Elk River Reactor Technical Specifications July 1, 1963

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ELK RIVER REACTOR

TECHNICAL SPECIFICATIONS

July 1, 1963

Rural Cooperative Power Association

Elk River, Minnesota

CONTENTS

1. INTRODUCTION

This document sets forth the technical specifications that shall govern the design and operation of the Elk River Reactor plant.

References to drawing numbers shall, unless otherwise noted, be to the drawings contained in Reference 1 of these Technical Specifications.

These Technical Specifications consist of (a) design specifications of physical features of the facility and (b) limitations governing operations of the facility to be observed by operating personnel. Except for specifications contained in Secs. 4.8 and 14, any specification of performance of the physical features of the facility shall be considered a design specification.

2. SITE

The specifications relating to the reactor site are as follows:

(1) The Elk River Reactor plant shall be built on a 240-acre site near the village of Elk River, Minnesota, approximately 30 mi northwest of Minneapolis and St. Paul, on the east bank of the Mississippi River. The plant shall be approximately 1050 ft south of the corporate limits of the village of Elk River and approximately 660 ft east of the east bank of the Mississippi River.

(2) The plant location and exclusion area shall be as shown on Fig. 1 of Reference 2.

(3) The minimum distance from the center of the reactor containment building to the boundary of the exclusion area shall be 1050 ft.

(4) The principal activities carried on within the boundaries of the exclusion area shall be restricted to:

(a) the generation of electric power at the Elk River plant of the Rural Cooperative Power Association,

(b) experimental work conducted on behalf of the U. S. Government in a concrete-block building owned by the Fluidyne Corporation 40 ft south of the reactor containment building, and

(c) operation of a sewage disposal plant for the village of Elk River 750 ft north of the reactor containment building.

3. CONTAINMENT

3.1 GENERAL SPECIFICATIONS

The general specifications relating to the reactor containment are as follows:

(1) A containment building shall enclose (a) all principal components of the reactor plant except the reactor control room, the superheater and its auxiliaries, and the turbine-generator and its auxiliaries; (b) all portions of the primary system; and (c) all auxiliary systems which contain primary coolant.

(2) The containment building shall be a concrete-lined gas-tight (as defined in Sec. 3.1 (5)) steel shell consisting of an upright cylinder with a hemispherical top and a semi-ellipsoidal bottom.

(3) The containment-building shell shall be constructed of carbon-silicon steel conforming to ASTM Specification A201-57T, Grade B, heat-treated in accordance with ASTM Specification SA-300.

(4) The containment-building shell shall be designed, constructed, and tested in accordance with Sec. VIII of the ASME Boiler and Pressure Vessel Code for:

(a) a maximum internal pressure of 21 psig at 220 F,
(b) a minimum internal pressure of -0.33 psig,
(c) a minimum free-air volume of 287,000 cu ft, and
(d) a welded-joint efficiency of 90 per cent.

(5) Whenever the containment-building truck door, airlocks, and duct dampers are closed, the fraction of the gaseous contents of the containment-building shell that escapes to the surrounding atmosphere in any 24-hour interval shall not exceed 0.1 per cent at an internal pressure of 21 psig.

(6) The nominal diameter and over-all height of the containment building shell shall be 74 ft and 115 ft, respectively.

(7) The bottom of the containment-building shell shall be approximately 18.5 ft below grade level, and the top of the containment-building shell shall be approximately 96.5 ft above grade level.

(8) Above grade level, the exterior of the containment-building shell shall be covered with a layer of flex-cell insulating material 2 in. thick and an outer coat of paint. The shell shall be protected from corrosion by a coat of bituminous enamel on both the inside and outside surfaces below grade. The flex-cell insulating material can be removed for repairs or during repairs to the insulation providing that the steel shell temperature will not decrease to less than 10 F.

(9) A water storage tank with a nominal capacity of 30,000 gal shall be installed in the topmost portion of the containment-building shell.

(10) The nominal thickness of the concrete lining of the containment-building shell shall be 2 ft in the cylindrical portion and the bottom portion and shall vary from 2 ft at the lower edge of the top portion to 4 in. at the edge of the 30,000-gal water storage tank.

3.2 PENETRATION SPECIFICATIONS

The specifications relating to the containment-building penetrations are as follows:

(1) The containment building shall be penetrated by two airlocks, one freight door, two ventilation ducts, two vacuum-breaker ducts, one emergency-and-test condenser vent pipe, two valve-operating shafts, 18 pipes, and approximately 274 electrical cables.

(2) Each airlock shall have two gasket-sealed doors arranged so that containment-building overpressure will further compress the gasket of any closed door it acts against.

(3) The two doors of each airlock shall be mechanically interlocked to ensure that at least one is closed and sealed under all conditions for which containment integrity provisions are required under Sec. 3.6. Door operators shall be coupled to pressure-relief valves to equalize pressure before a door is opened.

(4) The freight door shall be sealed with gaskets arranged so that containment-building overpressure, as well as the door-closing devices, will compress them.

(5) During reactor operation, removable concrete shield blocks shall be placed in front of the freight door.

(6) Each ventilation duct shall be seal-welded into the containment shell and shall be equipped with two in-line quick closing butterfly dampers held closed by springs whenever not held open by a pneumatic cylinder operated from the service-air supply system.

(7) Each vacuum-breaker duct shall be sealed with a gasketed disk arranged so that containment-building overpressure, as well as the disk, will compress the gasket.

(8) Each vacuum breaker shall be set so as to prevent the atmospheric pressure outside the containment building from exceeding the air pressure inside the containment building by more than 0.22 psi.

(9) The emergency-and-test-condenser vent pipe shall be welded to the containment-building shell and shall be sealed from the atmosphere of the containment building by the shell side of the emergency and test condenser, to which it shall be welded. Valves shall be arranged so that the escape of gases from the containment building into the shell side of the emergency and test condenser can be prevented.

(10) Each valve-operating shaft that penetrates the containment-building shell shall pass through a pipe that shall be welded into the shell and fitted with a packing gland at each end.

(11) Each pipe that penetrates the containment-building shell shall be welded to the shell. Valves on each pipe penetration shall be arranged so that neither the failure of any single valve to close properly nor a pipe rupture will prevent isolation of that line. The following lines are excluded: (a) containment ventilation inlet and outlet lines, (b) both vacuum breakers, and (c) the emergency and test condenser vent.

(12) Each electrical cable that penetrates the containment-building shell falls into one of three classes: multi-conductor control cables (Class A), coaxial cables (Class B), and all other cables (Class C). The method by which each cable in Classes A, B, and C shall be sealed is as follows:

(a) Each Class A cable shall pass through a Thomas and Betts water-tight strain relief connector mounted in a 1-1/4-in. IPS tapped hole in the containment-building shell and a second Thomas and Betts watertight strain relief connector mounted in a 1-1/4-in. IPS tapped hole in a plate 3 in. from the containment-building shell. The chamber between the containment-building shell and the connector mounting plate shall be filled with "NOVOID C" potting compound manufactured by the G & W Electric Specialty Company.

(b) Each Class B cable shall terminate on either side of the containment-building shell at an Amphenol Series 83 pressure-tight bulkhead fitting mounted in a gasketed Micarta plate. The Micarta plate in this type of penetration shall be of type 20201-2 Micarta conforming to ASTM grade G-7. The Micarta penetration shall be designed so that the stresses in the plate under the specified containment-shell design conditions do not exceed 1/6 of the ultimate strength of the material.

(c) Each Class C cable shall pass through pipe sleeves welded into the containment-building shell. In each such penetration, the cable shall enter the pipe sleeve through a Type EYS fitting with an O-Z watertight cable seal inside the containment-building shell and shall leave the pipe sleeve through an O-Z insulator bushing outside the containment-building shell. The sleeves shall be filled with Scotch-Cast Resin No. 4.

(13) Terminals of electrical cables shall be sealed in accordance with Detail 2 of Dwg 12E11809.

3.3 SPRAY-SYSTEM SPECIFICATIONS

The specifications relating to the containment-building spray system are as follows:

(1) The containment-building spray system shall consist of a standpipe extending from the basement of the containment building into the 30,000-gal water storage tank at the top of the containment building, a motor-operated valve at the lower end of the standpipe, and pipes leading from the gate valve to spray nozzles throughout the containment building.

(2) The containment-building spray-system standpipe shall extend into the 30,000-gal water storage tank far enough to ensure that the capacity of the tank below the top of the standpipe is approximately 15,000 gal.

(3) The containment-building spray-system motor-operated valve shall be the only impediment to the flow of water from the portion of the 30,000-gal storage tank above the top of the standpipe to all the containment-building spray-system nozzles.

(4) The containment-building spray system shall be designed to spray approximately 15,000 gal of water into the atmosphere of the containment building in approximately 15 min.

(5) The containment-building spray-system motor-operated valve shall be electrically operable from the reactor control console and mechanically operable from the decontamination room and from a point inside the containment building.

(6) The containment-building spray system shall be automatically actuated following a 10-min delay time by a pressure signal indicating that the containment-building pressure exceeds 2 psig, except that it shall be possible to prevent automatic operation of this system by manual action in the control room. Provisions for automatically actuating the building spray system shall not be required prior to primary-system pressurization with the reactor loaded.

3.4 TESTING SPECIFICATIONS

The specifications relating to tests of the containment building are as follows:

(1) The following tests of the containment building and penetrations shall be made:

(a) The leakage rate of the building shall be determined by pressurizing the building to no less than 10 psig. In this test, the locations to which pressure shall be applied shall include all penetrations and the valves, doors, and gaskets of penetrations subject to opening and normally or occasionally exposed to the internal free volume of the building.

(b) Gasketed closures and ventilation-system closures shall be tested by the soap-bubble, halide detector, helium detector, or other method of equivalent effectiveness in detecting and locating leaks. This test shall be performed using a pressure differential, and the results of this test shall be used as a guide in evaluating leakage.

(c) Electrical penetrations, including the Micarta plates, shall be tested by the halide or helium detector method with a driving pressure differential no less than 0.5 psi, or by the soap bubble method with a driving differential of no less than 5 psi. The results of this test shall be used as a guide in evaluating leakage.

(2) The tests specified in Sec. 3.4(1) shall be performed in accordance with the following schedule:

(a) Not more than six months before initial criticality, and thereafter at intervals no longer than 18 months, tests (a) and (b) shall be performed.

(b) After initial criticality, at intervals no longer than six months, tests (b) and (c) shall be performed.

(c) Any portion of the containment building which has been subjected to maintenance, repair, or other operations which would, during or after these operations, temporarily or permanently affect its performance shall, before any subsequent operation for which containment integrity is required, be tested in accordance with the relevant subparagraphs of Sec. 3.4(1).

(3) The double-gasket seal of the containment-building freight door shall, after each time the seal has been broken and reestablished and before any subsequent operation for which containment integrity is required, be pressure-tested at 21 psig to ensure its integrity.

(4) The containment-building spray system shall be tested at intervals no longer than 18 months to ensure

(a) that the motor-operated valve can be fully opened and closed from the reactor control console and from the decontamination room;

(b) that the motor-operated valve can be actuated by the automatic system; and

(c) that all spray-system piping and nozzles are unobstructed.

3.5 BUILDING ARRANGEMENT SPECIFICATIONS

The specifications relating to the internal arrangement of the containment building are as follows:

(1) The general arrangement of equipment in the containment building shall be as shown in Dwg. 12E-01504.

(2) The containment building shall have a main (top) floor, an intermediate floor, a basement, and a subbasement. Equipment on the main floor shall include the evaporators and two 20-ton air conditioners. The air conditioners shall draw recirculating air from and return it to the basement, intermediate, and main floors. One air conditioner shall draw approximately 3000 cfm of fresh air from the outside. The displaced building air shall be exhausted through a stack at the top of the building. The exhaust air shall pass through one coarse and one fine high-efficiency filter. The air conditioners shall be used for heating when required.

(3) Equipment on the intermediate floor shall include the boric-acid storage tank and its associated compressor, the shield-cooling-system expansion tank, and the makeup-injection-system pumps. The subcoolers and the decay-heat removal system heat exchanger shall be between the intermediate floor and the basement.

(4) Equipment in the basement and subbasement shall include the decay-heat removal system pump, the sump pumps, the motor control centers, switchgear, the startup heater, the retention-tank pump, the fuel-element storage-well pump, the purification-system pumps and heat exchangers, the shield-cooling-system pumps and heat exchanger, the ion exchangers, the retention tanks, and the control-rod drive mechanisms. The subbasement shall be reached by stairs from the basement.

3.6 SPECIFICATIONS RELATING TO CONTAINMENT INTEGRITY PROVISIONS

The specifications relating to containment integrity provisions are as follows:

(1) There shall be seven ERR containment integrity provisions, as follows:

Provision A: sealing all containment building access ports.

Provision B: preventing all access to and egress from the containment building except through airlocks.

Provision C: maintaining all systems for automatically closing containment-building penetrations in operating condition.

Provision D: maintaining the temperature of the steel shell of the containment building above 10 F.

Provision E: maintaining all emergency power supplies, monitors, and automatic emergency equipment associated with the containment building in operating condition.

Provision F: maintaining the 30,000-gal water storage tank full.

Provision G: maintaining at least 15,000 gal of water in the 30,000-gal water storage tank.

(2) Whenever primary-system pressure exceeds 250 psig, except in hydrostatic tests at primary-system temperatures below 212 F, and fuel is in the reactor, containment integrity provisions A, B, C, D, E, and F shall be in effect.

(3) Whenever primary system pressure exceeds 250 psig and irradiated fuel is within the containment building, containment integrity provisions A, B, C, D, and E shall be in effect.

(4) Whenever irradiated fuel is being handled, containment integrity provisions A, B, C, and E shall be in effect.

(5) Whenever any component is being handled in proximity to irradiated fuel, containment integrity provisions A, B, C, and E shall be in effect.

(6) Whenever fuel is in the reactor and any control rod is withdrawn, containment integrity provisions A, B, C, D, E, and F shall be in effect.

(7) Whenever primary-system pressure exceeds 250 psig in a hydrostatic test at primary system temperatures below 212 F and fuel is in the reactor, containment integrity provisions A, B, C, E, and G shall be in effect.

4. PRIMARY COOLANT SYSTEM

4.1 GENERAL SPECIFICATIONS

The general specifications relating to the primary coolant system are as follows:

(1) The primary coolant system shall consist of two loops connected to diametrically opposite sides of the reactor vessel, each of which shall include the tube side of an evaporator, the tube side of a subcooler, two isolation valves between the reactor vessel and the evaporator, and one isolation valve between the subcooler and the reactor vessel. The primary system arrangement, including piping, valves, measuring instruments, and connections to the system, shall be as shown on Dwg 12E09006, unless changed in accordance with Specification 4.1(7).

(2) Steam and water in each primary-coolant-system loop shall flow by natural circulation.

(3) Heat shall be transferred from the primary-coolant-system loops to the secondary-coolant-system loops by conductive evaporation of the secondary coolant on the shell side of the evaporators and by conductive heating of the secondary coolant on the shell side of the subcoolers.

(4) The primary coolant system shall not contain

(a) more than 5600 gal of water at 564 F, or

(b) any quantity of water whose stored energy exceeds that of 5600 gal of water at 564 F.

(5) All primary coolant system main-loop piping shall be of stainless steel, ASTM Type 304, Schedule 80, and shall be designed, fabricated, and tested in accordance with Sec. 1 of ASA Code B31.1 for Pressure Piping. Except as provided in Specification 14.3(12), all other primary system piping shall be designed, constructed, and tested in accordance with the ASA Code for pressure piping. All systems exposed to primary-system pressure shall be designed for a pressure no less than 1250 psig and for temperatures no less than the temperatures of the fluids they contain.

(6) Except as otherwise specified,

(a) All pressure vessels in the primary system shall be designed, fabricated, and tested in accordance with the ASME Boiler and Pressure Vessel Code, and

(b) All piping, valve bodies, and vessel surfaces in contact with primary coolant shall be of stainless steel.

(7) Additional penetrations to the primary-coolant-system shall be designed, fabricated, and tested in accordance with the applicable provisions of the relevant editions of the ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping except as otherwise specified for hydrostatic tests in Sec. 14.3(12). Such additions shall be limited to vent, drain, and instrument connections, the inside diameter of which shall be of no greater than 1 in.

4.2 REACTOR-VESSEL SPECIFICATIONS

The specifications relating to the reactor vessel are as follows:

(1) Construction of the vessel and modifications shall conform to the details shown on the following drawings:

	Drawing Number	Reference Number
Vessel Construction:		
general arrangement	4789-1-E	5
lower head assembly	4789-2-D	5
nozzle details	4789-3-D	5
misc. flange and nozzle details	4789-4-D	5
miscellaneous	4789-11-D	5
Proposed Modifications:		
16-in. nozzles	12-SK-078	6
8-in. nozzles	12-SK-080	6
10-in. nozzles	12-SK-081	6
inside edges of nozzles	Fig. E	7

(2) The design temperature and pressure of the reactor vessel shall be 650 F and 1250 psig, respectively.

(3) The rated operating temperature and pressure of the reactor vessel shall be 564 F and 1153 psig, respectively.

(4) The base metal of the shell courses and heads of the reactor vessel shall be manganese-molybdenum alloy steel conforming to ASTM Specification SA-302, Grade B.

(5) The base metal of the main closure flanges of the reactor vessel shall be of carbon steel conforming to ASTM Specification SA-105, Grade II.

(6) The roll-bonded internal cladding of the reactor vessel shall be of stainless steel conforming to ASTM Specification SA-264, Grade 3.

(7) The weld-overlay internal cladding of the reactor vessel shall be of either Type 309 or Type 308L stainless steel.

(8) The minimum thickness of the internal cladding of the reactor vessel shall be 0.109 in.

(9) The nominal inside diameter, inside height, and wall thickness of the reactor vessel shall be 7 ft, 25 ft, and 3 in., respectively.

(10) Above the horizontal midplane of the reactor core, the reactor vessel shall be fitted with 14 nozzles whose numbers, nominal sizes, and purposes shall be as follows:

number	nominal size, in.	purpose
4	12	top-head access
1	1	vent line (plugged)
2	1-1/2	liquid-level measurement
2	10	steam outlet
1	1-1/2	emergency coolant inlet
2	8	feedwater inlet
2	16	forced-circulation outlet (for future use)

(11) Below the horizontal midplane of the reactor core, the reactor vessel shall be fitted with 17 nozzles, whose numbers, nominal sizes, and purposes shall be as follows:

number	nominal size, in.	purpose
2	16	forced-circulation inlet (for future use)
13	4	control rod drive
1	2-1/2	liquid-poison injection
1	1-1/2	liquid-level measurement

(12) The top head of the reactor vessel shall be fastened to the shell of the reactor vessel by a flanged joint secured by 34 studs and sealed by a soft iron gasket.

(13 Operation of the reactor vessel within the design pressure-temperature conditions shall be limited to 250 full pressure-temperature cycles of loading.

Lesser or greater pressure variations shall be included in the allowable 250 cycles of loading, in accordance with the following table of cycle equivalents:

Operating pressure variations (1)	Equivalent "full range" cycles
100%	1.00
90%	0.70
80%	0.50
70%	0.25
60%	0.15
50%	0.05
40%	0.02
30%	0.01
20% or less	0

Hydrostatic test pressures (2)	Equivalent "full range" cycles
150%	1.8
140%	1.5
125%	1.1
115%	0.8
110%	0.7
100%	0.5

(1) Pressure variation expressed in per cent of 1153 psig.

(2) Hydrostatic tests, expressed in per cent of the design pressure (1250 psig), conducted at temperatures not below NDT temperature plus 60 F.

For the purpose of computing operating time, records of the megawatt days produced in the reactor will be kept.

(14) Records shall be kept of all operating cycles and hydrostatic tests to permit calculation of the equivalent number of full cycles to which the reactor vessel has been subjected during its service lifetime.

4.3 COOLANT SPECIFICATIONS

The specifications relating to the primary coolant are as follows:

(1) The primary coolant shall be light water and shall, during operation with the primary system pressure above 250 psig, conform with the following requirements:

maximum conductivity	1.0 micromho/cm
approximate pH	7.0

The chloride content of the primary coolant shall not exceed 0.1 ppm at any time.

(2) The primary coolant shall be pressurized by evaporation, by thermal expansion, or by hydrostatic testing techniques.

4.4 EVAPORATOR SPECIFICATIONS

The specifications relating to the evaporators are as follows:

(1) Each evaporator shall be designed, fabricated, and tested in accordance with Sec. VIII of the 1956 edition of the ASME Boiler and Pressure Vessel Code for a tube-side pressure of 1250 psig and a shell-side pressure of 1000 psig.

(2) The tubes of each evaporator and all other evaporator surfaces in contact with the primary coolant shall be of Type 304 stainless steel.

(3) All surfaces of each evaporator in contact with the secondary coolant, other than the tubes, shall be of carbon steel.

(4) Each evaporator shall be capable of transferring heat from the primary coolant to the secondary coolant at the minimum rate of 78,300,000 Btu/hr when saturated steam at approximately 564 F enters the tube side at the approximate rate of 126,500 lb/hr and when secondary-system feedwater at approximately 486 F enters the shell side at the approximate rate of 104,000 lb/hr.

4.5 SUBCOOLER SPECIFICATIONS

The specifications relating to the subcoolers are as follows:

(1) Each subcooler shall be designed, fabricated, and tested in accordance with Sec. VIII of the 1956 edition of the ASME Boiler and Pressure Vessel Code for a tube-side pressure of 1250 psig and a shell-side pressure of 1000 psig.

(2) The tubes of each subcooler and all other subcooler surfaces in contact with the primary coolant shall be of Type 304 stainless steel.

(3) All surfaces of each subcooler in contact with the secondary coolant, other than the tubes, shall be of carbon steel.

(4) Each subcooler shall be capable of transferring heat from the primary coolant to the secondary coolant at the minimum rate of 15,500,000 Btu/hr when condensate at approximately 564 F enters the tube side at the

approximate rate of 125,000 lb/hr and when secondary-system feedwater at approximately 350 F enters the shell side at the approximate rate of 104,000 lb/hr.

4.6 ISOLATION-VALVE SPECIFICATIONS

The specifications relating to the primary-coolant-system isolation valves are as follows:

(1) Each of the two isolation valves in each primary-coolant-system loop that are installed between the reactor vessel and the evaporator tube-side inlet shall be a manually operated 8-in. gate valve constructed of stainless steel.

(2) The isolation valve in each primary-coolant-system loop that is installed between the subcooler tube-side outlet and the reactor vessel shall be a manually operated 6-in. gate valve constructed of stainless steel.

4.7 SHIELDING SPECIFICATIONS

The specifications relating to the biological shielding of the primary coolant system are as follows:

(1) The biological shield shall entirely surround the reactor vessel and that portion above the vessel and enclosing the pipe penetrations shall be removable.

(2) The biological shielding surrounding the cylindrical portion of the reactor vessel shall consist of steel, lead, monolithic ordinary concrete, and ordinary concrete blocks.

(3) The biological shielding above the reactor vessel shall consist of removable heavy-concrete slabs; that below the reactor vessel shall consist of lead and monolithic heavy concrete.

(4) The minimum density of the ordinary concrete in the reactor-vessel biological shielding shall be 2.5 g/cc; that of the heavy concrete in the reactor-vessel biological shielding shall be 4.3 g/cc.

(5) Radiation shall not emerge from the reactor-vessel biological shielding intensely enough to expose personnel to doses or dose rates exceeding the permissible doses and dose rates set forth in Title 10 of the Code of Federal Regulations, Part 20.

(6) Each evaporator shall be enclosed on three sides by a concrete biological shielding wall approximately 10 ft high and 2 ft thick with a shadow-shielded entrance, and on the fourth side by the concrete lining of the containment building.

(7) Heat shall be removed from lead portions of the reactor-vessel biological shielding by a shield cooling system consisting of a shell-and-tube heat exchanger, two centrifugal shield-cooling pumps with discharge filters, ten parallel-connected tubes embedded in the lead of the reactor-vessel biological shielding, a surge tank, interconnecting piping, and valves. The shield-cooling-system coolant shall be demineralized light water.

(8) The shield-cooling-system heat exchanger shall be capable of transferring heat from the shield coolant to service water at the minimum rate of 350,000 Btu/hr when the shield coolant enters the shell side at approximately 120 F at the approximate rate of 35 gal/min and when service water enters the tube side at approximately 90 F at the approximate rate of 70 gal/min.

(9) Each shield-cooling-system pump and its associated filter shall be rated at 35 gal/min at a net developed head of 110 ft.

(10) Whenever the reactor is critical, the shield cooling system shall be operated so that the temperature of the shield coolant does not exceed 140 F.

4.8 SPECIFICATIONS GOVERNING OPERATING VARIABLES

The specifications governing primary-coolant-system operating variables are as follows:

(1) The bulk temperature of the primary coolant leaving the reactor core shall not exceed 569 F.

(2) The internal pressure imposed on the reactor vessel, including hydrostatic test pressure, shall not exceed 250 psig, unless the temperature of the reactor vessel as measured by the thermocouples attached to the vessel (shown on Dwg 12E-11810 of Reference 8) equals or exceeds the higher of these two temperatures:

(a) 160 F (initially).

(b) The sum of 160 F and the higher nil-ductility transition (NDT) temperature shift (in degrees Fahrenheit) of the materials A10511 and A302B, as correlated by Charpy V-notch impact tests on surveillance specimens removed from the reactor vessel.

The NDT temperature shift shall be the difference between the initial NDT temperature of the material samples placed in the irradiation capsules and the highest measured NDT temperature determined from the irradiated samples subsequently removed from the reactor vessel.

The sum of previous NDT temperature shifts, as periodically determined, shall be added to the 160 F to establish the safe pressurization temperature of the vessel for the interval of operation before the next NDT temperature shift is determined.

(3) Whenever the internal pressure of the reactor vessel is less than 955 psig, the temperature of the vessel shall not be allowed to change, except during a scram shutdown, at any instantaneous rate exceeding 2.5 F/min.

(4) Whenever the internal pressure of the reactor vessel equals or exceeds 955 psig, the temperature of the vessel shall not be allowed to change, except during a scram shutdown, at any instantaneous rate exceeding the rate R computed by this equation:

$$R = 483 - 0.348\ P,$$

where

R = maximum allowable reactor-vessel temperature change rate at pressure P, ^{o}F/hr, and

P = reactor-vessel internal pressure, psig.

(5) In routine startup and shutdown operations of the reactor, the temperature of the primary coolant shall be monitored and controlled so as to keep the reactor-vessel temperature change rate below the limits of Specifications 4.8(3) and 4.8(4).

(6) Whenever fuel is in the reactor and the head of the reactor vessel is in place, control rods shall not be withdrawn unless both primary coolant loops to the evaporators are open and in operating condition; except that during regularly scheduled tests, control rods may be so withdrawn with one loop isolated from the reactor vessel, provided that all control rods must be inserted during the operation of isolation valves and during periods of loop temperature changes.

(7) The nominal operating pressure of the reactor shall not exceed 1153 psig.

4.9 REACTOR VESSEL LEAKAGE MONITORING SPECIFICATIONS

The specifications relating to reactor-vessel leakage monitoring are as follows:

(1) Leakage from any point on the surface of the reactor vessel shall be detected and monitored by the following means;

(a) A locally indicating thermometer installed in the low point of the vent line from the cavity around the reactor vessel shall indicate rises above the normal temperature.

(b) Two drainable sight glasses, one in the low point of the vent line and the other in the drain line of the cavity around the reactor vessel, shall permit collection of water condensate from the air. Water collected in either sight glass shall be drained and analyzed to determine if coolant has escaped from the primary system.

(2) If the vent-line temperature or the cavity-drain-line water level is higher than normal, an alarm shall be actuated in the control room. Vent-line temperature shall be indicated in the control room and shall be recorded hourly until normal operating conditions are determined.

(3) If a reactor-vessel leak is detected, the reactor shall be shut down and maintained at atmospheric pressure.

5. PRIMARY COOLANT AUXILIARY SYSTEMS

5.1 PRESSURE-RELIEF-VALVE SPECIFICATIONS

The specifications relating to the primary-coolant-system pressure-relief valves are as follows:

(1) The primary coolant system shall be protected against overpressure by two spring-loaded pressure-relief valves designed, constructed, and tested in accordance with all applicable provisions of the ASME Boiler and Pressure Vessel Code.

(2) One primary-coolant-system pressure-relief valve shall vent primary steam to the interior of the containment building if the primary-coolant-system pressure should rise to or exceed 1240 psig. The other primary-coolant-system pressure-relief valve shall vent primary steam to the interior of the containment building if the primary-coolant system pressure should rise to or exceed 1250 psig.

(3) The minimum steam-flow capacity of each primary-coolant-system pressure-relief valve when fully open shall be 129,000 lb/hr.

5.2 PURIFICATION-SYSTEM SPECIFICATIONS

The specifications relating to the primary-coolant purification system are as follows:

(1) The primary-coolant purification system shall consist of a regenerative heat exchanger, a cooler, two shielded prefilters, two shielded ion-exchange columns containing anion and cation resins in mixed beds, two afterfilters, two centrifugal pumps, interconnecting piping, and valves.

(2) The prefilters, ion-exchange columns and afterfilters shall be connected between the cooler outlet and the pump inlet header so as to form two parallel branches each containing one prefilter, one ion-exchange column, and one afterfilter. Valves shall be installed so that either branch may be isolated from the remainder of the system.

(3) The pumps shall be connected in parallel between the ion-exchange column outlets and the return inlet of the regenerative heat exchanger. The design flow rate of the system with one pump operating shall be 9 gpm.

(4) Each ion-exchange column and its associated prefilter shall be designed to maintain the conductivity of the primary coolant under normal operating conditions at or less than 1 micromho/cm.

(5) The temperature of the water entering the ion-exchange columns shall not be allowed to exceed 130 F.

5.3 MAKEUP-SYSTEM SPECIFICATIONS

The specifications relating to the primary-coolant makeup system are as follows:

(1) The primary-coolant makeup system shall consist of a collecting tank, two parallel-connected positive-displacement injection pumps, a full-flow filter, interconnecting piping, and valves.

(2) The capacity of the collecting tank shall be at least 50 gal.

(3) Each injection pump shall be capable of injecting 12.5 gpm of water into the primary coolant system from a 3-ft suction head against positive discharge pressures up to 1250 psig.

(4) The collecting-tank inlet shall be connected to the 30,000-gal containment-building overhead-storage-tank supply line so that makeup water may be injected into the primary coolant system either from the overhead storage tank or from the RCPA treated-water supply system.

5.4 STARTUP HEATING SYSTEM SPECIFICATIONS

The specifications relating to the primary-coolant startup heating system are as follows:

(1) The primary-coolant startup heating system shall consist of a shell-and-tube heat exchanger, interconnecting piping, and valves.

(2) The startup-heating-system heat exchanger shall be designed, fabricated, and tested in accordance with Sec. VIII of the 1956 edition of the ASME Boiler and Pressure Vessel Code for a tube-side pressure of 1250 psig and a shell-side pressure of 700 psig.

(3) The startup-heating-system heat exchanger shall be interconnected with the primary coolant system, the boric-acid injection system, the decay-heat-removal system, and the RCPA 600-psig steam system so that heat from steam flowing through its shell side may be transferred to primary coolant circulated through its tube side and through the reactor vessel by the decay-heat-removal-system pump.

(4) The startup heater line shall contain a temperature switch which actuates the low-reactor-water temperature interlock specified in Sec. 8.3.5.

5.5 RECOMBINER SYSTEM SPECIFICATIONS

The specifications relating to the primary-coolant recombiner system are as follows:

(1) The primary-coolant recombiner system shall consist of a catalytic hydrogen-oxygen gas recombiner, a condenser, a gas drier, two parallel-connected activated-charcoal absorbers, interconnecting piping, and valves.

(2) The components of the primary-coolant recombiner system shall be arranged

(a) to collect the mixture of non-condensable gases, steam, and water vapor collected in the outlet channel section of the tube side of each evaporator,

(b) to expand the mixture across a needle valve at the recombiner inlet,

(c) to direct the mixture into the catalytic bed,

(d) to recombine the hydrogen and oxygen of the mixture to form steam,

(e) to condense the steam formed by recombination and that extracted from each evaporator channel,

(f) to direct the condensate and the water extracted from each evaporator channel to the primary-coolant makeup collecting tank, and

(g) to divert the non-condensable gases either past the fission-product monitor and thence to the stack or through the gas drier and the charcoal absorbers past the fission-product monitor and thence to the stack.

(3) Nominal flow rates in the primary-coolant recombiner system shall be as follows:

rate at which primary coolant is extracted from the lower channel section of each evaporator	1300 lb/hr
rate at which non-condensable gases are vented to the stack .	0.3 scfm
rate at which primary coolant is returned to the makeup collecting tank	5 gpm

5.6 CONTROL-ROD-SEAL COOLING SYSTEM SPECIFICATIONS

The specifications relating to the control-rod-seal cooling system are as follows:

(1) The control-rod-seal cooling system shall consist of

(a) valves and a piping network connecting the discharge side of the primary-coolant purification-system pumps to the reactor side of each control-rod drive-shaft water seal and to the bottom of each control-rod-drive mechanism lower housing, and

(b) valves and a piping network connecting each control-rod drive-shaft leakoff chamber to the primary-coolant makeup collecting tank.

(2) The control-rod-seal cooling system shall prevent the temperature of the water in each control-rod drive-shaft leakoff chamber from exceeding 125 F.

(3) A relief valve that shall discharge to the interior of the containment building shall prevent the pressure difference across each control-rod drive-shaft lip seal from exceeding 30 psi.

5.7 CONTROL-AIR SUPPLY SYSTEM SPECIFICATIONS

The specifications relating to the control-air supply system are as follows:

(1) The control-air supply system shall consist of a motor-driven low-pressure air compressor, an after cooler, an accumulator, distribution valves and piping, and at least one automatically actuated regulating valve connecting the system to an independent source of compressed air.

(2) The control-air supply-system air compressor shall have a rated capacity of 31 scfm of oil-free air at 100 psig and shall be of the single-cylinder, single-state reciprocating type.

(3) The control-air supply-system air compressor shall normally operate automatically to maintain the pressure in the control-air supply-system air accumulator between 80 and 100 psig.

(4) The valve between the control-air supply system and the service-air supply system shall regulate the control-air supply-system pressure sensed by the valve at 60 psig whenever the control-air supply-system air compressor is not maintaining the accumulator pressure above 60 psig.

6. SECONDARY COOLANT SYSTEM

6.1 GENERAL SPECIFICATIONS

The general specifications relating to the secondary coolant system are as follows:

(1) The secondary coolant system shall consist of the shell side of each subcooler, the shell side of each evaporator, two moisture separators, a coal-fired superheater, a turbine generator set having a 22 Mwe turbine name plate rating, a surface condenser, two parallel-connected condensate pumps, an air-ejector condenser, two condensate heaters, one deaerating condensate tank, two parallel-connected feedwater pumps, one feedwater heater, two feedwater control valves, interconnecting piping, and isolation valves. All secondary-system piping shall be designed, constructed, and tested in accordance with the ASA Code for Pressure Piping.

(2) The secondary coolant shall be light water and shall conform with the following requirements:

maximum conductivity (in evaporator shell)	500 micromhos/cm
maximum chloride content	0.10 ppm
maximum chloride content (in evaporator shell)	0.5 ppm
maximum silica content	0.01 ppm
maximum silica content (in evaporator shell)	3.0 ppm
maximum oxygen content	0.1 cc/liter
maximum oxygen content (outlet of deaerating cond. tank)	0.005 cc/liter
maximum total solids content	0.5 ppm
maximum total solids content (in evaporator shell)	250 ppm
maximum hardness	0.1 ppm
approximate pH	9.0

(3) The points at which portions of the secondary coolant may be discharged to the air inside the containment building shall be as follows:

(a) each of two evaporator pressure-relief valves;

(b) each of two evaporator drain valves;

(c) each of two subcooler drain valves;

(d) each of two moisture-separator vent valves;

(e) each of two evaporator-feedwater-valve bypass-line vent valves; and

(f) other miscellaneous points such as instrument drains, instrument vents, coolant-sampling connections, etc.

(4) The points at which portions of the secondary coolant may be discharged to the air outside the containment building shall be as follows:

(a) each of four superheater pressure-relief valves;

(b) each of four superheater blowdown valves;

(c) each of three superheater vent valves;

(d) each of two superheater steam line drain valves;

(e) the air-ejector condenser vent; and

(f) other miscellaneous points such as instrument drains, instrument vents, coolant sampling connections, etc.

(5) The points at which portions of the secondary coolant may be released to the liquid-effluent discharge lines leading off-site shall be as follows:

(a) each of two evaporator blowdown valves;

(b) each of four superheater blowdown valves; and

(c) other miscellaneous points such as piping drains, etc.

(6) Concentrations above natural background of radioactive materials in the secondary system coolant shall not exceed those levels specified in 10 CFR Part 20, Appendix "B", Table II and footnotes thereto.

(7) Pressure-relief valves in the secondary system shall be designed, fabricated, tested, and set in conformance with the ASME Boiler and Pressure Vessel Code. The secondary side of each evaporator shall have a relief valve set to relieve pressure at 975 psig.

6.2 OPERATING SPECIFICATIONS

The operating specifications relating to the secondary coolant system are as follows:

(1) The secondary-coolant-system flow rate shall be at all times sufficiently high to remove the heat generated in the reactor core and in the superheater without exceeding the temperature limits of Specification 6.2(6).

(2) The nominal secondary-coolant-system full-power flow rate shall be 225,000 lb/hr.

(3) Whenever the reactor is operating, the tube-side pressure in either evaporator and in either subcooler shall not exceed the corresponding shell-side pressure by more than 275 psi.

(4) Whenever the reactor is operating, the logarithmic mean temperature difference between the primary and secondary coolants in either evaporator shall not exceed 39 F, and the logarithmic mean temperature difference between the primary and secondary coolants in either subcooler shall not exceed 125 F.

(5) Maximum design and operating pressures and maximum relief-valve settings at various points in the secondary coolant system shall be as follows:

	maximum pressure, psig		
	design	operating	relief-valve setting
evaporator outlet	1000	910	975
superheater outlet	975	863	900
turbine inlet	900	850	960
subcooler inlet	1000	923	None

(6) Maximum design and operating temperatures at various points in the secondary coolant system shall be as follows:

	maximum temperature, ^{o}F	
	design	operating
evaporator outlet	575	536
superheater outlet	905	905
turbine inlet	900	900
subcooler inlet	575	350

(7) Minimum operating temperatures of the secondary coolant at various points in the secondary coolant system shall be as follows:

	minimum operating temperature, ^{o}F
deaerating condensate tank makeup inlet	50
shell-side inlet of each subcooler	70

(8) Whenever either or both of the feedwater pumps are supplying secondary coolant to either or both of the subcoolers, the total weight of the portion of the secondary coolant within the containment building shall not exceed 30,000 lb.

6.3 SECONDARY-COOLANT MAKEUP SYSTEM SPECIFICATIONS

The specifications relating to the secondary-coolant makeup system are as follows:

(1) The secondary-coolant makeup system shall consist of a water demineralizing plant, a storage tank, two pumps, interconnecting piping, and valves.

(2) The design capacity of the secondary-coolant makeup-system demineralizing plant shall be 30 gpm; that of the secondary-coolant makeup-system storage tank shall be 3000 gal; and that of each secondary-coolant makeup-system pump shall be 55 gpm.

(3) The secondary-coolant makeup water shall conform with the following requirements:

maximum conductivity	1 micromho/cm
maximum chloride content	0.1 ppm
maximum silica content	0.01 ppm
maximum total solids content	0.5 ppm
maximum hardness	0.1 ppm
approximate pH	9.0

7. REACTOR CORE

7.1 GENERAL SPECIFICATIONS

The general specifications relating to the reactor core are as follows:

(1) The reactor core shall consist of the fuel elements specified in Sec. 7.4, the control rods specified in Sec. 8.1.2, the shroud structure specified in Sec. 7.5, the supporting structure specified in Sec. 7.6, and the neutron source specified in Sec. 7.7.

(2) The reactor core components shall be arranged as shown in Allis-Chalmers Dwg. 12D-04000.

(3) The axis of the reactor core shall coincide with the vertical axis of the reactor vessel.

(4) A cylindrical stainless-steel thermal shield with a nominal thickness of 1 in., a nominal outside diameter of 83 in. and a nominal height of 144 in. shall surround all vertical sides of the reactor core.

(5) The neutrons from reactor core shall be moderated and reflected by the primary coolant. The nominal radial thickness of the primary-coolant reflector shall be 8 in.

(6) The reactor core shall include no more than 148 fuel elements, no more than 178 kg of uranium-235, no more than 13 kg of uranium-238, and no more than 3785 kg thorium-232.

(7) The steam-void coefficient of reactivity shall be negative.

(8) Except during the initial startup and testing of the reactor, the temperature coefficient of reactivity shall be negative whenever the reactor is critical.

(9) The primary coolant shall flow up through the core in one pass.

(10) The average fuel burnup shall not exceed 10,000 Mwd/T.

(11) The reactor core shall have predicted nuclear design characteristics as follows:

(a) The initial effective multiplication factor of the cold, clean core shall be 1.100.

(b) The initial effective multiplication factor of the hot core with equilibrium xenon and samarium shall be 1.018.

(c) The reactivity change at zero power due to change in reactor temperature from 68 F to 536 F shall be 0.013.

(d) The reactivity change due to the Doppler effect for a change of power from 0 to 58.2 Mw with no voids in the core shall be 0.003.

(e) The temperature coefficient of reactivity for changes in temperature of moderator and fuel shall be $-1.5 \times 10^{-4}/^{\circ}C$ at 536 F and $+6 \times 10^{-5}/^{\circ}C$ at 68 F.

(f) The void coefficient of reactivity shall be $-1.2 \times 10^{-3}/\%$ void at 0% void and $-1.9 \times 10^{-3}/\%$ void at 17.6% void.

7.2 THERMAL SPECIFICATIONS

The specifications relating to the thermal characteristics of the reactor core are as follows:

(1) The steady-state power level of the reactor shall not exceed 58.2 Mw.

(2) The maximum local heat flux in the reactor core shall not exceed 313,000 Btu/hr-ft^2.

(3) The burnout safety factor of the reactor core (defined as the ratio of the heat flux required to melt the cladding of any fuel element in the core to the actual steady-state heat flux at the same point) shall equal or exceed 3.2. The burnout heat flux shall be based upon The Correlation of Nucleate Boiling Burnout Data, ASME Paper No. 57-HT-21 by P. Griffith.

7.3 FUEL-MATERIAL SPECIFICATIONS

The specifications relating to the reactor fuel material are as follows:

(1) The fuel material of all 25 tubes of each regular fuel element and of the center tube of each spiked fuel element shall be a mixture of uranium dioxide and thorium dioxide in the following proportions, by weight: uranium dioxide, 4.60 ± 0.02 per cent; thorium dioxide, 95.40 ± 0.02 per cent.

(2) The fuel material of all but the center tube of each spiked fuel element shall be a mixture of uranium dioxide and thorium dioxide in the following proportions, by weight: uranium dioxide, 5.57 ± 0.02 per cent; thorium dioxide, 94.43 ± 0.02 per cent.

(3) All uranium in the fuel material shall be enriched to at least 93.13 per cent in uranium-235.

(4) The fuel material shall be mixed, milled, pressed, and sintered to form homogeneous cylindrical solid-solution ceramic pellets whose minimum density shall be at least 94 per cent of the theoretical, or maximum possible, density of the solution.

(5) Each fuel pellet shall have a nominal diameter of 0.407 in. and a nominal length of 0.500 in.

7.4 FUEL-ELEMENT SPECIFICATIONS

The specifications relating to the reactor fuel elements are as follows:

(1) Each fuel element, whether regular or spiked, shall consist of 25 fuel tubes in a 5 x 5 array, fuel-tube end plugs and fasteners, an upper grid plate, a lower grid plate, two fuel-tube spacers, an upper end fitting and a lower end fitting.

(2) Materials of fuel-element components shall be as shown on Dwg. 12D-02000.

(3) The configuration, arrangement, and principal dimensions of each fuel element shall be as shown on Dwg. 12D-02000.

(4) The center fuel tube of each fuel element shall be removable; all other tubes of each fuel element shall be fixed. The center fuel tube of each fuel element shall be replaceable with a removable burnable poison pin or a radiation test center pin.

(5) Each fuel tube shall have a nominal length (exclusive of end plugs) of 62 in., a nominal inside diameter of 0.410 in., and a nominal wall thickness of 0.020 in.

(6) Each fuel tube shall contain 120 fuel pellets of the type specified in Sec. 7.3 in an atmosphere of helium at a nominal pressure of 14.7 psia.

(7) The external dimensions of each removable burnable poison pin shall be those of a center fuel tube. Each removable burnable poison pin shall be fabricated of type 304 stainless steel containing from 1.2 per cent to 1.5 per cent natural boron by weight.

(8) The external dimensions of each radiation test center pin shall be those of a center fuel tube. Each radiation test center pin shall be fabricated of type 304 stainless steel and shall be capable of rigidly containing four control-rod poison-sections test coupons.

(9) Initially, the stainless steel of each fuel tube shall contain 600 ± 100 ppm of natural boron.

(10) The top surface of the upper end fitting of each fuel element shall be engraved (a) with a serial number, and (b) with a letter to indicate whether the element is regular or spiked. All characters engraved on each fuel element shall be at least 3/8 in. high.

7.5 SHROUD STRUCTURE SPECIFICATIONS

The specifications relating to the reactor-core shroud structure are as follows:

(1) The reactor-core shroud structure shall consist of a lower section made of Zircaloy-2 that extends from the core support plate to a horizontal plane approximately 64 in. above the core support plate and an upper section made of stainless steel that extends from the top of the lower section to a horizontal plane approximately 129 in. above the core support plate.

(2) The reactor-core shroud structure shall form the channels in which the fuel elements are placed and the channels in which the control rods move, as shown in Allis-Chalmers Dwg. 12D-04000.

(3) The nominal thickness of the Zircaloy-2 shroud-structure metal shall be 1/16 in.; that of the stainless-steel shroud-structure metal shall be 3/32 in.

7.6 SUPPORTING STRUCTURE SPECIFICATIONS

The specifications relating to the reactor-core supporting structure are as follows:

(1) The reactor-core supporting structure shall consist of a stainless-steel core support plate, a stainless-steel hold-down barrel, and various stainless-steel lugs and gussets arranged to hold the fuel elements and the shroud structure in their properly aligned positions and to transfer their weight to the lower head of the reactor vessel, as shown in Allis-Chalmers Dwg. 12D-01200.

(2) The nominal thickness of the core support plate shall be 6 in.

7.7 NEUTRON SOURCE SPECIFICATIONS

The specifications relating to the reactor-core neutron source are as follows:

(1) The reactor-core neutron source shall be of the antimony-beryllium type; it shall have a minimum strength of 1 curie. A plutonium-beryllium source may be used for critical testing at power levels less than 1 kw.

(2) The reactor-core neutron source shall be installed whenever the reactor core contains fuel and either fuel is being loaded or any control rod is withdrawn.

8. CONTROL AND SAFETY SYSTEMS

8.1 REACTIVITY CONTROL SYSTEM SPECIFICATIONS

8.1.1 General Specifications

The general specifications relating to the reactivity control system are as follows:

(1) The reactivity control system shall consist of the following components and subsystems, as hereinafter specified: the reactor control rods, the reactor control-rod drive mechanisms, the control-rod actuating system, and the boric-acid injection system.

(2) The maximum reactivity worth of control-system components whenever the reactor is in the cold, clean condition shall be as follows:

Component(s)	Maximum Reactivity Worth
All control rods	0.190
One control rod	0.045
Burnable poison in the fuel cladding	0.058
Burnable poison pins	0.058

(3) Under all conditions, full insertion of any 12 control rods shall render the core subcritical by at least 0.02.

(4) The boric-acid injection system shall be capable of reducing the reactivity of the core by a minimum of 0.02 below criticality with all control rods removed from the core.

(5) The rate at which reactivity may be increased by movement of control rods shall be limited by administrative control so as not to exceed 0.001/sec in any core with Keff greater than 0.98. The maximum capability of the control system in any core with Keff greater than 0.98 shall not exceed 0.002/sec.

8.1.2 Control-Rod Specifications

The specifications relating to the reactor control rods are as follows:

(1) The reactor core shall be designed for the installation and operation of 13 control rods. Each of 12 of the 13 control rods shall be designated a shim rod; the thirteenth rod shall be designated the regulating rod.

(2) At least 12 operable control rods shall be installed in the core whenever the core contains one or more fuel elements.

(3) Each control rod shall consist of a poison section, a follower section, and an extension section with the following characteristics:

poison section

cross-sectional configuration	cruciform
nominal length	58 in.
nominal thickness	1/4 in.
nominal width across opposite blade edges	14-7/8 in.
material	borated Type 304 stainless steel
natural-boron content	2.2% ± 0.2%

follower section

cross-sectional configuration	cruciform
nominal length	45 in.
nominal thickness	1/4 in.
nominal width across opposite blade edges	14-1/2 in.
material	Zircaloy-2

extension section

cross-sectional configuration	cruciform
nominal length	80-1/2 in.
nominal thickness	1/4 in.
nominal width across opposite blade edges	3 in.
material	Type 304 stainless steel
drive-coupling type	bayonet

(4) The control rods shall be fabricated in conformance with Dwg. 12D03000 of Reference 3.

8.1.3 Control-rod Drive-Mechanism Specifications

The specifications relating to the control-rod drive mechanisms are as follows:

(1) Each control rod shall be actuated by a drive mechanism consisting of a reversible gearhead motor, a magnetic clutch, an overrunning clutch, universal-jointed drive shafting, a drive-shaft seal assembly, a rack-and pinion assembly, a dashpot, a dashpot plunger, and a control-rod connector assembly. The general arrangement of these components shall be as shown on Fig. 3 of Reference 1 and Dwg. R-9-11-1024 of Reference 4.

(2) Each control-rod drive mechanism shall be capable of driving its associated control rod between its fully inserted and fully withdrawn positions (a nominal distance of 56 in.) at a rate that shall not exceed 34 in./min.

(3) The rack, seal shaft, pinion, rack roller, shaft roller, pinion coupling, and drive-shaft coupling of each control-rod drive mechanism shall be fabricated of 17-4 precipitation-hardened stainless steel. Machining and straightening of these components in fabrication shall be performed with the material in the solution-heat-treated condition. Except for the initial set of 14 mechanisms, which shall be as described in Reference 4, components shall be age-hardened at 1100 F $\pm$ 25 F to a Rockwell C hardness of 34 to 37, or equivalent; and the final bow of the rack shall not exceed 0.032 in.

(4) It shall be possible to remove any control rod and all components of any control-rod drive mechanism except the dashpot, the dashpot plunger, and the control-rod connector assembly without draining the reactor vessel.

(5) The dashpot and dashpot plunger of each control-rod drive mechanism shall not decelerate their associated control rod after a free fall from its fully withdrawn position at any rate exceeding 270 ft/sec^2.

(6) Interruption of the current flowing through the magnetic clutch of each control-rod drive mechanism shall disconnect the gearhead motor and allow the associated control rod to fall to its fully inserted position.

(7) The overrunning clutch of each control-rod drive mechanism shall make it possible to drive the associated control rod to its fully inserted position if, after interruption of the current flowing through the associated magnetic clutch, the control rod does not fall to its fully inserted position.

(8) Each control-rod drive mechanism shall be provided with coarse and fine position-indicator transmitters that sense the position of the associated control rod regardless of whether or not the associated magnetic clutch is energized.

(9) The position-indicator transmitters of each control-rod drive mechanism shall be electrically connected to receivers that drive coarse and fine dial position indicators on the reactor control console.

(10) The difference between the position of each control rod and the position of that control rod indicated on the reactor control console shall not exceed 0.1 in.

(11) Each control-rod drive mechanism shall be provided with limit switches, circuitry, and indicating lamps on the reactor control console so that

(a) a green lamp is illuminated whenever the associated control rod is within 1/2 in. of its fully inserted position, and

(b) a red lamp is illuminated whenever the associated control rod is within 1/2 in. of its fully withdrawn position.

(12) The interval between the instant at which a scram signal is initially generated and the instant at which any control rod affected by the scram signal that is not fully inserted begins to descend shall not exceed 0.15 sec.

(13) The interval between the instant at which a scram signal is initially generated and the instant at which control rods arrive within 1/2 in. of their fully inserted positions shall not exceed 2.0 sec.

(14) The control-rod drive mechanisms shall be provided with a timer capable of measuring the interval between actuation of the all-rod scram button and the arrival of any selected rod within 1/2 in. of its fully inserted position.

8.1.4 Control-Rod Actuating System Specifications

The specifications relating to the control-rod actuating system are as follows:

(1) The control-rod actuating system shall have two modes of operation: manual and automatic.

(2) The current flowing through each control-rod drive mechanism magnetic clutch shall be interrupted in response to any all-rod scram signal (see Sec. 8.3.1).

(3) The current flowing through the magnetic clutches of four control-rod drive mechanisms shall be interrupted in response to any four-rod scram signal (see Sec. 8.3.5).

(4) The regulating-rod drive mechanism shall be actuated so as to insert the regulating rod if the control-rod actuating system is in its automatic mode of operation, if the regulating rod is more than 21 in. from its fully inserted position, and if the primary pressure exceeds the primary-pressure set point.

(5) The regulating-rod drive mechanism shall be actuated so as to withdraw the regulating rod if the control-rod actuating system is in its automatic mode of operation, if the regulating rod is more than 1/2 in. from its fully withdrawn position, and if the primary pressure is less than the primary-pressure set point.

(6) The drive mechanisms of all shim rods that are more than 1/2 in. from their fully inserted positions shall be actuated so as to insert all such rods whenever the "all shim rods insert" button on the reactor control console is pressed.

(7) The drive mechanism of any shim rod that is more than 1/2 in. from its fully inserted position shall be actuated so as to insert that rod if the "shim-rod selector" switch on the reactor console is set to that rod and if the "shim-rod control" switch on the reactor console is turned to its "insert" position.

(8) The regulating-rod drive mechanism shall be actuated so as to insert the regulating rod if the regulating rod is more than 1/2 in. from its fully inserted position and if the "regulating-rod control" switch on the reactor control console is turned to its "insert" position.

(9) The drive mechanism of any shim rod that is more than 1/2 in. from its fully withdrawn position shall be actuated so as to withdraw that rod if the "shim-rod selector" switch on the control console is set to that rod, if the "shim-rod control" switch on the reactor console is turned to its "withdraw" position, if the regulating rod is not being withdrawn, and if either (a) a "withdraw permit" is in effect or (b) all other control rods are within 1/2 in. of their fully inserted positions and the "shim-rod test" key switch on the reactor console is set at its "test" position.

(10) The regulating-rod drive mechanism shall be actuated so as to withdraw the regulating rod if the "regulating-rod control" switch on the reactor console is turned to its "withdraw" position, if the "shim-rod control" switch on the reactor console is in its "off" position, and if either (a) a "withdraw permit" is in effect, or (b) all shim rods are within 1/2 in. of their fully inserted positions and the "regulating-rod test" key switch on the reactor console is set at its "test" position.

(11) The amount by which the control-rod actuating system can change reactivity when in its automatic mode of operation shall not exceed 0.005.

(12) The control-rod actuating system shall be designed so that only one control rod can be withdrawn at a time.

(13) The control-rod actuating system shall be designed so as to cause the mode of operation to revert from automatic to manual whenever the regulating rod is driven automatically to the limits specified in Secs. 8.1.4(4) and 8.1.4(5).

(14) The control-rod actuating system shall be designed so that in the manual mode of operation all 13 control rods can be positioned manually.

8.1.5 Boric-Acid Injection System Specifications

The specifications relating to the boric-acid injection system are as follows:

(1) The boric-acid injection system shall consist of an electrically heated boric-acid storage tank, a motor-driven high-pressure air compressor,

two air-operated boric-acid injection valves, one air-operated leakoff valve, a valve-operating air accumulator, and associated piping. These components shall be connected as shown on Sargent & Lundy Dwg. 12E09010.

(2) The boric-acid storage tank shall be of carbon steel lined with stainless steel and shall be designed, fabricated, and tested in accordance with Sec. VIII of the 1956 edition of the ASME Boiler and Pressure Vessel Code for a pressure of 2500 psig and a nominal capacity of 330 gal.

(3) The boric-acid storage tank shall be fitted with external electrical heating elements that shall be capable of maintaining the contents of the tank at approximately 200 F.

(4) The boric-acid injection-system air compressor shall be capable of delivering 10.7 scfm of air at 2000 psig.

(5) The boric-acid injection valves and the boric-acid injection-system leakoff valve shall be operable from the reactor control console, from the decontamination room, and from a point inside the containment building.

(6) The boric-acid injection valves and the boric-acid injection-system leakoff valve shall remain operable despite failure of all control-air supply systems external to the boric-acid injection system.

(7) It shall be possible to open both boric-acid injection valves and to close the boric-acid injection-system leakoff valve from the decontamination room and from a point inside the containment building whenever the pressure in the boric-acid injection-system valve-operating air accumulator equals or exceeds 40 psig.

(8) The pressure in the boric-acid injection-system air accumulator shall be at least 40 psig whenever the reactor core contains one or more fuel elements.

(9) The boric-acid injection system shall be capable of injecting enough boric acid into the primary coolant system to reduce reactivity by at least 0.145 within 20 sec after the boric-acid injection valves are opened.

(10) If, at any time more than 2 sec after an all-rod scram signal, any control rod is not within 1/2 in. of its fully inserted position, visible and audible alarms shall be actuated in the control room; and turning the "boric acid inject" switch on the reactor control console to its "on" position shall, unless the all-rod scram circuit has been reset, open both boric-acid injection valves and close the boric-acid injection system leakoff valve.

8.2 NUCLEAR INSTRUMENTATION SYSTEM SPECIFICATIONS

The specifications relating to the nuclear instrumentation system are as follows:

(1) The nuclear instrumentation system shall incorporate nine independent nuclear-instrumentation channels, whose designations, types, ranges, detector types, and detector sensitivities shall be as follows:

channel designation	channel type	approximate power-level range	detector type	detector sensitivity
N-1	startup	0.006 w-600 w	proportional counter	15 (counts/sec)/nv
N-2	startup	0.006 w-6000 w	proportional counter	15 (counts/sec)/nv
N-3	log-n & period	60 w-180 Mw	comp. ion. chamber	4.0×10^{-14} amp/nv
N-4	log-n & period	60 w-180 Mw	comp. ion. chamber	4.0×10^{-14} amp/nv
N-5	linear power	6 w-90 Mw	comp. ion. chamber	4.0×10^{-14} amp/nv
N-6	linear power	6 w-90 Mw	comp. ion. chamber	4.0×10^{-14} amp/nv
N-7	level safety	600 kw-90 Mw	uncomp. ion. chamber	4.4×10^{-14} amp/nv
N-8	level safety	600 kw-90 Mw	uncomp. ion. chamber	4.4×10^{-14} amp/nv
N-9	level safety	600 kw-90 Mw	uncomp. ion. chamber	4.4×10^{-14} amp/nv

(2) The devices and functions associated with each nuclear instrumentation channel shall be as shown on Sargent & lundy Dwg. 12E11926.

(3) An all-rod scram signal shall be generated by the manual actuation of all-rod scram switches in the control room and in the reactor building. Loss of electrical power to the magnetic clutches of the control-rod drive mechanisms shall cause a scram.

8.3 INTERLOCK SPECIFICATIONS

8.3.1 Period and Power-Level All-Rod Scram and Alarm Interlock Specifications

The specifications relating to the nuclear-instrumentation-system period and power-level all-rod scram and alarm interlocks are as follows:

(1) Channel N-1 shall actuate a control-room alarm and prohibit control-rod withdrawal whenever it senses a reactor period less than 7 sec, unless by-passed in accordance with Specification 8.4.2(1) or (2).

(2) Channel N-2 shall actuate a control-room alarm and prohibit control-rod withdrawal whenever it senses a reactor period less than 7 sec, unless bypassed in accordance with Specification 8.4.2(1) or (2).

(3) Channel N-3 shall actuate a control-room alarm and prohibit control-rod withdrawal whenever it senses a reactor period less than 15 sec, and shall actuate a control-room alarm and generate an all-rod scram signal whenever it senses a reactor period less than 3 sec, unless bypassed in accordance with Specification 8.4.2(3) or (4).

(4) Channel N-4 shall actuate a control-room alarm and prohibit control-rod withdrawal whenever it senses a reactor period less than 15 sec, and shall actuate a control-room alarm and generate an all-rod scram signal whenever it senses a reactor period less than 3 sec, unless bypassed in accordance with Specification 8.4.2(3) or (4).

(5) Channel N-5 shall actuate a control-room alarm whenever it senses a reactor power level less than 5 percent or greater than 110 percent of its power range setting, and shall actuate a control-room alarm and initiate an all-rod scram signal whenever it senses a reactor power level exceeding 125 percent of its power-range setting.

(6) Channel N-6 shall actuate a control-room alarm whenever it senses a reactor power level less than 5 percent or greater than 110 percent of its power range setting, and shall actuate a control-room alarm and initiate an all-rod scram signal whenever it senses a reactor power level exceeding 125 percent of its power-range setting.

(7) Channel N-7 shall actuate separate control-room alarms whenever it senses a reactor power level less than 10 percent, greater than 110 percent, and greater than 125 percent of the reactor full-power level.

(8) Channel N-8 shall actuate separate control-room alarms whenever it senses a reactor power level less than 10 percent, greater than 110 percent, and greater than 125 percent of the reactor full-power level.

(9) Channel N-9 shall actuate separate control-room alarms whenever it senses a reactor power level less than 10 percent, greater than 110 percent, and greater than 125 percent of the reactor full-power level.

(10) Circuitry associated with the signal outputs of Channels N-7, N-8, and N-9 shall be arranged so that a two-position key-operated switch will determine whether an all-rod scram signal will be generated whenever any one or whenever any two of the three channels senses a reactor power level greater than 125 percent of the reactor full-power level.

(11) Circuitry associated with the signal outputs of Channels N-5 and N-6 shall be arranged so that a two-position key-operated switch will determine whether an all-rod scram signal will be initiated whenever either (or both) of the two channels senses a power level greater than 125 percent of the power-range setting.

8.3.2 "Reactor Start" Interlock Specifications

The specifications relating to obtaining and revoking "reactor start" permits are as follows:

(1) The design of the reactor-start interlock shall require that the following conditions be met before a "reactor start" permit can be obtained:

(a) the reactor "power on" key switch shall be turned to its "on" position,

(b) the Channel N-1 and N-2 high-voltage switches shall be in their "on" positions,

(c) the Channel N-3 and N-4 period-scram bypass switches shall be in their "in circuit" positions,

(d) all 13 control rods shall be fully inserted,

(e) the containment-building airlocks shall be secured, and

(f) all all-rod scram circuits shall be clear.

(2) Whenever the six conditions set forth in Specification 8.3.2(1) have been met and the "reactor start" button has been pressed, it shall be unnecessary to meet the second, third, fourth, and fifth conditions set forth in Specification 8.3.2(1) thereafter, and the "reactor start" permit shall be automatically revoked thereafter only in response to any all-rod scram signal or whenever the reactor "power on" key switch is turned to its "off" position.

8.3.3 "Rod Withdraw" Interlock Specifications

The specifications relating to obtaining or revoking "rod withdraw" permits are as follows:

(1) The design of the rod-withdrawal interlock shall require that the following conditions be met before a "rod withdraw" permit can be obtained:

(a) the reactor period sensed by Channels N-1 and N-2 shall be greater than 7 sec, or the reactor-period interlock associated with either Channel N-1 or N-2 must be bypassed (see Sec. 8.4),

(b) the reactor period sensed by Channels N-3 and N-4 shall be greater than 15 sec, or the reactor-period interlock associated with either Channel N-3 or N-4 must be bypassed (see Sec. 8.4),

(c) the regulating-rod and shim-rod test switches shall be in their "off" positions,

(d) all four-rod scram circuits shall be clear, and

(e) a "reactor start" permit shall be in effect.

(2) Any "rod withdraw" permit shall be automatically revoked whenever any of the conditions set forth in Specification 8.3.3(1) no longer obtains.

8.3.4 "Automatic Control" Interlock Specifications

The specifications relating to obtaining and revoking "automatic control" permits are as follows:

(1) The design of the automatic control interlock shall require that the following conditions be met before an "automatic control" permit can be obtained:

(a) the regulating rod shall be within its automatic control travel range,

(b) the difference between the actual primary-system pressure and the pressure at which the pressure set-point potentiometer is set shall be less than or equal to 50 psi,

(c) the regulating-rod control switch shall be in its "off" position,

(d) a "rod withdrawal" permit shall be in effect, and

(e) the mode-of-operation switch shall be in its "automatic control" position.

(2) Any "automatic control" permit shall be automatically revoked whenever any of the conditions set forth in Specification 8.3.4(1) no longer obtains.

8.3.5 Specification Relating to the Causes of Four-Rod Scrams

Each of the following conditions shall cause a four-rod scram, unless the sensing or signalling circuitry has been bypassed in accordance with Sec. 8.4.3:

(1) high water level in the reactor vessel (7.7 to 9.6 ft above the top of the core),

(2) low water level in the reactor vessel(5 ft-4 in. above the top of the core),

(3) low reactor water temperature (less than 425 F),

(4) primary-system pressure greater than 1195 psig,

(5) opening of the reactor-vessel bottom-head warmup valve,

(6) secondary-system pressure greater than 880 psig,

(7) opening of the emergency-and-test-condenser condensate flow-control valve,

(8) tripping of the superheater auxiliaries,

(9) closure of the turbine stop valve, and

(10) pressing the four-rod scram button on the control panel.

8.3.6 Specification Relating To The Causes Of Containment Building Closure

Any high-radiation signal from the stack monitor (see Sec. 9.1), any high-pressure signal (i.e., one indicating a pressure greater than 2 psig) from the containment-building pressure monitor, and any high primary-system pressure signal (i.e., one indicating a pressure greater than 1210 psig) shall actuate the containment-building ventilation-duct damper closing mechanisms.

8.3.7 Specification Relating To The Causes Of Control-Room Alarms

Each of the following conditions, in addition to those specified in Sec. 8.3.1, shall actuate a separate control-room alarm:

(1) boric acid tank - low level,
(2) boric acid tank - low pressure,
(3) shield cooling system - high temperature,
(4) building area monitors - high radiation,
(5) secondary water monitor - high radiation,
(6) stack particulate monitor - high radiation,
(7) stack-monitor trouble,
(8) stack-gas monitor - high radiation,
(9) fission products monitor - high radiation,
(10) sewer monitor - high radiation,
(11) recombiner - high temperature,
(12) Evaporator 1 - downcomer low level,
(13) Evaporator 2 - downcomer low level,
(14) reactor spray valve - open,
(15) reactor spray system - off automatic,
(16) building airlocks - not secured,
(17) mobile monitor - high radiation,
(18) shield-cooling-system surge tank - low level,
(19) boric acid tank - low temperature,
(20) shield cooling system - low flow,
(21) instrument cabinet blower - power off,
(22) purification system - high conductivity,
(23) purification system inlet - high temperature,
(24) boric acid system - power off,
(25) decay heat cooler outlet - high temperature,
(26) reactor building - high pressure,
(27) unitiated return to manual control,
(28) evaporator blowdown line - high conductivity,
(29) decontamination room retention tank - high level,
(30) Retention Tank 1 - high level,
(31) Retention Tank 2 - high level,

(32) demineralized water storage water storage tank - high level
(33) demineralized water storage tank - low level,
(34) control air - low pressure,
(35) emergency condenser - low level,
(36) emergency condenser - high level,
(37) evaporator - high level,
(38) evaporator - low level,
(39) superheater outlet steam - high temperature,
(40) superheater interstage steam - low temperature,
(41) superheater auxiliaries trip,
(42) superheater fuel -system trip,
(43) reactor auxiliaries trip,
(44) main feed breaker trip,
(45) reserve feed breaker trip,
(46) superheater building main feed breaker trip,
(47) superheater building reserve feed breaker trip,
(48) reactor building main feed breaker trip,
(49) reactor building reserve feed breaker trip,
(50) Motor Control Center 1 feed breaker trip,
(51) Motor Control Center 2 feed breaker trip,
(52) Motor Control Center 3 feed breaker trip,
(53) Motor Control Center 3 transfer switch in emergency position,
(54) switchgear d-c power off.,
(55) engine generator cooling system - high temperature,
(56) engine generator oil - low pressure,
(57) engine generator start failure,
(58) inverter trip,
(59) battery charger trip,
(60) superheater pressure positive,
(61) superheater flame failure,
(62) reactor vessel - high level,
(63) reactor vessel - low level,
(64) reactor spray system - no flow,
(65) collecting tank - high level,
(66) collecting tank - low level,
(67) secondary power relief valve - open,
(68) boiler feedwater pump trip,
(69) turbine air ejector - high radiation,
(70) turbine air ejector - monitor trouble,
(71) building evacuation signal - actuated,
(72) building spray system -actuation of delayed trip timer,
(73) reactor cavity - high temperature and high leve, and
(74) AC-DC trouble on MG set.

8.4 INTERLOCK BYPASS SPECIFICATIONS

8.4.1 General Interlock Bypass Specifications

The general specifications relating to interlock bypasses are as follows:

(1) No interlock bypasses other than those specified in this section (Sec. 8.4) shall be provided in the design.

(2) Key lock switches shall be used for all interlock bypasses.

(3) Except as explicitly provided in Sec. 8.4, no interlock shall be bypassed in the course of any operation for which service of that interlock is required by these Technical Specifications.

8.4.2 Period Interlock Bypass Specifications

The specifications relating to the conditions under which the nuclear-instrumentation-system period interlocks may be bypassed (i.e., rendered inoperative by key switches) are as follows:

(1) The period interlocks associated with either Channel N-1 or Channel N-2 may be bypassed whenever the reactor power level exceeds the power level range of that channel.

(2) The period interlocks associated with Channel N-3 and Channel N-4 may be bypassed whenever the reactor power level exceeds 3 Mw.

8.4.3 Other Interlock Bypass Specifications

The specifications relating to the conditions under which interlocks other than those dealt with in Sec. 8.4.1 may be bypassed (i.e., rendered inoperative by key switches) are as follows:

(1) The containment-building airlock interlock, which ordinarily requires that one door of each airlock be closed and sealed before a "reactor start" permit can be obtained (see Sec. 8.3.2), may be bypassed at the discretion of the reactor shift supervisor, but only for as long as all installed control rods remain in their fully inserted positions.

(2) The reactor-vessel high-water-level scram interlock may be bypassed with the consent of the reactor shift supervisor during fuel loading, critical testing, or the quarterly check of the safety circuit.

(3) The reactor-vessel low-water-level scram interlock may be bypassed with the consent of the reactor shift supervisor during critical testing or the quarterly check of the safety circuit.

(4) The reactor-water low-temperature scram interlock may be bypassed with the consent of the reactor shift supervisor during fuel loading, critical testing, or the quarterly check of the safety circuit.

(5) The primary-steam high-pressure scram interlock may be bypassed with the consent of the reactor shift supervisor during initial startup tests or the quarterly check of the safety circuit.

(6) The reactor-vessel bottom-head warmup valve-open scram interlock may be bypassed with the consent of the reactor shift supervisor during fuel loading, critical testing, or the quarterly check of the safety circuit.

(7) The secondary-steam high-pressure scram interlock may be bypassed with the consent of the reactor shift supervisor during initial startup tests or the quarterly check of the safety circuit.

(8) The emergency-condenser open-outlet-valve scram interlock may be bypassed with the consent of the reactor shift supervisor during initial startup tests or the quarterly check of the safety circuit.

(9) The superheater-trip scram interlock may be bypassed with the consent of the reactor shift supervisor whenever the superheater power level is less than 50 per cent of full superheater power or during the quarterly check of the safety circuit.

(10) The turbine-trip scram interlock may be bypassed with the consent of the reactor shift supervisor whenever the turbine load is less than 4400 kwe or during the quarterly check of the safety circuit.

8.5 EMERGENCY POWER SUPPLY SYSTEM SPECIFICATIONS

The specifications relating to the emergency power supply system are as follows:

(1) Electrical power for starting up the reactor plant and the superheater shall be supplied by a 69 kv/480 v reserve auxiliary transformer energized by the RCPA 69-kv bus, but thereafter the reactor-plant and the superheater auxiliary load shall be transferred to the 13.8 kv/480 v auxiliary transformer energized by the generator driven by the nuclear steam supply system. If, after the reactor-plant and superheater auxiliary load have been so transferred, the 480-v bus feeding the reactor plant or the superheater is de-energized, the auxiliary load shall be automatically re-transferred to the reserve auxiliary transformer energized by the RCPA 69-kv bus.

(2) Electrical power for the following equipment shall be supplied from a 440-v distribution bus in Motor Control Center 3 (in the containment building) that shall be energized automatically by a 50-kw gasoline-engine-driven generator if both sources of auxiliary power specified in Specification 8.5(1) fail, or are isolated from Motor Control Center 3:

(a) all 13 control-rod drive mechanisms (1/6 hp each),

(b) both primary-coolant purification-system pumps (5 hp each),

(c) both shield-cooling-system pumps (5 hp each),

(d) the decay-heat-removal-system pump (15 hp),

(e) the stack monitor vacuum pump (1 hp),

(f) the 125-v battery-charger (20 kw), and

(g) the 480 v/120 v distribution transformer that supplies electrical power to the failure-free bus if the motor-generator set fails (10 kva).

(3) Electrical power for starting the 50-kw gasoline-engine-driven generator shall be supplied by a 12-v storage battery.

(4) The failure-free bus shall ordinarily be energized from the 125-v d-c distribution bus through a 125-v d-c--120-v a-c motor generator set, but if the motor-generator set fails, the failure-free bus shall be automatically connected to the 440-v distribution bus in Motor Control Center 3 through a 10-kva 440-v/120-v distribution transformer.

(5) The 125-v d-c distribution bus shall ordinarily be energized by the 20-kw battery-charger, but if the battery-charger fails, the d-c bus shall be energized by a 125-v battery bank kept floating on the 125-v d-c distribution bus. The 125-v d-c distribution bus shall be provided with a tie to the station battery.

(6) The a-c failure-free bus shall supply electrical power to the reactor control console, all nuclear instrumentation, radiation monitors, area monitors, primary pressure transmitters C-7 and H-51, secondary pressure transmitter H-50, and fire-eye equipment for the superheater.

(7) Electrical power for emergency lights in the containment building and the superheater building shall ordinarily be supplied from the 120-v/208-v distribution bus in the containment building, but if that distribution bus is de-energized, the emergency lighting load shall be automatically transferred to the 125-v d-c distribution bus.

(8) The 125-v d-c bus shall supply power to all control-rod drive mechanism magnetic clutches and the power necessary for remotely actuating the following systems:

(a) the boric-acid injection system,

(b) the emergency-and-test-condenser flow-control valve,

(c) the reactor-core emergency-cooling-system flow-control valve,

(d) the containment-building spray-system flow-control valve,

(e) the containment-building air inlet and outlet dampers, and

(f) the reactor control-room annunciators.

9. RADIATION MONITORING SYSTEMS

9.1 STACK MONITOR SPECIFICATIONS

The specifications relating to the stack monitor are as follows:

(1) The stack monitor shall incorporate three detectors: one for immediate, continuous detection of particulate beta-gamma activity; one for immediate, continuous detection of gaseous beta activity; and one for delayed, continuous detection of particulate beta activity. Each detector shall be provided with indicating and alarm devices and circuitry. The delayed particulate detector shall sense the activity of any trapped particles approximately 6 hr after the immediate particulate detector senses the activity of the same trapped particles.

(2) The stack monitor shall be installed near the containment-building exhaust duct and shall expose all three detectors to a continuous representative sample of the airborne particles and gases in the duct.

(3) The stack-monitor immediate particulate beta activity detector shall be capacle of detecting stack particulate beta activity levels that correspond to stack release rates of radioactive particles in the range from 7×10^{-6} to 1.7×10^{-1} microcurie/sec when a background of 20 cpm exists, and when using Tl^{204} as a standard.

(4) The stack-monitor immediate gaseous gamma activity detector shall be capable of detecting stack gaseous gamma activity levels that correspond to stack release rates in the range from 9×10^{-1} to 7×10^{3} microcuries/sec when a background of 250 cpm exists, and when using Ar^{41} as a standard.

(5) The stack-monitor delayed particulate beta activity detector shall be capable of detecting the stack particulate beta activity of long-lived isotopes (those whose half-lives exceed 6 hr) at levels corresponding to stack release rates in the range from 7×10^{-6} to 1.7×10^{-1} microcurie/sec when a background of 20 cpm exists and when using Tl^{204} as a standard.

(6) The stack-monitor immediate particulate beta activity detector shall generate a high-radiation signal whenever its associated devices and circuitry indicate an activity level corresponding to a stack radioactive-particle release rate of 2.5 microcuri/sec or less.

(7) The stack-monitor immediate gaseous gamma activity detector shall generate a high-radiation signal whenever its associated devices and circuitry indicate an activity level corresponding to a total stack release rate of 1680 microcuries/sec or less.

(8) The stack-monitor delayed particulate beta activity detector shall generate a high-radiation signal whenever its associated devices and circuitry indicate an activity level corresponding to a stack release rate of long-lived radioisotopes of 2.5 microcurie/sec or less.

(9) Any stack-monitor high-radiation signal shall actuate audible and visible alarms in the reactor control room and shall actuate the containment-building ventilation-duct damper closing mechanisms.

(10) The stack-monitor detectors and their associated devices and circuitry shall be operated in accordance with Specification 14.2(12).

(11) A one-point calibration check of the stack-monitor gaseous detector response shall be made daily. Whenever the check varies more than ± 15% from normal, the instrument settings shall be checked to verify that the proper high voltage is being applied to the detector tube. If reading is still not acceptable, the instrument shall be repaired. Characteristics of the detector tube shall be checked annually with a standard source and whenever the tube is changed.

(12) A one-point calibration check of the stack monitor particulate detectors shall be made daily. These detectors shall be recalibrated annually. Whenever the daily calibration check varies more than ± 15% from normal after correcting for background, the instrument shall be repaired.

(13) The annual average stack emission rate for radioactive noble and activation gases shall not exceed 1680 microcuries/sec; the instantaneous stack emission rate for these radioactive gases shall not exceed 16,800 microcuries/sec. The annual average stack emission rate for all other artificially produced radioisotopes, including radioactive halogens, shall not exceed 2.5 microcurie/sec; the instantaneous stack emission rate for these radioisotopes shall not exceed 2.5 microcurie/sec.

9.2 FISSION-PRODUCT MONITOR SPECIFICATIONS

The specifications relating to the fission-product monitor are as follows:

(1) The fission-product monitor shall incorporate a gamma-activity detector, energy-sensitive discriminating circuitry, and indicating and alarm devices and circuitry for the immediate, continuous detection of gaseous fission-product gamma activity.

(2) The fission-product monitor shall sense gaseous fission-product gamma activity in the recombiner-system-condenser vent line.

(3) The fission-product monitor shall discriminate against nitrogen-16 gamma activity and shall be capable of detecting gaseous fission-product gamma-activity levels greater than twice background.

(4) The fission-product monitor shall generate a high-radiation alarm signal whenever its associated devices and circuitry indicate a gross counting rate corresponding to a gamma activity level of 40 cpm above the normal background in the absence of fission products.

(5) Any fission-product-monitor high-radiation signal shall actuate audible and visible alarms in the reactor control room.

(6) The fission-product monitor shall be operated in accordance with Specification 14.2(13).

(7) A one-point calibration check of the fission-product monitor shall be made daily. This instrument shall be recalibrated annually. Whenever the daily calibration check varies more than $\pm$ 15% from normal after correcting for background, the instrument shall be repaired.

9.3 AREA MONITOR SPECIFICATIONS

The specifications relating to the area monitors are as follows:

(1) Each area monitor shall incorporate a gamma-activity detector, an electrometer, and indicating and alarm devices and circuitry for the immediate, continuous detection of gamma radiation fields.

(2) One area monitor shall sense gamma radiation near each of the following components: The bottom plug of the reactor-vessel cavity, the startup-heating-system heat exchanger, the shell-side (secondary) outlet of each evaporator, each subcooler, each ion-exchange column, the shield-cooling-system heat exchanger, the emergency and test condenser.

(3) One area monitor shall sense gamma radiation in each of the following spaces: the Dispatcher's Office, the reactor control room, and the main floor of the containment building.

(4) Each area monitor shall be capable of detecting gamma radiation levels exceeding 10 mr/hr.

(5) Each area monitor shall generate a high-radiation signal whenever its associated devices and circuitry indicate a gross gamma-radiation level that exceeds twice the background level in its vicinity during full-power operation.

(6) A high-radiation signal generated by any of the area monitors sensing gamma radiation near the components enumerated in Specification 9.3(2) shall actuate audible and visible alarms at the location of the monitor and in the reactor control room.

(7) A high-radiation signal generated by the area monitor sensing gamma radiation on the main floor of the containment building shall actuate audible and visible alarms at the location of the monitor and in the reactor control room.

(8) A high-radiation signal generated by either of the area monitors sensing gamma radiation in the Dispatcher's Office and in the reactor control room shall actuate audible and visible alarms at the location of the monitor.

(9) Each area monitor shall be operated in accordance with Specification 14.2(14).

(10) A check of the visible and audible alarm operation of the area monitors shall be made daily. A one-point calibration check of these monitors shall be made weekly, and they shall be recalibrated annually. Whenever the calibration check of an area monitor varies more than $\pm$ 20% from normal after correcting for background, the instrument shall be repaired.

9.4 SEWER MONITOR SPECIFICATIONS

The specifications relating to the sewer monitor are as follows:

(1) The sewer monitor shall incorporate a gamma-activity* detector and indicating and alarm devices and circuitry for the immediate, continuous detection of water-borne gamma activity*.

(2) The sewer monitor shall sense the gamma activity* of water flowing through the service-water return line immediately outisde the containment building.

(3) The sewer monitor shall be capable of detecting water-borne gamma-activity* concentrations that exceed 5×10^{-6} microcurie/cc when a background of 250 cpm exists and when using Zn^{65} as a standard.

(4) The sewer monitor shall generate a high-radiation signal whenever its associated devices and circuitry indicate gamma-activity* concentrations greater than 2×10^{-4} microcurie/cc.

(5) Any sewer-monitor high-radiation signal shall actuate audible and visible alarms in the reactor control room.

(6) The sewer monitor shall be operated in accordance with Specification 14.2(15).

(7) A one-point calibration check of the sewer monitor shall be made daily. This monitor shall be recalibrated annually. Whenever the calibration check varies more than $\pm$ 15% from normal after correcting for background, the instrument shall be repaired.

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* As used here, the term "gamma activity" shall denote only the activity of radionuclides that emit gamma rays with energies exceeding 1 Mev.

9.5 SECONDARY-SYSTEM WATER MONITOR SPECIFICATIONS

The specifications relating to the secondary-system water monitor are as follows:

(1) The secondary-system water monitor shall incorporate a gamma-activity detector and indicating and alarm devices and circuitry for the immediate, continuous detection of water-borne gamma activity.

(2) The secondary-system water monitor shall sense the gamma activity of water flowing through a line connecting the discharge side of the condensate pumps to the hotwell of the turbine condenser.

(3) The secondary-system water monitor shall be capable of detecting water-borne gamma activity levels of 5×10^{-6} microcurie/cc when a background of 250 cpm exists and when using Zn^{65} as a standard.

(4) The secondary-system water monitor shall generate a high-radiation signal whenever its associated devices and circuitry indicate a water-borne activity level that exceeds 6×10^{-5} microcurie/cc.

(5) Any secondary-system water monitor high-radiation signal shall actuate audible and visible alarms in the reactor control room.

(6) The secondary-system water monitor shall be operated in accordance with Specification 14.2(16).

(7) A one-point calibration check of the secondary-system water monitor shall be made daily. This monitor shall be recalibrated annually. Whenever the calibration check varies more than $\pm$ 15% from normal after correcting for background, the instrument shall be repaired.

9.6 PLANT PARTICULATE MONITOR SPECIFICATIONS

The specifications relating to the plant particulate monitor are as follows:

(1) The plant particulate monitor shall incorporate two detectors: one for immediate, continuous detection of particulate beta-gamma activity and one for delayed, continuous detection of particulate beta activity. Each detector shall be provided with indicating and alarm devices and circuitry. The delayed detector shall sense the activity of any trapped particles approximately 6 hr after the immediate detector senses the activity level of the same trapped particles.

(2) It shall be possible to operate the plant particulate monitor at any of three locations in the basement of the containment building and at one location on the main floor of the containment building.

(3) The plant particulate monitor shall expose both detectors to a continuous representative sample of the airborne particulate activity in its vicinity.

(4) The plant-particulate-monitor immediate particulate beta-gamma activity detector shall be capable of detecting particulate beta-gamma activity levels that exceed 1 x 10^{-11} microcurie/cc when a background of 25 cpm exists and when using Tl^{204} as a standard.

(5) The plant-particulate-monitor delayed particulate beta-gamma activity detector shall be capable of detecting particulate beta-gamma activity levels that exceed 1 x 10^{-11} microcurie/cc when a background of 25 cpm exists and when using Tl^{204} as a standard.

(6) The plant-particulate-monitor immediate beta-gamma activity detector shall generate a high-radiation signal whenever its associated devices and circuitry indicate a concentration of air-borne radioisotopes exceeding 3 x 10^{-9} microcurie/cc.

(7) The plant-particulate-monitor delayed beta-activity detector shall generate a high-radiation signal whenever its associated devices and circuitry indicate a concentration of air-borne radioisotopes exceeding 1 x 10^{-10} microcurie/cc.

(8) Any plant-particulate-monitor high-radiation signal shall actuate audible and visible alarms in the reactor control room.

(9) The plant particulate monitor shall be operated in accordance with Specification 14.2(17).

(10) A one-point calibration check of the plant particulate monitor shall be made daily. This monitor shall be recalibrated annually. Whenever the calibration check varies more than ± 15% from normal after correcting for background, the instrument shall be repaired.

9.7 TURBINE AIR-EJECTOR MONITOR SPECIFICATIONS

The specifications relating to the turbine air-ejector monitor are as follows:

(1) The turbine air-ejector monitor shall incorporate a beta-gamma activity detector and indicating and alarm devices and circuitry for the immediate continuous detection of gaseous beta-gamma activity.

(2) The turbine air-ejector monitor shall sense the activity of a continuous representative sample of the air flowing through the turbine air-ejector-condenser vent line.

(3) The turbine air-ejector monitor shall be capable of detecting gaseous beta-gamma activity levels in the air-ejector-condenser vent line that would result in a stack release rate of 1 microcurie/sec.

(4) The turbine air-ejector monitor shall generate a high-radiation signal whenever its associated devices and circuitry indicate an activity level that would result in a stack release rate of 6 microcuries/sec or less.

(5) Any turbine-air-ejector-monitor high-radiation signal shall actuate audible and visible alarms in the reactor control room.

(6) The turbine air ejector monitor shall be operated in accordance with Specification 14.2(18).

(7) A one-point check of the turbine air-ejector monitor shall be made daily. This monitor shall be recalibrated annually. Whenever the calibration check varies more than ± 15% from normal, after correcting for background, the instrument shall be repaired.

9.8 ENVIRONMENTAL MONITORING SPECIFICATIONS

Five air particulate sampling stations shall be operated to measure radioactivity concentrations in the environment. Samples shall be collected and analyzed at least monthly.

9.9 RADIATION-MONITORING INDICATING AND RECORDING SPECIFICATION

The measurements from all radiation monitors required by these Technical Specifications except (a) the area monitors in the Dispatcher's Office and the reactor control room, (b) the delayed particulate stack monitors, and (c) the particulate sampling stations specified in Sec. 9.8. shall be indicated and automatically recorded in the reactor control room.

10. RADIOACTIVE-WASTE DISPOSAL SYSTEMS

10.1 LIQUID-RADIOACTIVE-WASTE DISPOSAL SYSTEM SPECIFICATIONS

The specifications relating to the liquid-radioactive-waste disposal system are as follows:

(1) The liquid-radioactive-waste disposal system shall consist of

(a) two liquid-waste retention tanks in the containment building,

(b) a centrifugal retention-tank pump in the containment building,

(c) two submersible centrifugal sump pumps in the containment building,

(d) a portable demineralizer in the containment building that incorporates an ion-exchange column containing anion and cation resins in a mixed bed and a centrifugal circulating pump,

(e) a decontamination-room liquid-waste retention tank in the pipe tunnel,

(f) a centrifugal decontamination-room liquid-waste retention-tank discharge pump in the pipe tunnel,

(g) interconnecting piping, and

(h) valves.

(2) The capacity of each containment-building liquid-waste retention tank shall be at least 3000 gal.

(3) The containment-building retention-tank pump shall be rated at 50 gpm against a net developed head of 50 ft of water.

(4) Each of the two containment-building sump pumps shall be rated at 20 gpm against a net developed head of 15 ft of water.

(5) The portable demineralizer shall be designed to reduce the specific conductivity of 5 gpm of influent by a factor of at least 5 while operating against a net developed head of 30 ft of water.

(6) The capacity of the decontamination-room liquid-waste retention tank shall be at least 350 gal.

(7) The decontamination-room liquid-waste retention-tank discharge pump shall be rated at 5 gpm against a net developed head of 10 ft of water.

(8) Piping and valves shall be arranged so that the containment-building sump pumps may discharge either to the containment building liquid-waste retention tanks or to the service-water return line upstream of the sewer monitor.

(9) Piping and valves shall be arranged so that the containment-building retention-tank pump may discharge to either containment-building liquid-waste retention tank or to the service-water return line upstream of the sewer monitor or to a tank truck connection.

(10) The decontamination-room retention-tank discharge pump shall discharge to either containment-building liquid-waste retention tank or to the service-water return line upstream of the sewer monitor.

(11) The containment-building sump pumps, the containment-building retention-tank pump, and the decontamination-room retention-tank discharge pump shall be operated in accordance with Specification 14.1(6). Before discharging liquids from the containment-building sump, the containment-building retention tank, or the decontamination-room retention tank, the concentration of radioactivity of the liquids to be discharged shall be sampled and analyzed. No liquids shall be discharged from the containment-building sump or any of the retention tanks into the service-water return line which would result in discharges of water from the facility in concentrations in excess of the maximum permissible concentration given in Table II of Appendix B to 10 CFR 20 as determined at the point of discharge.

(12) The connections between the containment-building retention tanks and the portable demineralizer shall be of the "quick-disconnect" type.

(13) The design of the liquid-radioactive-waste disposal system shall be such that liquids can be discharged from the containment-building sump, the containment-building retention tank, and the decontamination-room retention tank only by pumping.

10.2 GASEOUS-RADIOACTIVE-WASTE DISPOSAL SYSTEM SPECIFICATIONS

The specifications relating to the gaseous-radioactive-waste disposal system are as follows:

(1) The containment-building gaseous-radioactive-waste disposal system shall consist of

(a) that portion of the recombiner system containing the gas drier, the two parallel-connected activated-charcoal absorbers, interconnecting piping and valves;

(b) the upper and lower reactor-cavity vent lines and valves;

(c) that portion of the containment-building ventilation system containing the exhaust-duct prefilter assembly, the exhaust-duct absolute filter assembly, the exhaust-duct blower and the portion of the exhaust duct in which the two butterfly dampers are installed; and

(d) the stack monitor.

(2) The activity of gaseous radionuclides in the air inside the containment building shall not exceed the maximum permissible concentrations listed in Column 1 of Table I of Appendix B to 10 CFR 20.

10.3 SOLID-RADIOACTIVE-WASTE DISPOSAL SPECIFICATIONS

The specifications relating to the disposal of solid radioactive wastes are as follows:

(1) Solid radioactive wastes shall not be stored indefinitely at the reactor site.

(2) Solid radioactive wastes shall be packaged and shipped from the site for processing and ultimate disposal.

(3) Solid radioactive wastes other than radioactively contaminated clothing awaiting shipment from the reactor site shall be stored in the containment building. Radioactively contaminated clothing awaiting shipment from the reactor site shall be stored in closed barrels in the decontamination room.

(4) All shipments of solid radioactive wastes from the reactor site shall be governed by Specification 14.1(6).

section 11

pp 11-1 — 11-3

NSIC - F1.10.1

section 3

Figs 3.1 — 3-7

NSIC F-1.10.1

11. VENTILATION SYSTEM

11.1 GENERAL SPECIFICATIONS

The general specifications relating to the containment-building ventilation system are as follows:

(1) The containment-building ventilation system shall consist of an outside-air intake duct incorporating two butterfly dampers of the type specified in Specification 3.2(6), two packaged air-conditioning units, an exhaust-duct prefilter assembly, an exhaust-duct absolute filter assembly, an exhaust-duct blower, and an exhaust duct incorporating two butterfly dampers of the type specified in Specification 3.2(6).

(2) The outside-air intake duct shall provide an air path through the containment-building shell from a point 50 ft above grade immediately outside the containment-building shell to the inlet plenum of one of the packaged air conditioning units.

(3) The outside-air intake-duct butterfly dampers shall be between the outside atmosphere and any equipment or device within the containment building.

(4) The outside-air intake duct shall contain flow control dampers to permit adjusting the ratio of outside-air intake flow rate to inside-air recirculation flow rate when the intake-duct butterfly dampers are open.

(5) Each packaged air-conditioning unit shall be rated to discharge conditioned air to the containment building at the rate of 8000 cfm against a net pressure of 1 in. of water.

(6) The exhaust-duct prefilter assembly shall incorporate one filter conforming to the following requirements:

nominal height	4 ft
nominal width	4 ft
nominal thickness	1 in.
efficiency	98% of all particles 12 microns
rated flow capacity	3500 cfm
nominal pressure drop at rated flow capacity	0.25 in. H_2O

(7) The exhaust-duct prefilter assembly shall completely cover the exhaust duct opening inside the containment building.

(8) The exhaust-duct absolute filter assembly shall incorporate four filters each conforming to the following requirements:

nominal height	2 ft
nominal width	2 ft
nominal thickness	1 ft
efficiency . . . . 99.85% of all particles	5 microns
rated flow capacity	1100 cfm
nominal pressure drop at rated flow capacity	1 in. H_2O

(9) The exhaust-duct absolute-filter assembly shall be installed within the exhaust duct downstream from the prefilter assembly and shall filter all containment-building exhaust gases, all recombiner condenser vent gases, and all reactor cavity vent gases before their discharge to the atmosphere outside the containment building.

(10) The exhaust-duct blower shall be rated to discharge air at the rate of approximately 3600 cfm against a net pressure of 1-3/4 in. of water.

(11) The exhaust-duct blower shall be installed within the exhaust duct downstream from the absolute filter assembly and downstream from the stack-monitor sample point.

(12) The exhaust duct shall provide the only continuous air path from the exhaust-duct prefilter assembly, through the exhaust-duct absolute-filter assembly, through the exhaust-duct blower, to a point 96.5 ft above grade immediately outside the containment shell.

(13) The exhaust-duct butterfly dampers shall be between all equipment or devices within the containment building and the outside atmosphere.

(14) The exhaust duct extending from the absolute filter assembly to the blower suction shall be capable of withstanding an inside-to-outside pressure difference of $\pm$ 4 in. H_2O.

(15) The exhaust duct extending from the blower discharge to the butterfly dampers shall be capable of withstanding an inside-to-outside pressure difference of $\pm$ 4 in. H_2O.

11.2 OPERATING SPECIFICATIONS

The specifications relating to operation of the containment-building ventilation system are as follows:

(1) The intake-duct and the exhaust-duct butterfly dampers shall be operable from the reactor control console.

(2) The intake-duct and exhaust-duct butterfly dampers shall automatically close on receipt of any stack-monitor high-radiation signal specified in Secs. 9.1(6) through 9.1(9) or on receipt of any pressure signal specified in Sec. 8.3.6.

(3) The intake-duct and exhaust-duct butterfly dampers shall close on loss of control power.

(4) The rate at which air flows from the containment building through the exhaust duct to the surrounding atmosphere shall not exceed 3500 cfm.

12. EMERGENCY COOLING AND DECAY HEAT REMOVAL SYSTEMS

12.1 REACTOR-CORE EMERGENCY COOLING SYSTEM SPECIFICATIONS

The specifications relating to the reactor-core emergency cooling system are as follows:

(1) The reactor-core emergency-cooling spray ring shall be capable of spraying water over the top of the core through 12 spray nozzles in a pattern determined on the basis of the cooling-water requirements of each part of the core in a manner so that the cladding of not more than 8.62% of the fuel rods would melt in the event of a loss-of-coolant type accident. During fuel rotation or critical testing at atmospheric pressure, a reactor-core emergency-cooling spray ring having 24 nozzles may be installed in lieu of the above described spray ring. At other times, when either spray ring is not installed, the control rods shall be fully inserted and the reactor water level maintained above the reactor water low level alarm point. This spray ring shall be capable of spraying water over the top of the core through 24 spray nozzles in a manner so that cladding of not more than 8.62% of the fuel rods would melt in the event of a loss-of-coolant type accident.

(2) Four of the five pneumatically operated stop valves shall operate as follows whenever their respective control switches on the reactor control console are in their "automatic" positions:

(a) Valve N-11 (reactor feedwater makeup admission valve) shall close to block the normal makeup injection path if a reactor-vessel low-water-level scram signal is generated.

(b) Valve N-9 (makeup collecting tank fill valve) shall open to allow water from the 30,000-gal overhead storage tank to enter the makeup collecting tank if a reactor-vessel low-water-level scram signal is generated and if the water level in the makeup collecting tank level is below that which automatically closes Valve N-9.

(c) Valve N-10 (reactor-spray-ring high-pressure demineralized water admission valve) shall open to allow makeup water from the makeup-injection-pump discharge header to enter the reactor-vessel spray ring if a reactor-vessel low-water-level scram signal is generated and if at least one makeup injection pump is operating.

(d) Valve N-3 (reactor-spray-ring low-pressure demineralized-water admission valve) shall open to allow water from the 30,000-gal overhead storage tank to enter the reactor-vessel spray ring if a reactor-vessel low-water-level scram signal is generated and if the pressure within the reactor vessel is less than 25 psig.

(3) Pneumatically operated stop valve N-10 (reactor-spray-ring high-pressure demineralized water admission valve) shall open to allow makeup water from the makeup-injection-pump discharge header to enter the reactor-vessel spray ring whenever its control switch on the reactor control console is turned to its "open" position and an all-rod scram signal is generated.

(4) Pneumatically operated stop valve N-4 (reactor-spray-ring service-water admission valve) shall open to allow service water from the containment-building service-water supply header to enter the reactor-vessel spray ring if its control switch on the reactor control console is turned to its "open" position and if the pressure within the reactor vessel is less than the service-water supply-header pressure.

(5) The primary-coolant-system makeup-injection-pump control system shall start either or both of the pumps whose control switches on the reactor control console are in their "automatic" positions whenever a reactor-vessel low-water-level scram signal is generated.

(6) The reactor-core emergency cooling system shall be designed to provide approximately 15,000 gal of water from the 30,000-gal overhead storage tank for injection into the reactor vessel through the reactor-vessel spray ring whenever a reactor-vessel low-water-level scram signal is generated.

(7) Water shall flow through the reactor-vessel spray ring whenever the conditions of Specifications 12.1(3) (c) and 12.1 (3) (d) are met at the minimum rates of 12 gpm and 20 gpm, respectively.

(8) The reactor-core emergency cooling system shall be capable of supplying water to the reactor core through the spray ring for at least 500 min.

12.2 EMERGENCY AND TEST CONDENSER SYSTEM SPECIFICATIONS

The specifications relating to the emergency and test condenser system are as follows:

(1) The emergency and test condenser system shall consist of piping connecting each primary-coolant-system steam line to the tube-side inlet of the emergency and test condenser, a condensate flow-control valve, the emergency and test condenser, piping connecting the tube-side outlet of the emergency and test condenser to each primary-coolant-system feedwater return header, a shell-side service-water-level control valve, and service-water piping.

(2) All emergency and test condenser piping except the shell-side service-water piping shall be of stainless steel, ASTM Type 304, Schedule 80, and shall be designed, fabricated, and tested in accordance with the ASA Code for Pressure Piping.

(3) The emergency and test condenser shall be designed, fabricated, and tested in accordance with Sec. VIII of the 1956 edition of the ASME Boiler and Pressure Vessel Code for a tube-side pressure of 1250 psig, a shell-side internal pressure of 0 psig, and a shell-side external pressure of 21 psig.

(4) The surfaces of the emergency and test condenser in contact with the primary coolant, other than the tubes, shall be of Monel metal.

(5) The tubes of the emergency and test condenser shall be of an alloy containing 70 per cent copper and 30 per cent nickel.

(6) All surfaces of the emergency and test condenser in contact with the shell-side service water, other than the tubes, shall be of carbon steel.

(7) The emergency and test condenser shall be capable of transferring heat from the primary coolant to service water at the minimum rate of 154,000,000 Btu/hr when the primary coolant as saturated steam at approximately 564 F enters the tube side at the approximate rate of 250,000 lb/hr and when service water at temperatures ranging from 50 F to 90 F enters the shell side at a minimum rate of 136,000 lb/hr.

(8) The condensate flow-control valve shall open automatically whenever the primary pressure exceeds 1210 psig and whenever the condensate flow-control valve switch on the reactor control console is turned to its "open" position.

(9) The condensate flow-control valve shall be fitted with an adjustable mechanical valve-stem travel limiter.

(10) The shell-side service-water level-control valve shall be actuated by a pneumatic level-control system so as to keep the shell-side service water at an essentially constant level above the tubes.

12.3 DECAY-HEAT REMOVAL SYSTEM SPECIFICATIONS

The specifications relating to the decay-heat removal system are as follows:

(1) The decay-heat removal system shall consist of a shell-and-tube heat exchanger, a centrifugal pump, interconnecting piping, and valves.

(2) The decay-heat removal system shall be interconnected with the primary coolant system so that heat from primary coolant circulated by the decay-heat-removal-system pump from the reactor vessel through the tube side of the decay-heat-removal-system heat exchanger may be transferred to service water flowing through the shell side of the decay-heat-removal-system heat exchanger.

(3) The decay-heat-removal-system heat exchanger shall be designed, fabricated, and tested in accordance with Sec. VIII of the ASME Boiler and Pressure Vessel Code for a tube-side pressure of 1250 psig and a shell-side pressure of 1250 psig.

(4) The decay-heat-removal-system heat exchanger shall be capable of transferring heat from the primary coolant to the shell-side service water at the minimum rate of 3,000,000 Btu/hr when primary coolant enters the tube side at approximately 120 F at the approximate rate of 300 gpm and when service water enters the shell side at approximately 90 F at the approximate rate of 600 gpm.

(5) The decay-heat-removal-system pump shall be rated at 300 gpm against a net developed head of 45 ft of water.

(6) The decay-heat-removal-system pump shall be controled electrically from the reactor control console and from points inside the containment building.

(7) It shall be possible to operate the decay-heat-removal-system pump and its controls from either of two independent sources of electric power.

13. FUEL STORAGE

13.1 SPECIFICATIONS GOVERNING THE STORAGE AND HANDLING OF FRESH FUEL

The specifications governing the storage and handling of fresh fuel are as follows:

(1) Fresh fuel elements shall be stored in cabinets with hinged covers that can be locked. The cabinets shall be installed in the containment building as shown in Dwg. 12D-08000.

(2) It shall not be possible to assemble a critical mass by placing fresh fuel elements in the fresh-fuel-element storage rack in any number or arrangement whatever. The fresh-fuel-element storage rack shall hold fuel elements in fixed positions.

(3) The fresh-fuel-element storage cabinets shall be used only for the storage of fresh fuel elements of the design specified in these Technical Specifications, clean dummy fuel elements, fresh fuel center pins, clean poison pins, and the plastic bags and desiccant bags necessary for moisture control.

(4) Fresh fuel elements shall be moved within the containment building either by the building crane and a special fuel-element hoist device, or by the fuel-transfer equipment shown in Dwg. 12D-08000.

13.2 SPECIFICATIONS GOVERNING THE STORAGE AND HANDLING OF IRRADIATED FUEL

The specifications governing the storage and handling of irradiated fuel are as follows:

(1) Irradiated fuel elements shall be stored in the rack at the bottom of the fuel-element storage well shown in Dwg. 12D-08000.

(2) It shall not be possible to assemble a critical mass by placing fresh or irradiated fuel elements in the rack at the bottom of the fuel-element storage well in any number or arrangement whatever. The rack in the fuel-element storage well shall hold fuel elements and other stored components in fixed positions.

(3) The rack at the bottom of the fuel-element storage well shall be used only for the storage of irradiated fuel elements, fresh fuel elements, dummy fuel elements, loaded and empty fuel-element center-pin storage cans, and the reactor-core neutron source.

(4) Water in the fuel element storage well shall be cooled by a storage-well cooling system consisting of a full-flow filter, a shell-and-tube heat exchanger, a circulating pump, portable demineralizer connections, interconnecting piping, and valves.

(5) The storage-well cooling-system heat exchanger shall be capable of transferring heat from storage-well water to service water at the minimum rate of 1,000,000 Btu/hr when storage-well water enters the tube side at approximately 120 F at the approximate rate of 100 gpm and service water enters the shell side at approximately 90 F at the approximate rate of 250 gpm.

(6) The storage-well cooling-system circulating pump shall be rated at 100 gpm against a net developed head of 105 ft of water.

(7) A system shall be provided for flooding the shield cavity above the reactor vessel and the fuel-element storage well with demineralized water. The depth of the water in the shield cavity and the fuel-element storage well shall be maintained so as to limit the radiation dose rates to those permitted by 10 CFR, Part 20.

14. ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS

14.1 SPECIFICATIONS RELATING TO WRITTEN PROCEDURES

The specifications relating to written procedures are as follows:

(1) The plant shall be unloaded, started up, operated, maintained, tested, shut down, and reloaded in accordance with approved detailed written procedures conforming to these Technical Specifications and to all applicable Federal and State regulations.

(2) Foreseeable plant emergencies shall be dealt with in accordance with detailed written procedures conforming to these specifications and to all applicable Federal and State regulations.

(3) Unforeseeable plant emergencies shall be dealt with as the protection of the public, plant personnel, and property necessitates; in no circumstances shall the protection of the public, plant personnel and property be subordinated to the observance of written procedures.

(4) Each member of the operating organization shall be familiar with the written procedures for which he has partial or complete responsibility.

(5) The operating organization shall conduct drills often enough to ensure their proficiency in emergency operations and their familiarity with written procedures.

(6) Detailed written radiological-safety procedures conforming to Title 10 of the Code of Federal Regulations, Part 20; Chapter 0524 of the AEC Manual; and NBS Handbooks 59 and 69 shall be enforced at all times.

(7) All written procedures and all changes in written procedures shall be approved by the Chicago Operations Office of the USAEC.

14.2 ADMINISTRATIVE RULES

(1) At least two persons, one of whom shall be AEC licensed to operate the Elk River Reactor, shall be in the reactor control room (a) whenever the reactor is critical, (b) whenever reactor conditions are changing so as to affect reactivity, or (c) whenever reactor components are being modified while one or more fuel elements are in the reactor. At such times a reactor shift supervisor, who shall be AEC licensed to operate the reactor shall be present at the plant. Such reactor shift supervisor may be considered to be the AEC licensed operator when he is in the control room. At all other times, the reactor control room shall be locked or attended by at least one person, and the reactor controls shall be locked. The plant shall be attended while reactor fuel is present.

CHANGE SHEET

July 1, 1963

Following is a list of the differences between the proposed RCPA Technical Specifications dated July 1, 1963 and Appendix "A" to DPRA-3 as revised February 7, 1963.

Page 1-1, Para. 1 Introduction

Delete the first paragraph and insert the following: "This document sets forth the technical specifications that shall govern the design and operation of the Elk River Reactor plant."

Page 2-1 - no change

Page 3-1 - no change

Page 3-2 - no change

Page 3-3 - no change

Page 3-4 - no change

Page 3-5, Para. (c)

Delete this paragraph and insert the following:
"(c) Electrical penetrations, including the Micarta plates, shall be tested by the halide or helium detector method with a driving pressure differential no less than 0.5 psi, or by the soap bubble method with a driving differential of no less than 5 psi. The results of this test shall be used as a guide in evaluating leakage."

Page 3-6 - no change

Page 3-7 - no change

Page 4-1 - no change

Page 4-2, Para. 4.2(3)

Delete Para. 4.2(3) and insert the following:
"The rated operating temperature and pressure of the reactor vessel shall be 564F and 1153 psig, respectively."

Page 4-3 - no change

Page 4-4 - Delete all references to pressures in the two columns titled "Operating pressure variations" and "Hydrostatic test pressures".

Delete reference (1) and insert a new reference (1) as follows: "(1) Pressure variation expressed in percent of 1153 psig."

Delete the following sentence, "For the purpose of computing operating time as defined in this paragraph, all periods of time during which the reactor vessel pressure exceeds 250 psig shall be included.", and insert in its place the following sentence, "for the purpose of computing operating time, records of the megawatt days produced in the reactor will be kept".

Page 4-5, <u>4.4 Evaporator Specifications</u>

Delete para. (4) and insert in its place the following:
"(4) Each evaporator shall be capable of transferring heat from the primary coolant to the secondary coolant at the minimum rate of 78, 300, 000 Btu/hr when saturated steam at approximately 564 F enters the tube side at the approximate rate of 126, 500 lb/hr and when secondary-system feedwater at approximately 486 F enters the shell side at the approximate rate of 104, 000 lb/hr."

Page 4-5 & 6, <u>4.5 Subcooler Specifications</u>

Delete para. (4) and insert in its place the following:
"(4) Each subcooler shall be capable of transferring heat from the primary coolant to the secondary coolant at the minimum rate of 15, 500, 000 Btu/hr when condensate at approximately 564 F enters the tube side at the approximate rate of 125, 000 lb/hr and when secondary-system feedwater at approximately 350 F enters the shell side at the approximate rate of 104, 000 lb/hr."

Page 4-7, Sec. 4.8 Para (1)

Delete 544 F and parenthesis around 569F

Page 4-8, Para (7)

Delete 922 psig and parenthesis around 1153 psig.

Page 4-9 - no change

Page 5-1 - no change

Page 5-2 - no change

Page 5-3 - no change

Page 5-4 - no change

Change Sheet

Page 6-1, Para (1) third line --

Delete the words "a 25 Mwe turbine-generator set" and insert the words "a turbine generator set having a 22 Mwe turbine name plate rating".

Page 6-1, Para (2)

Delete this paragraph and add the following:
"(2) The secondary coolant shall be light water and shall conform with the following requirements:

maximum conductivity (in evaporator shell)	500 micromhos/cm
maximum chloride content	0.10 ppm
maximum chloride content (in evaporator shell)	0.5 ppm
maximum silica content	0.01 ppm
maximum silica content (in evaporator shell)	3.0 ppm
maximum oxygen content	0.1 cc/liter
maximum oxygen content (outlet of deaerating cond. tank)	0.005 cc/liter
maximum total solids content	0.5 ppm
maximum total solids content (in evaporator shell)	250 ppm
maximum hardness	0.1 ppm
approximate pH	9.0

Page 6-2 - no change

Page 6-3, Para (5) and (6)

Delete paragraphs (5) and (6) and insert the following:
"(5) Maximum design and operating pressures and maximum relief-valve settings at various points in the secondary coolant system shall be as follows:

	maximum pressure, psig		
	design	operating	relief-valve setting
evaporator outlet	1000	910	975
superheater outlet	975	863	900
turbine inlet	900	850	960
subcooler inlet	1000	923	None

"(6) Maximum design and normal operating temperatures at various points in the secondary coolant system shall be as follows:

	maximum temperature, °F	
	design	operating
evaporator outlet	575	536
superheater outlet	905	905
turbine inlet	900	900
subcooler inlet	575	350

Change sheet

Page 6-3, Para (7)

Change the minimum operating temperature at the shell-side inlet of each subcooler from 300 to 70.

Page 6-4 - no change

Page 7-1 - no change

Page 7-2 - no change

Page 7-3 - no change

Page 7-4 - no change

Page 8-1, Para (2)

Change the Maximum Reactivity Worth of one control rod from 0.035 to 0.045.

Page 8-1, Para (5)

Delete this paragraph and insert in its place the following:
"(5) The rate at which reactivity may be increased by movement of control rods shall be limited by administrative control so as not to exceed 0.001/sec in any core with Keff greater than 0.98. The maximum capability of the control system in any core with Keff greater than 0.98 shall not exceed 0.002/sec.

Page 8-2 - no change

Page 8-3 - no change

Page 8-4, Para (6)

Last line -- delete the words "all rods insert" and insert the words "all shim rods insert".

Page 8-5 - no change

Page 8-6 - no change

Page 8-7, Sec. 8.2

Delete the tabulation for N-1 through N-9 and insert the following tabulation in its place.

Change Sheet

channel designation	channel type	approximate power-level range	detector type	detector sensitivity
N-1	startup	0.006 w-600 w	proportional counter	15 (counts/sec)/nv
N-2	startup	0.006 w-6000 w	proportional counter	15 (counts/sec)/nv
N-3	log-n & period	60 w-180 Mw	comp. ion. chamber	4.0×10^{-14}amp/nv
N-4	log-n & period	60 w-180 Mw	comp. ion. chamber	4.0×10^{-14}amp/nv
N-5	linear power	6 w-90 Mw	comp. ion. chamber	4.0×10^{-14}amp/nv
N-6	linear power	6 w-90 Mw	comp. ion. chamber	4.0×10^{-14}amp/nv
N-7	level safety	600 kw-90 Mw	uncomp. ion. chamber	4.4×10^{-14}amp/nv
N-8	level safety	600 kw-90 Mw	uncomp. ion. chamber	4.4×10^{-14}amp/nv
N-9	level safety	600 kw-90 Mw	uncomp. ion. chamber	4.4×10^{-14}amp/nv

Page 8-8, Para (5)

Delete this paragraph and insert in its place the following paragraph: "(5) Channel N-5 shall actuate a control room alarm whenever it senses a reactor power level less than 5 percent or greater than 110 percent of its power range setting, and shall actuate a control room alarm and initiate an all-rod scram signal whenever it senses a reactor power level exceeding 125 percent of its power-range setting."

Page 8-8, Para (6)

Delete this paragraph and insert in its place the following paragraph: "(6) Channel N-6 shall actuate a control-room alarm whenever it senses a reactor power level less than 5 percent or greater than 110 percent of its power range setting, and shall actuate a control room alarm and initiate an all-rod scram signal whenever it senses a reactor power level exceeding 125 percent of its power-range setting."

Page 8-8, Para (11)

Add the following paragraph:
"(11) Circuitry associated with the signal outputs of Channel N-5 and N-6 shall be arranged so that a two-position key-operated switch will determine whether an all-rod scram signal will be initiated whenever either (or both) of the two channels senses a power level greater than 125 percent of the power range setting."

Page 8-9, Sec 8.3.3

Change the words "obtaining the revoking" to "obtaining or revoking".

Page 8-10, Sec 8.3.5

Para (2) -- change the dimensions "1 ft to 3.7 ft" to read "5 ft-4 in.".
Para (4) -- change 975 psig (1195 psig) to read 1195 psig.
Para (6) -- delete 650 psig (880 psig) and add 880 psig.

Change Sheet

Page 8-11 - no change

Page 8-12 - Add the following item:

(74) AC-DC Trouble on MG Set

Page 8-13, Sec 8.4.3, Para (1) and (2)

Change references to "Reactor Shift Supervisor" to read "reactor shift supervisor"

Page 8-14

Change all references to "Reactor Shift Supervisor" to read "reactor shift supervisor".

Page 8-15, Para (5), line 3 --

Change the words "125-v battery kept floating" to read "125-v battery bank kept floating".

Page 8-16 - no change

Page 9-1, Sec 9.1

Para (1), line 3 -- change "beta-gamma activity" to read "beta activity".

Para (3), line 1 & 2 -- change "beta-gamma activity" to read "beta activity".

Para (3), line 3 -- delete "5×10^{-5} to 5×10^{-1} microcuri/sec" and insert in its place "7×10^{-6} to 7×10^{-1} microcurie/sec when a background of 20 cpm exists, and when using Tl^{204} as a standard".

Para (4), line 1 & 2 -- change "beta-gamma activity" to read "gamma activity".

Para (4), line 3 -- delete "30 to 3×10^{4} microcurie/sec" and insert in its place "9×10^{-1} to 7×10^{3} microcurie/sec when a background of 250 cpm exists, and when using Ar^{41} as a standard".

Para (5), line 4 -- delete the fourth line and insert in its place "range from 7×10^{-6} to 1.7×10^{-1} microcurie/sec when a background of 20 cpm exists and when using Tl^{204} as a standard".

Para (6), line 1 -- change "beta-gamma activity" to read "beta activity".

Para (6), line 4 -- delete line four and insert in its place "of 2.5 microcurie/sec or less.

Change Sheet

Page 9-1, Sec 9.1

Para (7), line 1 -- change "beta-gamma activity to read "gamma activity".
line 3 -- change 600 to read 1680.
line 4 -- after "curies/sec." add the words "or less".

Para (8), line 4 -- change the fourth line to read "2.5 microcurie/sec or less".

Page 9-2, 9.1

Para (13) -- change 60 to read 1680; change 600 to read 16,800; change 5×10^{-4} to read 2.5 and change 5×10^{-3} to read 2.5.

Para (14) -- delete this paragraph in its entirety.

Page 9-2, Sec 9.2

After paragraph (3) add the following paragraph from page 9-2-A:
"(4) The fission-product monitor shall generate a high-radiation alarm signal whenever its associated devices and circuitry indicate a gross counting rate corresponding to a gamma activity level of 40 cpm above the normal background in the absence of fission products."

Page 9-2-A - Delete this page

Page 9-3 - no change

Page 9-4, Sec 9.4

Para (3) -- delete line two and insert in its place "concentrations that exceed 5×10^{-6} microcurie/cc when a background of 250 cpm exists and when using Zn^{65} as a standard."
Para (4) -- change 2×10^{-5} to read 2×10^{-4}

Page 9-5

Para (3) -- change last line to read "borne gamma activity levels of 5×10^{-6} microcurie/cc when a background of 250 cpm exists and when using Zn^{65} as a standard."

Page 9-6

Para (4) -- change last line to read "exceed 1×10^{-11} microcurie/cc when a background of 25 cpm exists and when using Tl^{204} as a standard.

Change Sheet

Page 9-6

Para (5) -- change "beta-activity" to read "beta-gamma activity". After "microcurie/cc" in the last line, add the words "when a background of 25 cpm exists and when using Tl^{204} as a standard.".

Para (6) -- change 3×10^{-10} to read 3×10^{-9}.

Page 9-7

Para (4), last line -- add the words "or less".

Page 10-1 - no change

Page 10-2

Para (9), last line -- add the words "or to a tank truck connection".

Para (11) -- delete the last sentence beginning on line 10.

Page 10-3 - no change

Page 11-1 - no change

Page 11-2 - no change

Page 11-3 - no change

Page 12-1

Para (1) -- delete this paragraph and insert in its place the following new paragraph:
"(1) The reactor-core emergency-cooling spray ring shall be capable of spraying water over the top of the core through 12 spray nozzles in a pattern determined on the basis of the cooling-water requirements of each part of the core in a manner so that the cladding of not more than 8.62% of the fuel rods would melt in the event of a loss-of-coolant type accident. During fuel rotation or critical testing at atmospheric pressure, a reactor-core emergency-cooling spray ring having 24 nozzles may be installed in lieu of the above described spray ring. At other time, when either spray ring is not installed, the control rods shall be fully inserted and the reactor water level maintained above the reactor water low-level alarm point. This spray ring shall be capable of spraying water over the top of the core through 24 spray nozzles in a manner so that the cladding of not more than 8.62% of the fuel rods would melt in the event of a loss-of-coolant type accident."

Para (2) -- delete this paragraph in its entirety.

Para (3) & (4) -- change paragraph (3) to (2) and (4) to (3).

Change Sheet

Page 12-2

Para (5) through (8) -- change Para (5) to (4), (6) to (5), (7) to (6) and (8) to (7).

Para (9) -- change (9) to (8) and delete the words "Regardless of its mode of operation".

Page 12-3

Para (7) -- delete all parenthesis and the following numbers: 168,500,000 Btu/hr, 536 F, 258,000 lb/hr and 149,000 lb/hr.

Page 12-4

Para (6), last line -- change "a point" to read "points".

Page 13-1 - no change

Page 13-2 - no change

Page 14-1, Sec 14.1

Para (1) -- change the word "loaded" to "unloaded" and the word "unloaded" to "reloaded".

Para (4) and (5) -- change the word "staff" to read "organization".

Para (7) -- delete this paragraph and insert the following paragraph: "(7) All written procedures and all changes in written procedures shall be approved by the Chicago Operations Office of the USAEC."

Page 14-1, Sec 14.2

Delete paragraph (1) and insert the following paragraph: "(1) At least two persons, one of whom shall be AEC licensed to operate the Elk River Reactor, shall be in the reactor control room (a) whenever the reactor is critical, (b) whenever reactor conditions are changing so as to affect reactivity, or (c) whenever reactor components are being modified while one or more fuel elements are in the reactor. At such times a reactor shift supervisor, who shall be AEC licensed to operate the reactor shall be present at the plant. Such reactor shift supervisor may be considered to be the AEC licensed operator when he is in the control room. At all other times, the reactor control room shall be locked or attended by at least one person, and the reactor controls shall be locked. The plant shall be attended while reactor fuel is present."

Change Sheet

Page 14-2

Change all references to "Reactor Shift Supervisor" to read "reactor shift supervisor".

Para (10) -- change "650 psig (870 psig)" to read "650 psig" and change "975 psig (1195 psig) to read "1195 psig".

Page 14-3 - no change

Page 14-4

Change all references to "Reactor Shift Supervisor" to read "reactor shift supervisor".

Para (22) -- delete this paragraph and insert in its place the following: "(22) All modifications in facility design shall be approved by the Chicago Operations Office of the U. S. Atomic Energy Commission."

Para (23) -- delete this paragraph and insert in its place the following: "(23) All changes in operating organization shall be approved by the Chicago Operations Office of the U. S. Atomic Energy Commission."

Para (26), line 2 -- change "Operations Supervisor or the Project Manager" to read "reactor shift supervisor".

Para (26), line 4 -- change "Operations Supervisor" to read "Nuclear Plant Manager" and delete the words "and with the assistance of a Reactor Physicist".

Page 14-5

Para (27) -- delete this paragraph in its entirety.

Para (28) -- change to paragraph (27)

Para (29) -- change to paragraph (28)

Para (c) -- after the words "trip levels" insert the words "on Channels N-5 and N-6".

After item (c), add the following new item:
"(d) The key-operated switch specified in Sec 8.3.1(11) shall be positioned so that an all-rod scram signal is generated whenever either one of Channels N-5 or N-6 senses a reactor power level greater than 125 percent of power range setting while the reactor power is less than 3 Mw."

Para (30) -- change to paragraph (29)

Change Sheet

Page 14-6

Para (14), Item (b) -- delete this item and insert the following item in its place:
"(b) borescopic examination of the 10 inch nozzles, direct examination of the weld-overlay cladding at the vessel shell flange and the top head flange; and".

Page 14-7 - no change

Page 14-8, Sec 14.4

Delete Sec 14.4 in its entirety, including paragraphs (1) and (2) on page 14-8, and paragraphs (3) through (10) on pages 14-9, 14-10 and 14-11.

Reference Page:

Reference 1 - change to read:
"1. Final Hazards Report for the Elk River Reactor at Elk River, Minnesota, Allis-Chalmers Manufacturing Company, Washington D. C., July 8, 1960, as revised November 15, 1962.

Reference 9 - Delete this reference in its entirety.

(2) No person shall enter the reactor control room whenever the reactor core contains one or more fuel elements, except with the permission of the reactor shift supervisor.

(3) Keys for all reactor-plant bypass switches shall be available only to the reactor shift supervisor, and shall be used only under the direct supervision of the reactor shift supervisor. Keys to other control devices shall be available to the operating personnel only on the authority of the reactor shift supervisor.

(4) All reactor-plant valves that are locked in position shall not be operated except upon order of the reactor shift supervisor.

(5) At least one nuclear instrumentation channel shall be sensing and indicating the reactor neutron flux whenever the reactor core contains one or more fuel elements.

(6) The mechanical interlock between the containment-building personnel airlock doors may be disabled only when none of the containment integrity provisions specified in Sec. 3.6 are required.

(7) Upon the occurrence of abnormal behavior of the reactor, including its controls and safety systems, action shall be taken promptly to secure the safety of the plant (including, if necessary, shutting down the reactor) and to determine and eliminate the cause of the abnormal behavior.

(8) The containment-building spray-system motor-operated valve shall be manually operated only (a) for testing in accordance with Specification 3.4(4) or (b) upon direct order from the reactor shift supervisor.

(9) The 12 fuel elements immediately adjacent to any control rod shall be removed from the core before that control rod is removed. The fuel-element positions from which the 12 fuel elements were removed shall remain empty of fuel elements until the control rod is re-installed or replaced.

(10) The control-rod actuating system shall not be placed in its automatic mode of operation except when the primary-coolant-system pressure is greater than 650 psig and less than 1195 psig.

(11) The boric-acid injection valves and the boric-acid-injection-system leakoff valve shall be operated only upon direct order from the reactor shift supervisor.

(12) Whenever the containment-building vent-duct butterfly dampers are open, the stack monitor shall be operating, except during periods of monitor maintenance or repair not exceeding 24 hr, at which time either the immediate particulate or the immediate gaseous beta-gamma activity detector must be operating and capable of actuating the containment building ventilation duct damper closing mechanisms.

(13) Whenever the recombiner condenser is receiving gases from the primary system and venting noncondensable gases to the stack, the fission-product monitor shall be operating, except during periods of monitor maintenance or repair not exceeding 24 hr. During any such period of maintenance or repair, the three detectors of the stack monitor must be operational.

(14) All containment-building area monitors shall be operated whenever irradiated fuel elements are in the building or whenever the reactor-core neutron source is in the building, except those shut down briefly for maintenance or repair. Adequate spare parts shall be on hand to allow repairs to be made promptly. Whenever one or more of the area monitors are shut down, portable radiation detection instruments shall be employed by all personnel entering the area or areas usually monitored by the shut-down area monitor or monitors.

(15) Whenever water is flowing from the containment building through the service-water return header, the sewer monitor shall be operating, except during periods of monitor maintenance or repair not exceeding 24 hr. A grab sample shall be taken every 8 hr during any such maintenance or repair period.

(16) Whenever the main turbine condensate pumps are circulating condensate from secondary steam generated in the evaporators, the secondary-system water monitor shall be operating, except during periods of monitor maintenance or repair not exceeding 24 hr. During any such maintenance or repair period, the turbine air-ejector monitor shall be operable.

(17) The containment-building plant particulate monitor shall be operated whenever irradiated fuel elements are in the building or whenever the reactor-core neutron source is in the building, except during periods of monitor maintenance or repair not exceeding 24 hr. During any such maintenance or repair period, the stack particulate monitors shall be operable. Continuous samples shall be taken with a portable filter air sampler. The filters shall be changed every 8 hr.

(18) The turbine air-ejector monitor shall be operated whenever the turbine air ejector is operating and secondary steam generated in the evaporators is passing through the turbine inlet valve, except during periods of monitor maintenance or repair not exceeding 24 hr. During any such maintenance or repair period, the secondary-system water monitor shall be operable.

(19) The control switches for the reactor-feedwater makeup admission valve, the makeup-collecting-tank fill valve, the reactor spray-ring high-pressure demineralized-water admission valve, and the reactor spray-ring low-pressure demineralized-water admission valve shall be in their "automatic" positions whenever the reactor core contains one or more fuel elements, except when they are being operated upon order of the reactor shift supervisor or when their associated valves are being serviced or repaired.

(20) The valve-stem travel limiter of the emergency-and-test-condenser condensate flow-control valve shall be positioned to limit travel between the fully closed position and the half-open position, except during the tests to be made as part of initial startup and testing program for which a greater opening is specified, and except during reactor plant cool-down when the primary coolant temperature is below 325 F and all control rods are within 1/2-in. of their fully inserted positions.

(21) The number of fuel elements of the reactor site but not in the fresh fuel-element storage cabinets, the fresh-fuel element shipping containers, the irradiated fuel-element storage rack, the irradiated fuel-element shipping cask, or the reactor core shall not exceed eight.

(22) All modifications in facility design shall be approved by the Chicago Operations Office of the U. S. Atomic Energy Commission.

(23) All changes in operating organization shall be approved by the Chicago Operations Office of the U. S. Atomic Energy Commission.

(24) Should it be suspected that any automatic closing of the containment-building ventilation-system valves was caused by a fission-product release from the reactor, the reactor shall be shut down, and unless it has been determined that the automatic closing was not caused by a fission-product release from the reactor, the reactor shall not be operated, except for further investigation of the cause of the closure, until the fuel elements have been inspected for defects.

(25) Maintenance operations on the primary system, reactor components, or emergency systems while fuel is in the reactor shall be conducted under the supervision of the reactor shift supervisor. Maintenance involving the opening of systems containing radioactive materials shall be conducted under the surveillance of a Health Physics Representative.

(26) Reactor tests involving the bypassing of interlocks shall be authorized by the reactor shift supervisor. Reactor tests involving the measurement of nuclear parameters of the reactor core shall be conducted on the authority of the Nuclear Plant Manager.

(27) In the course of all operations in which the reactivity of the reactor core may vary, a minimum of two neutron detectors shall indicate (a) a count rate of at least 2 counts/sec and (b) a count rate of at least twice the level of instrument noise.

(28) The following conditions shall obtain in reactor startup operations:

(a) all nuclear instrumentation (Channels N-1 through N-9) shall be operative when the startup is initiated;

(b) the key-operated switch specified in Sec. 8.3.1(10) shall be positioned so that an all-rod scram signal will be generated whenever any one of Channels N-7, N-8, and N-9 senses a reactor power level greater than 125% of the reactor full-power level while reactor power is less than 3 Mw; and

(c) neutron-flux scram trip levels on Channels N-5 and N-6 shall be depressed during startup and operation at low reactor power.

(d) The key-operated switch specified in Sec. 8.3.1(11) shall be positioned so that an all-rod scram signal is generated whenever either one of Channels N-5 or N-6 senses a reactor power level greater than 125 percent of power range setting while the reactor power is less than 3Mw.

(29) At least two of Channels N-7, N-8, and N-9 shall be operative during generation of electric power. During any operation in which one of these channels is inoperative, the scram selector switch shall be positioned so that a scram signal from either operative channel will cause a scram.

14.3 SPECIFICATIONS RELATING TO TESTS AND MEASUREMENTS

The specifications relating to various tests and measurements are as follows:

(1) Rod drop tests shall be conducted before approaching criticality during each cold startup, and the time required for each control rod to fall from its fully withdrawn position to within 1/2-in. of its fully inserted position shall be measured. Rod drop tests will not be required more often than once a month unless the reactor vessel head has been removed or maintemance work performed on the control-rod drive mechanisms.

(2) Whenever the reactor core contains one or more fuel elements, the radiation-monitoring-system monitors shall be checked for proper operation with their internal self-checking devices at least once a day.

(3) Whenever the reactor core contains one or more fuel elements, the radiation-monitoring-system monitors shall be tested for proper operation with a calibration standard or independent samples at least once a week.

(4) Each portable radiation detector shall be tested for proper operation and calibration every two months.

(5) The boric-acid injection valves and the boric-acid system leakoff valve shall be tested for proper operation from the reactor control room and from the decontamination room at least once a year.

(6) The automatic transfer switch that connects the 10-kva transformer to the failure-free bus if the motor-generator fails (see Sec. 8.5) shall be tested for proper operation (when the reactor is shut down) at least once every three months.

(7) The automatic transfer switch that connects the emergency lighting load to the 125-v d-c distribution bus if the 120-v/208-v bus is de-energized shall be tested for proper operation at least once every three months. The 50-kw generator shall be started at least once a week.

(8) The containment-building ventilation-duct-damper automatic-closure circuit shall be tested for proper operation at least once very three months.

(9) The reactor-core emergency cooling system automatically operated valves and their associated devices shall be tested for proper operation (when the reactor is shut down) approximately once a year.

(10) Each four-rod scram circuit shall be tested for proper operation before each approach to criticality that follows a shutdown period more than 24 hr long.

(11) Each all-rod scram circuit shall be tested for proper operation before each approach to criticality that follows a shutdown period more than 15 min long.

(12) The primary-coolant-system piping shall be tested hydrostatically at 1375 psig whenever a new penetration is made or an old penetration is plugged, and at 1250 psig whenever other minor repairs are made.

(13) The reactor vessel shall be tested hydrostatically at 1375 psig whenever a new penetration is made or an old penetration is plugged, and at 1250 psig after any gasketed joint has been opened and re-sealed. All hydrostatic tests shall be performed with the vessel at a temperature no lower than the nil-ductility transition temperatures specified in Sec. 4.8(2).

(14) The reactor vessel shall be inspected approximately every full-power year of operation to determine its structural adequacy under operating conditions. The inspection shall include at least the following:

(a) borescopic examination of the weld-overlay cladding of the 8-in. and 16-in. nozzles (The area examined shall include the inside surface of each nozzle for an axial distance of approximately 6 in. and the weld-overlay cladding of the vessel shell surrounding each nozzle within a distance of approximately twice the nozzle wall thickness as measured from the inside edge of the nozzle);

(b) borescopic examination of the 10-in. nozzles, direct examination of the weld-overlay cladding at the vessel shell flange and the top head flange; and

(c) direct examination of the weld-overlay cladding at the inside edges of the 12-in. nozzles.

All surfaces to be inspected shall first be cleaned to remove any deposits built up during reactor operation. Any unusual conditions detected during the internal examination shall be noted and recorded for comparison with original conditions and with following inspections and to permit evaluation of any observed defects.

(15) Vessel-material specimens shall be periodically removed from the reactor vessel to determine the shift in their nil-ductility transition (NDT) temperature. Vessel-material specimens shall include samples of the following carbon steels used in fabricating the vessel shell, nozzles, and flanges: ASTM-A-302 Grade B, and ASTM-A-105 Grade II.

Forty capsules shall be mounted around the inner periphery of the thermal shield, centered on the axial midplane of the core. Thirty-two of the 40 capsules shall contain samples of ASTM-A-302 Grade B carbon steel, of which 20 capsules shall each contain 10 Charpy-V-notch samples and the other capsules shall each contain one standard tensile specimen and six miniature tensile specimens. Each of the other eight capsules shall contain 16 samples of ASTM-A-105 Grade II carbon steel, of which 10 shall be Charpy-V-notch impact specimens and six shall be miniature tensile specimens.

Specimens of ASTM-A-302 Grade B carbon steel shall be tested to determine the normal (unirradiated) properties of the material. These specimens shall be made from a material whose chemical and physical properties and heat treatment are similar to those of the vessel material.

Irradiated capsules containing ASTM-A-105 Grade II and ASTM-A-302 Grade B material shall be removed from the vessel and disassembled in accordance with the following schedule:

(a) The first impact and tensile specimen capsule shall be removed after one full-power year or approximately 2 x 10^4 thermal megawatt-days of operation.

(b) The remaining impact and tensile specimen capsules shall be removed one at a time, coincident with plant shutdown, or at some convenient period approximating 2 x 10^4 thermal megawatt-days of operation.

After capsule removal and disassembly, the integrated neutron dose shall be determined by radioactivity detection of the flux wires taken from the capsule. The flux-wire data shall be used to confirm the calculated dosage rates. The removal frequency shall be modified if the measure dosage does not agree with the calculated dosage.

Charpy tests shall be performed on the impact specimens to establish NDT temperature correlations. The yield point, tensile strength and elongation shall be determined from the tensile speciments.

(16) The flow-control valve of the emergency and test condenser and its actuating circuits shall be tested at least once per year.

(17) Power level nuclear instrumentation of the reactor shall be calibrated by heat balance at least once every three months.

(18) The operation of the containment-building vacuum breakers shall be tested at least once every 18 months.

(19) Safety-system interlocks for which tests have not been specified elsewhere in these Technical Specifications shall be tested at least once every three months.

(20) After about three months of power operation of the reactor, all control rods shall be removed from the reactor and inspected visually for cracks.

(21) Tensile and impact specimens of boron stainless-steel control-rod blade material shall be placed in the reactor so as to be irradiated whenever the reactor is critical. Tensile and impact specimens shall be removed at intervals of approximately 1×10^4 thermal megawatt-days of operation. These specimens shall be tested to determine the impact value, the yield point, the tensile strength, and the elongation of the irradiated material. The control-rod blades shall be used in the reactor only for as long as they do not contain observable cracks or for as long as tests of the blade-material specimens specified in this paragraph indicate that they have minimum ductility corresponding to an elongation at fracture of 3.4 per cent.

(22) After approximately three months, but not later than approximately six months, of operation at least three control-rod drive mechanisms shall be removed from the reactor for inspection. These mechanisms shall be disassembled and the components of each mechanism fabricated of 17-4 PH stainless steel shall be inspected visually for defects.

REFERENCES

1. Final Hazards Report for the RCPA Elk River Reactor at Elk River, Minnesota, Allis-Chalmers Manufacturing Company, Washington, D. C., July 8, 1960, as revised November 15, 1962.

2. Technical Specifications for the RCPA Elk River Reactor at Elk River, Minnesota (Revised), Allis-Chalmers Manufacturing Company, Washington, D. C., September 1, 1961.

3. Report on Elk River Reactor Control Rod Blades, Allis-Chalmers Manufacturing Company, Washington, D. C., May 4, 1961.

4. 17-4 PH Material in the Elk River Reactor, Allis-Chalmers Manufacturing Company, Washington, D. C., May 4, 1961.

5. Elk River Reactor Vessel Data Book (Revised) (ACNP-62506), Allis-Chalmers Manufacturing Company, Washington, D. C., March 8, 1962.

6. Letter from Allis-Chalmers Manufacturing Company to the U. S. Atomic Energy Commission, subject: "ELK RIVER REACTOR SURVEILLANCE PROGRAM", June 28, 1962.

7. Procedure for Rounding the Inside Edges of Certain Nozzles of the Elk River Reactor Pressure Vessel, Allis-Chalmers Manufacturing Company, Washington, D. C., June 8, 1962.

8. Letter from Allis-Chalmers Manufacturing Company to the U. S. Atomic Energy Commission, subject: "LOCATION OF THERMOCOUPLES ELK RIVER REACTOR VESSEL (DRAWING 12E-11810)", June 13, 1962.