos packing between the lower bonnet bushing and the lantern ring and five more rows between the lantern ring and upper gland. The bolted bonnet flanged-type joint was sealed with a Flexitallic gasket style CG.

Operating difficulties were encountered after a short while when the stellite back seat in the bonnet and the stellite bushings in the stuffing box severely gouged the valve stems and eventually caused one to jam in the stuffing box in one of the 16-inch valves. The valves have been modified by changing the material of the bushings and stem overlay. Now the bushings are made of bronze, and the stem overlaid with Colmonoy No. 6.

Any leakage past the first series of rings is drained off through ports cut through the lantern ring to two drain lines welded into the bonnet. Drain connections are also provided at the bottom of each valve body which permit the valves to be completely drained when desired.

The valve motor operators are Limitorque Type SMA manufactured by the Philadelphia Gear Works with magnetic brakes and slide-wire transmitters for remote position indication. The valve operators are powered by high-torque low-starting-current electric motors of squirrel-cage design.

1.62 Flow Control Valves. The primary-coolant flow rate in the primary loops is controlled by motor-operated gate valves. Each loop contains an 8- and a 16-inch valve located at the outlets of the 8- and 16-inch flow-measuring tubes. The 8-inch valves are used for controlling flow rates over the range of 200 to 2200 gpm, and the 16-inch valves are used for a 2000 to 11,000-gpm range. The valves are automatically controlled from the control center to maintain the desired flow rate or may be controlled manually from the control center, the reactor building instrument panel, or with hand wheels on the motor operators. The valves and motor operators are similar to those described for the heat-exchanger temperature control system.

1.63 Check Valves. A 12-inch-swing check valve is installed at the discharge of each primary-coolant pump to prevent recirculation during one pump operation. The valve seats are angled back 15° from vertical so that the gate is slightly open when the pumps are off. This allows a flow of at least 200 gpm to pass by without having to raise the seat and thus permits thermal circulation of the primary coolant when the pumps are off. The check valves are cast stainless steel and were manufactured by the Cooper Alloy Corporation. They are designed for 2500 psi, 670°F operating conditions, and were hydrostatically tested at 5000 psi. All wetted parts are Type 304L stainless steel with deposited stellite wearing surfaces on the seats and disks. Access to the valve internals for inspection or repair is through a threaded cap in the top of the valve body. A seal ring welded to the cap contains leakage past the threads.

[a] Now a division of Manning, Maxwell, and Moore.

1.64 Blowdown Valve. Excess water in the primary system is relieved by an automatically controlled 2-1/2-inch single-port blowdown valve with a 1/2-inch orifice. A 2-inch blowdown line is taken off the 16-inch inlet piping at the bottom of the reactor vessel and run through the reactor pit wall to the blowdown valve which is located on the south wall of the process pit. The valve discharges to a 3-inch line which runs outdoors underground and terminates above ground about 80 ft east of the reactor building. The valve is a Fisher Governor Company diaphragm motor, Design H, single-port valve with all wetted parts of Type 316 stainless steel with stellite wearing surfaces. It is of the "normally closed" type, is designed for on-off service only, and is operated with 50-psi air. Valve operation may be automatically controlled by the pressurizer liquid level control system on the control-center instrument panel or manually controlled from the reactor building or control-center panels.

1.65 Blowoff Valve. Excess pressure is blown off the pressurizer by a blowoff valve located in the upper southwest corner of the process pit. The valve is similar to the blowdown valve discussed above. It also exhausts outdoors through a 3-inch line paralleling the vent line from the blowdown valve. Operation of this valve may be controlled automatically by the pressurizer pressure control system or manually from the control center or reactor building control panels.

1.7 Flow Tubes

The flow in each loop is measured by either of two flow measuring tubes installed in parallel downstream of the primary pumps. Eight-inch tubes are used to measure flows up to 2200 gpm, and 16-inch tubes are used for rates between 2000 and 11,000 gpm. The tubes are Gentile Type D flow tubes with internal piezometer rings manufactured by the Foster Engineering Company of Union, New Jersey. The tubes are constructed of Type 304L stainless steel and are welded into the primary system piping. Differential head capacity relationships may be calculated from the following equations which were based on the factory certified head capacity curves for these tubes.

$$\text{gpm} = 141.5 \sqrt{\frac{H}{\text{sp gr}}} \quad \text{for the 8-inch tubes}$$

$$\text{gpm} = 707.0 \sqrt{\frac{H}{\text{sp gr}}} \quad \text{for the 16-inch tubes}$$

where

gpm = flow in gal per min

H = differential in inches of water

sp gr = specific gravity

1.8 Piping

1.81 General. The primary-loop piping is Type 304L spun-cast stainless steel, Schedule 160 for 10-, 12-, and 16-inch pipe and Schedule 140 for 8-inch pipe. Pipes four-inches and smaller are seamless-drawn 304L stainless steel. All piping connections in the primary system are welded except for flanged connections to two inspection spool pieces and the reactor inlet.

The spun-cast pipe was fabricated by the United States Pipe and Foundry Company at Burlington, New Jersey. Tensile-test specimens and rings for flattening tests were cut from each end of each length of cast pipe. Each casting was 100 percent radiographed and 100 percent dye-penetrant inspected with no repairing of defects permitted. Tension tests were required at room temperature and at 670°F. The following were the minimum physical properties specified:

<u>At Room Temperature</u>				
<u>TS</u>	<u>YS</u>	<u>EL</u>	<u>RA</u>	<u>Charpy V-Notch</u>
70,000	30,000	35.7 percent	40 percent	50 ft-lbs

At 670°F, the tensile strength specified was 51,000 psi, and yield strength 18,300 psi.

As actually fabricated, the manufacturer was able to greatly exceed these requirements. Following is an average representative value of the physicals for all heats poured:

<u>At Room Temperature</u>				
<u>TS</u>	<u>YS</u>	<u>EL</u>	<u>RA</u>	<u>Charpy V-Notch</u>
85,000	40,000	60 percent	60-70 percent	Over 120 ft-lbs

At 670°F, tensiles attained were about 61,000 psi, and yields at about 22,000 psi. 0.03 percent maximum carbon content was specified; the average of all heats ran about 0.025 percent. The pipe was machined inside and out. A 125-microinch finish was specified. Final outside finish was about 125 microinches, but the inside was finished to an almost mirror-like surface. Each length of pipe was hydrostatically tested to 5900 psi at the manufacturer's plant.

Spun-cast pipe was selected for Spert III because of the significant cost advantage of using cast pipe and because of favorable results from metallurgical examinations of the material. Essentially, no operating experience was available at that time on the performance of cast piping in elevated pressure and temperature service. On the basis of the Spert experience, it can be concluded that centrifugally cast stainless steel pipe has given adequate service under cyclic operation and probably is equal to high pressure stainless steel piping fabricated by other methods [5].

The fittings used with the spun-cast pipe were also cast 304L stainless steel and were manufactured by the Cooper Alloy Corporation [a] of Hillside, New Jersey. In general, specification requirements for the fitting were the same as for the pipe, except that certain defects could be repaired by welding. A 150-microinch finish was specified, and 125-microinch finish was attained. Tensile specimens were not cut from the fitting, but were made separately from the metal from the same ladle and poured at the same time as each casting. Results of room temperature physicals were not quite as high for the cast fittings; but at 670°F, the results were about the same as for the cast pipe. The fittings were also hydro-tested at 5900 psi.

[a] Now a division of Manning, Maxwell, and Moore.

The heavy-wall pipe and fittings of cast material were welded together satisfactorily both in the shop and field. A U-type bevel joint preparation proved most satisfactory. The first three closure passes were heliarc'd with the inside of the pipe charged and purged with inert gas. Then, in most instances, the filler passes were made with electric-arc welding, and the final finishing or covering pass was heliarc'd. All welds both in shop and field were 100 percent radiographed, class two or better. Intermittent checks were also made as thought necessary with dye penetrant. Repair of defects revealed by radiographing was permitted.

1.82 Inspection Spools. The first two flanged inspection spool pieces are installed in the east piping loop. The spool pieces may be removed to allow inspection for evidence of thermal stresses, cracking, and corrosion on the inside of the pipe. One spool is located in an 8-inch reactor outlet line and the other is in the 10-inch east heat exchanger outlet line.

1.83 Temperature-Measuring Stations. Three pipe wall temperature-measuring stations are provided to determine the temperature distribution across the pipe walls for use in calculation of thermal stresses. Each station consists of three iron-constantan thermocouples for measuring the temperatures of the outside wall, the center of the pipe wall, and a point 3/16 inch from the inside wall. (The stations are located in the 8-inch inlet line at the southeast side of the reactor, in the 16-inch east reactor inlet line, and in the 16-inch line downstream from the east heat-exchanger mixing tee.)

1.84 Mixing Nozzles. Mixing nozzles are installed in each loop at the junction of the 10-inch heat-exchanger outlet line and the 16-inch heat-exchanger bypass line. The nozzles are formed of 3/8-inch stainless steel plate and are installed in the 16- by 10-inch reducer, extending into the 16-inch tee. In effect, the 3/8-inch plate extends the 10-inch line from the exchanger on through the reducer and into the upstream side of the tee side outlet where the bypass coolant enters. The nozzles are intended to reduce the thermal stresses in the pipe as the annular space between the nozzle and the reducer provides a gradual temperature gradient zone.

1.9 Expansion

As mentioned previously, 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. As the reactor vessel is mounted in a fixed position, the reactor inlet and outlet piping positions are fixed. Therefore, the piping loops "grow" in a northerly direction toward the north end of the process pit as the system is heated. Measurements taken during initial testing of the system indicate that the piping loops grow two inches in length as the system is heated from 70 to 650°F.

1.10 Insulation

All equipment and piping in the primary system are insulated to minimize heat losses. The reactor vessel is insulated with 4 inches of Pittsburgh Corning Corporation Foamglass while the remaining equipment and piping 7 inches in diameter and over are insulated with 3 inches of 85 percent magnesia insulation. Primary piping less than 7 inches in diameter is insulated with 2 to 2-1/2 inches of 85 percent magnesia.

2. SECONDARY-COOLANT SYSTEM

2.1 General Description

The secondary-coolant system consists of two systems. Each system contains a secondary-coolant pump, the shell side of the heat exchanger, the necessary equipment for heat-exchanger level control, and drain and vent valves for the various components of the system. The west system contains an auxiliary line for filling the primary system with water.

The inlets to the secondary pumps are arranged for pumping either deionized water or potable water. Deionized water from the deionized water storage tank enters the pumps through a 4-inch gate valve, and the potable water enters through a removable flanged-spool piece and a 4-inch gate valve. The spools are to be removed and blind flanges installed when using deionized water. This precaution will prevent entry of any potable water into the deionized water system. The pump discharge lines are routed to the respective heat-exchanger shells. These 4-inch heat-exchanger supply lines contain check valves and automatic heat exchange level control valves. In order to protect the secondary-coolant pumps from overheating during periods of low flow or no flow, a 1/2-inch bypass return line, connected to the pump discharge piping, allows some of the water to return to the storage tank through flow-limiting orifices installed in the line. The 1/2-inch shutoff valves which are upstream of the orifices in the return lines must be closed during potable water operation to prevent contamination of the deionized water in the storage tank.

The west loop contains a 4-inch shutoff valve upstream of the level control valve and an auxiliary line for filling the primary system. Shortly ahead of the shutoff valve, the 2-inch primary-system filling line is taken off the 4-inch discharge line. This line contains a shutoff valve, an angle valve, and a check valve and is connected into a 16-inch primary line near the base of the reactor.

The steam generated in the heat exchanger shell is exhausted to the atmosphere through 10-inch vent lines. Flanges are provided at the shell nozzle for installation of rupture discs. The discs may be installed during liquid-to-liquid operation but must be removed during boiling-liquid operation. Moisture separating devices are installed inside the shell to reduce liquid-droplet carry over. During liquid-to-liquid operation, the heated water overflows through 4-inch lines to an outdoor underground drainage header and to the leaching pond. Shutoff valves are installed at the shell nozzles. The heat exchanger shell and secondary-coolant lines are drained to the sump.

2.2 Pumps

Each secondary-coolant system has a single-stage centrifugal pump for supplying cooling water to the shell side of the heat exchangers. The pumps were manufactured by the Pacific Pumping Company Type LSCOM and are designed to deliver 250 gpm against a 200-ft head at 60°F. All wetted parts are of 316 stainless steel. The pumps shafts are equipped with Type nine John Crane Teflon and stainless steel mechanical seals. The General Electric motors are 20 hp, 3500 rpm 440/3/60, drip proof, 40°C, with standard class A insulation. Hydraulic characteristics for the pumps are shown in Figure 22. These curves are based on data from factory tests on the pumps.

Fig. 22 Hydraulic characteristics of secondary pumps.

2.3 Piping and Valves

All lines in the secondary-coolant system to the upstream side of the heat-exchanger level control valves are Schedule 40, seamless or welded stainless steel. The valves, 2 inches and smaller, are 600-lb forged stainless steel, and the valves, 2-1/2 inches and larger, are 150-lb ASA cast stainless steel.

The heat exchanger supply piping from the upstream side of the level control valves, the vent and overflow piping of the heat exchangers, and drain to the sump are Schedule 40, seamless carbon steel. The valves 2 inches and smaller are 600-lb ASA forged steel, and the valves 2-1/2 inches and larger are 150-lb cast carbon steel.

VI. PROCESS INSTRUMENTATION AND CONTROL

1. GENERAL DESCRIPTION

A wide variety of industrial-type instrumentation is used for sensing and controlling plant parameters. These include pressure, temperature, level, flow, and conductivity instrumentation.

The plant instrumentation is designed so the parameters of the system (ie, pressure, temperature, level, and flow) can be controlled either from the reactor building or the control center 1/2 mile away. Because of the distance, electrical transmission is required for transmission of signals to the control center. The signals are transmitted over multiconductor cables between the two areas. Pneumatic instruments at the reactor building are equipped with transducers to convert pneumatic signals to electrical signals for transmission to the control center. A simplified block diagram of the instrumentation is shown in Figure 23. This diagram indicates the process variables to be measured and/or controlled, the type of instrument, and the physical location of the instruments.

763-275

Fig. 23 Simplified block diagram of process instrumentation.

2. PRESSURE

The pressures monitored are listed in Table II. The pressure is controlled by the pressurizer heaters and the "blowoff" valve. The controller regulates pressure by controlling the two lower heaters when the pressure deviates 5 psi from the control point, the four heaters above the lower two at 10 psi from the control point, the next four heaters at 15 psi from the control point, and the upper six heaters at 25 psi from the control point. The controller regulates overpressure by controlling the blowoff valve at 130 psi above the control point. An alarm is set at 2660 psi which also de-energizes the pressurizer heaters. The blowoff valve may be manually operated by a spring return switch.

TABLE II

PRESSURE INSTRUMENTS

Variable Measurement	Range (psi)	Method of Transmission	Alarm	Reactor Building Instrument Panel	Control Center Instrument Panel
Primary Coolant Pump Discharge	0 to 3000	Pneumatic		PI	
Heat Exchanger Inlet Pressure	0 to 3000	Pneumatic		PI	
Heat Exchanger Differential Pressure (Primary)	0 to 20	Electronic		PI	PR
Secondary Coolant Pump Discharge Pressure	0 to 150	Pneumatic		PI	
Reactor Vessel Inlet Pressure	0 to 3000	Electronic		PR 2 to 1500 psi Ranges	PR 2 to 1500 psi Ranges
Reactor Vessel Differential Pressure	0 to 100	Electronic		PR	PR
Heat Exchanger Shell Pressure	0 to 150	Electronic		PI	PI
Primary Coolant Pump Suction Pressure	0 to 3000	Electronic	Variable	PI	PR
Reactor Vessel Pressure	0 to 3000	Electronic	Variable	PI	PR
Pressurizer Pressure	0 to 3000	Electronic	2660 psi	PI 3 to 1000 psi Ranges	PRC 3 to 1000 psi Ranges
Pressure-Process Differential Pressure	0 to 150	Electronic	100 psi	PR	

P = Pressure
 R = Recorder
 I = Indicator
 C = Controller

The pressurizer-process differential pressure is used to ensure that the valve in the connecting line between the pressurizer and reactor system is not closed during operation. In the event that the differential pressure exceeds 100 psi, an alarm annunciates and shuts off the pressurizer heaters.

3. TEMPERATURE

The temperatures monitored are listed in Table III. The process coolant temperature is regulated by automatically or by manually controlling the flow through the heat exchangers.

TABLE III
TEMPERATURE INSTRUMENTS

Variable Measurement	Range (°F)	Alarm	Reactor Building Instrument Panel	Control Center Instrument Panel
Reactor Inlet Primary Coolant Temperature	0 to 800		TR	
Reactor Outlet Primary Coolant Temperature	0 to 800	655°F	TR	TR
Reactor Differential Primary Coolant Temperature	0 to 150			TR
Heat Exchanger Primary Coolant Inlet Temperature	0 to 800			TR
Heat Exchanger Primary Coolant Outlet Temperature	0 to 800		TR	TRC
Heat Exchanger Primary Coolant Differential Temperature	0 to 150			TR
Heat Exchanger Secondary Coolant Temperature	0 to 800		TR	
Secondary Coolant Pump Discharge Temperature	0 to 800		TR	
Primary Pump Bearing Temperatures	0 to 800	195°F	TR	
Pressurizer Steam Region Temperatures	0 to 800	675°F	TR	TR
Pressurizer Water Region Temperatures	0 to 800	675°F	TR	TR
Pressurizer Skin Temperatures	0 to 720		TR 7 to 120°F Ranges	
Heat Exchanger	0 to 20 Differential	Variable		TR

T = Temperature
 R = Recorder
 C = Controller

The pressurizer water and steam temperatures are monitored and set to alarm if the temperature exceeds 675°F. At this temperature, the pressurizer heaters are de-energized.

A "ΔT downscale scram" system is used to scram the reactor in case the heat exchanger primary-coolant outlet (mixed) temperature deviates downscale from a preset level. This system monitors the heat exchange outlet temperature and compares it to a predetermined temperature which is set manually. Should the temperature drop by a set amount, the reactor control rods are scrambled automatically.

4. LEVEL

The levels monitored are listed in Table IV. The 6-foot pressurizer level controller regulates the water in the pressurizer by controlling the makeup pump at the 1.5-ft pressurizer level and the blowdown valve at the 2.5-ft level. In addition, a "hi-level" alarm is activated at the 3-ft level and a low level alarm at the 1-ft level. The alarm signals de-energize the two upper banks of heaters. The levels refer to the zero level of the differential pressure cell which corresponds to the top of the heaters.

TABLE IV

LEVEL INSTRUMENTS

Variable Measurement	Range	Alarm	Reactor Building Instrument Panel	Control Center Instrument Panel
Pressurizer Level	0 to 12 feet		LR	LR
Pressurizer Level	6 feet	1 and 3 feet	LR	LRC
Reactor Vessel Level	100 inches		LI	LR
Heat Exchanger Secondary Coolant Level	6 inches	Adjustable	LI	LI
Deionized Water Storage Tank Level	0 to 12,500 gallons	Adjustable	LIC	LI

L = Level
 R = Recorder
 I = Indicator
 C = Controller

A 0- to 12-ft level recorder is used to observe the water level when the water level is beyond the range of the 6-ft pressurizer level controller. This recorder de-energizes the lower two banks of heaters when the water level drops to 4.25 feet above the bottom of the level indicator.

In addition, there is a spring-loaded hand control switch that manually controls the blowdown valve, and a three-position switch that switches the makeup pump from "off" to "auto" to "manual". The manual positions of both switches are spring loaded.

5. FLOW

The flow measurements monitored are listed in Table V. The primary coolant flow is controlled by dissipating the pump head through the flow control valves. Small flow rates, 0-2200 gpm, are controlled by the 8-inch bypass flow valves while larger flow rates are controlled by the main 16-inch flow valves.

TABLE V
FLOW INSTRUMENTS

<u>Variable Measurement</u>	<u>Range</u>	<u>Alarm</u>	<u>Reactor Building Instrument Panel</u>	<u>Control Center Instrument Panel</u>
Main Primary Coolant Flow	0 to 11,000 gpm		FI	FRC
Bypass Primary Coolant Flow	0 to 2,200 gpm		FI	FRC
Secondary Coolant Flow	0 to 250 gpm		FR	
Evaporative Cooler Flow	0 to 300 gpm	Adjustable	FI	
Demineralizer Water Flow	0 to 100 gpm		FR	

F = Flow
R = Recorder
I = Indicator
C = Controller

6. CONDUCTIVITY

The conductivity of the demineralized water is monitored at the demineralizer and the storage tank by a recorder controller. The recorder controller is equipped with a conductivity cutoff control contact and an alarm contact. When the conductivity of the water becomes higher than the recorder setpoint, the cutoff contact activates a valve which stops the flow of the water to the demineralized water storage tank. Also, the alarm is activated.

7. INTERLOCK SYSTEM

In order to minimize the possibility of pressurizer damage, the following instruments are interlocked with the pressurizer heaters in such a way that they must be on and operating before the heaters can be energized:

- (1) All pressurizer level instruments
- (2) Pressurizer heater power recorder
- (3) West loop sixteen point temperature recorder
- (4) Pressure/process differential pressure
- (5) Pressurizer pressure controller.

VII. SPERT III CONTROL AND TRANSIENT ROD DRIVES

1. MECHANICAL DESIGN CRITERIA

Reactor start-up and control are accomplished using four groups of two control rods each. A cruciform transient rod, centrally located in the core, is used to rapidly introduce reactivity in the reactor, thereby initiating power excursions. Complete descriptions of these core components are to be found in Section IV-2.

The active core selected for the initial operation of the Spert III facility was 36 inches in length. However, future testing of other cores of undetermined length was planned. Since, at that time, the maximum expected active core height was 42 inches, it was decided to incorporate provisions in the control rod drives for testing 42-inch cores if the additional length of drive did not compromise the design for the 36-inch active core drive. Subsequent detailed design did, in fact, accomplish this desired feature for the control rod drives. However, due to the desire to keep all of the rod drives essentially mechanically interchangeable and the special requirements for removing transient rod poison from the effective core region with the attendant longer stroke required for the transient rod drive, the possible use of the present transient rod drive for a 42-inch active core is dependent upon a special fuel assembly design and a relaxation of the poison removal distance for the transient rod.

While it was not a design parameter, it was understood from the beginning that the primary function of this project was to perform reactor behavior and safety experiments and that the control rod drive design was to be accomplished with the minimum development work possible using the maximum "off-the-shelf" commercial hardware. Exotic design and radical departure from proven systems were to be avoided if possible.

The following is a tabulation of the criteria for design and operation of the rod drives for the control rods and the transient rods:

(1) Controlled rod movement travel:

(a) Control rod 45 inches

(b) Transient rod 45 inches

(2) Accuracy of controlled rod position indication:

(a) Control rod ± 0.01 inch

(b) Transient rod ± 0.01 inch

(3) Speed of controlled movement:

(a) Control rods -- 3 speeds, approximately 17.4-in./min,
11.5-in./min, 5.6-in./min

(b) Transient rod -- 3 speeds, approximately 17.4-in./min,
11.5-in./min, 5.6-in./min

- (4) Emergency insertion (scram) times as measured from initiation of scram signal to insertion through full travel to contact with shock absorber.
 - (a) Control rod 0.4 sec
 - (b) Transient rod 0.2 sec
- (5) All rod drives must be operable and meet all other design criteria through the full operating range of the reactor system which include operation from 0 psig through 2500 psig, 70°F through 670°F, and no flow through 20,000 gpm upflow and all possible variations and combinations of these process variables.
- (6) "Bounce" of control rods and transient rod after scrambling will not be permitted.
- (7) Leakage of coolant through any seals or connections for the drive units must be limited to a maximum of 5 lb/hr for each drive unit.
- (8) Control rods are to be run individually if desired or synchronized in gang operation.

2. DESIGN SAFETY CRITERIA

Design safety criteria were established to provide as nearly as possible a drive system that would fail-safe both from a mechanical and from a control circuit standpoint under any foreseeable condition that might arise even during a "major disaster" in the reactor building. The following is a tabulation of the various safety criteria:

- (1) After full insertion of the rods, it shall not be possible to raise the rods through any means except by deliberate operator function.
- (2) Under no conditions of rod position or mechanical equipment condition, eg, loss of motor brakes, power, air, gears, or couplings, either singly or in a series of multiple failures, shall the control rods raise without an intentional act of the reactor console operator.
- (3) In the event of loss of external driving forces, sufficient energy must be contained in the drive units to effect a scram.
- (4) Adequate mechanical and/or control devices must be incorporated in the drives to prevent mechanical damage or hazardous reactor operation due to overtravel of the rods in either direction or any other operator controlled function.

- (5) The transient rod drive must not permit an inadvertent scram of the transient rod.
- (6) Positive indication of control rod position must be provided. While it is desirable to know the position of the control rod itself at all times, the mechanical and electrical problems attendant to such a system of indication due to the emergency scram feature and the reactor vessel environment indicated that considerable expense and delay for development would be required. As an alternate, it was deemed acceptable to use secondary indication of the control and transient rod positions during all times of controlled movement by indicating the position of the controlling portion of the drive units in mechanical contact with the control and transient rod parts. During emergency scrams, the mechanical contact would be broken, thus resulting in no indication of control rod position until contact was reestablished or unless the rods were at seat position.

3. DESCRIPTION OF DRIVE UNITS

3.1 Design Bases

In keeping with the original philosophy of designing a reliable drive unit in the shortest possible time and without unduly delaying the overall project, the drive units were mounted externally to the reactor vessel in a vertical position on the top head. Vessel design, recognized as one of the more difficult and time-consuming facets of the overall project, was to proceed concurrently with the design of the drive units. Any other method of providing drive units would have delayed the vessel and/or vessel support design until firm space commitments for the drive units were established. Further, the accessibility of the drive units for maintenance and handling increased the advantage of a top head drive system, especially since the reactor was to be installed in a pit and utilized only a bare minimum of personnel shielding around the reactor vessel.

The requirement for sufficient self-contained stored energy to accomplish an emergency scram indicated a need for a stored energy system capable of being triggered remotely and giving up sufficient energy to effect a scram within the allotted time. It was necessary from both the operational viewpoint and from the mechanical stress viewpoint to keep the control and transient rod acceleration essentially the same for all conditions of reactor environment. The use of external drives thus dictated the design of a remotely controlled variable stored energy source. The most effective means to accomplish these various ends is by a compressed gas system. Since stored energy is required at zero reactor pressure, a system related to but independent of the reactor primary system was required. (See Section VIII-3.13 for a description of this system.) A compressed air-driven piston proved to be the most economical and most available source of motive power.

The release of the stored energy to effect a scram required reliable operation at all times with the added feature of rapid operation under widely varying load conditions. In addition, means had to be provided to hold the scram-air piston in

contact with the Acme lead screw. Space limitations imposed by the control rod positions in the core prohibited the use of magnetic couplers. Compressed air in conjunction with a rapid dump valve on the underside of the piston was used to meet these requirements.

The requirement for fail-safe speed control, precise control rod mechanical positioning, and the self-locking feature to prevent accidental withdrawal of the control rods due to electrical or mechanical failures dictated the choice of an ac electric motor-driven lead screw with self-locking Acme threads.

The design of the reactor core established the control rod drive locations on a radius of 6.710 inches with the transient rod drive located in the center. The drives were then designed to contain the majority of the components within a 6.5 inch diameter. Miscellaneous connections, minor equipment devices, and the drive motors were so arranged as to prevent interference among the drives when installed on the vessel head.

The containment of the reactor coolant at the drive rod seal assembly allowed the use of conventional carbon steel for all basic components of the drives with the exception of the stainless steel seal assembly. Since the drive units are separated from the reactor core by approximately 9 ft of water, the radiation level at the drives has no noticeable effect on the drive unit components.

3.2 Description of Drive Unit

3.21 General Description. The drive unit, as shown in Figures 24 and 25, may be visualized as a vertical column 6.5 inches in diameter and 14 ft 8 inches long. A 4-inch-diameter pneumatic piston is connected to the upper end of the drive rod. The rapid control rod scrams or rod insertions are obtained by exhausting compressed air below the piston, allowing compressed air above the piston to drive the rods into the core.

Fig. 24 Drive-unit assembly.

Fig. 25 Drive-unit installation.

Slow rod travel is obtained by running the drive motors in either direction. The motors are geared to an Acme screw-nut combination in the drive screw section located above the air cylinder. Air pressures on the piston maintain a positive upward force on an extension connected to the Acme nut. The nut extension enters the air cylinder from the drive screw section. Directly below the air cylinder section is the shock absorber which absorbs the energy of the control rods during air scrams.

The drive rod seal for retaining the reactor coolant is located below the shock absorber. Detailed descriptions of the drive components are presented below, beginning at the top of the drive unit. Differences between the control and transient rod drives will be discussed at the point where specific differences occur.

3.22 Detailed Description

(1) Drive Motor. Under normal operating conditions, with the air pressure controls functioning properly, the drive motors are only required to have sufficient power input to overcome the drive system friction, the net upward force on the piston to maintain contact and the force to produce acceleration and overcome system inertia. However, as a safety requirement in case of air-system failure or in the case of operational convenience for the transient rod drive, the drive motors are sized to provide adequate power to drive the rods into the reactor under any and all operating conditions of the reactor. Accordingly, each drive unit is powered by a 480-volt, three-phase, drip-proof, three-speed, squirrel-cage induction-gear motor with a rating of one horsepower at 1780 rpm, $2/3$ horsepower at 1180 rpm, and $1/3$ horsepower at 580 rpm. An integral gear reduction of 25.6:1 reduces the output speeds to 69.5 rpm, 46.2 rpm, and 22.65 rpm. This results in drive rod speeds of 17.4 in./min, 11.5 in./min, and 5.6 in./min, respectively. Each motor is equipped with a spring-set, magnetic-release disc brake.

The drive motors receive power either from commercial power sources or from a standby emergency generator which is kept in operation ready to go "on-line" at all times while the reactor is running.

(2) Drive Gear Box. The gear box as shown in Figure 26 contains a pair of 1:1 right-angle bevel gears which transmit power from the gear motor to the lead screw. It also contains a servo generator geared to the lead screw in such a ratio that one revolution of the generator is equivalent to 0.01 inch of rod movement. The signal from the servo generator drives a slave servo generator digital counter combination located on the control console. This counter indicates control or transient rod position in the core when the air piston is in contact with the lead screw nut extension.

(3) Drive Screw Section. Converting rotary motion of the gear motor to linear motion for rod position is the function of the Acme screw and nut located in the drive screw section as shown in Figure 27. The screw is so restrained as to produce only rotary motion, and the nut is so restrained as to produce only linear motion. The Acme thread is used to allow the screw-nut combination to be self-locking; ie, the screw will not rotate due to upward thrust on the nut. Experimental measurements have been made to confirm this. With a total force of 7420 lbs on the rod (2500 psi system pressure plus 300 psi "hold" air pressure), the magnetic brake was released and there was no movement of the screw.

Fig. 26 Gear-box section.

Fig. 27 Drive-screw section.

Thus, under no conditions of rod position or mechanical equipment condition (loss of motor brakes, power, air, gears, or couplings) will the control rods raise without an intentional act of the reactor console operator. This screw has a 1-1/2 inch diameter and four threads per inch. The total rod travel possible is approximately 46 inches. The nut is restrained from rotating by a pair of rollers which travel in guides mounted inside the drive screw section walls. Attached to the nut is the nut extension which extends downward through an air seal into the air cylinder section. During motor-driven rod positioning, the air piston is held in contact with the lower end of this extension by air pressure under the piston. During air scrams, contact is broken at this point. This contact is indicated electrically by a pair of snap-action switches. The contact switch feature is common to both the control and transient rods.

In the case of the transient rod, a 28-volt, solenoid-operated latch is also mounted on this extension and is used to prevent the accidental release of this rod. The latch contains a set of pivoted "fingers" which act as shear pins between the top of the piston and the Acme nut extension forming a mechanical linkage between these parts. Because of the relatively small force exerted by the solenoid, these fingers can only be retracted from the latch plate when there is no load on the latch. This is the normal condition when air pressures are correctly balanced. If there is load on the latch, an abnormal operating condition exists which must be corrected before the transient rod is dropped. Miniature snap-action switches indicate the position of the latch fingers as either "locked" or "unlocked".

In order to bring electric power to the latch, the latch switches, and the contact switches, a stationary, segmented commutator is mounted in the drive screw section, and leads are brought to it inside the nut extension. Each commutator has a total of seven contacts, six of which are used on the control rods, and all seven are in use on the transient rod. Transient rod switches are wired to a common ground, and the control rod switches are grounded individually. Double wiper contacts are mounted on an assembly bolted to the Acme nut. Leads from the stationary commutator segments are gathered at a multicontact connector attached externally to the drive unit wall.

Control rod safety interlock control is accomplished by four snap-action switches mounted parallel to the commutator on the outside of the drive-unit housing, and these are actuated by a projection (torpedo) on the wiper assembly. These switches are as follows: (a) top-limit (upper-limit) switch at a 45-inch position from physical bottom; (b) an interlock at 6 inches; (c) down-auto-stop switch at approximately 1 inch; and (d) the lower-limit switch located about 1/8 inch above the physical bottoming of the drives.

It should be noted that the reference to physical "rock bottom" of the drive units is a reference to bottoming of the shock absorbers. Physically, these switches are mounted in a position to accomplish the desired function at the desired relative location and are not actually mounted at the specified distances from the reference point. All of these switch indications are primary indications of the position of the face of the nut extension.

The interlock switch works in conjunction with a shock-extended switch in the air cylinder section and prevents withdrawal of the rods beyond the 6-inch position in the event that the shock absorber does not rise to the extended

position upon rod withdrawal. The down-auto-stop switch prevents the insertion of the rods beyond this point by the motor at speeds other than slow.

Transient rod safety and interlock switches are identical to the control rod switches.

(4) Air Cylinder Section. Located directly below the drive screw section is the air cylinder section. The control rod air cylinder section is shown in Figure 28, and the transient rod air cylinder section is shown in Figure 29. This contains a 4-inch-diameter air piston which is connected to the drive rod. During reactor operation at zero reactor vessel pressure, compressed air at approximately 150 psi is maintained above the piston and air at 250 psi maintained below. Air scrambling is accomplished by the rapid exhausting of the lower air. Since it is desired that the transient rod be driven at a greater speed than that necessary for the control rods, somewhat greater air pressure in the upper cylinder is used. During reactor operation with vessel pressure above atmospheric, the air pressure above the piston must be increased in order to counteract the force acting upon the 1-1/2-inch-diameter drive rod as it enters the vessel.

The primary purpose of the air pressure below the piston is to provide energy to lift the control rods and maintain contact of the piston with the nut extension while providing a source of energy that can be quickly released enabling the resulting unbalanced system to effect a scram. In order that the amount of energy to be released be kept as low as possible and as nearly constant as possible for uniform scram times, variations of the vessel pressure are compensated by varying the air pressure on top of the cylinder. Consequently, the pressure below the piston remains constant for any vessel pressure.

Space limitations and the corresponding mechanical design problems, as well as the desirability of maintaining reasonably uniform scram times over the wide range of reactor operating pressures, dictated that the air pressure above the piston be varied linearly in proportion to the reactor vessel pressure with an initial pressure above the piston of 125 psig and a final pressure of approximately 450 psi at 2500-psi vessel pressure.

To maintain a reasonable factor of safety in the stressed portions of drive units affected by acceleration and deceleration loads during scram operations, air pressure above the piston could be adjusted in proportion to 300-psi steps in reactor vessel pressure. These adjustments would necessarily have to be made remotely since vessel pressure changes could occur during nuclear runs.

In order to reduce the number of functions required of the reactor operator, a pneumatic system was designed and built to automatically sense the vessel pressure and to increase the upper air pressure as a linear function of vessel pressure. This automatic system is used for the four control rod drives. Since the transient rod is not used for scrambling and need not be inserted for safety reasons, the transient rod upper air is adjusted by the operator just prior to ejection by means of remotely controlled valving. The control rod pneumatic system is described in greater detail in Section VIII-3.13.

A snap-action switch, mounted externally at the lower end of the air cylinder and actuated by the piston moving a pin mounted in the air cylinder wall, informs the operator when the piston has been driven to its "seat" or

Fig. 28 Control-rod air-cylinder section.

Fig. 29 Transient-rod air-cylinder section.

full-down position. During the period of time after the piston has been driven to its seat position by air scrambling and before the nut extension has reached the piston to hold it in place, if the upper air were to be lost, the piston and control rods could withdraw from the reactor until contact with the nut extension were made. In order to eliminate this possibility, a piston lock device is installed at the lower end of the air cylinder. The piston lock consists of a spring-loaded latch bar which is similar to the common door latch. The latch box assembly is mounted externally to the air cylinder section, and the bar enters a hole in the drive unit wall. Force of the piston passing the latch retracts the latch until the latch bar reaches a latch bar groove in the piston. This groove is so located that the latch bar will not engage until the piston is approximately 1/8 inch from the mechanical "rock bottom". An electric solenoid is used to retract the latch bar when it is desired to raise the piston. However, as long as there is load on the latch bar either from vessel pressure or air pressure under the piston without the opposing force of the nut extension, the solenoid is unable to provide the force to withdraw the latch bar. Miniature snap-action switches indicate the latch-bar position as "locked" or "unlocked".

(5) Shock-Absorber Assembly. Due to considerable kinetic energies generated in the scrambling of the rods, a shock-absorber unit, shown in Figures 30 and 31, was designed to apply a relatively constant decelerating force to the rods over a total travel of 6 in. This unit is mounted directly below the air-cylinder section and is actuated by the air piston coming in contact with the shock-absorber mechanism.

The shock-absorber mechanism consists of six 1-inch-diameter bronze plungers joined together on a ring and arranged radially around the drive rod.

Fig. 30 Shock absorber unit.

These plungers operate in a fluid-filled housing containing a cylinder block with six cylinders. Each constant-diameter cylinder has six orifice holes spaced equally throughout its length and connected to a fluid reservoir. This allows a nearly constant decelerating force to be applied to the rod throughout the 6-inch stroke. A coil spring returns the piston group to the upper position upon rod withdrawal. To prevent rod bounce,

Fig. 31 Parts of shock absorber unit.

the coil spring is designed to lift only the moving parts of the shock absorber and will not lift the control rods. An externally mounted switch similar to the piston seat switch is actuated when the shock mechanism has reached the extended position. This switch is used in conjunction with the interlock switch discussed in the Drive Screw Section.

Silicone fluid, consisting of 2 parts 350 cp viscosity and 1 part 1000 cp viscosity, is used for the shock-absorbing fluid. This fluid has the combination of a relatively high flash point and relatively stable viscosity over the operating range of 100 to 200°F. Although the lubricity of the silicone fluid and its flash point ($\approx 600^\circ\text{F}$) are not fully satisfactory, this fluid is the best commercially obtainable for this service, and it has an adequate performance.

(6) Drive-Rod-Seal Assembly. The drive-rod-seal assembly, as shown in Figure 32, bolts to the reactor top head and isolates the reactor vessel environment from the drive units proper. The seal housing is constructed of stainless steel to resist corrosion of the reactor water and is equipped with four large openings at the upper end to permit cooling of the drive rod and adjustment of the packing glands.

The packings consist of two sets of standard commercial Teflon filled asbestos packing. Each set consists of five rings 1-1/2-inch-ID by 2-1/4-inch-OD with a stack height of 1-1/2 inches. The lower set of packings retains the vessel pressure under normal conditions with the top set acting as an emergency backup in case the lower set fails. A bleed line between the two sets of packings is passed through a leak detector.

The leak detector is orificed to permit normal leakage to be bled off with no indication. Excessive leakage, indicative of a seal failure, moves a plunger in the leak detector cutting off the bleed line and actuating a snap-action switch. This switch operates a light on the reactor console showing seal failure. The upper set of packings then receives full reactor pressure. A lower guide bushing on the seal assembly helps maintain concentricity of the packings around the drive rods.

Fig. 32 Drive-rod-seal assembly.

The success or failure of the seal unit is predicated upon the characteristics of the drive rod. The rod used is fabricated from a mild steel tube of 1-1/2-inch-diameter with 1/4-inch wall thickness. Applied externally is a 0.030-inch coating of hard facing material, a nickel-chromium-boron alloy. This coating is ground to an 8-microinch finish and has a hardness of 60 Rockwell "C".

Rather than subject the hard-faced tube to the tensile stresses occasioned during rod scrams, these forces are transmitted from the piston to the control rods or transient rod through a 4130-alloy steel rod mounted inside the drive rod tube. This rod is prestressed so that the outer tube is in compression at all times and does not contribute to the support of the rods in the reactor core.

4. CONTROL AND TRANSIENT ROD OPERATION

4.1 General

For all practical purposes, the operation of the control rod and transient rod drive units is identical except that the piston latch has been omitted from the transient rod inasmuch as no reactor hazard is involved in the event of malfunction permitting the transient rod to rise.

In order to present the operation of the drive units in a logical sequence, the following conditions are presupposed to exist:

- (1) Emergency power generator is running
- (2) The air system is fully charged and ready to operate
- (3) The control rods are at "seat position" (seat switch on) with the piston latch in place (piston latched) and the nut extension in contact with the piston (contact switch on)
- (4) Transient rod latch is closed (transient rod latch switch on).

4.2 Rod Withdrawal

In normal sequence, the following operations occur for withdrawal of control rods:

- (1) Air is introduced above the piston by means of the top air-controller
- (2) Air is introduced below the piston through a solenoid valve in the fixed air system
- (3) To withdraw the control rods, the piston latch is unlocked and the nut extension moved up by motor power
- (4) The piston latch is released to be in position for operation at any time
- (5) Assuming a signal has been received by the interlock switch from the shock absorber extend switch signifying that the shock absorber has extended, withdrawal may continue to the desired level as indicated on the digital counter on the console.

4.3 Rod Insertion

Normal insertion is as follows:

- (1) Fast-speed insertion by motor power to the down-auto-stop position, approximately 1-1/2 inches above lower limit. Slow-speed insertion by motor power to lower limit position. The seat switch, lower limit switch, and piston-locked switch indicate that the rods are fully inserted, that the piston is locked down, and that the drives are at lower limit.

Scram insertion is as follows:

- (2) For scram operation, two pilot-operated dump valves are opened on the air beneath the piston; the solenoid valve supplying air beneath the piston is closed; and the air beneath the piston is rapidly dumped. The air above the piston drives the piston (and control rods) to the seat. Loss of contact between the nut extension and the piston initiates an automatic fast run-down of the drives to the down-auto-stop position; then a slow-speed run-down to lower limit. The seat switch, contact switches, lower limit switch, and piston-locked switch indicate that the rods are fully inserted, that the drive is in contact with the piston, that the drives are at lower limit, and that the piston is locked down. Following reset of the scram circuit, the drives are ready for rod withdrawal as outlined above.

4.4 Safety Features

Table VI lists possible drive system faults and/or faulty operations and the safety devices built into the system to counteract these faults.

TABLE VI

SPERT III CONTROL ROD DRIVE SAFETY BACKUPS

Fault	Automatic Corrective Action
Loss of commercial electric power	(a) Automatic air scram (b) Automatic switching to emergency power (c) Automatic fast run-down of drives by emergency power
Loss of emergency electric power only	(a) Automatic fast run-down of drives with commercial power
Loss of all electrical power	(a) Automatic air scram (b) Automatic piston latch (c) Scram air holds piston down
Loss of piston contact	(a) Automatic air scram (b) Automatic fast run-down of all drives
Loss of air above piston	(a) Automatic fast run-down of drives by motor power
Loss of all electric power, co-incident with loss of all air	(a) Acme self-locking screw prevents further rod rise and maintains status quo (b) Motor brake backs up self-locking Acme screw
Failure of shock absorber to extend	(a) Interlock prevents raising of rods more than 6 inches
Failure of primary drive rod seal	(a) Second set of seals becomes operative (b) Warning light on console signifies that primary seal has failed
Breakage of mechanical drive parts above Acme screw (motor couplings, broken gears, etc)	(a) Self-locking Acme screw prevents further rise of rods (b) Manual air scram possible
Failure of air system to follow reactor pressure	(a) Warning light on console indicates over or under pressure of the scram air (b) Automatic fast run-down of drives
Failure of operator to observe upper and lower limit of rod travel	(a) Top and lower limit switches de-energize drive motor contactors
Failure to obtain contact on all pistons	(a) Permission to withdraw rods is automatically denied. All contact switches must be "on" to have motor power

TABLE VI (Cont.)

SPERT III CONTROL ROD DRIVE SAFETY BACKUPS

Fault	Automatic Corrective Action
Attempt by operator to unlatch piston with a net upward driving force on piston (no piston contact)	(a) Friction on the latch will prevent its withdrawal
Attempt by operator to drive rods at top speed to lower limit	(a) Down auto-stop switch prevents rod insertion to lower limit position except at slow speed (b) Lower limit switch cuts power from motor
Various faults in control wiring	(a) Continuity circuits necessary to turn on main power are provided
Loss of air in air reservoir or low reservoir pressure	(a) Warning light on console indicates low pressure and reactor should be shut down at earliest time which does not abort test
Rapid drop of reactor inlet temperature	(a) ΔT scram circuit scrams reactor if reactor inlet temperature is some set temperature (adjustable) cooler than primary coolant temperature
Loss-of-flow of primary coolant	(a) Reactor scram by loss-of-flow scram circuitry
Grounding of any point in the normally above-ground "floating" system circuitry	(a) Reactor scram and fast run-down of control rod drives accomplished by protective circuitry

VIII. CONTROL SYSTEM DESIGN

1. INTRODUCTION

This section of the report is devoted to a discussion of the various components of the Spert III 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 for the control system are that no hazard to personnel shall arise from control system operation and that risks to reactor equipment shall be minimized. The control system must provide proper manipulation of the control units and must furnish information on the operations performed. It also must indicate equipment failures or improper operation, and be designed in such a manner that any component failure which constitutes loss of control will automatically scram the reactor.

2. DESIGN PHILOSOPHY

The control system design reflects the philosophy of operation of the Spert reactors. The purpose of Spert III is to provide a facility in which experimental programs can be carried out to obtain kinetic information for the advancement of pressurized-water reactor technology and safety. One of the main experiments that will be performed is a power excursion initiated by reactivity perturbations. Two primary methods will be used. First, a poison "transient" rod may be ejected from the core, thus adding reactivity as a step function. Secondly, the control rods or transient rod may be moved with a time-dependent motion, thus adding reactivity as a ramp function.

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 shortness of their duration lead to the selection of a simple control system. Because of the short duration of the tests, the individual functions required during a transient test, such as starting data recording and photographic equipment, ejecting the transient rod, and scrambling control rods, are programmed on a sequence timer and the test is initiated by starting the timer. During a test, the two certified reactor operators and all other persons in the control room have the authority and responsibility to scram the reactor in the event of any unanticipated situation.

3. DESCRIPTION OF CONTROL SYSTEM

This section of the report describes the motors used to drive the control and transient rods, the power source for the motors, the motor speed selectors, the inhibitions placed on the power source, the various switches that are located on the control and transient rods, inhibitions placed on the movement of the control and transient rods, the circuitry used for running a transient, the shut-down circuits, the timer circuits, the control rod position indicator, the control rod air systems, the alarm systems, and the control console.

3.1 Drive Motors and Power Requirements

The basic power requirements of the control system are the five drive motors. These are 480-V, 3-phase, 60-cycle, 3-speed, squirrel-cage induction motors equipped with magnetic brakes. The motors have two stator windings and taps for 4-, 6-, and 12-pole operation. This provides synchronous speeds of 1800, 1200, and 600 rpm. Slip at rated load is 20 rpm. The current ratings of each motor are 2.2-, 2.3-, and 1.3-A for fast, medium, and slow speeds, respectively. Breakdown torque at fast speed is 430 percent of rated torque, so that the maximum power requirement for all four motors, assuming 75 percent efficiency, is 4.3 kW. The power system must be able to provide 73-kVA for starting all four motors. Voltage must be maintained at 60 percent of normal in order to carry the magnetic relays and contactors of the control system. Commercial 480-V, 3-phase power is supplied to the control system through a 100-A branch air circuit breaker, which is supplied through a 600-A feeder circuit breaker from a 750-kVA transformer bank. Emergency power is provided by a gasoline-engine-driven generator which is paralleled to the 100-A branch air circuit breaker through an automatic transfer switch. The generator, rated at 25 kW at 80 percent power factor, is powered by a four-cylinder, 60 hp overhead valve gasoline engine. The generator is of the three-phase, revolving-field type with a direct connected exciter and a voltage regulator.

3.2 Motor Power

Power from the branch circuit breaker and from the standby generator is brought to two manually operated disconnect switches. A phase-sequence and undervoltage relay is connected to the load side of each disconnect switch through potential transformers (Figure 33). The control system proper is connected to one or the other of these power sources through a pair of magnetic contactors which are mechanically interlocked to prevent simultaneous operation. Power for the drive motors is taken directly from these power contactors through a group of contactors which provide three speeds forward and reverse for each of the five motors. The speed control contactors select from the nine leads of each motor the three which are appropriate for a desired speed. These contactors are not required to make or break live circuits; speeds are selected only with the motors stopped. Mechanically interlocked contactor pairs are used for starting, stopping, and reversing the motors and for releasing the magnetic brakes. Each motor circuit is kept independent of the other, insofar as possible, in order to minimize the occurrence of malfunctions capable of interfering with the insertion of more than one control rod.

A 5-kVA and a 2-kVA transformer are connected open-delta through the main contactors to 3-phase power, and are used to provide 120-V power for all control relays and contactor coils. Another transformer, rated at 0.5-kVA,

763-B-7276

Fig. 33 Drive motor power system.

provides 28-V dc from a single-phase, full-wave bridge selenium rectifier. This 28-V dc operates the indicating lights and the transient rod latch solenoid.

3.3 Motor Speed Selector

The motor speed selector system is used to select the control rod rate of travel. Through different winding combinations, it is possible to obtain fast, medium, or slow speeds (17.4-, 11.5-, and 5.6-in./min, respectively).

Each drive motor has nine leads brought out from two windings. One of the windings is an ordinary Y-connected, 4-pole winding. The other winding has six leads which can be connected to give either 12-pole or 2 paralleled 6-pole, Δ -connected windings. In addition, each motor is equipped with a spring-set, magnetically released disc brake; the windings of which must be energized continuously with 480-V single-phase during motor operation.

Each of the five motors has independent circuitry beyond the insert and withdraw relays in the control system. The transient rod drive controls are completely independent. By means of selector switches, all the control rod drives can be ganged to operate in unison; or as many control rod drives that are selected can operate as one drive, and no drive can be operated in any other way. Thus, three pushbuttons on the console are sufficient to select fast, medium, or slow speed for the rod drives.

The motor speed selector relay system is shown in Figure 34. In order to select fast speed, the control-rod-fast-speed pushbutton is depressed. This will pick up the control-rod-fast-speed relay provided that the control-rod-drive-running relay is de-energized, which indicates that the control rods are not being moved. This interlock is used so that speeds cannot be changed while the control rods are moving, thus allowing smaller contacts to be used (because of no switching under load) on the terminal and winding contactors. Once the control-rod-fast-speed relay is picked up, it is locked in by a shunting circuit, provided that the medium or slow speed relays are not picked up.

If the control-rod-slow-speed relay is picked up, the control-rod-winding-selector relays will be energized if the control-fast-run-down relay is energized, indicating there is no signal for a fast run-down of the drives. The control-rod-winding-selector and control-rod-terminal-selector relays will be energized if the control-rod-medium-speed relay is picked up. These relays actuate the same named contactors.

3.4 Power Control

Commercial power is admitted to the control circuitry through the main power contactors which are operated by the power control circuitry. In case of commercial power failure, the power control circuitry is designed to automatically switch to standby power.

Commercial power of proper voltage and phase sequence at the main potential transformer actuates the main-phase-failure relay and thereby admits power to the main-power keyswitch, as shown schematically in the upper diagram of Figure 35. Closing the main-power keyswitch picks up the on-main-power relay, which then picks up the main-power contactors thus energizing the control system, provided the plug-continuity relay is energized, and the on-standby-power relay and contactor are de-energized. The former relay

CONTROL ROD SPEED CONTROL RELAY CIRCUITS

THE ABOVE GROUP OF CONTROL ROD SPEED CONTROL CIRCUITS IS USED FOR 4 CONTROL ROD DRIVES.

THE CONTROL ROD DRIVES MAY BE SWITCHED ON OR OFF INDIVIDUALLY BY THE OPERATOR, BUT THOSE THAT ARE RUNNING ALL RUN AT THE SAME SPEED AND DIRECTION. A SIMILAR GROUP OF SPEED CONTROL CIRCUITS (NOT SHOWN) IS USED FOR THE TRANSIENT ROD DRIVE.

SPEED CONTROL CONTACTOR CIRCUITS TYPICAL OF 5

Fig. 34 Motor speed selector system.

763-A-7277

Fig. 35 Power control system.

763-A-7290

prevents energizing the circuitry if any power plug to the drive systems is not properly engaged, and the latter interlock prevents the on-main-power relay from energizing if standby power is being used. However, once the control system is in operation, failure of a power plug will not kill commercial power (it will cause an annunciator alarm) as the circuitry is paralleled through an on-main-power relay. In the lower portion of Figure 35, the main-power-memory relay prevents the on-standby-power relay from picking up along with the on-main-power relay. Once the on-main-power relay is safely picked up, a pair of its normally closed contacts prevents the on-standby-power relay from being picked up, and a pair of its normally open contacts picks up the main-power-memory relay. The standby-power OK relay is energized provided that the standby-phase-failure relay is energized, thus indicating proper standby voltage (90 percent or normal voltage) and phase sequence.

If commercial power should fail, standby power would automatically come on. The on-main-power relay would de-energize and close its contacts causing the on-standby-power relay to energize through the main-power-memory relay which is locked in by a holding circuitry. Energizing of the on-standby-power relay and de-energizing of the main-power contactors (from loss of main power) causes the standby-power contactors to close. If the standby-power OK relay is energized, standby power will be admitted to the control system circuitry.

No provisions are made for automatic return to main power on termination of the main power outage; but if standby power should fail, main power will be picked up.

The plug-continuity and main-power-bypass keyswitch provide a means of picking up main power despite an open cable connector or, if main power is not available, a means of bypassing the preference for main power in order to pick up standby power in emergency circumstances.

3.5 Control Rod Control Switches

Mechanically actuated switches are provided in the control system to indicate rod positions, air pressures, and to energize indicating lights. Only the location of the switches will be given; the purpose will be described in the next section (Section VIII-3.6).

The pistons of the four control rods and the transient rod each actuate a seat switch when the rods are fully inserted into the core and a contact switch when the pistons are in contact with the Acme nut extension of the rod drives. An actuator on each shock absorber assembly actuates a shock extended switch when the shock absorbers are fully extended.

An adjustable torpedo on the Acme nut of each drive actuates switches on the drive housing at the following positions:

Lower limit (lower-limit switch)

One and one-half inches above lower limit (down-auto-stop switch)

Six inches above lower limit (up-auto-stop switch)

45 inches above lower limit (top-limit switch)

The transient rod drive is equipped with a mechanical latch which couples the Acme nut extension to the piston to prevent premature dropping of the rod. Two switches are mounted on the latch; one is actuated when the latch is open and the other when closed (transient-rod-unlocked and transient-rod-locked switches). The control rod drives have mechanical latches which lock the control rod pistons in the down position. Two switches indicate the position of the latch (control-rod-locked and control-rod unlocked switches). Five air-pressure switches are included in the system which, when actuated by air pressure, close relays in the main-air-pressure OK, scram-air OK, control-rod-hold-air-on, fire-air-on, and transient-rod-hold-air-on circuits. Each drive motor is equipped with a thermal switch which opens when the temperature of the motor becomes dangerously high and sounds an alarm on the console. There are two identical sets of electrical cables and compressed air hose connectors for each of the rod drives: one for operating the drives on the reactor and the other for testing the drives in dry dock.

As a general rule, the normally open contacts in all of the above switches are used to operate relays which actuate console indicator lights and control system interlocks.

As exceptions to the general rule, the motor-over-heat switch relay operating contacts are normally closed. The control-rod-up-auto-stop switches do not operate relays or indicating lights (the transient-rod-up-auto-stop switch does operate a relay); their sole function is to provide normally closed contacts for use directly as interlocks with the shock-extended relays in the rod withdrawal circuits to prevent any rod being withdrawn above an unextended shock absorber. The down-auto-stop switch operates a relay, but there is not a corresponding console indicator light. The air pressure switches operate relays, while the scram and fire air pressures circuits have gauges on the console and annunciators that are connected to the scram-air OK and main-air OK relays.

3.6 Control and Transient Rod Relay System

Except for automatic shutdown in certain hazardous situations, the control of the Spert III reactor is entirely manual. In order to prevent damage to the control rod drive system or to the reactor core, certain interlocks have been designed into the system. These interlocks are described below.

3.61 Miscellaneous Relays. Figure

36 shows the wiring of the up-auto-stop switches and control-rod-series-shock-extended relay. When the control rods are first withdrawn, the up-auto-stop switches allow the control rods to be withdrawn, approximately 6 inches above the lower limit. If any of the shock absorbers do not extend, indicating that they are not ready to decelerate a control rod, the control-rod-series-shock-extended relay will not be picked up, and further control rod withdrawal will be prevented. However, if the control-rod-shock-extended relays are energized, the control-rod-series-shock-extended relay is energized and control rod withdrawal may be continued.

Fig. 36 Miscellaneous relays.

3.62 Control Rod Relays. Primary control of the drive motors is by the control-rod-insert-withdraw switch on the control console. This is a pistol grip

switch which has a spring return from "withdraw" to "off" position. The insert position is maintained by a detent until manually returned to off. Also, the control rods may be withdrawn by a timer-operated ramp circuit which is intended for ramp-type power excursions. A fast run-down circuit provides automatic insertion of the control rod Acme screws after the control rods have been scrambled. The same action occurs for the transient rod after the transient rod has been fired and seated.

For manual or ramp operation, any desired number of the control rod drives may be operated in unison through the use of the drive selector switches on the console and corresponding relays in the starting contactor circuits.

No interlocks restrict the operation of the control-rod-insert relay. However, the control-rod-withdraw relay has several restrictions (Figure 37). In order to energize the control-rod-withdraw relay, the following conditions have to be met. The control-rod-series-contact relay signifies all the control rods are in contact with their Acme screws; the scram-air OK relay indicates that the scram air pressure is within the prescribed limits; and the standby-power OK relay indicates correct standby voltage and phase sequence. The above relays will be energized if their conditions are met, but they can be bypassed by energizing the bypass relay by means of the bypass keyswitch. This bypassing procedure is for dry dock test operation or for coupling or uncoupling the control rods from their drive screws. It is not for reactor operation. The transient-rod-contact relay has to be energized, indicating contact between the transient rod and its Acme screw. The transient-rod-latch-locked relay has to be energized, indicating that the transient rod is locked to the Acme screw. The transient-rod-fire-air-on relay and the transient-rod-arming relay are de-energized, thus indicating that the transient rod does not have air to the top side of the piston and that the transient rod is not in a position to be fired. If a ramp transient were being run, a timer-section relay would be used in conjunction with the control-rod-ramp-selector switch in order to withdraw the control rods for a given time element. This timer section circuit bypasses the control-rod-withdraw switch. (See Figure 40, page 80).

In order to energize the control-rod-withdrawal contactors, the control-rod-drive-selector relay and the control-rod-withdrawal relay must be energized. The control-rod-series-shock-extended relay, which must also be energized, can be bypassed by use of the bypass relay (the purpose of which is to be able to withdraw the drive without withdrawing the control rod poison). The control-rod-fast-rundown relay is energized indicating no fast rundown of the drives. (If the control-rod-fast-rundown relay were to de-energize, a fast rundown of the control rod drives would be initiated.) The control-rod-emergency-stop relay, control-rod-top-limit relay, and the control-rod-insert contactor and relay must be de-energized. The control rods can now be withdrawn.

The control-rod-insert contactors are energized if the control-rod-drive-selector relay and control-rod-insert relay are energized and if the control-rod-down-auto-stop relay, control-rod-emergency-stop relay, and control-rod-lower-limit relay are de-energized. A control rod fast run-down signal bypasses the first two relays. The down-auto-stop relay prevents insertion at high or medium speed beyond one and one-half inches above lower limit. It may be bypassed by selecting slow speed which energized the control-rod-winding-selector relay, thus allowing the control rods to be inserted to lower limit.

Fig. 37 Control and transient rod relay system.

If the control-rod-emergency-stop relay is picked up through actuation of the control-rod-emergency-stop-toggle switch, which is spring returned, neither the control-rod-insert contactor nor the control-rod-withdraw contactor can be energized.

3.63 Transient Rod Relays. Since insertion of the transient rod removes poison, thus adding reactivity to the core, it has many inhibitions in its insertion as shown in Figure 37. Also, in order to ensure that the transient rod is not prematurely dropped, there are certain interlocks in its withdrawal circuit. In order to withdraw the transient rod, the following conditions must be fulfilled: The transient-rod-insert-withdraw switch is moved to the withdraw position. The transient-rod-contact relay, indicating contact between the piston and the Acme screw, must be energized. The transient-rod-latch-locked relay must be energized, signifying that the transient rod is locked to the Acme screw drive. The transient-rod-top-air-on relay must be de-energized, indicating that there is no air on the top side of the transient rod piston; and the transient-rod-armering relay must be de-energized, thus indicating the transient rod is latched and not ready to be fired. The transient-rod-hold-air-on relay must be energized, indicating that there is air on the bottom side of the piston. The control-rod-series-contact relay and the control-rod-top-air OK relay must be picked up, signifying there is contact between the control rods and their drives and that there is sufficient air to scram the control rods. The latter interlock may be bypassed for rod coupling or uncoupling by closing the bypass keyswitch which energizes the bypass relay. If all of these conditions are fulfilled, the transient-rod-withdraw relay may be energized.

In order to energize the transient-rod-withdraw contactors, for the actual movement of the rod, the following conditions must be fulfilled: The transient-rod-up-auto-stop relay is de-energized in order to allow withdrawal of the transient rod from lower limit. At shock extend position, this relay is energized; and if the shock absorber is extended, the transient-rod-shock-extend relay will be energized.

If the shock absorber does extend, the transient rod withdrawal may be continued if the following conditions are satisfied: The transient-rod-fast-run-down relay is de-energized; the transient-rod-upper-limit relay is de-energized, indicating that the rod is not at upper limit; the transient-rod-withdraw relay is energized; and the transient-rod-insert relay and contactors are de-energized. Meeting the above conditions will permit energizing the transient-rod-withdraw contactor which will cause the withdrawal of the transient rod.

In order to insert the transient rod, the following conditions have to be met: The transient-rod-insert-withdraw switch is moved to the insert position. The transient-rod-armering relay must be de-energized, thus indicating that transient rod piston is locked to the Acme screw and cannot be fired. The transient-rod-hold-air-on relay and the control-rod-top-air OK relay are energized, indicating that the transient rod piston will be held up against its Acme screw by the hold air and that the control rods can be scrammed. The control-rod-series-contact relay must be energized, indicating all of the control rods are in contact with their Acme screws. The transient-rod-insert relay will then be energized. The transient rod may be inserted for ramp transients through the timer-section-7 relay and the transient-rod-ramp switch which bypasses the transient-rod-insert-withdraw switch. Once the transient-rod-insert relay is picked up, the transient rod may be inserted at any speed to one and one-half

Fig. 38 Transient rod relay system.

inches above lower limit. At this point, the transient-rod-down-auto-stop relay is energized allowing further insertion of the transient rod. The transient rod may be inserted to lower limit by selecting slow speed. This energizes the transient-rod-winding-selector relay and allows the transient-rod-down-auto-stop interlock to be bypassed which permits the transient-rod-insert contactors to be picked up. Insertion at slow speed is continued to lower limits, at which point the transient-rod-lower-limit relay is energized, thus prohibiting further insertion.

Step changes in reactivity are obtained by ejecting the transient rod from the core. Following this type of transient, the piston would be seated while the Acme screw would be at the transient rod position before the rod was fired. Automatic run-down circuits return the Acme screw to the piston by means of the transient-rod-fast-run-down and -insert circuits. The transient-rod-fast-run-down relay is energized by de-energizing the transient-rod-contact relay and energizing the transient-rod-seat relay (Figure 38). The transient-rod-fast-run-down relay parallels the transient-rod-insert relay in the transient-rod-insert-contactor schematic (Figure 37) and allows the Acme screw to return to the down-auto-stop position, thereby energizing the down-auto-stop relay. At this point the transient-rod-insert relay is de-energized. Since the transient-rod-down-auto-stop relay bypasses the transient-rod-hold-air-on relay and the control-rod-series-seat relay bypasses the control-rod-series-contact relay, turning the transient-rod-insert-withdraw switch to the withdraw position will energize the transient-rod-insert relay. The transient rod may then be inserted in slow speed to the lower limit position as described previously. (The control-rod-series-lower-limit relay is provided in the transient-rod-insert circuitry to simplify the rod coupling procedure.)

3.7 Transient Relay Systems

Many relays were mentioned in the previous section with little or no description of purpose. These will be described below. After the transient rod and control rods have been withdrawn to the desired position, the transient rod's arming procedure begins. This is accomplished by moving the transient-rod-arming and unlock-keyswitch to the arming position (Figure 38). The transient-rod-arming relay will be picked up and sealed through the normally closed disarm switch if the following conditions are met: the standby-power OK relay is energized; the control-rod-fast-run-down relay is energized (indicating that there is no signal for a fast run-down); and the control-rod-scram-air OK relay, control-rod-series contact relay, transient-rod-shock-extended relay, transient-rod-contact relay, and the transient-rod-hold-air-on relay are energized. Energizing the transient-rod-arming relay causes the transient-rod-top-air-fill valve to open and the exhaust valve to close. The fire air pressure may now be adjusted from the control console by the fire-air-increase-decrease switch which controls the transient-rod-fire-air-control valves. This adjustment is not permitted unless the transient rod piston is latched to the drive.

By moving the transient-rod-arming and unlock keyswitch to the unlock position, the transient-rod-latch-control relay will be energized, provided the transient-rod-arming and fire-air-on relays have been energized. Once the relay is made, it is locked in unless the Disarm button is pushed. The contactors on the transient-rod-latch-control relay will energize the transient-rod-latch solenoid, thus unlocking the piston from the Acme screw. The latch movement

will actuate the transient-rod-latch-unlocked switch which will energize its relay. The transient-rod-armed relay will then be energized if the transient-rod-hold-air-on, transient-rod-latch-unlocked, and transient-rod-fire-air-on relays are energized. The transient may then be initiated by starting the timer. When timer-section-4 relay is energized, the transient-rod-fire relay is energized provided that the transient-rod-arming and transient-rod-armed relays are energized. The transient-rod-fire relay energizes the transient-rod-hold-air-exhaust contactors which closes the hold-air-inlet solenoid valves and opens the hold-air-exhaust solenoid valves, thus allowing the transient rod to be ejected from the core.

Once the transient-rod-fire relay has been energized, it is locked in until the transient rod drive reaches lower limit. This ensures that the solenoid valves are not reset before contact has been restored between the piston and Acme screw. This is necessary since resetting the valves prematurely would drive the piston up against the Acme screw which, in all probability, would damage the screw. The transient-rod-fire-air-control relay is also locked in by the transient-rod-fire relay which maintains fire air on top of the piston until the Acme screw reaches lower limit. Once the transient rod has been fired and is seated (transient-rod-seat relay energized). The Acme screw will be inserted by the energizing of the transient-rod-fast-run-down relay. The motion can automatically be stopped by opening the transient-rod-emergency-stop switch, by the energizing of the transient-rod-down-auto-stop relay when the drive reaches one and one-half inches above lower limit, or when the transient rod Acme screw comes in contact with the piston, thus energizing the transient-rod-contact relay. When the transient rod reaches the down-auto-stop position the operator selects slow speed and inserts the transient rod to lower limit. Upon reaching lower limit, the transient-rod-fire relay is de-energized causing the following: the transient-rod-latch-control relay de-energizes and closes the latch; the transient-rod-hold-air-inlet solenoid valve opens; the transient-rod-hold-air-exhaust contactor de-energizes causing its exhaust valve to close; and the transient-rod-fire-air-control relay de-energizes, causing the fire-air-inlet valve to close and the transient-rod-fire-air-release contactors to de-energize causing its solenoid to open. The transient rod's circuitry has now completed its function for running a step transient.

3.8 Scram, Ramp, and Run-down Circuits

The scram circuits for the reactor control system are shown in Figure 39. Since the scram relays remain energized after resetting, any interruption of power will de-energize the relays and initiate a scram. A scram also may be initiated by depressing one of five control-room-scram push buttons or one of three strategically located reactor-building-scram push buttons. Once the relays have been de-energized, they remain locked out by two pairs of normally open contacts wired in series. The scram relays may be re-energized by pushing the normally open scram-reset push button provided that the control-rod-lower-limit-series relay is energized. The control-rod-lower-limit-series relay prevents resetting the scram before the Acme screws are at lower limit.

Scramming the control rods vents the hold air from the bottom of the control rods. This is accomplished by de-energizing the scram relays which de-energizes the control-rod-hold-air relays and causes the following: (a) the control-rod-hold-air-inlet valves to close and (b) the control-rod-hold-air contactors to de-energize, causing the control-rod-exhaust valves to open. In order to test these

Fig. 39 Scram circuits.

valves, two control-rod-test-fire push buttons are provided on the console for each of the control rods, one for each dump valve. Pushing one of the buttons will drop out one of the control-rod-hold-air contactors, provided the scram, control-rod-contact, control-rod-hold-air, and the control-rod-shock-extended relays are energized. The other test fire push button operates the second dump valve.

With the Spert III plant operating at high temperature and pressure, it is conceivable that a malfunction of the primary system heat-exchanger temperature control circuit would cause the operation of the temperature control valves such that the coolant-moderator temperature would drop causing a reactivity insertion to the reactor. To protect the reactor facility from this type of operational accident, a differential temperature (ΔT) scram system has been installed. The ΔT scram system consists of a fast response temperature sensing device located in each loop of the primary coolant stream, downstream of the heat exchanger mixing tee. The output signal from the sensor is fed to an adjustable down-scale, setpoint recorder located in the Spert III control room. The setpoint is connected to the reactor control rod scram circuit such that the reactor is scrammed when the setpoint is reached. The recorder's reference temperature indicator is adjustable over the full range of temperatures obtainable at Spert III. The setpoint is usually set 10°F below the reference temperature indicator after the primary coolant-moderator temperature equilibrium has been established and before nuclear operations begin.

At any time during which the reactor is operating at an appreciable power level, the potential exists for a loss-of-flow accident through a reduction in pumping power, an accidental primary flow valve closure, or other means of losing primary flow. In order to protect the reactor core from fuel damage resulting from this type of accident, a loss-of-flow scram circuit was installed in the reactor scram system. It consists of two basic circuits. The first circuit has four voltage relays, each of which is connected across the power input to a pump. The second circuit consists of a differential pressure (ΔP) measurement across the core. The ΔP signal is recorded in the control room on an instrument with an indicator and a variable setpoint. The loss-of-flow scram relay is operated by the four voltage relays which are in series with the ΔP 's indicator microswitch. If the voltage to any pump is lost or if the ΔP across the core drops such that the indicator lowers to the setpoint (activating the microswitch), the loss-of-flow scram relay will de-energize initiating a scram. To permit zero flow operations or a loss-of-flow type test, a keyswitch was installed on the console to bypass the loss-of-flow scram relay.

Because of the long cable runs (≈ 3000 ft) between the reactor building and the control center and the nature of the Spert III control system, significant capacitive coupling exists between various system elements and ground. Therefore, it was possible that control system relays may fail to respond to the operator demand if a particular shorted condition should arise between one side of a relay and ground. To prevent such unplanned behavior of this type, a protective system (the ground monitor scram) has been added to the control wiring. This system consists of a low impedance grounding mechanism which detects the common ground current and the voltage difference between the control system and ground. Two meters on the control console monitor this current and voltage. In the event of any voltage or current abnormality, a reactor scram will be

To prevent possible damage to the reactor core or facility from a control rod withdrawal accident or other type of accidental power rise, a power level scram circuit has been installed in the Spert III control system. This circuit consists of two ion chambers and their associated electronics. The output of the chambers is amplified and fed into a tripping circuit. If the reactor power increases such that the output of the chamber exceeds a preset level, a scram will be initiated. This scram mode is operative during all nuclear operations except when the timer is operating or when deliberately bypassed with a specific keyswitch. With these two exceptions, which permit power excursion tests and high power runs, full protection is provided from the control rod withdrawal accident or any unexpected increase in power which could arise from routine operation of the facility.

The ramp and control rod fast run-down circuits are shown in Figure 40. Ramp transients are run as well as step transients. The ramp transients are of two types. The first is the control rod ramp in which the control rods are withdrawn at a constant speed thereby adding reactivity at a time-dependent rate. The second method is the transient rod ramp in which the transient rod is inserted at a constant speed, thereby removing poison and increasing the reactivity. The type of ramp is selected with the ramp-selector switch and the timer-selector switch. Turning the ramp-selector switch to the control rod position will initiate, in conjunction with the timer-selector switch, the control rod withdrawal as described in the Control Rod Movement Section (Section VIII-3.62).

The control rod fast run-down circuitry is also shown in Figure 40. When the control-rod-fast-run-down relay drops out, it energized the control-rod-insert

Fig. 40 Ramp and run-down circuits.

contactors, thereby inserting the control rods. Once the control-rod-fast-run-down relay is de-energized, it is locked out until the control rods reach the down-auto-stop position and energize all four relays. A fast run-down can also be stopped or prevented by turning on the bypass keyswitch which energizes the bypass relay (this function is used during rod coupling or testing) or by energizing all four emergency-stop relays by activation of their switches. A fast run-down is also called for when the timer-section-5 relay is energized or if the control-rod-fast-run-down push button is pushed. The timer-section-5 relay is used for a fast run-down at the termination of a transient if desired.

To initially energize the control-rod-fast-run-down relay for reactor operation, each of the control-rod-down-auto-stop relays must be energized. Once the control rods have been withdrawn above the down-auto-stop positions, the following conditions are necessary: control-rod-series-contact relay energized; control-rod-top-air-OK relay energized; scram relays energized, indicating that the reactor is not in a scrammed condition, control-rod-fast-run-down relay energized, indicating that the reactor is not in a fast-run-down condition; control-rod-timer-fast-run-down relay de-energized, indicating that the timer is not calling for a fast-run-down; standby-power-OK relay energized, indicating that emergency power is "on line" and ready to be used; and on-standby power relay de-energized indicating that standby power is not being used to run the reactor. The control-rod-fast-run-down relay is then energized and ready to initiate a fast-run-down of the control rods.

3.9 Sequence Timer Circuits

The control system is provided with two Multiflex timers, manufactured by Eagle Signal Corp., for sequence programming of experiments. Two similar units are provided; one has a range of 0 to 30 sec and the other 0 to 300 sec. Precise timing is available for sequences up to 30 sec. Sequences requiring greater than 30 sec can be programmed on the 300 sec timer but with less precision. Timer settings can be made with an accuracy of 0.25 percent of full scale.

The basic timer circuit is shown in Figure 41. The 30-sec timer and the 300-sec timer are started by a keyswitch which energizes a clutch solenoid to start the timers. The stop push button is used to de-energize the clutch solenoids and reset the timers to their original position. Section-1-timer contacts are used as a holding circuit, enabling a momentary switch to be used for starting so that the timer does not automatically repeat its cycle after resetting.

Before a test, the reactor operator selects the timer to be, the functions (Console Timer Section 1, etc) desired

Fig. 41 Sequence timer circuits.

for the test (ie, tape start, oscillograph start, transient rod fire, control rod scram, oscillograph stop, and tape stop) and the time at which these functions will perform.

The reactor control system is not operated by the timer directly, but by seven timer-controlled relays at the control console. These relays are designated Console Timer Section 1, Console Timer Section 2, etc, and are controlled by correspondingly numbered contact pairs of the timers. Timer relays 2, 5, and 6 are shown in Figure 41. The selector switch controls the 30- or 300-sec timer contacts to the console timer section relays, thus energizing the relays when the timer contacts are closed. Once the contacts of the transient rod fire relay are closed, they bypass the timer stop button during a programmed transient power excursion. Timer Section 7 relay is permanently assigned to control ramp transients.

3.10 Control Console

The control is built up from six standard prefabricated 22-inch wide metal sections and two 45° "pie" sections. The two pie sections are inserted between the second and third, and the fourth and fifth rectangular sections so that the overall appearance is roughly that of a quadrant of a circle. The operator is seated in front of the third and fourth sections. Figure 42 is a photograph of the complete reactor control console.

Located to the right of the sixth rectangular section are two 22- by 72-inch storage racks which contain recording oscillographs and intercom systems.

On top of the third and fourth rectangular section of the console are the annunciator units which will be explained in Section VIII-3.12. Located underneath these are the operational log and linear power recorders. Beneath the operational log recorder in the third section is the control rod drive switches such as position indicator, limit switches, insert-withdraw handle, speed selector, etc, and the bypass and latch keyswitches.

Beneath the operational linear power recorder is the scram button, reset button, fast run-down button, top air pressure gauges, a count-rate meter, and the transient rod limit switches, speed selector, and insert-withdraw handle.

To the left of the operator in the first section is the power display scope and various radiation meters. The second section contains a temperature recorder, intercom switches, various power keyswitches, control rod and transient rod main seal leak detector lights, and a clear light which is activated by a series of contacts from the control rod contact series, control rod seat series, control rod piston locked series, and the transient rod lower limit relays. Lighting of the clear light indicates all rods are locked in the fully inserted position.

The fifth and sixth sections contain the TV, scalars, timers, and equipment needed to run transients.

3.11 Rod Position

Telesyn self-synchronous motion-transmitting systems are geared to the Acme screws of the transient rod drive and the four control rod drives. These

Fig. 42 Control console.

drive Veeder-Root digital counters on the control console and indicate rod drive positions in hundredths of inches.

3.12 Alarm and Warning System

Warning lights and horns incorporated in the control system indicate reactor start-up and dangerous situations. A schematic diagram of the system is shown in Figure 43.

To notify personnel in the reactor area of reactor start-up, two warning horns and eight flashing red lights are provided in the area. One horn is located inside the building and one outside. Six red lights are located on the exterior of the building and two inside the building. Withdrawal of any rod will de-energize its seat relay which will energize the reactor-on relay or dry-dock-on relay if so connected. The reactor-on relay will, in turn, energize the warning-flasher relay to flash the warning lights and a horn relay which sounds the horns. The horns will sound for approximately 30 sec at which time a horn-off-delay relay is picked up to silence the horns. The horns also may be sounded by pushing the horn push button on the console with or without console power on. If the control rod drives on the reactor head are being tested in the dry dock position, two yellow lights in the dry dock area will be flashing, but no warning horn is sounded.

Fig. 43 Alarm circuits.

The rod jam relay circuitry is shown in Figure 43. The purpose of the circuit is to indicate when a stuck rod has occurred. When a control rod has been scrammed, the control-rod-series-contact relay will be de-energized, thus closing the contacts in the rod jam circuit. Normally, the control rods will drop into place, thus opening the control-rod-series-seat relay contacts in the rod jam circuit. However, if the control-rod-series-seat relay is not energized within one second after the control-rod-series-contact relay has been de-energized, the rod jam annunciator will be picked up, thus indicating that a stuck rod has occurred. The circuitry for the transient rod is essentially the same, except that a normally open pair of contacts from the transient-rod-fire relay parallel a pair of normally closed contacts of the transient-rod-contact relay.

A motion detector circuit was designed and installed in the control system which causes an alarm in the event of an unintentional motion of any control rod drive shaft and is called the brake-failure arm. Cam-operated switches were mounted on each control rod drive rod motor shaft and circuitry was provided such that a change of any switch condition from open to closed or vice versa causes an alarm on the annunciator panel unless the motion was intentional. Whenever any rod is being moved by operator action the detection circuit is automatically bypassed for that rod. Operation of the circuit may be checked by removing the bypass with a test switch on the console. Any withdrawal or insertion will then cause an alarm if the circuit is operating properly.

An annunciator panel on the control console provides alarm signals for certain abnormal conditions in the control system. When an alarm occurs, the contact closes, lighting a bulb, and activating the annunciator buzzer. Pressing the alarm reset push button silences the buzzer and leaves the bulb lighted. Clearing the alarm condition cuts out the light and returns the unit to normal.

Alarm circuits included on the annunciator panel are shown in Figure 43. Power for the annunciator is obtained from the control-center building through a set of contacts on the main power keyswitch. The plug continuity, standby power failure, ΔT scram, and scram air-pressure alarm circuits are powered from the keyswitch. All other alarm circuits are not powered unless the power-on relay is energized.

The following abnormal conditions will actuate an alarm:

- (1) Plug interlock (plug continuity) -- this alarm indicates no main power available at the control system relay rack, or one or more of the drive power plugs are not plugged into their proper receptacles.
- (2) Power failure -- this alarm indicates an interruption of the main power source or a drop out of the main power low voltage-phase failure relay.
- (3) Scram air -- this alarm will come on wherever the alarm pressure switches in the scram air pressure controller indicate that scram air is outside the normal control limits (too high or too low).
- (4) Emergency power failure -- this alarm will come on whenever the control power keyswitch is on and the emergency power source is not available.

- (5) Air reservoir low -- this indicates the control rod air reservoir pressure is below the setpoint.
- (6) Drive motor overheat -- this alarm indicates any one of the five thermal overload switches on the drive motors has closed.
- (7) Rod jam -- this alarm comes on whenever any one of the following situations is present:
 - (a) Transient rod indicates loss of contact, but no transient rod seat indication.
 - (b) Fire relay energized, but no transient rod seat indication.
 - (c) Control rods indicating loss of series contact, but no series seat indication.
- (8) Drive rod seal -- this alarm is provided to indicate failure of the primary seals on the rod drives.
- (9) Brake failure -- this alarm indicates unrequested movement of control rod drives.
- (10) ΔT scram -- this alarm indicates the system temperature in one or both loops has dropped below the setpoint.

3.13 Rod Drive Air System

The control rod air system supplies compressed air to the transient rod and the control rods to scram the rods or to initiate transient power excursions. The compressed air for the system is divided into three types: (a) hold air, (b) control rod top air, and (c) transient rod top air. A schematic flow diagram of the air system is shown in Figure 44.

Compressed air for this system is supplied by a high pressure, two-stage, water-cooled compressor rated at 0.63 cfm at a pressure of 1000 psig. Intake air for the compressor is supplied from dried 125-psig service air (see Section IX-3); discharge is to a 42-ft³ air receiver. Pressure in the receiver is maintained by start-stop operation of the compressor.

(1) Hold Air. The hold air for the four control rods and one transient rod is supplied from a 15-ft³, 300-psig reservoir which is connected to the 1000-psig reservoir through a pressure regulator valve. The air from the 300-psig reservoir is reduced to 225 psig through a pressure regulator and acts on the bottom side of the air pistons in the drive units to position the rods against the drive screws. The hold air is admitted by means of five solenoid-operated valves and exhausted for reactor scram by means of ten pilot-operated dump valves (two for each rod drive).

(2) Control Rod Top Air. Air for the top side of the control rod air pistons is supplied from the 1000-psig air receiver at a controlled pressure. Pressure control is accomplished by means of two pressure transmitters, a pneumatic-set recorder-controller, a tri-act controller, and two diaphragm control valves. This instrumentation system is designed to continuously detect, record, and control the air pressure required to scram the pistons as a

Fig. 44 Flow diagram of the control rod air system.

function of reactor vessel pressure, in accordance with a predetermined ratio as shown in Figure 45.

The top air pressure is the controlled variable, while reactor vessel pressure is the independent variable. The output from the reactor vessel pressure transmitter (PT-II) automatically adjusts the control point of the controlled variable on the pneumatic-set recorder-controller. Ratio adjustment of the controller is effected by positioning a pneumatic-set dial, calibrated in inches of setpoint movement per pound change in input air pressure. The output from the pneumatic-set recorder-controller is fed to a tri-act controller; output from tri-act controller is parallel to two diaphragm control valves. When the pneumatic circuits in the tri-act controller are in balance, the output from the controller to the valves is steady, and both

Fig. 45 Top air pressure required as a function of vessel pressure.

valves are closed. As the reactor vessel pressure changes, the system becomes unbalanced, and the controllers act to open the appropriate valve to bring the system back to the control point.

High and low pressure alarm switches are provided for an alarm at the control center whenever the output pressure from the pneumatic-set recorder-controller exceeds the preset limits. These switches are also interlocked with the rod drive motors, causing fast run-down of the control rods, when these conditions exist.

(3) Transient Rod Top Air. Air for the top side of the transient rod air piston is supplied from the 1000-psig air reservoir through a pressure regulator set at 750 psi. Two solenoid valves are controlled by a switch on the reactor console. These valves are used to manually adjust the top air pressure to the desired pressure. Downstream of these valves are two solenoid valves which fill or exhaust the top air cylinder as commanded by the control system.

Since placing top air on the transient rod prepares it for firing, this action is controlled through a keyswitch arming sequence. Permission to add top air is denied through this keyswitch unless the following conditions are met: The transient rod shock absorber must be extended; there must be contact between the transient rod and its drive; the transient-rod-hold-air relay must be energized; all control rods must be in contact with their drives; control rod top air must be O.K.; the emergency power generator must be on; and the control rods cannot be in a fast run-down mode.

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 and one 500 yards southeast of the control center designated as No. 2 well pump. The No. 1 well pump is a 20-stage, 10-inch submersible type with a capacity of 400 gpm at a 500-ft head. The No. 2 well pump is a 15-stage, 10-inch line shaft type with a capacity of 550 gpm at a 545-ft 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 gallons.

Water is distributed within the Spert area by two 400-gpm, parallel-connected, booster pumps, which, in conjunction with a pressure control valve, maintain a 70-psig pressure in the distribution system. Under normal conditions, only one booster pump is in service; however, if required the second pump can be put into service.

The water treating equipment for the Spert III Facility is located in the reactor building and consists of that equipment necessary to ensure an adequate supply of high purity water for both the primary- and secondary-coolant systems. The equipment has been sized to permit filling the storage tank in approximately 2 hr. The storage tank is of sufficient capacity to furnish a continuous supply of secondary-coolant water at a rate of 360 gpm for a period of 30 min, based on 60-MW reactor operation for 30 min. All wetted surfaces in the system are stainless steel or an inert organic material so that no minerals or ions are contributed to the water. Well water at 70 psi flows through a softener, a mixed bed demineralizer, and then to the storage tank.

The primary-coolant system is filled with demineralized water from storage utilizing one of the two secondary-coolant pumps. Makeup to the primary system during operation is by means of a 5-gpm makeup pump.

1.2 Softener

A zeolite softener, using a carbonaceous zeolite and operating on the sodium cycle, has been installed in the system to remove water hardness, thereby permitting regeneration of the demineralizer with sulfuric acid. The inter-connecting line, between the softener and the demineralizer has a capacity integrating flow meter with alarm to determine the bed life of the unit.

Table VII is a summary of engineering and operating data for the softener and Table VIII presents a typical analysis of the raw well water.

1.3 Demineralizer

The demineralizer is a single-column, mixed-bed unit of conventional construction with sufficient capacity to demineralize 100 gpm of softened water. The ion-exchange bed will remove all ionic impurities from the water. The effluent flow from the unit has a specific conductance of 2 μ mhos/cm or less, 0.5 gpm total solids maximum, and a pH of about 7.0.

TABLE VII

ENGINEERING AND OPERATING
DATA FOR SOFTENER

Shell	42 in. OD
Shell height	72 in.
Design pressure	75 psi
Resin:	
Type	High capacity nuclear sulphonic polystyrene
Ft ³ of resin	22.5
Bed depth	30 in.
Resin rating:	
kilograins/ft ³	26
kilograins/regeneration	585
lb salt/ft ³	11.7
lb salt/kg	0.45
Regenerant	NaCl
Gal/regeneration	64,500
Maximum flow rate	90 gpm

TABLE VIII

CHEMICAL ANALYSIS
OF SPERT WELL WATER

	ppm
Ca	39
Mg	14
Fe	0.14
Mn	0.1
Na	8.8
K	2.7
B	0.05
SiO ₂	26
NO ₃	1.2
HCO ₃	158
Cl	16
SO ₄	19
F	0.1
Hardness as CaCO ₃	155
Dissolved Solids	205
Specific Conductance at 25°C	332 μ mho/cm
Turbidity	Low
pH	8.2

Breakthrough of the column is determined by a conductivity cell located in the demineralizer-effluent line. A conductivity recorder-controller located in the reactor building control panel, in conjunction with a solenoid valve in the inlet line, automatically stops the flow of water to storage when the conductivity reaches a preset maximum. There are no provisions for the purification or cleanup of the in-pile water. The water will remain in the system until such time as low pH or high activity dictates it be drained to the leaching pond.

The engineering and operating data of the demineralizer unit are summarized in Table IX.

1.4 Storage Tank

Demineralized water is stored at ground level in a 12,000-gal horizontal storage tank located outside the reactor building. This tank is of carbon steel construction, with a 3/16-inch natural-rubber lining and is externally insulated with 2 inches of fiber glass. The water temperature is thermostatically controlled to prevent freezing. The heat is supplied by two 6-kW electric immersion heaters located in each end of the tanks. The liquid level in the tank will be indicated and controlled from the reactor building control panel. A high-low alarm and liquid-level indication are also provided at the control center.

1.5 Chemical Addition

Hydrazine is added to the demineralized water to reduce the oxygen content. It is added to the water in the storage tank through the hydrazine addition station. The concentration of the hydrazine in the demineralized water is maintained at 50 ppm.

1.6 Analysis of Reactor Water

A typical analysis of the water used in the primary coolant system is summarized in Table X.

TABLE IX

ENGINEERING AND OPERATION SUMMARY -- DEMINERALIZER	
Shell	48 in. OD
Shell height	84 in.
Lining and thickness	3/16 vulcanized rubber (in.)
Design pressure	100 psi
Cation resin:	
Type	High capacity nuclear sulphonc polystyrene
Ft ³	18
Regenerant	93% H ₂ SO ₄
Anion resin:	
Type	High capacity nuclear sulphonc polystyrene -- strongly basic type I
Ft ³	30
Regenerant	50% Na OH
Resin ratings:	
Cation	
kilograins/ft ³	13.5
kilograins/regeneration	243
lb acid/regeneration	90
Anion	
kilograins/ft ³	8
kilograins/regeneration	240
lb caustic/regeneration	150
Gal/regeneration	20,000
Maximum flow rate	100 gpm

TABLE X

CHEMICAL ANALYSIS OF PRIMARY COOLANT WATER	
	ppm
Ca	0.06
Mg	0.02
Fe	0.05
Mn	< 0.001
Al	0.09
Cr	0.008
B	0.03
Si	2
Ni	0.01
Co	< 0.005
Cd	< 0.01
Zn	< 0.02
Cl	0.22
Total Solids	0.1
Dissolved Solids	23
Specific Conductance	33 μ mho/cm
pH	9.5

2. AUXILIARY-COOLANT SYSTEM

The auxiliary-coolant system is used to cool the four primary circulating pumps. The cooling water is pumped from a cooling tower, through the pumps, and back to the tower.

2.1 Cooling Tower

The tower is an induced-draft, double-flow, single-cell-type unit, manufactured by the Marley Company. It has a capacity to cool 300 gpm of water from 100 to 80°F with a 60°F wet bulb air temperature. The tower is constructed of weather- and corrosion-resistant materials and is mounted on a concrete basin.

2.2 Circulating Pump

The pump is a Layne and Bowler Pump Co., two-stage pump driven by a 7-1/2-hp motor. It has a capacity of 300 gpm at a 60-ft head and is located on the north side of the cooling tower. The pump is wired to start automatically when any of the primary-coolant pumps are started, or it can be started manually.

2.3 Piping

The circulating pump discharge line runs 4 ft underground and through the east building foundation. Inside the building, the cooling water is distributed to the four primary-coolant pump motors. The return lines for the cooling water run nearly parallel to the supply lines. The return piping to the tower is equipped with a tower bypass line. The purpose of the bypass line is to provide a means for keeping all or part of the return flow from going through the tower during periods of extreme cold weather operation. The tower and pump basin may be drained to a pit located just east of the tower.

3. COMPRESSED AIR SYSTEM

Three compressed air systems are provided at the reactor building. A low pressure compressor supplies air for a service air system and an instrument air system. A high pressure compressor supplies air for a control rod air system (see Section VIII-2.13 for description).

3.1 Service Air System

The service air compressor supplies air at the reactor building for general use. The compressor is a two-stage, air-cooled unit with an intercooler between stages, rated at 50 cfm at a pressure of 125 psig.

The compressed air is piped through a water-cooled, horizontal pipe line type after-cooler, then to an air receiver which has a capacity of 70 ft³. Pressure in the receiver is maintained by stop-start operation of the compressor. Air from the receiver is distributed about the building at 125 psig with outlets provided at all levels.

3.2 Instrument Air System

Air for the pneumatic instruments, diaphragm control valves, and cylinder-operated valves in the reactor building is supplied from the 125 psig service air receiver. The air is piped from the receiver through a prefilter, an air dryer, and then to the distribution lines. The pressure is reduced to 30 psig for distribution about the building.

The air dryer is an automatic dual tower type with electric reactivation on an 8-hr-reversal cycle. Reactivation is accomplished by means of electric heaters embedded in each tower. Each bed is capable of drying 125 lb/hr of 90°F saturated air to an average moisture content of less than 1.0 grain of water vapor per pound of dry air on a continuous basis.

To ensure an adequate supply of instrument air in case of failure in the plant service air system, a pressure control valve, actuated by a pressure switch in the receiver, has been installed in the service air distribution line. This valve closes when the receiver pressure drops to 60 psig.

4. WASTE DISPOSAL

4.1 General

Under normal operating conditions, the only radioactive waste to be handled is the primary-coolant water drained from the system for maintaining water purity or that drained for the maintenance of mechanical components of the system. The activity of the water will be due primarily to the presence of corrosion and/or erosion products in the water. The activity level will be low enough to permit discharge of the water directly to the leaching pond. Since the Spert III reactor is an experimental facility, it is conceivable that rupture and/or melting of the fuel plates will occur during transient tests thereby releasing substantial quantities of fission products into the water. Provision is made to handle high level contamination should it be necessary. The equipment provided and method of operation for handling waste is described in the following section.

4.2 Reactor Building Sump

All building and primary system drains extend to a 3- by 3- by 6-ft stainless steel lined sump located in the process pit area. A 75-gpm vertical submerged sump pump transfers the waste water from the sump to either the leaching pond or the hot waste storage tanks as desired. The pump is automatically controlled by a float switch actuated by the liquid level in the sump.

4.3 Hot Waste Storage Tank

An 8000-gal, underground, hot waste storage tank is located outside the reactor building for the collection of highly contaminated waste water. In the event the activity level of the primary-coolant water is above dumping tolerance, the entire contents of the system may be pumped into the tank for decay and/or holdup until disposal action can be taken. Two alternatives exist as to the disposal of the hot waste once collected in the tank: (a) hold the waste until the activity level has decayed to the extent it can be discharged to the leaching pond or (b) concentrate the hot waste by evaporating the water. The hot concentrate may remain in the tank or be transferred to the Idaho Chemical Processing Plant (ICPP) for permanent storage. Evaporation may be accomplished in the tank by two 30-kW electric immersion heaters.

4.4 Leaching Pond

Since there are no rivers near the Spert III site for waste water disposal, it is necessary to dispose of the water by allowing it to percolate through the

ground until it reaches the water table. A percolation rate of 2 gal/ft²/day was determined by laboratory tests of ground cores obtained from the site. Based on this percolation rate, a leaching pond having an approximate area of 10,000 ft² is provided. Although no activity is expected to accumulate in the pond, it is enclosed by a fence to prevent animals or unauthorized persons from entering the area.

4.5 Auxiliary Leaching Pond

Drains from the water softener and deionizer, are piped to an auxiliary leaching pond having an approximate area of 1000 ft². The piping is vitrified clay to withstand the corrosive effect of the dilute acid and caustic used for regeneration of the ion exchange beds in the demineralizer.

5. HOT STORAGE FACILITIES

Two facilities, dry storage holes and a flooded hot storage well, are provided for the storage of radioactive fuel elements.

5.1 Dry Storage Holes

Eight dry storage holes are embedded in the floor at the south end of the building, east of the reactor. The holes are 6-inch-diameter pipe, 16-ft deep, and plugged with lead covers. Each hole has a nonvalved drain line to the building sump. Fuel elements that are hot and used infrequently will be stored in these holes.

5.2 Hot Storage Well

A hot storage well is located in the southeast corner of the main building. It is a concrete pit 8- by 6- by 16-1/3 ft deep, lined with 16-gauge stainless steel. The well is surrounded by a concrete curb and removable hand rail and is covered with a removable aluminum cover. Deionized water, used for radiation shielding, is supplied through a 1-inch valve. A filter system is installed to keep the water clean. The water in the well may be drained to the building sump through a 2-inch valve.

6. AUXILIARY REACTOR SHUTDOWN SYSTEM

An auxiliary method of reactor shutdown is available in case the mechanical control rods cannot be inserted into the core. This emergency shutdown is accomplished by pumping a neutron poison (boric acid) solution into the reactor system.

The system consists of a 1000-gallon, Type 304 stainless steel storage tank, 3 ft 10 inches in diameter by 13 ft high, connected by a 1-inch, Type 304 stainless steel pipe to the 5-gpm makeup pump. In order to prevent any leakage of the boric acid solution into the makeup pump, a system of four air-operated valves is used and is shown in Figure 46. When the auxiliary shutdown system is required, the valves are energized by a keyswitch at the Spert III control room

located at the control center. This allows the boric acid to feed by gravity to the positive displacement makeup pump which pumps it into the system. The amount of boric acid required to compensate for the reactivity in the E-core is shown in Figure 47 while the addition time is shown in Figure 48.

Fig. 46 Auxiliary shutdown system.

Fig. 47 Amount of boric acid required to compensate for reactivity.

Fig. 48 Time required to poison the Spert III reactor system.

XIII. REFERENCES

1. R. E. Heffner and T. R. Wilson, Spert III Reactor Facility, IDO-16721 (October 1961).
2. J. E. Houghtaling and J. A. Norberg, Addendum to the Spert III Hazards Summary Report -- Low-Enrichment Oxide Core, IDO-17003 (to be issued).
3. C. R. Montgomery et al, Summary of the Spert I, II, III Reactor Facilities, IDO-16418 (November 1957).
4. F. Schroeder, Spert III Hazards Summary Report, IDO-16425 (January 1958).
5. L. G. Gale, Spert III Spun Cast Pipe Experience, IDO-16887 (May 1963).
6. R. E. Heffner et al, Spert III Thermowell Failure and Replacement, IDO-16741 (February 1962).
7. R. E. Heffner et al, Spert III Pressurizer Vessel Failure, IDO-16743 (January 1962).
8. H. Chelemer and L. S. Tong, "Engineering Hot-Channel Factors for Open-Lattice Cores", Nucleonics, 20 (September 1962) pp 68-73.
9. A. H. Spano (ed.), Quarterly Technical Report Spert Project, July, August, September 1959, IDO-16606 (August 1960) p 15.

APPENDIX A -- DESIGN DATA SUMMARY

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX A -- DESIGN DATA SUMMARY

A summary of the characteristics of the reactor and major plant equipment is tabulated in this section. The nuclear characteristics of the reactor as presented represent those used for nuclear design of the core. The numbers quoted are calculated values and have not been verified by experiment. The core has been designed for the following reactor operating conditions: 40-MW nuclear power at 2500 psig and 550°F. A fuel loading sufficient to provide an excess reactivity of about 3,48\$ for experimental purposes under these conditions was selected for the design criteria.

The information included here, such as core loading, core composition, core volume, heat-transfer area, is, therefore, based upon the 64 fuel element core assembly. It is important that the reader realize that this loading is not necessarily representative of the core loading which will be used throughout the experimental program. The core loading for a particular series of experimental investigations can only be determined by experiment and will be dependent upon the particular reactor parameters which are under investigation.

1. GENERAL REACTOR DESIGN DATA

Type -- pressurized-water experimental reactor suitable for boiling or pressurized water experiments
Function -- experimental, reactor transient behavior and safety studies
Moderator -- H₂O
Coolant -- H₂O
Neutron energy -- thermal
Core type -- heterogeneous, rod type
Heat power (steady state max) -- 60 MW plant limited, 40 MW core limited
Power density (at 40 MW) -- 308 kW/liter of UO₂
Design pressure and temperature (max) -- 2500 psig at 650°F

2. REACTOR COMPONENT DESIGN DATA

Core: (Operational Loading)

Configuration -- cylinder (approximately)
Diameter -- $\approx$ 30 inches
Active height -- 38.3 inches
Volume -- 2.21×10^4 in.³
Composition:
 Water -- 6.7 weight percent
 Stainless steel -- 22.5 weight percent
 UO₂ -- 70.8 weight percent
 U -- 62.4 weight percent
 U-235 -- 3.0 weight percent
Fuel load, U-235 -- 57.4 kg
Number of fuel assemblies, all types -- 64

Fuel Elements:

Type -- rectangular, rod type
Number (Operational Loading)
25-rod assembly -- 52
16-rod assembly -- 4
Control rod assembly -- 8
Overall Dimensions:
25-rod assembly -- 2.975 by 2.975 by 52.750 inches
16-rod assembly -- 2.476 by 2.476 by 52.750 inches
Control rod fuel section -- 2.496 by 2.496 by 45-41/64 inches
Fuel content:
25-rod assembly -- 962.5 g U-235
16-rod assembly -- 616 g U-235
Control rod assembly -- 616 g U-235
Pitch -- 0.585 inch (square)
Flow area (inside)/element:
25-rod assembly -- 4.29 in.²
16-rod assembly -- 3.16 in.²
Control rod assembly -- 2.80 in.²
Heat transfer area (active fuel length)
25-rod assembly -- 1402 in.²
16-rod assembly -- 897 in.²
Control rod assembly -- 897 in.²

Fuel Rods:

Type -- cylindrical
Materials:
Fuel tube -- stainless steel, Type 348
Pellets -- 4.8 percent enriched sintered UO₂
Fuel pellet dimensions:
Long pellet -- 0.420 inch OD by 0.766 inch
Short pellet -- 0.420 inch OD by 0.511 inch
Rod dimensions:
Diameter -- 0.466 inch OD
Wall thickness -- 0.020 inch
Overall length -- 40.8 inches (excluding end plugs)
Active length -- 38.3 inches
U-235/rod -- 38.5 g
Gas gap -- 0.003 inch helium

Control Rods:

Type -- rectangular; upper-section is absorber material, lower section is a fuel subassembly
Number -- 8
Composition:
Absorber section -- 1.35-weight-percent boron-10 in Type 18-8 stainless steel; 0.186-inch-thick hollow square box
Fuel section -- see fuel assembly section above
Overall dimensions:
Absorber section (nom) -- 2.496 by 2.496 by 45-31/32 inches
Fuel section (nom) -- 2.496 by 2.496 by 45-41/64 inches

Travel -- 45 inches maximum
Average scram time -- 0.350 sec
Withdrawal Rates:
Fast speed -- 17.4 in./min
Medium speed -- 11.5 in./min
Slow speed -- 5.6 in./min

Transient Rod:

Type -- cruciform; lower section, absorber material; upper section, 18-8 stainless steel
Number -- 1
Composition -- 1.35 weight percent boron-10 in 18-8 stainless steel
Dimensions:
Thickness of blades -- 3/16 inch
Blade width -- 5-1/8 inches
Absorber-section length -- 38 inches
Travel -- 45 inches max
Average drop time -- 0.2 sec

Control Rod Drives:

Type -- Acme nut and screw, air pressure maintains rod in contact with screw and scrams rods
Number -- 5
Motor type -- 480 V, 3 ϕ , 3 speed, constant torque
Motor rating -- 1 hp

Reactor Vessel:

Construction:
Shell -- layer type
Top head -- forged, full-opening flanged hemispherical
Bottom head -- layer type
Inside diameter -- 48 inches
Shell thickness (including clad) -- 3.25 inches
Head thickness -- 3.5 inches
Materials:
Outer shellplate -- VMS-1146 plates 1/4 inch thick
Inner shellplate -- ASTM A-225 Gr B with 304L stainless steel clad
Top head -- VMS-1146 with manually applied Type 309 Cb stainless steel clad
Clad thickness -- 0.125 inch
Design pressure:
Working pressure -- 2500 psig
Instrument shock pressure -- 3500 psig
Design temperature -- 700°F
Overall length, includes head, 19 ft 11-1/8 inches
Maximum diameter of head flange -- 5 ft-7 inches
Number of thermal shields -- 5 (including reactor core flowskirt)
Total thickness of shield -- 5.75 inches
Average length of shields -- 45.5 inches
Inside diameter of inner shield -- 32 inches

Weight of vessel:

Shell	55,784 lb
Head	10,606 lb.
Bottom tee	5,580 lb
Shield	11,430 lb

Total. 83,400 lb

Insulation -- 4 inches of foamglass

3. THERMAL DESIGN DATA -- 64-ASSEMBLY CORE

Coolant -- deionized water

Heat power (max) -- 40 MW core limited

Coolant flow (max) -- 20,000 gpm

Coolant passes through core -- one

Velocity in core (max flow) -- 23.6 ft/sec

Heat transfer area -- 580.9 ft²

Average heat flux (core) -- 2.35×10^5 Btu/hr/ft² at 40 MW

4. PRIMARY-COOLANT SYSTEM DESIGN DATA

Primary Pumps:

Type -- single stage, single suction, double volute, canned rotor

Number -- 4

Capacity, each -- 5000 gpm

Net differential head -- 328 ft of H₂O at 650°F, 2500 psig

Motor:

Rating -- 500 hp

Type -- 480 V, 3Ø, 60 cycle

rpm -- 3550

Weight, each -- 8500 lb

Material -- stainless steel, Type 304L

Heat Exchangers:

Type -- U-tube, evaporative

Number -- 2

Heat transfer capacity, each -- 30 MW

Tube side fluid -- primary coolant, deionized H₂O

Shell side fluid -- deionized H₂O

Design pressure:

Tube side -- 2500 psig

Shell side -- 75 psig

Design temperature:

Tube side -- 668°F

Shell side -- 320°F

Heat transfer surface -- 1940 ft²

Number of tubes -- 367 U's

Tube size, OD -- 5/8 inch, 16 BWG

Tube straight length -- 191 inches

Tube pitch -- triangular

Pressure drop -- 10 psi (tube side)

Materials:

Tubes -- stainless steel, Type 304L

Tube sheet -- A-266-5S, CL 2, clad with stainless steel, Type 304L

Channel -- A-266-5S, CL 2, clad with stainless steel, Type 304L

Channel cover -- A-266-5S, CL 2, clad with stainless steel,
Type 304L

Shell -- A-212, Gr A

Weight:

Full -- 43,200 lb

Empty -- 23,400 lb

Tube bundle -- 11,600 lb

Pressurizer Vessel:

Number -- 1

Design -- Section VIII, ASME Boiler and Pressure Vessel Code

Design pressure -- 2750 psig

Design temperature -- 700°F

Dimensions:

Overall height -- 15 ft 2-11/16 inches

Overall diameter -- 3 ft 3.3 inches

Volume -- $\approx 83.0 \text{ ft}^3$

Shell thickness -- 3.15 inches

Head thickness -- 2.9 inches

Materials:

Backing -- A-212, Gr B, FBX

Clad -- nickel 200 (ASTM B-162), 0.100 inch

Heaters -- electric immersion, 480 V, 3Ø:

Number -- 16

Capacity, each -- 12 kW

Total capacity -- 192 kW

Weight -- $\approx 26,000 \text{ lb}$

Primary-System Piping:

Type -- centrifugally cast stainless steel as per ASTM-A-362-52T,
modified

Wall thickness:

10 to 16 inches -- Schedule 160

8 inches -- Schedule 140

Material -- stainless steel, Type 304L

Insulation -- 3 inches of 85 percent magnesia

Makeup Pump:

Number -- 1

Type -- Triplex, direct flow, vertical reciprocating pump

Piston diameter -- 7/8 inch

Stroke -- 2-1/2 inches

Capacity rating (max):

gpm -- 5

Head -- 3000 psi

Motor:

Type -- 480 V, 3Ø, 60 cycle, gear motor
hp -- 10
rpm -- 280

5. SECONDARY-COOLANT SYSTEM

Pumps:

Type -- single stage, single suction, vertical split case, centrifugal pump
Number -- 2
Capacity, each -- 250 gpm
Total discharge head -- 200 ft of H₂O at 70°F
Motor:
Rating -- 20 hp
Type -- 480 V, 3Ø, 60 cycle
rpm -- 3500
Weight -- 800 lb
Materials -- stainless steel, Type 316

Piping:

All piping to the upstream side of the heat-exchanger, level-control valves:
Type -- seamless or welded
Wall thickness -- Schedule 40
Material -- stainless steel
All other piping:
Type -- seamless
Wall thickness -- Schedule 40
Material -- carbon steel

6. AUXILIARY EQUIPMENT

Water Treatment System:

Softener:

Type -- sodium zeolite
Capacity/cycle -- 50,000 gal
Flow rate (max) -- 90 gpm

Deionizer:

Type -- mixed bed
Capacity/cycle -- 20,000 gal
Flow rate (max) -- 100 gpm
Effluent water specific conductance -- µmho/cm

Deionized water storage tank:

Construction -- carbon steel, rubber-lined, shallow-dished heads
Capacity -- 12,000 gal
Outside diameter -- 108 inches
Length straight section -- 300 inches

Raw Water Cooling System:

Cooling Tower:

Type -- double flow, forced-draft evaporative

Water flow rate -- 300 gpm

Fan motor -- 5 hp, 480 V, 3 ϕ

Overall dimensions:

Length -- 16 ft 0 inch

Width -- 8 ft 6 inches

Cooling Water Pump:

Type -- close couples, vertical turbine pump

Number -- 1

Capacity -- 300 gpm

Total discharge head -- 60 ft H₂O

Motor:

Type -- 480 V, 3 ϕ

Rating -- 7-1/2 hp

rpm -- 1770

Poison Shutdown System:

Poison -- boric acid solution

Addition method -- reactor primary system makeup pump

Reactivity compensated by poison addition -- 17.80\$

Boric acid concentration required -- 4.97 g/l

Time for addition -- 107 minutes

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX B -- ENGINEERING CALCULATIONS

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX B -- ENGINEERING CALCULATIONS

Hydraulic and heat transfer calculations have been made for the core area and heat exchanger of the Spert III system.

1. HYDRAULICS

Hydraulic calculations and experiments have been performed on the Spert III core components and heat exchanger. A summary of the results is presented in the following pages.

1.1 Core Area

The hydraulic characteristics of a 25-rod Type "E" dummy fuel assembly were determined. The test was conducted in a low pressure, low temperature hydraulic test loop. The fuel assembly pressure loss was determined by monitoring the static pressure below and above the fuel assembly with a mercury manometer and a differential pressure gauge. The flow was determined by use of a sharp-edged orifice and monitored with an oil manometer. The experimental overall pressure-loss, assembly-flow relationship is shown in Figure B-1.

Calculations were made to determine the pressure drop as a function of flow rate for the control rods and 16-rod fuel assemblies. The calculated relationships are shown in Figure B-2.

1.2 Heat Exchangers

Calculations were made to determine the tube side pressure drop as a function of flow for various conditions of temperature and pressure. Figure B-3 shows the calculated values for

Fig. B-1 Pressure drop through 25-rod fuel assembly.

Fig. B-2 Pressure drop through control rod and 16-rod fuel assembly.

Fig. B-3 Pressure drop at tube side of heat exchanger.

2500 psig and various temperatures. A limited number of experimental data points are included for comparison. Figure B-4 presents the calculated shell side pressure drop as a function of the heat load when the exchanger is in evaporative service.

Fig. B-4 Pressure drop at shell side of heat exchanger.

2. HEAT TRANSFER

Some calculations have been made on the heat-transfer characteristics of the core and heat exchangers. The calculations on the core include the heat-transfer coefficient and hot channel factor. Performance characteristics of the heat exchangers were calculated for various operating conditions.

2.1 Core Area Heat-transfer Coefficients

Heat-transfer coefficients for various mass flow rates and temperatures have been calculated using the Dittus-Boelter equation. The results are shown in Figure B-5. The heat-transfer

Fig. B-5 Heat transfer coefficient.

The total hot channel factors for the heat flux, film temperature drop, and enthalpy rise are defined as the maximum-to-average ratios of these quantities. The heat flux and film temperature drop consider the local maximum at a point (the "hot spot"), and the enthalpy rise involves the integrated maximum value along a channel (the "hot channel").

Each of the total hot channel factors is the product of a nuclear hot channel factor and an engineering hot channel factor. The nuclear hot channel factor is dependent on the nuclear flux distribution and is not considered in this study. The engineering hot channel factor depends on the fabrication tolerances and flow conditions. The deviations in fabrication occur in a statistical manner; therefore, the subfactors related to these variations are amenable to statistical treatment. The subfactors related to flow conditions are not of statistical nature.

In combining the contributing subfactors to obtain an overall hot channel factor, the subfactors that are evaluated by statistical techniques are combined in a statistical manner.

The engineering hot channel factors were determined using the dimensions reported in the fuel rod and assembly specifications. When quantities were multiplied or divided, the differential error method was used to determine the combined error. The dimensions are as follows:

Fuel Rod Diameter (348 SS), d	0.4660 ± 0.0025 inch
Fuel Density (UO ₂)	10.50 ± 0.10 g/cc
Fuel Pellet Diameter	0.4200 ± 0.0005 inch
Fuel Cross Sectional Area	0.1385 ± 0.0003 in. ²
Active Fuel Length	38.300 ± 0.150 inch
Fuel Enrichment, e	4.80 ± 0.05 percent

coefficients calculated with the Dittus-Boelter equation are within 5 percent of those calculated with the modified Colburn equation.

2.2 Engineering Hot Channel Factor

Engineering hot channel factors for the Spert III Type "E" core have been determined by the method of Chelemer and Tong [8]. In this method, hot spot contributions are combined statistically. Instead of multiplying all hot spot factors together, the statistical approach is cognizance that the probability is vanishingly small that all the adverse effects will occur at the same spot or in the same channel in the core.

Fuel Pin Pitch, p

0.585 $\pm$ 0.005 inch

Coefficient of Thermal Expansion
(348 SS)

10.5 x 10⁻⁶ per °F

The fuel rod diameter and the fuel density dimensions are supposed to be determined with a confidence level of 95 percent. This makes the standard variable equal to 3.92:

The standard deviation was determined by the following formula:

$$\sigma = \pm \text{Tolerance}/Z$$

where

σ = standard deviation

Z = standard variable.

2.21 Fabrication Tolerances.

(1) Subfactors for Pellet Diameter, Density, Enrichment, and Eccentricity. The heat flux engineering hot channel subfactor is

$$F_{ql} E = (w'e)_{\text{max local}} / (w'e)_{\text{nom}}$$

$$F_{ql} E = (w'_{\text{max local}} / w'_{\text{nom}}) (e_{\text{max local}} / e_{\text{nom}})$$

where

$F_{ql} E$ = hot channel subfactor for heat flux

w' = pellet weight per unit length

e = enrichment.

This hot channel subfactor accounts for variations in the pellet diameter, density, and enrichment.

If the variations in w' and e are small and normally distributed, $F_{ql} E$ will be normally distributed with a value of σ given by the equation:

$$\sigma_{ql}^2 = (\sigma_{w'} / \mu_{w'})^2 + (\sigma_e / \mu_e)^2$$

where

σ = standard deviation

μ = mean.

For this core,

$$\sigma_{w'} = 0.27/3.92$$

$$\mu_{w'} = 23.84$$

$$\sigma_e = 0.05/3.92$$

$$\mu_e = 4.80$$

$$\sigma_{q1} = \left[\left(\frac{0.27}{23.84 \times 3.92} \right)^2 + \left(\frac{0.05}{4.80 \times 3.92} \right)^2 \right]^{1/2} = 0.00394$$

$F_{q1}E = 1 + 3 \sigma_{q1} = 1.012$ at 99.9 percent confidence level. This value also applies to the film-drop subfactor $F_{\theta 1}E$. It is also assumed that the enthalpy rise factor is equal to the heat flux subfactor, $F_{\Delta H 1}E = F_{q1}E$. This assumption is made because $F_{\Delta H 1}E$ is determined from the average e and w' of the complete reactor core. These values are determined from the measurements of each individual fuel pin. Since these measurements are not available, it is assumed that $F_{\Delta H 1}E = F_{q1}E$; however, this assumed value should be conservative.

(2) Subfactors for Fuel Rod Diameter, Pitch, and Bowing. The heat flux engineering hot channel subfactor is dependent on the fuel pin diameter because the heat flux is inversely proportional to the surface area.

$$F_{q2}E = (1/d)_{\max \text{ local}} / (1/d)_{\text{nom}} = d_{\text{nom}} / d_{\max \text{ local}}$$

This equation leads to

$$\sigma_{q2} = \sigma_d / \mu_d$$

For this core,

$$\sigma_d = 0.0025/3.92$$

$$\mu_d = 0.4660$$

$$\sigma_{q2} = \frac{0.0025}{0.4660 \times 3.92} = 0.00137$$

$F_{q2}E = 1 + 3 \sigma_{q2} = 1.004$ at 99.9 percent confidence level.

The enthalpy rise engineering hot channel subfactor is effected by variations in rod pitch and bowing. Although these variations are statistical, the average deviation of hot channel pitch must be estimated on the basis of engineering judgement because sample measurements of the pitch variations in the grid-type assembly have not yet been made. The hot channel subfactor is

$$F_{\Delta H2}^E = \frac{\left[\frac{(p^2 - \pi d^2/4)^{5/3}}{d^{2/3}} \right]_{\text{nom}}}{\left[\frac{(p^2 - \pi d^2/4)^{5/3}}{d^{2/3}} \right]_{\text{hot channel}}}$$

$$F_{\Delta H2}^E = \frac{\left[\frac{[(0.585)^2 - \pi(0.233)^2]^{5/3}}{(0.467)^{2/3}} \right]}{\left[\frac{[(0.570)^2 - \pi(0.235)^2]^{5/3}}{(0.467)^{2/3}} \right]}$$

$$F_{\Delta H2}^E = 1.20$$

The film rise engineering hot channel subfactor is the same as the enthalpy rise subfactor except it is evaluated at the hot spot instead of the hot channel. The film rise engineering hot channel is

$$F_{\theta 2}^E = \left(\frac{0.053}{0.602} \right) \left(\frac{0.604}{0.0330} \right) = 1.61$$

2.22 Flow Conditions.

(1) Inlet Flow Distribution. The inlet flow distribution was determined by using data from flow measurements on the Spert III Type "C" core [9].

Inlet factor flow shortage = 6.1 percent

Flow distribution factor = 2.5 percent

Total hot channel subfactor = 15 percent = (6.1) (2.5)

Hot channel subfactor = 1.15

(2) Flow Redistribution. Computer calculations show that a value of 1.05 for this subfactor is a conservative value for a large PWR. Since no other data are available, this value will be used for this core. For transient conditions, this value will increase.

(3) Flow Mixing Inside Assembly. Since flow mixing inside a large PWR reduces this value by only a factor of 0.95, it will be taken as 1.00 for the Spert III Oxide Core.

(4) Heat Transfer Correction. The pitch-to-rod diameter of the Type "E" core is 1.29; and for this large a value, the heat transfer coefficient is usually higher than that used to calculate $F_{\Delta H2}^E$. Therefore, $F_{\Delta H2}^E$ will be smaller than that calculated, but a subfactor of 1.25 will be used to allow for surface scale and crud deposits.

2.23 Summary. The values calculated for the various subfactors that contribute to the total engineering hot channel factors for heat flux, enthalpy rise, and film temperature drop are summarized in Table XI.

TABLE XI
ESTIMATED HOT CHANNEL FACTORS -- TYPE "E" CORE

Type	Subfactor	Heat Flux Factor	Enthalpy Rise Factor	Film Drop Factor	Probability of Not Being Exceeded 99.9% Confidence
Statistical	Pellet diameter, density enrichment, and eccentricity	1.012	1.012	1.012	0.9986
Statistical	Rod diameter, pitch, and bowing	1.004	1.20	1.61	0.9986
Nonstatistical	Inlet flow distribution	---	1.15	1.15	1.0
Nonstatistical	Flow redistribution	---	1.05	1.05	1.0
Nonstatistical	Flow mixing	---	1.00	1.00	1.0
Nonstatistical	Heat transfer correction	---	---	1.25	1.0
Nonstatistical	Total engineering hot channel factors	1.026	1.47	2.46	≈ 1.0

2.3 Heat Exchangers

The specifications issued for the purchase of the heat exchanger required heat transfer capacities of 30 MW with inlet water temperatures of 400 or 600°F and 1 MW with inlet temperatures of 270 to 660°F. Calculated performance data for these conditions submitted by the manufacturer with the bid for construction of the exchangers are given in Table XII. Performance characteristics were also calculated by the manufacturer for other operating conditions. Calculated flow requirements and outlet temperatures as a function of heat load and inlet temperature are shown in Figures B-6 through B-8 for heat loads of 1, 10, and 30 MW.

TABLE XII

CALCULATED HEAT EXCHANGER PERFORMANCE DATA
(Extrapolated from A. O. Smith Curves)

Power Q (MW)	Primary-Water Flow (gpm)	Primary-Water Temperature	
		Inlet Temperature (°F)	Outlet Temperature (°F)
1	500	247	235
5	750	245	235
	1000	244	235
	1250	243	236
	1500	242	236
	1750	241	236
	2000	240	236
	500	301	227
10	750	284	235
	1000	276	238
	1250	277	240
	1500	266	240
	1750	264	242
	2000	262	244
	500	368	218
15	750	334	236
	1000	314	241
	1250	304	244
	1500	296	246
	1750	292	250
	2000	288	253
	500	438	216
20	750	385	236
	1000	355	245
	1250	341	250
	1500	328	254
	1750	322	258
	2000	318	261
	500	506	214
25	750	436	237
	1000	396	249
	1250	378	257
	1500	361	262
	1750	351	266
	2000	345	269
	500	579	213

TABLE XII (Cont.)

CALCULATED HEAT EXCHANGER PERFORMANCE DATA
(Extrapolated from A. O. Smith Curves)

Power Q (MW)	Primary-Water Flow (gpm)	Primary-Water Temperature	
		Inlet Temperature (°F)	Outside Temperature (°F)
30	750	486	238
	1000	436	253
	1250	414	263
	1500	394	270
	1750	380	274
	2000	373	277
	500	660	211
35	750	538	239
	1000	476	257
	1250	450	269
	1500	426	278
	1750	409	282
	2000	401	285

Fig. B-6 Flow requirements and outlet temperatures as a function of inlet temperature -- heat load of 1 MW.

Fig. B-7 Flow requirements and outlet temperatures as a function of inlet temperature -- heat load of 10 MW.

Fig. B-8 Flow requirements and outlet temperatures as a function of inlet temperature -- heat load of 30 MW.

**PHILLIPS
PETROLEUM
COMPANY**

ATOMIC ENERGY DIVISION