



UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545
NORTHERN STATES POWER COMPANY

DOCKET NO. 50-130

PROVISIONAL OPERATING LICENSE

License No. DPR-11

3-12-64

1. This provisional operating license applies to the controlled recirculation boiling water reactor owned by the Northern States Power Company (hereinafter referred to as "Northern States"). The reactor which is part of the Pathfinder Atomic Power Plant is located approximately five and one-half miles northeast of Sioux Falls, South Dakota. The reactor is described in the licensee's application for operating license dated March 30, 1959, and amendments thereto dated July 6, 1959, August 7, 1959, November 5, 1959, November 20, 1959, December 18, 1959, August 24, 1960, November 7, 1960, January 17, 1961, May 22, 1962, June 12, 1962, February 22, 1963, April 24, 1963, May 15, 1963, May 29, 1963, June 11, 1963, June 18, 1963, August 14, 1963, August 28, 1963, October 21, 1963, October 24, 1963, October 29, 1963, October 30, 1963, December 18, 1963, and February 6, 1964, (hereinafter collectively referred to as "the application").
2. Subject to the conditions and requirements incorporated herein, including the Technical Specifications hereto, the Commission hereby licenses Northern States:
 - A. Pursuant to Section 104(b) of the Act and 10 CFR 50, to possess, use and operate the reactor as a utilization facility.
 - B. Pursuant to the Act and 10 CFR 70, to receive, possess and use in operation of the reactor at any one time:
 - (1) 800 kilograms of contained uranium-235;
 - (2) 128 grams of plutonium encapsulated as two 1-curie and one 6-curie plutonium-beryllium neutron sources.
 - C. Pursuant to the Act and 10 CFR 30, to receive, possess and use in operation of the reactor at any time:
 - (1) 10,000 curies of antimony-124 as an antimony-beryllium neutron source;
 - (2) Three sealed sources of cobalt-60 not to exceed 100 millicuries each for calibration of film badges and instruments and for the testing of radiation monitors and measurement of liquid levels in tanks and pipes;

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- (3) 300 microcuries of cesium-137 to be used as a laboratory standard;
- (4) 50 microcuries of iron-59 to be used as a laboratory standard;
- (5) 100 microcuries of strontium-90 to be used as a laboratory standard;
- (6) 300 microcuries of cobalt-60 to be used as a laboratory standard;
- (7) 0.002 microcuries of americium-241 for calibration of instruments;
- (8) 50 millicuries of krypton-85 for calibration of gaseous activity monitors;
- (9) 300 microcuries of chromium-51 in a solution of CrCl_3 to be used as a laboratory standard.

D. Pursuant to the Act and 10 CFR 30, to possess, but not to separate such byproduct material as may be produced by operation of the reactor.

3. This license shall be deemed to contain and be subject to the conditions specified in Section 30.32 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, Title 10, Chapter 1, CFR, and to be subject to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission, now or hereafter in effect, and to the additional conditions specified below:

- A. Northern States shall not operate the reactor at power levels in excess of one (1) megawatt thermal.
- B. Northern States shall not install the proposed nuclear superheater fuel in the reactor without prior written authorization by the Commission.
- C. Technical Specifications

The Technical Specifications contained in Appendix "A" hereto are hereby incorporated into this license. Except as otherwise permitted by the Act and the rules, regulations, and orders of the Commission, Northern States shall operate the reactor in accordance with the Technical Specifications. No changes shall be made in the Technical Specifications unless authorized by the Commission as provided in 10 CFR 50.59.

D. Records

In addition to those otherwise required under this license and applicable regulations, Northern States shall keep the following records:

- (1) Reactor operating records, including power levels and periods of operation at each power level;
- (2) Records showing radioactivity discharges into the air or water beyond the effective control of Northern States as measured at or prior to the point of such release or discharge;
- (3) Records of radioactivity levels at both on-site and off-site monitoring stations;
- (4) Records of emergency shutdowns and inadvertent scrams including the reasons therefor;
- (5) Records of safety system component tests and measurements performed pursuant to the Technical Specifications;
- (6) Records of maintenance operations involving substitution or replacement of reactor equipment or components;
- (7) Records of all facility tests and measurements performed.

E. Reports

In addition to reports otherwise required under this license and applicable regulations:

- (1) Northern States shall make an immediate report in writing to the Commission of any indication or occurrence of a possible unsafe condition relating to the operation of the reactor, including, without implied limitation:
 - (a) Any substantial variance disclosed by operation of the reactor from the performance specifications set forth in the Hazards Summary Report;
 - (b) Any accidental release of radioactivity, whether or not resulting in property damage or personal injury or exposure above permissible limits.
- (2) Within 60 days after (a) completion of initial core loading and associated critical testing and (b) completion of Phase II of the Power Operation Test Program Northern States shall submit a written report to the Commission of the results pertinent to safety of the tests and operations conducted,

including a description of changes made in the facility design, performance characteristics and operating procedures.

(3) Within 30 days after the completion of six months of operation of the reactor (calculated from the date of completion of Phase II of the Power Operation Test Program), and at the end of each six-month period thereafter Northern States shall submit a written report to the Commission which summarizes the following:

- (a) Total number of hours of operation and total energy generated by the reactor;
- (b) Number of shutdowns of the reactor with a brief explanation of the cause of each shutdown;
- (c) Operating experience including levels of radioactivity in principal systems; routine releases, discharges, and shipments of radioactive materials; a description of tests performed in the reactor; and the results of any test analyses completed during the period in the reactor including results of tests required by the Technical Specifications; a summary of experiments conducted; number of malfunctions in the control and safety systems with brief explanations of each; and a discussion of data obtained relating to superheater operation;
- (d) Principal maintenance performed and replacements made in the reactor and associated systems including a report on various tests performed on components of the reactor and associated systems;
- (e) A description of the leak tests performed pursuant to the Technical Specifications and the results of such tests including a description of any necessary corrective measures taken to meet the requirements of the Technical Specifications for assuring the specified containment leak tightness;
- (f) Significant changes made in operating procedures and in plant organization;
- (g) Radiation levels recorded at both on-site and off-site monitoring stations.

4. Pursuant to Section 50.60 of 10 CFR 50, the Commission has allocated to Northern States for use in the operation of the reactor 758.4 kilograms of uranium-235 contained in uranium at the isotopic ratios specified in the application. Estimated schedules of special nuclear material transfers to Northern States and returns to the Commission are contained in Appendix "B" attached hereto, which amends the allocation contained in

Construction Permit No. CPPR-8. Transfers by the Commission to Northern States in accordance with column (2) in Appendix "B" will be conditioned upon Northern States' return to the Commission of material substantially in accordance with column (3) of Appendix "B".

5. This license shall be effective as of the date of issuance and shall expire eighteen (18) months from said date, unless extended for good cause shown, or upon the earlier issuance of a superseding operating license.

FOR THE ATOMIC ENERGY COMMISSION

Original signed
by R. Lowenstein,

Director
Division of Licensing and Regulation

Attachments:

1. Appendix "A"
2. Appendix "B"

Date of Issuance: MAR 12 1964

APPENDIX "B"

TO

NORTHERN STATES POWER COMPANY

PROVISIONAL OPERATING LICENSE

Estimated Schedule of Transfers of Special Nuclear
Material from the Commission to Northern States
Power Company (NSP) and from NSP to the Commission

(1)	(2)	(3)		(4)	(5)
Date of Transfer (Fiscal Year)	Transfers from AEC to NSP Kilograms U-235	Returns by NSP to AEC Kilograms U-235		Net Yearly Distribution Kilograms U-235	Cumulative Distribution Kilograms U-235
		Cold	Hot		
Thru 10/18/63	490.0	-	-	-	490.0
1964*	80.6	195.2	-	(114.6)	375.4
1965	243.8	13.1	41.9	188.8	564.2
1966	243.8	53.2	-	190.6	754.8
1967	154.0	15.4	219.3	(80.7)	674.1
1968	187.0	54.2	183.8	(51.0)	623.1
1969	196.4	-	61.1	135.3	758.4
1970	87.2	17.8	188.0	(118.6)	639.8
	<u>1,682.8</u>	<u>348.9</u>	<u>694.1</u>		

*3rd and 4th quarters FY 1964 only

APPENDIX "A"
NORTHERN STATES POWER COMPANY
PATHFINDER ATOMIC POWER PLANT
TECHNICAL SPECIFICATIONS

APPENDED TO PROVISIONAL OPERATING
LICENSE DPR-11, DATED MAR 12 1964

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1.0 INTRODUCTION

1.1 Scope

1.1.1 These Technical Specifications set forth operating limits and requirements and principal design features which affect safety of the Pathfinder Atomic Power Plant.

1.2 Definitions

The following terms are defined to clarify the intent of the various provisions given within these Technical Specifications.

1.2.1 Power Operation - is any operation with the reactor vessel closure bolted in place when reactor criticality is possible. Reactor criticality is to be considered possible if the core is loaded with a quantity of fuel equal to or greater than a critical mass of fuel, power is available to the control rod drive motors and more than one rod is latched to its drive.

1.2.2 Refueling Operation - is any operation with any of the reactor vessel closures open during which either core alterations are being made, or other operations which might directly or indirectly increase the reactivity of the core are in progress.

1.2.3 Shutdown - is any condition not covered by Power Operation or Refueling Operation except that Cold Shutdown shall be specifically defined as indicated in Section 1.2.4 below.

1.2.4 Cold Shutdown - is a reactor condition involving either no fuel in the reactor or a condition meeting the following requirements:

(a) The control rods are fully inserted in the core, and their power circuit is locked by means of the key switch in the off position to prevent withdrawal. The key to the switch must be in the possession of the Shift Supervisor or higher plant management.

(b) The reactor coolant system is at atmospheric pressure.

1.2.5 Scram - is any automatic or manual action which de-energizes the magnetic clutches on the boiler control rod drives and causes run-in of the drives for the boiler and superheater control rods until the associated rod has reached its bottom limit.

1.2.6 Main Steam Isolation Scram - is a reactor shutdown as in 1.2.5 above with closure of the main steam isolation valve and the main steam isolation bypass valve.

1.2.7 Runback - is any automatic or manual action which drives in all boiler control rods until correction of the condition which initiated the action.

1.2.8 Leakage Rate - is defined as the percent of the contained atmosphere (weight basis) which escapes per day (24 hours) under the defined pressure conditions through any leaks in the containment building and its components including any extension of the containment boundary and all isolation valves and their associated piping.

2.0 SITE

2.1 Location

The Pathfinder reactor plant shall be located near the Big Sioux River, approximately 5-1/2 miles northeast of the center of the city of Sioux Falls, and 2-1/2 miles west of the town of Brandon in Minnehaha County, South Dakota.

2.2 Exclusion and Restricted Areas

The distance from the centerline of the reactor building to the boundary of the exclusion area shall be at least 2250 feet. The distance from the centerline of the reactor building to the boundary of the restricted area shall be at least 135 feet. The restricted area which includes the cooling tower, switch yard, and other equipment shall be enclosed by a fence.

2.3 Principal Activities

The principal activity carried on within the exclusion area shall be the operation of the reactor and associated power generating equipment. Other activities which may be carried on at the site shall be controlled by Northern States Power Company and may include maintenance of buildings, roads, grounds, and equipment; transmission and distribution of power; and operation of an Atomic Information Center.

3.0 REACTOR CONTAINMENT

Reactor containment shall consist of an externally insulated cylindrical steel vessel, hereinafter referred to as the containment vessel or enclosure. The insulation shall be approximately one inch thick with a waterproof exterior coating. The integrity of the insulation shall be sufficient to maintain the reactor building metal temperature above 55°F during power operation conditions.

3.1 The reactor containment shall enclose the reactor, the recirculation loops, shielding, and other components arranged as shown in Figures 3.3 and 3.4 of ACNP 5905 dated January 15, 1962. Electrical circuits and hydraulic and pneumatic lines used for the purpose of controlling the reactor or controlling or actuating safety and emergency systems shall be separated from all high pressure piping by substantial structural features so that rupture of a high pressure pipe would not impair the function of the control systems.

3.2 Containment Vessel-Design Parameters shall be as follows:

Internal pressure, psig	78
External pressure, psig	7.3
Temperature, (Coincident with design internal pressure), °F	342
Minimum building temperature, °F	10
Wind load, psf	30
Snow load, psf	35
Earthquake factor, C (UBC code)	0.05
Permissible air leakage rate at 78 psig at ambient temperature, percent per day of contained atmosphere (including penetrations)	0.2
Diameter, feet	50
Height, feet	120.5
Approximate free volume, cubic feet	145,000

3.3 Construction

The principal material of construction shall be SA-212 Grade B, firebox quality steel produced to SA-300 specifications. Design, construction, and testing shall be in accordance with ASME Unfired Pressure Vessel Code, Section VIII, as modified by the applicable nuclear code as of 1960 for the conditions specified in Section 3.2. The parent metal and weld metal shall have an NDT temperature of less than 25°F.

3.4 Penetrations

Penetrations through the reactor building shall conform to the following specifications:

3.4.1 Electrical Cables

Penetrations shall be provided for a maximum of eighty-five electrical cables. About seventy-two of these penetrations shall be used initially and the remainder shall be spares for

future use. The electrical penetrations shall be functionally as shown in Allis-Chalmers Drawing No. A-87219-D dated January 4, 1963.

3.4.2 System Piping

Penetrations shall be provided for a maximum of 19 system piping lines. The piping and instrument line penetrations shall be welded sleeves varying in diameter from 1-1/4 to 36 inches, with the lines rigidly welded to the sleeves except where temperature or pipe movements preclude rigid connections, in which case bellows seals shall be provided. All piping penetrations shall be designed with adequate reinforcement and support to prevent expansion or reaction from a ruptured pipe from causing damage to the penetration or the containment vessel. The stresses resulting from such reactions shall not exceed the allowable stresses for the materials used in the penetration and containment building. Each valve in the inlet and outlet of the ventilation system shall be provided with individual actuators. Valves which serve as isolation valves shall be located with respect to the containment vessel as shown schematically in Figure 2.1 of ACNP 5905 dated January 15, 1962. All such valves which may be open during operations requiring containment integrity shall be designed to permit remote manual closing of the valves from the control room. The following isolation valves shall be designed to be actuated automatically and manually from the control room. The mode of operation of these isolation valves shall be as follows:

<u>Penetration</u>	<u>Mode of Operation</u>
a. Reactor Water Purification	air to open, spring to close
b. Recirculation Pumps Seal Leak-off	air to open, spring to close
c. Shield Pool Cooling Outlet	air to open, spring to close
d. Main Steam Main Steam Isolation Valve	electric to open and close
Main Steam Isolation Bypass Valve	air to open, spring to close
e. Heating Steam	air to open, spring to close
f. Safety Valve Discharge	air to open, spring to close
g. Sump Pump Discharge	air to open, spring to close
h. Safety Valve Drain	air to open, spring to close
i. Ventilation Inlet and Outlet	(see Sec. 3.4.6)
j. Instrument and Service Air	air to open, spring to close

3.4.3 Personnel Access

There shall be a normal airlock (size 7 ft high x 3 ft wide) and an emergency airlock (size 30" diameter) for personnel access and egress. Each airlock shall withstand a 78 psig building pressure. Each airlock door shall open inward toward the reactor building. Airlock doors shall be mechanically interlocked so that only one door can be opened at a time. The emergency airlock shall be provided with a means for closing the inside door from the outside of the reactor building. Shafts or other movable mechanical devices penetrating the airlocks shall pass through packed fittings which provide a seal between the inside and the outside of the reactor building.

3.4.4 Equipment Transfer

The equipment transfer door shall be eleven feet in diameter and secured by a bolted O-ring gasketed joint arranged so that pressure inside the containment building tends to compress the gasket.

3.4.5 Fuel Transfer

The fuel transfer tube shall connect the shield pool to the fuel storage pool. A gate valve shall be used to close the transfer tube.

3.4.6 Isolation Valve Operators

All system piping isolation valves except the main steam isolation bypass valve shall be designed to close faster than the main steam isolation valve which shall close in less than 19 seconds. The main steam isolation bypass valve shall have a controlled closure time of about 2 minutes; backup control devices shall initiate instantaneous closure within 2-1/2 minutes.

The main steam isolation valve shall have an electric powered valve operator. The electrical power shall be available from the emergency power system.

Both of the main steam line valves shall close without operator attention upon Main Steam Isolation Scram signals as described in Section 6.1.4.

There shall be two valves in each ventilation duct. One valve in each duct shall be nitrogen operated in both directions. The nitrogen shall be supplied by two 1000 cubic inch accumulators at a minimum pressure of 400 psig. The second valve in each duct shall be air-to-open and spring-to-close. The ventilation valves shall close in less than 15 seconds. The inlet and outlet ventilation valves shall close automatically upon Reactor Building Isolation signals as described in Section 6.1.5.

3.5 Reactor Building Spray System

The reactor building spray system shall be designed to spray water in the reactor building at a minimum rate of 47 gpm with 78 psig pressure in the reactor building. The system shall be designed with the following features:

a. Number of Spray Nozzles	1
b. Nozzle Pressure, psia	118
c. System Actuation	Automatic, Manual Backup
d. Signal Used to Actuate	High Containment Vessel Pressure
e. Signal Trip Setting	Not more than 6 psi above atmospheric
f. Water Supply	Circulating Water System

The automatic actuation system shall include a time delay device allowing a time delay between actuation signal and opening of the valve of not more than five minutes. Manual controls shall be provided which allow opening and closing of the supply valve at any time.

3.6 Containment Integrity Requirements

3.6.1 Containment integrity provisions shall include the following:

- (a) Maintenance of the reactor building so that the leakage rate shall not exceed 0.2 percent per day of the contained atmosphere at an internal pressure of 78 psig at ambient temperature.
- (b) Sealing of all building access and equipment transfer ports including the equipment door and fuel transfer valve.
- (c) Maintaining all systems for automatically closing containment building penetrations in operating condition.
- (d) Maintaining temperature of the steel shell of the containment above 55°F.
- (e) Preventing all access to and egress from the containment building except through airlocks in which the door interlocks are operable.
- (f) Maintaining all emergency power supplies, monitors, and automatic emergency equipment associated with the containment building and building spray system in operating condition.

3.6.2 Whenever primary system pressure exceeds 250 psig and fuel is in the reactor, whenever fuel is in the reactor and any control rod is withdrawn, or whenever any component is being handled in proximity to irradiated fuel within the reactor building, the containment integrity provisions specified in this section shall be in effect; except that whenever (1) irradiated fuel is being handled with the primary system at atmospheric

pressure and with control rods disconnected from control rod drive mechanisms the gate valve in the fuel transfer tube may be opened and (2) whenever the fuel involved in any type of operation defined by this section shall have been irradiated to less than 100 megawatt days per metric ton of uranium the provisions of subparagraph 3.6.1(d) shall not be required.

3.7 Containment Testing

3.7.1 Preoperational Containment Initial Testing

- (a) The reactor building shall be initially tested by a pneumatic pressure test at 125% of the design pressure (nominal 97-1/2 psig test pressure).
- (b) A leak-detection test shall be conducted at a pressure not less than 5 psig for all reactor building welds, all shell penetrations, gasketed joints, and isolation valves using the soap-bubble method, or the Halide leak detector. All discernible leaks revealed by these test methods shall require repair and retest.
- (c) The initial integrated leakage rate test shall be conducted at 100% of the design pressure.
- (d) The maximum allowable leakage rate of the Reactor Building at design pressure shall not exceed 0.2% of the contained air in twenty-four hours. The actual measured leakage rate derived from the test conducted with air shall be corrected for the containment conditions postulated by the maximum design accident.
- (e) The accuracy of the leakage rate measuring system, in any test, shall be verified by superimposing a controlled leakage rate (measured through a gas flow meter) upon the existing vessel leakage rate, or by other means of equivalent accuracy, and continuing the test a sufficient period of time to measure the composite leakage rate.
- (f) The tests required by Sections 3.7.1(b) and (c) shall be conducted after all construction work affecting containment leak tightness is complete and within one year preceding initial operations requiring containment integrity as defined in Section 3.6.

3.7.2 Containment Periodic Testing (During Operation Below 1 Mw(th))

- (a) All penetrations and gasketed closures shall be subjected to a leak detection test either separately or as a building test, at a pressure not less than 5 psig every year, using the soap-bubble technique or other methods at equivalent sensitivity.

- (b) Equipment transfer door shall be subjected to similar leak detection test following each closure prior to plant startup.
- (c) All airlocks shall be subject to leak detection tests every six months.
- (d) All isolation valves with direct communication to outside atmosphere, including the valves for the ventilation openings and building vacuum relief, shall be subjected to a leak detection test at a pressure of no less than 5 psig every four months using soap bubble technique or other methods of equivalent sensitivity. The normal mode of valve operation shall be employed to close the isolation valves prior to the performance of the leak detection tests.

All automatic controls and instrumentation associated with these isolation valves shall be tested every three months.

- (e) All containment isolation valves and check valves considered as necessary for containment but not included under 3.7.2(d) and the automatic valves of the spray system shall be tested every year or at each major refueling, whichever occurs sooner, to verify the operability of the valves. The automatic controls and its instrumentation associated with these valves shall be tested at approximately quarterly intervals.

Defective operation shall require repair and retests.

All discernible leaks revealed (from tests of (a), (b), (c), and (d)) subsequent to the preoperational tests shall require repair and retests.

- (f) Integrated leakage rate tests of the Reactor Building shall be performed at least once every 18 months.

In order that the integrated leakage rate test be representative of the "as is" condition of containment building, no preliminary preparation of the leak tight condition of the containment building shall be performed which would influence the results of the scheduled integrated leakage rate test. Closure of the isolation valves of the Reactor Building penetrations for the purpose of the tests, shall be effected by the normal means provided for operation of valves.

Leaks detected in the containment boundary, or in the isolation valves directly prior to or during the test which require repairs to enable the integrated leakage rate testing to proceed, may be repaired provided such repairs are reported as part of the record of the leakage rate results.

The integrated leakage rate test may be conducted at 100% of design pressure. The corrected leakage rate shall not exceed the maximum allowable leakage rate specified under 3.7.1(d).

In the event the maximum allowable leakage rate is exceeded as determined at any time by the test results of 3.7.2(e), a retest shall be made following repairs of leaks in the containment building boundary.

A proposed schedule and specification for periodic leakage rate retesting for operation above 1 Mw(th) shall be submitted to Division of Licensing and Regulation for approval prior to operation above 1 Mw(th).

- (g) The time periods specified in this Section 3.7.2 shall be the maximum interval of time elapsed between successive tests. These tests shall be performed initially after all construction work affecting the function to be tested is complete and within three months preceding initial operations requiring containment integrity as defined in Section 3.6; provided that tests also specified under Section 3.7.1 may be performed initially at times required by that section.

4.0 Reactor and Power Systems Equipment

The reactor and power system equipment shall consist of the reactor vessel, the primary coolant recirculating system, the shutdown cooling system, emergency cooling system, pressure relief system, the primary steam and associated power producing equipment and the interconnecting piping and valves. This system shall be arranged as shown in Drawing 43-500-997 in ACNP-5905 dated January 15, 1962. This section will specify the important mechanical design features and operating limits of variables affecting these systems.

4.1 Reactor Vessel

The reactor vessel shall be designed, fabricated, installed, and tested in accordance with Section VIII and IX of the ASME Pressure Vessel Code as of 1955 as modified by Special Code Cases 1270N, 1271N, and 1273N.

4.1.1 Design Features shall be as follows:

Nominal Length, overall, inches	433
Nominal Inside diameter, inches	132
Nominal Wall thickness, excluding cladding, inches	3 (except head = 2 3/8)
Cladding thickness, minimum inches	1/4 (except head = 1/8)
Design pressure, psig	700
Design temperature, °F	500
Approximate Initial nil ductility transition temperature, °F	10

4.1.2 Principal Materials of Construction shall be as follows:

<u>Component</u>	<u>Material</u>	<u>Specification</u>
Vessel shell and head	Steel	ASTM A212 Grade B
Flanges and Nozzles	Steel	ASTM A-105 Gr 11, ASTM A155 Gr KC-70
Cladding	Stainless steel	Types 304L and 309
Head Studs and Nuts	Steel	ASTM A437

4.1.3 Reactor Vessel Penetrations

The penetrations of the reactor vessel and reactor head shall include only those listed below:

<u>Penetrations</u>	<u>Quantity</u>	<u>L.D.(inches)</u>	<u>Location</u>
1. Nozzles to Recirculation Pumps	3	20-1/2	Below Core
2. Nozzles from Recirculation Pumps	3	20-1/2	Below Core
3. Steam Outlet Nozzle	1	16-1/2	Below Suphtr
4. Feedwater Nozzle	1	9-3/4	Below Core
5. Boiler Core Instrument Nozzle	1	6-3/8	Above Core
6. Lower Liquid Level Nozzle	1	2-3/8	Below Core
7. Control Rod Drive Nozzles	20	3-1/2	Vessel Head
8. Superheater Core Instrument Nozzle	1	1-1/2	Vessel Head
9. Upper Liquid Level Nozzle	1	1/2	Vessel Head

4.1.4 Vessel Closure

The top head closure shall be a bolted flange with a 7 foot-7 inch clearance diameter. The seal shall be made by two corrugated, type 316 stainless steel jacketed, soft iron core gaskets. The space between gasket rings shall be connected to a leak monitoring system.

4.1.5 Vessel Support

The reactor vessel shall be supported by five columns fabricated of high strength T-1 steel. The vessel shall be held in a centered position by four keys welded to the vessel wall and engaged in channels fastened to the concrete shield.

4.1.6 Vessel Internals

The vessel internals shall consist of 412 superheater fuel assemblies and 96 boiler fuel elements, boiler fuel boxes, superheater structural assembly, control rods, shroud structure, and structures which support and retain fuel elements.

These vessel components shall be as shown in Figure 1.1 of ACNP-5905, dated January 15, 1962. The core region shall be surrounded by a light water reflector region approximately 32 inches thick.

4.1.7 Control Rod Guidance

The boxes which contain the boiler fuel elements shall form the guide channels for the cruciform control rods. These channels shall be fabricated to provide 0.113 inches clearance between the control rod and the guide channel when the control rod is centered in the guide channel.

The superheater control rods shall move in guide tubes. There shall be 0.065 inches between the guide tube and the control rod when centered.

4.1.8 Reactor Vessel and Steam Line Samples

There shall initially be at least fifteen capsules located in the vessel containing specimens of the reactor vessel material. The capsules shall be located in sample holders which are part of the steam separator assemblies. The samples shall be located at two radial distances from the core. The inner position shall provide an accelerated test of radiation effects.

Three capsules will be removed during the first year of equivalent full power operation for examination. The results of these examinations shall determine the schedule for removal of the other samples.

A test program shall be conducted to evaluate adverse effects of neutron irradiation of main steam line material.

4.1.9 Operating Requirements

The rate of change of temperature in the reactor vessel wall, flanges, and nozzles shall not exceed 200°F per hour as determined by at least 2 thermocouples in the reactor vessel wall. Reactor vessel pressurization in excess of 20% of normal operating pressure shall not be allowed to occur at temperatures below the maximum established nil ductility transition temperature plus 60°F. A determination of the shift in the nil ductility transition temperature shall be made at least once each year.

4.2 Primary Coolant System

4.2.1 Recirculation Loops

The reactor recirculation system shall consist of three loops, each containing a vertical mixed flow centrifugal pump. The pump shaft sealing arrangement shall be designed to prevent loss of recirculation water during all operating conditions. Seal water shall be supplied by either or both of two positive displacement gland seal booster pumps.

4.2.1.1 Piping

The recirculation piping shall be designed, fabricated and inspected in accordance with ASA B 31.1 1955 Code for Pressure Piping and applicable nuclear case rulings. The design pressure and temperature shall be 700 psig and 500°F respectively. The inlet and outlet piping from the reactor to the three recirculation pumps shall be 22 inches O.D. x 5/8 inch thick ASTM-A155 Gr. KC-70 carbon steel internally clad with 1/8 inch thick type 304L stainless steel.

4.2.1.2 Valves

Butterfly valves shall be located at the suction and discharge of each of the recirculation pumps. The valve operators shall be of the electrical motor, gear drive type which fails in position on loss of power. The pump discharge valve operators and controls shall be designed to limit total rate of flow increase to less than 455 gpm per second. (This shall correspond to a maximum reactivity addition rate of less than 5 cents/sec.)

The discharge valves shall be interlocked to prevent closure to below 45 percent open when the associated pump is running.

Control interlocks shall be designed to prevent controlled increase of recirculation flow coincident with control rod withdrawal.

Operation of pump discharge valves may be used to control recirculation flow. Reactivity addition by recirculation flow control shall not be continued for more than 10 seconds in any one 20 second interval.

4.2.1.3 Pumps

The recirculation pumps shall be designed to deliver 21,600 gpm against a 71 foot head. Each pump shall be driven by a 400 HP induction motor. The pump casing shall be made of ASTM A-351-57T grade CF-8 stainless steel.

4.2.2 Biological Shielding

Each recirculation pump and motor shall be in a separate shielded compartment in the reactor building.

4.2.3 Reactor

The volume of water in the reactor at normal operating level shall be about 13,000 gal. The volume of water in the reactor when the reactor is flooded shall be about 20,000 gal.

4.3 Safety Relief Valves:

Number	4
Type	Spring-Loaded, Bellows Sealed
Maximum Setting of First Valve, Psig	600
Maximum Pressure Setting of Remaining Valves, Psig	620

Safety Relief Valves: (Cont'd.)

Design Capacity per Valve, Pounds per Hour	177,500
Nozzle Area, Square Inches	6.38
Rupture Disc Design Pressure, Psi	250

4.4 Reactor Power Operation Cooling:

Coolant Material	Demineralized Water
Type of Cooling System	Forced Recirculation
System Pressurization	Boiling Water
Minimum Loops Operating Concurrently (or Equivalent)	1
Number of Passes Through Core	1
Flow Direction Through Boiler Core	Upward

4.5 Reactor Shutdown Cooling:

Design Pressure, (Standby Cooler)	150
Psig (Shield Pool Cooler)	50
Design Temperature, (Standby Cooler)	350
°F (Shield Pool Cooler)	212
Number Pumps	2
Number Heat Exchangers Available	2
Heat Removal Capacity (Standby Cooler)	19.5 x 10 ⁶
Btu/Hr	
Heat Removal Capacity (Shield Pool Cooler)	1.5 x 10 ⁶
Btu/Hr	

4.6 Reactor Emergency Cooling:

Emergency Condenser:

Type	Shell and Tube
Minimum Capacity, Btu/Hr	17 x 10 ⁶
Minimum Cooling Time Available from Water Storage, Hours	24
Coolant	Shield Pool Water
Design Pressure of Shell, Psig	55
Design Temp of Shell, °F	300
Design Pressure of Tubes, Psig	660
Design Temp of Tubes, °F	900
Actuating Signal	Isolation Scram
Maximum Time to put System in Full Operation Following Signal, Seconds	30 sec

4.7 Operating Requirements

During all reactor Power Operation in excess of 1 MW(th) the minimum recirculation flow shall be at least (1) 297 gpm per MW(th), or (2) flow equivalent to design flow of one recirculation loop.

The maximum operating pressure and temperature shall be 700 psig and 500°F. The controlled rate of change of temperature in the reactor vessel shall be limited to 200°F/hr. All other components in the system shall be capable of following this temperature change rate. The safety relief valves shall be set appropriately for all planned reactor operating pressures so that the allowable pressure (plus 10%) in the reactor system is not exceeded. The emergency condenser shall be operable and ready for service at all times during power operation at levels above 1 Mw(t). The shutdown cooling system shall be operable and ready for service during refueling operations if required for decay heat removal.

The primary coolant shall be sampled and analyzed at least daily during periods of power operation. The primary coolant shall be analyzed whenever the conductivity increases unexpectedly.

The following are absolute limits which if exceeded shall necessitate reactor shutdown. Corrective action shall necessarily be taken at more stringent limits to minimize the possibility of these absolute limits ever being reached.

Conductivity (Micromoho/cm) maximum	5
maximum transient*	10
pH (Lower and Upper Limits)	4.0 and 10.0
Chloride Ion (ppm) maximum	0.1
maximum transient*	1.0
Iodine (microcurie/ml)	20
Boron, ppm, maximum except during experiments below 1 Mw(th)	100

4.8 Biological Shield

The reactor vessel shall be located in a cavity formed by the biological shield of standard stone aggregate concrete approximately 10 feet thick.

There shall be about a one-foot air space between the concrete shield and the reactor vessel. Air shall be circulated in this space. Cooling coils in the concrete shield shall be designed to prevent the shield temperature from exceeding 180°F.

*Conductivity and chloride concentration is expected to increase temporarily after startups from cold shutdown. The time delay before the transients reach their peak will depend on the flow rate. The maximum transient values for conductivity and chlorides here stated are the maximum permissible and apply only to a period not to exceed 24 hours after reaching 20% rated power subsequent to a cold shutdown.

The top of the reactor shall be shielded by water in the shield pool. There shall be enough water in the shield pool to protect personnel working over the pool.

4.9 Power System Equipment and Associated Facilities

4.9.1 Electrical System

(a) Auxiliary Power

The auxiliary power system shall be the normal source of power to the plant under Power Operation, Refueling, and Shutdown conditions.

For continuity of station auxiliary power, the system design shall consist of four independent sources of power; (1) The 13.8 kv generator with voltage stepped down for station service, (2) the 115 kv system available through a reserve station auxiliary transformer, (3) an automatic starting standby 125 kw diesel-generator and (4) an emergency 500 kva manually switched interconnection outdoor with a local 12.5 kv distribution line. The station a-c service system shall be divided into three voltage classes: 2400 volt, 480 volt and 120/208 volt.

The 2400 volt system load shall be divided between two independent busses which may be manually transferred from the 13.8 kv stepdown transformer to the 115 kv stepdown transformer or vice versa.

The 480 volt system load shall also be divided between two busses normally independent and fed from separate 2400/480 volt 1200 KVA, stepdown transformers. One of these transformers shall be fed from one of the 2400 volt busses, the other transformer shall be fed from the other 2400 volt bus. In case of loss of either of these two sources to the 480 volt system, a 480 volt bus tie breaker shall automatically close to feed both 480 volt busses from the remaining source.

The station 208/120 volt system load (consisting primarily of lighting and fractional horsepower motors) shall connect to a single bus normally fed from one of the 2400 volt busses via a 2400 - 120/208 volt 300 KVA, stepdown transformer with provision for automatic transfer to a second stepdown transformer fed from the other 2400 volt bus.

(b) Emergency Power

The diesel generator shall provide emergency 480 volt power for essential auxiliary equipment in the event of a complete 480 volt station auxiliary power failure. The diesel generator shall start automatically after loss of power to both of the 480 volt buses.

The diesel generator loads shall be as follows:

- Boron Injection System
- Standby Lighting
- Turbine Auxiliary Equipment
- Reactor Building Spray Control
- Motor-Generator and associated reactor safety circuits

The station battery shall supply power through the motor-generator to the circuits which are normally supplied by the 480 volt bus. The emergency A.C. and D.C. loads shall include:

- Switchgear
- Radiation Monitoring
- Containment Building Isolation Control
- Nuclear Instrumentation
- Annunciators
- Main Steam Isolation Valve
- Control Rod Controls
- Pertinent Recorders
- Waste Controls
- Reactor Building Transmitters
- Emergency Lights
- Turbine Controls
- Reactor Pressure Control

The station battery shall have sufficient capacity to carry the emergency D.C. load for at least 2 hours.

The diesel generator shall be tested monthly for load carrying capacity and automatic operation.

The motor-generator set shall be tested weekly for automatic operation.

A 500 kva emergency backup source connection shall be available to the 480 volt bus during power operation.

4.9.2 Main Condenser

The main condenser shall have sufficient capacity to condense full reactor steam flow while the turbine is being bypassed. The condenser design features shall be as follows:

Type	Cross Flow Surface Condenser with Deaerating Hot Well
Condensing Surface Area, Square Feet	67,500
Design Condensing Pressure, Inches Hg Absolute	1.5
Btu per Hour at 1.5 Inches Hg Absolute	451,250,000
Air Ejector Capacity	187.5 lbs/hr air and vapor

4.9.3 Turbine By-pass Control System (Dump Valve)

(a) Design Features shall be as follows:

Flow Capacity at 500 psig, Pounds per Hour	700,000
Maximum Time, Full Valve Stroke, Seconds	Approximately 0.5

4.9.4 Condensate and Feedwater System

Two, 1450 gpm, full-capacity condensate pumps of conventional design shall be provided to pump condensate from the condenser through the feedwater heaters and filters to the suction of the reactor feed pumps.

Two full-capacity filters, designed to each pass 1400 gpm, shall be provided for removal of solids (Turbine-condenser system corrosion products).

There shall be two, 1500 gpm, horizontal, centrifugal, motor driven reactor feed pumps. Feedwater shall then pass through a feedwater control valve, high-pressure feedwater heater, and check valve.

4.9.5 Main Steam Piping

The main steam line shall be a 16-inch schedule - 60 pipe fabricated from A-335 Grade P-11 steel. The design pressure and temperature of the main steam line shall be 600 psi and 825°F. All connections to the main steam line shall be consistent with this pressure and temperature rating.

4.9.6 Cooling Water System

The cooling water system shall utilize circulating water to remove heat from the following pieces of equipment:

Diesel Generator
Oil Coolers
Rotary Vacuum Pump
Gland Steam Condenser
Reactor Purification Cooler
Pool Coolers

Two cooling water pumps, each rated at 1500 gpm, shall be provided to circulate water through the above equipment.

4.9.7 Fire Protection System

There shall be hose stations, extinguisher stations, and overhead sprinklers in the plant and hose stations outside the plant. All areas of the plant shall be covered by at least one station. The hose stations and sprinklers shall be connected to a fire protection header.

Water shall be supplied to the header by two cooling water pumps, a diesel fire pump, and a well pump as backup.

There shall be chemical and CO₂ extinguishers in the plant to supplement the water system.

4.9.8 Ventilation System

The induced draft fans shall exhaust air to the stack from the reactor building, turbine building, and the fuel handling building.

The following air exhaust ducts shall contain absolute filters while in operation:

1. Condenser off-gas to I.D. plenum.
2. Condenser vacuum pump discharge to turbine bldg. vent exhaust.
3. Decontamination room exhaust fan discharge to I.D. plenum.
4. Hot chemistry lab exhaust fan discharge to atmosphere.

The administration building air exhausts to the accessible area of the turbine building.

(a) Turbine Building

Turbine building ventilation air shall be delivered to the operating floor, equipment rooms, and shops. Turbine building ventilation air shall be exhausted from the accessible clean area to the shielded area.

The turbine building ventilation air shall be either 100% outside air or a mixture of outside air and recirculated air from the accessible area of the turbine building.

(b) Fuel Handling Building

The fuel handling building ventilation system shall be divided into three sections: (1) The main ventilation section shall deliver air to all sections of the building except the flash tank vault. (2) A separate exhaust fan shall draw the air from the decontamination area as required. (3) The air for the flash tank vault shall be supplied through a duct from the shielded area of the turbine building.

The ventilation air for the fuel handling building shall be 100% outside air with no recirculation.

4.9.9 Instrument and Service Air System

Instrument and service air shall be supplied by two air compressors, each rated at 317 scfm. Instrument air shall also pass through a dryer.

4.9.10 Fuel Handling

The fuel handled or stored as described in this section shall be limited to fuel as described in these specifications.

(a) Fuel Transfer

Irradiated fuel elements and control rods shall normally be transferred between the reactor shield pool and the fuel storage pool through a transfer tube located near the bottom of the pools. New fuel shall be transferred either through the transfer tube or through the personnel airlock.

(b) New Fuel Storage

New fuel shall normally be stored in steel racks in the storage vault. The boiler fuel racks shall be arranged in three double rows with a 23-inch aisle between double rows. A maximum of 32 new boiler elements may be in temporary storage in the reactor building during Refueling Operation.

The maximum calculated flooded K_{eff} for the boiler fuel storage shall be less than 0.80.

The maximum calculated flooded K_{eff} for the superheater storage shall be less than 0.70.

Movement of new fuel elements from the storage area for refueling or inspection shall be under qualified supervision.

The new fuel storage vault shall contain a maximum of 156 boiler elements and 550 superheater elements. There shall be approximately 173 lbs. of UO_2 in each boiler element and approximately 0.33 lbs. of UO_2 in each superheater element.

The boiler fuel element racks shall have 26 spaces for fuel elements separated by 3/8-inch thick steel partitions.

There shall be 5 superheater fuel racks. Each fuel rack shall have 10 tiers. Each tier shall have 11 spaces for superheater fuel. The minimum spacing of superheater fuel shall be 1-5/8 inches center-to-center. The tiers shall be approximately 8 inches apart.

A maximum of 64 superheater elements may be in temporary storage in the reactor building during Refueling Operation.

The door of the new fuel storage vault shall be locked, except when fuel handling activities or inspections require that it be unlocked, and the key shall be under strict supervisory control.

(c) Irradiated Fuel Storage

The fuel storage pool shall be 21 feet deep in the fuel storage area and 29 feet deep in the cask loading area. There shall be 9 double rows or 262 storage spaces provided by steel storage racks securely anchored to the concrete pool. Each space shall be able to contain one boiler element, a basket of up to 16 superheater elements, sources, or other materials. The racks shall be arranged in double rows separated by approximately 7-3/4 inches. The channel between rows shall be too narrow to pass more than one boiler element at a time.

Neighboring fuel elements in the rows shall be separated by 3/8-inch thick stainless steel. The calculated maximum reactivity for a new 3.2% enriched

spike element passing between the racks at the most reactive location shall be 0.82 or less. The calculated reactivity of a basket of 16 superheater elements shall be less than either 2.2% or 3.2% enriched boiler elements.

Temporary storage for 8 boiler elements or superheater baskets shall be available in two fuel transfer boxes in both the fuel storage pool and the shield pool. The calculated maximum K_{eff} of these boxes with new 3.2% elements shall be 0.76 or less.

The fuel storage area shall be monitored for radiation whenever fuel is handled. The monitor over the storage pool shall be set to actuate local and control room alarms.

4.9.11 Turbo-Generator

The turbine shall be a tandem-compound, double-flow unit. The turbine steam seals shall be designed to prevent radioactive leakage to the turbine room. Non-radioactive steam shall be supplied to the seals by the gland steam evaporator or a fossil fueled gland steam generator.

The generator shall be a direct coupled, hydrogen cooled generator rated as follows:

88,235 kva 85% pf 3600 rpm 3 phase 60 cycle 13,800 volts

5.0 Reactor Core and Controls

The reactor core shall be as described in this section. The location and arrangement of such components shall be as shown in Figure 1.1 of ACNP 5905, dated January 15, 1962 and Figure 1 of these specifications. The method of positioning each particular component within the core and the design features of each component shall be as specified in subsections 5.1, 5.2, 5.3, 5.4, and 5.5.

The nuclear characteristics including the reactivity characteristics, control rod drives, and boron injection system are also specified in this section.

5.1 Boiler Control Rods

There shall be 16 boiler control rods of cruciform shape located in the core as shown in Figure 1. Each boiler control rod shall be cruciform in shape as shown in Figure 1.17 of ACNP 5905, dated January 15, 1962. The rod shall be of all welded construction and shall be as described below:

Poison Material in Rods	2.0% Natural Boron in 304 SS
Nominal Active Length, Inches	72
Nominal Width, Inches	10-7/16
Nominal Blade Thickness, Inches	1/4

5.2 Superheater Control Rods

There shall be four superheater control rod support yokes, as shown on Figure 1.19 of ACNP 5905 of January 15, 1962, located in the core as shown in Figure 1. The superheater control rods shall be as arranged in Figure 1.18 of ACNP 5905, dated January 15, 1962, ~~and~~ shall be as described below:

Number of Rods per Assembly	12
Poison Material in Rods	2.0% Natural Boron in 304 SS
Nominal Active Length, Inches	72
Nominal Rod Diameter, Inches	3/4
Cladding	304L SS
Nominal Cladding Thickness, Inches	0.060

5.3 Poison Shims

Poison shims may be used for reactivity adjustment and initial physics requirements. Up to 70 shims may be placed between boiler boxes. These shims shall be prevented from moving sideways or upward by slots in the

top and sides of the boiler boxes and the hold down structure. The poison shims shall be:

Material	0.2% Natural Boron in 304 SS
Nominal Length, Inches	74
Nominal Width, Inches	2-25/32
Nominal Thickness, Inches	0.10

5.4 Core Composition

The data presented in this section consists of design features of the fuel which shall make up the physical composition of the core as arranged in Figure 1.

Total Number of Boiler Assembly Locations	96
Number of 2.2 w/o U-235 Boiler Fuel Assemblies, Max.	96
Number of 3.2 w/o U-235 Boiler Fuel Assemblies, Max.	32
Nominal Total Weight of U-235 in 96 - 2.2 w/o Fuel Assemblies, Kg.	145
Nominal Total Weight of U-235 in 64 - 2.2 w/o Fuel Assemblies and 32 - 3.2 w/o Assemblies, Kg.	167
Max. Number of Superheater Fuel Elements in Core	412
Enrichment of Superheater Elements, w/o U-235	93
Nominal Total Weight of U-235 in 412 Superheater Fuel Elements, Kg.	50

5.4.1 Boiler Fuel Assembly

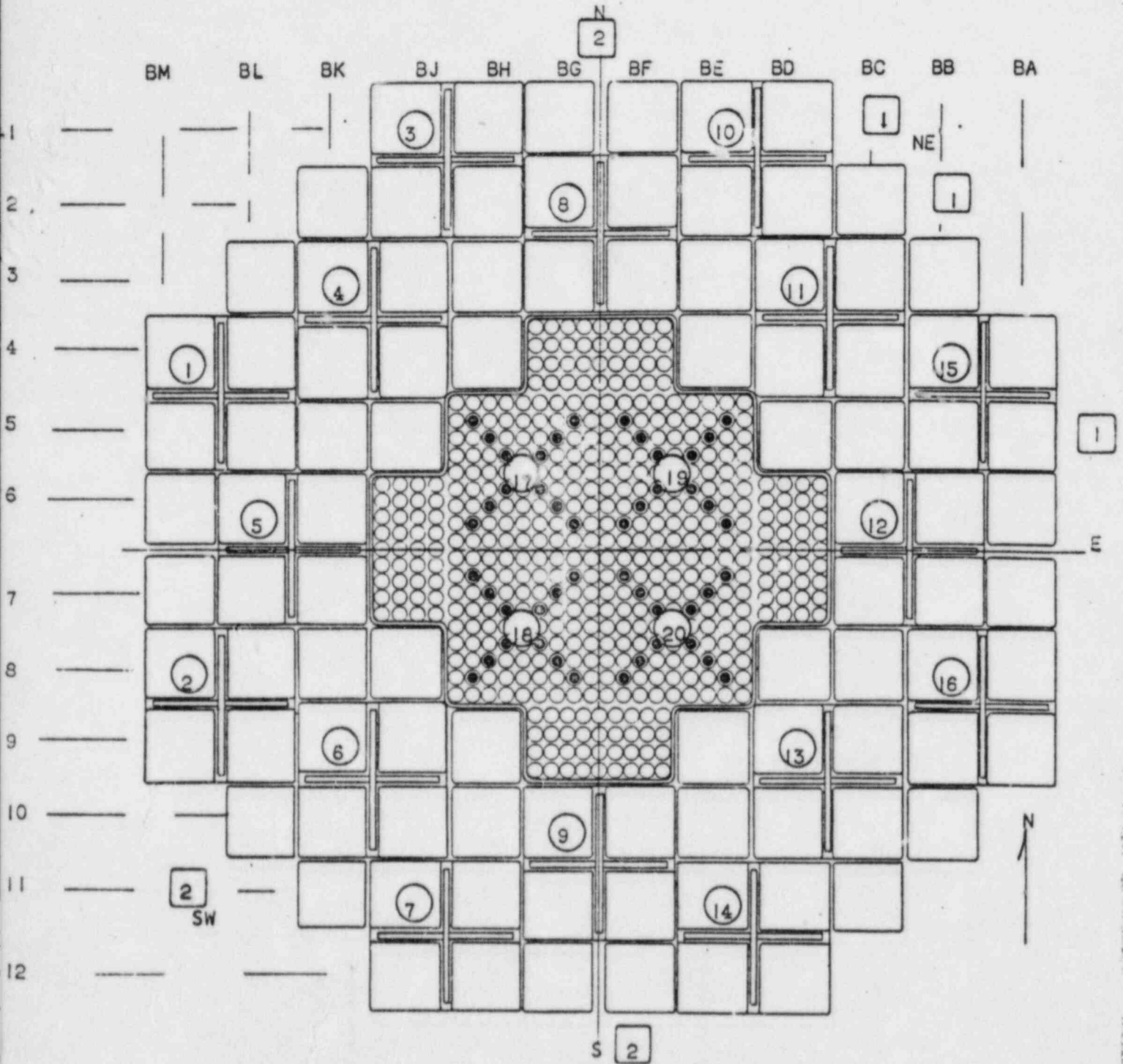
The boiler fuel assembly shall be as in Figure 1.3 in ACNP 5905 dated January 15, 1962, with the fuel rods as specified below:

(a) General

Nominal Square Bundle Outside Dimension, Inches	4.735
Geometry, Fuel Rod Array	9 x 9
Number of Rod Sections per Rod	4
Fuel Rods per Bundle	81
Number of Tube Sheets per Assembly	3

Fig. CORE PLAN AND CHAMBER LOCAT. 45

○ = Control Rod Numbers
 □ = Number of Chamber Locations Available



Three startup channels - Two shall be diametrically opposite
 Two linear channels shall be diametrically opposite
 Two Power channels shall be diametrically opposite
 Diametrically opposite shall be: N-S, NE-SW, E-SW, NE-S, N-SW

(b) Fuel Rod Cladding (Dimensions at Room Temperature)

Material	Zircaloy-II
Clad Thickness of Upper Sections, Inches	0.026
Clad Thickness of Lower Sections, Inches	0.028

(c) Fuel Rod (Dimensions at Room Temperature)

Rod Diameter of Upper Section, Inches	0.367
Rod Diameter, Lower Sections, Inches	0.408
Fuel Pellet Diameter, Upper Sections, Inches	0.310
Fuel Pellet Diameter, Lower Sections, Inches	0.348
Fuel Pellet Density, gm/cc	10.41

5.4.2 Superheater Fuel Elements

The superheater fuel elements shall be as arranged in Figure 1.5 of ACNP 5905, dated January 15, 1962. The fuel tubes shall be as specified below:

Fuel Tubes

Outer O D, Inches	.839
Outer I D, Inches	.769
Inner O D, Inches	.630
Inner I D, Inches	.560
UO ₂ in Cermet, w/o	17.5
Poison Tube O D, Inches	.467
Poison Material, w/o Natural Boron Carbide in Alumina	0.35

Cladding Material 316L SS

Cladding Thickness

Fuel Tubes, Inches	.0075
Poison Tubes, Inches	.028

Up to 72 superheater fuel assemblies without boron in the poison pin may be used for flux shaping.

5.4.3 Boiler Fuel Boxes

The boiler fuel boxes shall be as follows:

Material	Zircaloy-II
Number of Single Boxes	32

Number of Quad Boxes	16
Wall Thickness (Nominal), Inches	.156

5.5 Sources

5.5.1 Initial Start-up Source

During initial fuel loading and low-pressure testing, one 6-curie plutonium-beryllium neutron source shall be used. This source shall be positioned in a superheater process tube. At all times during the use of this source, the maximum thermal flux shall be limited to 5×10^{10} nv at the position of the source.

5.5.2 Operating Source

Type	Antimony Beryllium
Quantity	1
Location	Superheater
Minimum Initial Strength	5×10^9 n/sec.

(a) Physical Description

The beryllium annulus shall be part of a special inner insulating tube of the superheater. The antimony rod shall be located in the center of the beryllium cylinder. The dimensions of the assembly shall be as follows:

Total Length of Insulating Tube, Inches	194 11/32
Insulating Tube O.D., Inches	1
Total Length of Antimony, Inches	76 7/16
Length of Beryllium Annulus, Length	73 3/8

5.6 Principal Calculated Thermal Hydraulic and Nuclear Characteristics

The core shall have the following calculated design parameters. The operating limitations upon the core are given in Section 5.7.

5.6.1 Principal Calculated Thermal and Hydraulic Characteristics of the Core Loading at Full Power

(a) Core Power at Rated Steam Flow, MWT	190
(b) Boiler Peaking Factors (To be applied to Heat Flux)	
Radial	2.00
Axial	1.35
Fuel Loading	1.03
Fuel Eccentricity	1.25
Combined	<u>3.47</u>

(c) Heat Flux and Fuel Center Temperature at 190 MW(th) Maximum Steady-State Power	
Max. Boiler Heat Flux, BTU/hr ft ²	447,000
Max. Superheater Heat Flux, BTU/hr ft ²	219,000
Boiler Cladding Surface Temperature, °F	514
Max. Superheater Cladding Temperature, °F	1270
Max. Linear Heat Generation Rate of Boiler Fuel, kw/ft	14.0
Max. Superheater Fuel Temperature, °F	1280
(d) Burnout Ratio,* Minimum	1.9
(e) Max. Fuel Cladding Stress, Psi (55% of Yield Strength)	12,790
(f) Average Core Power Density, Kw/ft ³	1280
(g) Total Recirculating Flow Rate Max., gpm	65,000
(h) Boiler Flow Rate, Percent of Total Recirculating Flow Rate	89.7
(i) Superheater Flow Rate, Percent of Total Recirculation Flow Rate	6.0
(j) Core Inlet Conditions	
Inlet Velocity:	
Maximum, Ft/Sec.	14.2
Minimum, Ft/Sec.	12.5
Inlet Subcooling, Btu/Lb	3.8
(k) Reactor Boiler Core Pressure Drop at 60,000 gpm Flow, Psi	13.6
(l) Boiler Exit Bulk Temperature at 190 MW(th), °F	489
(m) Superheater Exit Bulk Temperature** °F	725

* Calculated at 600 psig by using:

$$\frac{\phi_{BO}}{10^6} = 0.0536 \left[\frac{H}{10^3} \right]^{-4.12}$$

This correlation, with the exponential derived from laboratory data has been biased by a factor of .75 to cover all data points.

** The superheater exit bulk temperature shall increase to a maximum at a core life of approximately two months and then decrease. The maximum shall not be greater than 750°F.

(n) Steam Volume Fraction	
Average Core Exit Void Fraction	.43
Maximum Exit Void Fraction	.56
Average Void Fraction Over Core Length	.31
(o) Minimum Core Inlet Pressure, psig	586.4

5.6.2 Principal Calculated Nuclear Characteristics of the Core

The calculated physics parameters shall conform to the values tabulated below:

(a) Minimum Negative Temperature Coefficients ($\Delta k_{eff}/^{\circ}F$)

<u>Temperature</u>	<u>Coefficient</u>
68 ^o F	-3.0×10^{-5}
68 ^o F at end of life	-1.2×10^{-5}
450 ^o F	-8.1×10^{-5}
450 ^o F at end of life	-4.0×10^{-5}

(b) Minimum Negative Void Coefficient ($\Delta k_{eff}/1\%$ exit voids)

<u>Reactor Power</u>	<u>Delta k/1% v</u>
Hot, zero	$-.3 \times 10^{-3}$
50% power	-1.0×10^{-3}
100% power	-1.6×10^{-3}

(c) Minimum Negative UO₂ (Doppler) Coefficient ($\Delta k_{eff}/^{\circ}F$)

<u>Reactor Power</u>	<u>Delta k/^oF</u>
Cold, zero voids	-1.22×10^{-5}
Hot, zero voids	-1.15×10^{-5}
50% Power	-0.97×10^{-5}
100% Power	-0.92×10^{-5}

(d) Minimum Pressure Coefficient ($\Delta k_{eff}/psi$)

<u>Pressure</u>	<u>At 20% Power</u>	<u>At 100% Power</u>
400 psi	$+3.3 \times 10^{-5}$	$+7.5 \times 10^{-5}$
600 psi	$+1.5 \times 10^{-5}$	$+4.6 \times 10^{-5}$
800 psi	$+0.98 \times 10^{-5}$	$+3.3 \times 10^{-5}$

- (e) Calculated Core Reactivity
- | | |
|--|------------|
| Cold, clean, SH flooded | K = 1.1232 |
| Full power, equilibrium Xe and Sm,
43% exit voids | K = 1.0184 |
| Temperature defect, delta k/k
(including doppler) | - .0245 |
| Superheater draining, delta k/k | + .0051 |
| Voids (43% exit voids), delta k/k | - .0329 |
| Xenon, delta k/k | - .0265 |
| Samarium, delta k/k | - .0102 |
- (f) The maximum reactivity addition rate shall be 5 cents/sec.
- (g) Worth of Liquid Poison -The reactivity worth of the liquid poison system shall be:
- | | |
|--|--------|
| Reactor-vessel open to shield pool,
delta k _{eff} | - 0.07 |
| Normal water level in reactor,
delta k _{eff} | - 0.30 |
| Time to inject 1000 gal. min. | 18 |
| Max. time to inject enough solution to
obtain 4% negative reactivity, minutes | 5 |
- (h) Fuel Burnup
- Average Mwd/tonne of contained U for the first core 7800
- Maximum Mwd/tonne of contained U 22,000

5.7 Principal Core Operating Limitations

5.7.1 Reactor Power Level

(a) Refueling

Partial core tests may be run, but the reactor power shall be limited to 1.0 Mwt, exclusive of core decay heat. When using the plutonium-beryllium source, the power shall be limited as described in 5.5.1

(b) Reactor Operation

The reactor shall be operated within the following limits:

Minimum Core Burnout Ratio,* steady state	1.9
Transient Minimum Burnout Ratio** for maximum heat flux conditions	1.5
Maximum Steady State Heat Flux BTU/hr ft ²	447,000
Maximum Steady State Fuel Clad Stress, Percent of Stress Yield	55
Superheater Power Maximum Steady State Power Fraction***	0.17
Boiler Power Maximum Steady State Power Fraction of 190 Mwt	0.86
Maximum Steady State Power Level, Mw(th)	190
Maximum Steady State Value of Core Power Density, Total Core Power Divided by Total Core Volume, kw/ft ³	1280
Minimum Reactor Pressure at Rated Power, Psig	500
Maximum Steam Temperature, °F	750
Maximum Reactivity (delta k/k) in Steam Voids	.035
Maximum Reactor Pressure During Power Operation, Psig	660
Minimum Recirculation Flow Rate,**** gpm/MW(th) above 1 MW(th)	297
Maximum MWD/tonne of Contained Uranium for an Individual Fuel Rod	22,000

* Burnout ratio is refined as the ratio of burnout heat flux to actual heat flux at a point in the core and shall relate to the burnout correlation stated in Sec. 5.6.1.

** Evaluated at 115% of rated power and compensated for total measurement error.

*** The superheater power fraction shall increase to a maximum at a core life of approximately two months and then decrease. The maximum shall not be greater than .17.

**** And as specified in Section 4.7.

The control rod withdrawal rate during power operation shall be such that the average rate of reactor power increase is less than 50 MW(th)/minute when power is less than 120 MW(th) and less than 20 MW(th)/minute when power is between 120 and 190 MW(th).

5.8 Control Rod Drives

Twenty control rod drives shall be mounted on the vessel head. The drives shall be of the rack and pinion type, driven by an induction gear-motor. The reactor shall be allowed to continue to operate if not more than one control rod drive is inoperative because of failure, the cause of failure ascertained, and the cause of failure is not apt to progress to other drives, and the shutdown margin stated in 5.9.2 can be met with the remaining drives.

There shall be sixteen cruciform control rods in the boiler region. There shall be four rod-type control assemblies in the superheater region.

Each control rod shall be driven by the gear-motor through a magnetic clutch and a cam clutch. The boiler control rod drive magnetic clutches shall release and allow the boiler rods to fall into the core on a reactor scram signal. A scram signal shall also initiate "run back" on all control rod drive motors.

The control rod position shall normally be transmitted to the control room by a Selsyn-type electrical servo transmitter. The position indication system shall have a change of position accuracy of $\pm 1/2$ inch.

The boiler rods shall be accelerated during a scram by a compressed spring when the drive rack is in the "full out" position. The drive rack and attached control rod shall be decelerated during the last six inches of the downward stroke by a hydraulic dashpot. The magnetic clutches for the superheater control rod drives shall be powered by a separate 24 volt DC supply and shall not de-energize upon a scram signal.

Control rod drive features shall be as follows:

<u>Type</u>	<u>Rack and Pinion</u>
Normal Stroke Length, Inches	73.0
Maximum Withdrawal Velocity In/Sec.	
Outer Boiler and Superheater	1.2
Inner Boiler	.4
Minimum Insertion Velocity In/Sec.	
Outer Boiler and Superheater	1.0
Inner Boiler	0.3

*Maximum Scram Insertion Time,
Sec., 90% of Stroke

2.0

5.9 Control Rod System

5.9.1 Control Rod Performance

The following tests shall be performed at each major refueling shutdown.

- (a) Continuous withdrawal and insertion of each drive over its stroke to verify velocities stated in section 5.8.
- (b) Withdrawal of each rod to check the functioning of the position indication system.
- (c) Scram of each drive from the fully withdrawn position. Maximum scram time from rod release to 90 percent of insertion shall not exceed 2.0 seconds.

Each drive shall be moved to determine by motor current that drive friction is normal at each major refueling but not less frequently than quarterly during the period of the provisional operating license.

5.9.2 Core Shutdown Margin Verification

The reactivity of the core loading shall be such that it is always possible to maintain k_{eff} at less than 0.997 with the most valuable reactivity-worth control rod completely withdrawn from the core in any operating condition.

The core shutdown margin shall be verified by a demonstration that the reactor is subcritical with the most valuable reactivity-worth rod fully withdrawn, the superheater drained, and a rod or a rod group withdrawn to a position known to contribute 0.003 k_{eff} or more to the effective multiplication.

This verification shall be performed prior to startup after any shutdown in which the system has cooled sufficiently to be opened to atmospheric pressure and any of the following events have occurred since the previous verification:

- (a) Fuel has been added or repositioned in a way which is not definitely known to reduce reactivity; or
- (b) A control rod has been changed and presence of poison has not been verified; or

* The time interval after scram signal causes the boiler control rod clutches to release until the rod has traveled the specified distance of its full stroke length.

- (c) Poison shims have been relocated or removed from the core; or
- (d) 31,000 MWD(th) have been generated by the plant since the previous margin demonstration.

During power operation, if core reactivity increases for no explainable reason, the reactor shall be brought to the cold shutdown condition.

5.9.3 Control Rod Drive Temperature

The rod drive rack housings shall be filled with water and submerged in the reactor shield pool. The housing and condensate temperature shall be approximately the temperature of the shield pool water, however the shield pool need not be flooded during reactor operation if the reactor coolant is below 200°F.

5.9.4 Control Rod Latching Checks

The control rod latch shall be designed to prevent the removal of the latching tool unless the rod is completely latched or unlatched to the drive.

The operation of the latch shall be verified by observing the drive motor current difference while moving the drive before and after the rod is attached.

The operation of the latch shall also be verified by observation of the nuclear instrumentation response to rod withdrawal.

5.9.5 Control Rod Exercising During Sustained Power Operation

Each control rod, which is either partially or completely withdrawn, shall be exercised at least once per two week period.

5.9.6 Abnormal Behavior of the Control Rod System

An immediate and thorough investigation shall be made of the occurrence of any abnormal behavior of the control system to determine the cause and safety significance of the occurrence. The reactor shall be shut down unless:

- (a) It is determined by the investigation that any malfunction which has occurred neither impairs the ability to control the reactor nor indicates the imminent impairment of the performance of additional components of the control rod system.

- (b) The core shutdown margin requirement (described in 5.9.2 above) can be met with the remaining operable control rods. Evaluation of this requirement shall be based on previous experimental measurements.

5.9.7 Minimum Accuracy of Rod Position Indicating System

The position indicating system is of a selsyn type and indicates the drive position over full range of the drive with a change of position accuracy of $\pm 1/2$ inch. If the position indicator for a particular control rod malfunctions, the rod shall not be moved except for scram or complete insertion by runback until the indication has been restored.

5.9.8 Operating Requirements

The maximum-burnup boiler and superheater control rod shall be removed from the core at one-half design burnup, 0.5 and 0.25 a/o burnup respectively and examined for any detrimental effects.

The first inspection shall be made not more than one year after reaching full power. Subsequent inspections shall be made of at least one rod at each major refueling.

The surface of the control rods shall be examined. Particular care shall be taken in the area of maximum burnup and in weld areas. Any discontinuity shall be examined at higher magnification. The superheater rod shall be checked for excessive swelling.

Any rod showing indications of surface cracks or other defects which are judged to adversely affect reactor operation shall be replaced in the core by a new rod. Further, all rods of the type showing the defect shall be removed and examined in the prescribed manner.

5.10 Boron Injection System Design

Material	$\text{Na}_2\text{B}_8\text{O}_{13}\cdot 16\text{H}_2\text{O}$
Available Quantity of Solution, Gal.	1000
Total Weight of Boron, Pounds	230
Maximum Total Injection Time, Minutes	20
System Actuation:	Remote Manual

A 1000 gallon tank containing a boron solution shall be located in the reactor building and shall be operable during reactor operation. The solution shall provide a minimum of 7 percent negative reactivity when

the reactor is open to the shield pool. The seal pumps shall be used to inject the solution into the reactor. The boron injection shall be manually controlled from the main control room. The system shall serve as back-up for the normal control system. When the reactor is in operation, room temperature shall maintain the boron solution above 58°F.

5.11 Boron Injection System

The boron injection system shall be available for operation at all times during Refueling Operation and Power Operation. The reactor shall be shut down in any situation where the poison solution tank level drops below an equivalent 1000 gallons of .23 lbs. boron/gal. solution. The minimum temperature of the solution shall be 58°F. The minimum worth of the liquid system (based on normal reactor water level) shall be 30% delta k_{eff} . The liquid poison system shall be used at any time when subcriticality cannot be assured by the normal shutdown mechanism. Injection shall be continued until a minimum shutdown margin of 0.01 delta k_{eff}/k_{eff} is assured in the most reactive core. The maximum time to inject enough boron to obtain 4% negative reactivity shall be five minutes. The system components shall be checked for operability at least once every two months of Power Operation. The reactor shall not be operated after poison has been injected until the boron concentration in the reactor water has been reduced to 100 ppm or less.

5.12 Reactivity Coefficients

The reactivity coefficients shall meet the following requirements:

- 5.12.1 The effect upon reactivity of increasing voids at constant pressure shall always be negative.
- 5.12.2 The moderator temperature coefficient (inferred from critical control rod position) shall always be negative.
- 5.12.3 The overall effect of increasing reactor power at constant pressure shall be a loss of reactivity whenever the reactor is operating so as to produce a net steam flow.

5.13 Reactivity Additions During Core Alterations

The limits and requirements which apply to reactivity additions are as follows:

- 5.13.1 Any refueling operation which may increase reactivity shall utilize the procedure outlined in 5.9.2 before and after the alteration to verify the core shutdown margin of 0.3% delta k_{eff}/k_{eff} with the superheater voided and the

most valuable rod completely withdrawn. All rods shall be fully inserted during fuel additions. Checks shall be made at frequent intervals during core alterations to assure that the core shutdown margin requirement is being met:

- 5.13.2 Core alterations which increase reactivity shall be limited between subcriticality checks to fuel loading increments which do not exceed one-half the reactivity addition for criticality or the placement of one boiler fuel bundle. At no time will the shutdown margin be knowingly allowed to be less than 0.3% $\Delta k_{eff}/k_{eff}$ with the control rod of highest reactivity worth fully withdrawn from the core and the superheater voided.

5.14 Reactivity Additions During Power Operation

Routine control rod withdrawal sequences shall be established for use during normal Power Operation. These shall be in a sequence of steps involving withdrawal of only one control rod at a time.

The maximum reactivity insertion rate when the k_{eff} of the core is greater than .997 shall be 5 cents/sec and this insertion rate shall continue for no more than 10 sec in any 20 second interval.

6.0 PLANT SAFETY AND MONITORING SYSTEMS

This section specifies the general arrangement, design features, and operating requirements of the plant safety and monitoring systems. These systems shall include the nuclear instrumentation, process instrumentation which initiates safety system actions, reactor safety system, control rod withdrawal permissive system, and the plant radiation monitoring systems. The term "rated power" as used in this section shall mean a reactor power level not exceeding the highest level of which all the operating limitations of Section 5.7 are met.

No bypass of the automatic safety functions, including interlocks, shall be provided except as specified in this section. Except as explicitly provided in this section, no such function shall be bypassed in the course of any operation for which the service of that function is required by these Technical Specifications.

6.1 Reactor Safety System

6.1.1 General Features

The reactor safety system shall consist of sensing devices to monitor reactor-associated parameters of plant operation and related circuitry designed to initiate appropriate safety action when operating parameters exceed applicable setpoint-boundary conditions. The related circuitry shall be designed to initiate runback, scram, main steam isolation scram, closure of containment penetration valves, operation of the emergency condenser, or interlock action to prevent control rod withdrawal as appropriate safety actions.

Control switches shall be located in the control room to permit manual initiation of scram, main steam isolation scram, closure of containment penetration isolation valves, and operation of the emergency condenser. The design shall also include control room provisions to manually control operation of the reactor building spray system, emergency condensate supply valves, and the boron injection system.

6.1.2 Scram Control System

This section specifies the general arrangement and design features of the scram control system. Principal components of this system shall be two input logic circuits and two silicon-controlled-rectifier type scram-clutch power supplies. The input logic circuits which control the excitation to each supply shall receive input signals from the flux-level trip circuits of nuclear channels 5, 6, 7 and 8; from 2/3 logic circuit which receives short period trip signals from channels 1, 2 and 3; and from process monitoring instrumentation.

This system shall be designed to cause reactor scram upon receipt of high flux level trip signals from any two of channels, 5, 6, 7 and 8; short period trip signals from any

two of channels 1, 2, and 3; or a trip signal from any one process monitoring instrument. The system shall be so designed that when reactor power is below a preset low level, power channel 7 shall insert one of two coincident signals required to cause scram.

The scram control system shall also have a backup system which enables operation of either input logic circuit to cause reactor scram by interrupting a 120 volt ac power to both clutch power supplies. The backup system shall include four backup circuits. Each backup circuit shall consist of a relay driver which controls a relay capable of interrupting the 120 volt ac power to both clutch power supplies. Each input logic circuit shall control two of the described backup circuits. The backup system design shall include provision for testing without causing reactor scram. Such test circuitry shall permit momentary bypass of no more than two backup relays at one time and shall include design features of spring action to prevent sustained bypass by operator error.

6.1.2.1 System Response Times

The maximum response time from generation of signal by detectors of channels 5, 6, 7 or 8 until rod motion is detected by operation of the limit switch shall be 300 milliseconds. The maximum response time of the scram control system shall be 30 milliseconds.

6.1.3 Nuclear Instrumentation

The nuclear instrumentation consists of eight channels. The instrumentation shall monitor reactor power from source level to 150% rated power. The channels and associated safety circuits shall be as specified in Table 1.

6.1.3.1 Channels 1, 2, and 3

Channels 1, 2, and 3 shall provide logarithmic neutron flux level and period information for the reactor safety system and indication from source level to a level covered by operating channels 5 and 6. The principal components of each channel shall be a neutron detector, high voltage supply, pulse amplifiers, log count rate meter and period meter. The detectors shall be BF₃ gas-filled proportional counters with minimum rated sensitivity of 12 cps/nv thermal. Each channel shall have two trip circuits for short period and one for low count rate output signals.

6.1.3.2 Channel 4

Channel 4 shall provide logarithmic neutron flux level and period information for the reactor safety system and indication from approximately 10⁻³% to 150% rated power. The principal components of the

Table 1 NEUTRONIC INSTRUMENTATION SHALL AS FOLLOWS:

Channel Type	Number of Channels	Range	Safety Circuits
Start-up BF ₃	3	Source to 10 ⁵ cps	<p><u>Rod withdraw interlocks -</u> Permit rod withdraw at ≥ 2 cps, 2/3</p> <p><u>Scram and Alarm -</u> Minimum period not less than 4 sec, 2/3 coincident logic No coincident logic req'd for alarm</p>
Intermediate CIC Log - N	1	.001% to 150% rated power	<p><u>Rod withdraw interlock -</u> Permit rod withdraw at $\geq .001\%$ of rated power (key switch bypass available)*</p> <p><u>Runback and Alarm -</u> Minimum period not less than 5 sec (key switch bypass available)*</p>
Intermediate CIC LINEAR FLUX	2	.001% to 150% rated power	<p><u>Runback and Alarm -</u> Maximum indicated level not greater than 110% meter indication on any selected range Runback on 2/4 logic with ch 5-6-7-8</p> <p><u>Rod Withdraw Interlock -</u> If either channel is "Low Out of Range," set point not less than 5% of selected range, except on selected ranges at or less than 10⁻³% rated power range.</p> <p><u>Scram and Alarm -</u> Max indicated level not greater than 115% meter indication on any selected range. Scram on 2/4 logic with ch 5-6-7-8</p>
Power IC	2	1% - 150% rated power	<p><u>Scram logic switch -</u> Logic change from 1/3 to 2/4 coincidence (ch 5-6-7-8) when ch 7 $> 10\%$ rated power</p> <p><u>Runback and Alarm -</u> Level not more than 110% rated power Runback logic 2/4 coincidence ch 5-6-7-8</p> <p><u>Scram and Alarm -</u> Level not more than 115% rated power. Scram logic 2/4 coincidence ch 5-6-7-8</p>

* See sec. 6.2.1 for operating requirements on bypass key switches

channel shall be a neutron detector, detector voltage power supply, log-N meter, period meter and log-N recorder. The detector shall have a gamma-compensated ion chamber with sensitivity rated at 4×10^{-14} amperes per nv. This channel shall have two trip circuits for short period and one for low level output signals.

6.1.3.3 Channels 5 and 6

Channels 5 and 6 shall provide linear flux level and selected range information for the reactor safety system and indication from approximately 10⁻³% to 150% rated power. The principal components of each channel shall be a neutron detector, detector voltage power supply and picoammeter with console-mounted range switch. The detectors shall be gamma-compensated ion chambers with sensitivity rated at 4×10^{-14} amperes per nv. A recorder shall record range switch position and linear power indication of the selected channel. Both channels shall sound an alarm when the indication is low out-of-range. The low out-of-range alarm may be bypassed during startup by switching the channel to the lowest useful range. Each channel shall have one trip circuit for low level and three for high level output signals.

6.1.3.4 Channels 7 and 8

Channels 7 and 8 shall provide linear neutron flux level information for the reactor safety system and indication from approximately 1% to 150% rated power. The principal components of each channel shall be a neutron detector, detector power supply, and linear power meter. The detectors shall be uncompensated ion chambers with sensitivity rated at not less than 2.6×10^{-14} amperes per nv. A comparator shall sound an alarm when the difference between channel 7 and 8 power level indication exceeds the comparator set point of not less than 10% of full scale. Each channel shall have two trip circuits for high level output signals and channel 7 shall have an additional trip circuit for low level output signal.

6.1.3.5 General Service Recorder and Scaler

A general service recorder with selector switch shall be provided to record log count rate or period from channels 1, 2, or 3; period from channel 4; or power level from channels 7 or 8.

A scaler with selector switch shall be provided to measure count-rate from discriminator pulse-output of channels 1, 2, or 3.

6.1.4 Main Steam Isolation Scram

Main steam isolation scram conditions shall initiate scram through operation of the scram control system. By use of related circuitry these same conditions shall also initiate the following actions:

- (1) Close the main steam line isolation valves
- (2) Sound reactor building evacuation alarm
- (3) Operate emergency condenser
- (4) Close reactor safety valve discharge isolation valve (With a delay of up to one minute).

A main steam isolation scram may be initiated manually from the control room and shall be initiated automatically if any of the following conditions exist when the main steam isolation valve is open.

<u>Condition</u>	<u>Set Point</u>
Loss of circulating water pumps	Breaker trip or loss of bus power
Turbine building ventilation exhaust radiation high	not more than 10 times normal background or 10 mr/hr, whichever is larger
Air ejector exhaust radiation high	not more than 10 times normal full power background
Main steam line radiation high	not more than 10 times normal full power background
Low reactor water level	not less than -3 feet*
Reactor building pressure	5 ± 1 psig

6.1.5 Reactor Building Isolation

Reactor building pressure of 5±1 psig or higher shall isolate the reactor building by closing the following isolation valves:

1. Purification System Isolation Valves
2. Shield Pool Cooling System Isolation Valve
3. Recirculation Pump Seal Water Return Isolation Valve
4. Sump Isolation Valve
5. Heating System Condensate Return Isolation Valve
6. Ventilation Cooling Coil Water Return Isolation Valve

* Note: Reference water level zero is 4 feet above fuel

7. Ventilation Inlet and Outlet Isolation Valves
8. Safety Valve Isolation Valve
9. Main Steam Isolation Valves

These last three valves shall also close on radioactivity levels as discussed in Table 6 of Section 6.4 and in Section 6.1.4.

6.1.6 Reactor Scram Mode Selection

The reactor safety system shall include a "flood-operate" mode switch which permits the selection of two modes of operating conditions of the system. The following conditions shall automatically initiate reactor scram and shall not be bypassed by the "flood-operate" mode switch.

<u>Condition</u>	<u>Set Point</u>
Reactor Pressure Low	not less than 500 psig (Manual startup bypass cleared automatically when pressure rises to normal.)
Reactor Pressure High	not more than 700 psig
Reactor water level low	not less than - 2 feet*
Reactor water level low (backup)	not less than - 3 feet*
Reactor building pressure high	not more than 5 ± 1 psig
Reactor safety valves open	
Reactor building ventilation exhaust radiation high	not more than 10 times normal full power background or 10 mr/hr, whichever is larger
Nuclear instrument scrams	(see Table 1)
Control power key switch	Off
40 lb instrument air header pressure low	not less than 25 psig

* Note: Reference water level zero is 4 feet above fuel.

When the mode switch is in the "flood" position, the reactor shall scram when nuclear channel 5 or 6 is switched to a range for which the full scale indication is greater than 4% rated power. The following conditions shall automatically scram the reactor when the mode switch is in the "operate" position and may be bypassed by the mode switch in the "flood" position.

<u>Condition</u>	<u>Set Point</u>
Turbine stop valves tripped	
Main steam isolation valve closing (bypassed for exercising)	
Turbine inlet valves and dump valve both closed	
Improper power to flow ratio	normalized ratio not greater than 1.15
Water in main steam line	float switch in drain line
Reactor water level high	not more than + 5 ft*

6.1.7 Runback

The following conditions shall automatically initiate runback with the mode switch in "flood" or "operate" position.

<u>Condition</u>	<u>Set Point</u>
Loss of voltage on both 480 v bus sections	(loss of power to bus)
Loss of all three recirculation pumps	breaker operation
Reactor water level low	not less than - 15 inches*
Fuel transfer valve open	
Nuclear instruments	See Table 1

* Note: Reference water level zero is 4 feet above fuel.

Reactor water level less than 11 feet shall cause runback when the mode switch is in the "flood" position. The following conditions shall initiate runback when the mode switch is in the "operate" position and may be bypassed by the mode switch in the "flood" position.

<u>Condition</u>	<u>Set Point</u>
Low feedwater temperature	not less than 275°F
Low feedwater pressure	not less than 600 psig after a 15 sec delay
Reactor water level high	15 inches or less above normal set point
Dump valve hydraulic oil pressure low	not less than 1800 psig

6.1.8 Control Withdrawal Permissive System Interlocks

- (a) Interlocks shall prevent control rod withdrawal when two or three startup channels (1, 2, and 3) read less than 2 counts per second. This interlock may be automatically bypassed when Log-N channel (4) reads greater than 0.001% rated power or manually bypassed by key-locked switch.*
- (b) Interlocks shall prevent control rod withdrawal when either operating channel (5 or 6) indicates less than 5% on any range, and may be bypassed during startup by switching both channels down to approximately 10^{-3} % rated power or lower ranges.
- (c) Interlocks shall also prevent withdrawal of inner boiler rods when outer boiler rods are not fully withdrawn. This interlock may be automatically bypassed when reactor water temperature is at least 300°F and may be manually bypassed by key-locked switch* for precritical rod operation checks, and specially supervised tests.

6.1.9 Control Room

- 6.1.9.1 Control of the reactor and most of the other plant systems and equipment shall be centralized in the control room located in the administration building.

* See Section 6.2.1 for operating requirements on bypass key switches

- 6.1.9.2 The control room shall be designed and shielded to permit continuous occupation up to two hours following an accident releasing 100% of noble gases, 50% of the halogens, and 1.0% of the solids in the end-of-core-life maximum fission products.
- 6.1.9.3 Sufficient protective clothing, supplied air masks, and portable radiation monitoring equipment shall be stored in the control room and available for use in emergency.
- 6.1.9.4 Emergency lighting and communication equipment shall be provided for the control room. Two-way communication between the control room and in-plant communications posts and between the control room and off-site locations shall be possible.
- 6.1.9.5 In addition to those specifically described in other sections of this specification, instruments shall be located in the control room to indicate reactor operating parameters of main steam flow, temperature and pressure; feedwater flow, temperature and pressure; purification flow; reactor water level; reactor vessel saturated steam pressure; instrument supply air pressure; turbine main-condenser vacuum and condensate conductivity; dump valve and turbine inlet valve positions; and main generator electrical load.
- 6.1.9.6 An annunciator system shall be located in the control room to indicate off-normal plant conditions. Electrical system, fire alarms, pumps, turbine system, ventilation system, and nuclear steam supply system annunciators shall be provided. The nuclear system annunciator points shall include those listed in Table 2.

6.2 Reactor Safety System Operating Requirements

6.2.1 Bypasses

The specifications relating to the design of reactor safety system bypasses and the conditions under which system functions may be bypassed are contained in this section.

TABLE 2

NUCLEAR SYSTEM ANNUNCIATOR POINTS

Superheater outlet temperature - High
Reactor feedwater temperature - Low
Superheater outlet pressure - High
Reactor pressure - High
Turbine 102% overspeed - Tripped
Turbine trip-stop valves - Tripped
Reactor Control - Runback
Reactor Control - Scram
Nuclear instrumentation reactor period - Short
Main steam isolation valve by-pass flow - Low
Main steam isolation valve - Tripped-Closed
Main steam isolation valve - Loss of Power
Superheater outlet temperature - Low
Superheater outlet pressure - Low
Main steam safety valves - Open
Reactor recirculating pump motors bearing temperature - High
Reactor recirculating pump motors - Overload
Reactor recirculating pump motor temperatures - High
Reactor recirculating discharge valves - Loss of Power
Reactor water level - High
Reactor water level - Low
Reactor feedwater temperature - High
Reactor recirculating water temperature - Low
Reactor feedwater temperature control set point - Low
Reactor building shield pool seals - Leaking
Reactor building air lock doors - Open
Reactor building pressure - High
Reactor vent temperature - High Radiation - High, Isolation - Trip
Reactor control rod drive motors - Loss of Power
Reactor control rod drive motors - Overload
Reactor control rod drive motors - Reverse Phase
Reactor control rod drive seals leakage flow - High
Reactor control rod clutch power - Trouble
Main steam isolation valve interlock switch - Out-of-Position
Nuclear instrumentation power range flux channels 7 & 8 - High differential
Nuclear instrumentation - Trouble
Reactor pressure control pressure error - High - Low
Nuclear instrumentation short period - Runback Trip
Nuclear instrumentation channels 5 & 6 - Runback Trip
Nuclear instrumentation channels 7 and 8 - Runback Trip
Reactor control loss of feedwater - Runback Trip
Reactor control ejector exhaust high activity - Isolate - Scram
Reactor control loss of circulating water - Scram Trip
Reactor control main steam dump failure - Scram Trip
Nuclear instrumentation channels 5 and 6 - Scram Trip
Nuclear instrumentation channels 7 and 8 - Scram Trip

Key lock switches shall be used for all bypasses listed in this Section 6.2.1.

1. Source Indication Bypass

This key switch may be used to bypass the rod withdraw permissive interlocks associated with nuclear channels, 1, 2, 3, and 4 to permit maintenance of channel 4 when reactor power is above the startup channel range, provided the minimum operability requirements for nuclear channels are otherwise satisfied.

2. Short Period Runback Bypass

This key switch may be used to bypass channel 4 period runback in the boiling range of reactor operation, and permit maintenance of channel 4 provided the minimum operability requirements for nuclear channels are otherwise satisfied.

3. Inner Boiler Rods Permissive Interlock Bypass

This key switch may be used only to permit precritical rod operation checks and to permit the conduct of specially supervised tests.

4. Main Steam Isolation Valve Exercise Bypass

This key switch may be used only to bypass the valve-closing scram interlock to permit exercising the main steam isolation valve. Use of this bypass shall require direct supervision by the Shift Supervisor if reactor is operating above 20% rated power.

5. Bypass Key Control

The keys to key locked bypass switches shall normally be kept in a locked key cabinet under the direct supervision of the Shift Supervisor or higher plant management. The keys shall be captive in the switch lock when the bypass is in effect.

6.2.2 Scram Control System Testing

- (a) The scram control system, exclusive of backup relay system, shall be tested with the installed test provisions at least once per shift on each shift in which reactor operation occurs and prior to each reactor startup. Such test shall utilize installed test switches to verify design operation of each input logic circuit and associated clutch power supply by simulation of each input signal and combination of input signal that should cause scram action.
- (b) The scram control system backup relay system shall be functionally tested prior to each startup except that more than one such test in any twenty-four hour period shall not be required. Such test shall utilize installed test circuitry to simulate scram signals at the input to the relay drivers and verify that the relay scram contacts operate in accordance with design function.
- (c) Redundant components of the scram control system, exclusive of backup relay system, shall be inspected prior to each reactor startup if such inspection has not been performed within the previous 1000 hours of circuit operation. This requirement shall apply to redundant diodes of the two input logic circuits, redundant transistors of the logic circuit emitter followers, and redundant transistors of the rectifier-control excitation circuits. Such inspection shall require tests as in 6.2.2 (a) or equivalent to verify design function of each component with the redundancy of the associated component temporarily disabled.

6.2.3 Nuclear Instrumentation Testing and Operating Requirements

- (a) At least one startup channel shall monitor neutron flux during shutdown if the reactor contains fuel.
- (b) At least two startup channels shall monitor neutron flux during reactor startup.
- (c) When operating in the startup channel range, coincident withdrawal of more than one startup channel detector drive shall not be permitted and control rod withdrawal shall not be permitted while a detector is being withdrawn.

- (d) A minimum of both operating channels (5 and 6) or one operating channel and channel 4 shall be required to monitor neutron flux in the intermediate range. During any period in which channel 4 is used to satisfy this minimum requirement, the short-period trip normally used for runback action shall be reconnected to have scram capability.
- (e) A minimum of three of channels 5, 6, 7 and 8 shall be available to monitor neutron flux in the power range. When one of these channels is removed from service for maintenance or repairs a scram signal from that channel shall be inserted at scram control input logic circuits to maintain an effective 1 of 3 noncoincidence scram logic under this condition, except that coincidence logic may be used only to permit testing required by paragraphs i and j.
- (f) Consistent with the requirements of 6.2.3(a) thru 6.2.3(e) nuclear channels may be removed from service for maintenance, testing or repairs.
- (g) When channel repairs are made which could significantly affect circuit time constants, response time checks shall be performed.
- (h) Detailed calibration procedures shall be performed on each channel control room chassis circuit prior to reactor startup, if such procedures have not been performed within the previous 1000 hours of channel operation.
- (i) Channel calibration checks shall be performed on each channel with installed function switch circuitry prior to reactor startup. During reactor operation such checks shall be repeated at least once every shift on those channels actively monitoring flux, except that if such checks be required during the conduct of particular reactor testing under the provisions of the power operation test program which extend beyond the period of one shift, the checks may be postponed until the completion of the particular test.
- (j) The set point and operation of each nuclear channel trip circuit shall be verified previous to each reactor startup. During reactor operation such checks shall be repeated once every shift on those channels actively monitoring flux, except that if such checks be required during the conduct of particular reactor testing under the provisions of the power operation test program, which extend beyond the period of one shift, the checks may be postponed until the completion of the particular test.

- (k) Coincidence logic shall not be used in the intermediate range.
- (l) In the event modules are replaced or repaired, calibration tests shall be conducted to verify channel operation before the channel is returned to service.

6.2.4 Process Instrumentation

This section specifies the operating requirements for instrument channels that meter process parameters and operate the safety system trip signal devices whose set points are specified in Sections 6.1.4, 6.1.6, 6.1.7 and 6.1.8. The provisions of this section shall exclude the radiation monitoring instrumentation for which operating requirements are specified in Section 6.3.4.

- 6.2.4.1 All process instrumentation components which develop input signals to the reactor scram safety system shall be tested at least each time the system is depressurized in accordance with the established preventive-maintenance methods if such tests have not been conducted within the preceding 90 days. Such tests, to the extent practicable, shall include application of calibration input signals at the transducer of each integral signal channel to verify operability, accuracy and set point of each component unit in the channel from point of input to final trip device.
- 6.2.4.2 All process channels applicable to safety system scram control surveillance of plant operation at rated conditions above twenty percent power shall be tested at least once every three months. Such tests shall consist of insertion of simulated input signals to verify operation and set point of the final trip device.
- 6.2.4.3 All process instrument channels applicable only to safety system scram control surveillance of plant startup operation to rated conditions below twenty percent power shall be tested as specified in 6.2.4.2 prior to reactor startup, or before placing the safety system mode switch in the position where such protection is designed to operate, if such tests have not been conducted within the previous thirty days.

6.2.4.4 Bypassing individual process channels from the scram control system during reactor operation shall be permissible when such channel is backed up by another channel providing the same safety function provided that:

- (a) Procedures for testing and bypassing have been approved by the Operations Committee.
- (b) All bypasses for testing and maintenance shall be approved and recorded in the reactor log by the Shift Supervisor.
- (c) The Shift Supervisor witnesses the installation and removal of the bypass.

6.2.4.5 Bypassing individual process channels from the scram control system during reactor operation shall be permissible in the event such channel is not backed up by another channel performing the same safety function provided that:

- (a) The provisions of 6.2.4.4 (a), (b), and (c) shall apply.
- (b) During any period such a channel is bypassed, a licensed reactor operator is assigned primary responsibility to perform the function of the bypassed function and to take any required safety action in the event of off-normal operating conditions.
- (c) No more than one such channel may be bypassed at any time.
- (d) If the maintenance or testing for which the bypass is installed cannot be accomplished in less than eight hours, the reactor shall be brought to a condition under which such protection is not required.

6.2.4.6 Bypasses required for special reactor tests may be installed to permit such tests provided that such bypasses and test procedures shall be approved by the Operations Committee.

6.2.5 In-core Monitoring System

6.2.5.1 Description and Purpose

An in-core flux monitoring system shall be installed in the superheater region and shall be used to determine superheater flux distribution, the effects of specified rod programs on flux distribution and dynamic coupling between superheater and boiler cores.

The monitoring system shall consist of three vertical strings of three ion chambers each with a flux wire thimble included in the assembly. The 9 ion chambers shall monitor neutron flux from about 10% to 120% rated power, with rated chamber sensitivity of 5×10^{-13} amps per nv and rated range of 10^{12} to 10^{14} nv. There shall be three picoammeters arranged to indicate relative flux level at any three chambers in any one vertical string, any one vertical plane, or any one diagonal plane. The picoammeters shall be readable from the control room and located in or adjacent to the control room. Switching provisions for chamber selection shall be located at the picoammeters.

The detector assemblies shall be designed to operate at rated reactor conditions. The detector and wire thimble assemblies shall be located in dummy superheater assemblies, one near the center of the superheater and the other two approximately diametrically opposite near the superheater periphery. The chambers of each string shall be spaced about eighteen inches apart and shall be located such that the center chamber is at about core midplane.

6.2.5.2 Operating Requirements

At least five of the nine ion chambers shall be operable when reactor power is above 50% rated power. At least seven of the chambers shall be operable during testing to verify design calculations. Chambers shall be calibrated by use of flux wires as required for design verification tests and thereafter shall be calibrated in accordance with established schedules to limit indicated drift to less than 10% of indicated flux levels.

6.3 Plant Monitoring System

The plant monitoring systems include the process gaseous, liquid, and ventilation monitoring systems, the area monitoring system, and environmental monitors.

6.3.1 Process Radiation Monitoring Systems

The process monitoring systems consist of the stack monitors, air ejector monitor, off-gas monitor, main steam line monitor, 7-liquid monitors, and 4-ventilation monitors. Monitors may temporarily be taken out of service for maintenance calibration and repairs in accordance with the requirements of Tables 6 and 7. Spare parts shall be on hand to allow necessary repairs to be made promptly.

(a) Stack Monitor

The stack effluent gases shall be monitored by a gaseous and a particulate monitor. Each monitor shall measure the concentration of activity in the stack gas by monitoring a representative side-stream sample of the gas. The stack gaseous monitor shall be set to isolate the gaseous waste disposal system from the atmosphere and to isolate the reactor purification system if the instantaneous stack release limits are exceeded.

The particulate monitor shall measure stack particulate activity with a moving filter paper mechanism and there shall be a carbon filter in the particulate monitor sample return line to the stack. The filter shall be analyzed for halogen activity at least weekly to verify that the release rate of halogens is below the release rate limit. If the filter indicates an activity of above 50 percent of the average release rate for halogens, the particulate filter shall also be analyzed for halogen activity.

(b) Air Ejector Monitor

The air ejector monitor shall detect activity in the air ejector exhaust. It shall take approximately three minutes for radioactivity from the reactor to appear at the air ejector monitor during full power operation.

(c) Off-gas Monitor

The off-gas monitor shall detect activity in the off-gas discharge to the stack downstream of the holdup tanks and filter. The off-gas monitor shall initiate isolation of the gaseous waste disposal system from the atmosphere

at an activity level no higher than that which would result in the stack instantaneous release rates being exceeded. The set point shall be no higher than 90% of full range of the instrument.

(d) Main Steam Line Monitor

The main steam line monitor shall detect activity in the main steam line entering the turbine building basement. This monitor shall initiate a main steam isolation scram on abnormally high radiation levels. In no event shall the monitor be set to initiate action at greater than ten times the normal operating background level.

(e) Liquid Monitors

There shall be liquid monitors at the following locations:

1. Turbine Building Cold Sump
2. Liquid Waste Discharge Line
3. Two at the Liquid Waste Holdup Tanks
4. Outlet of the Waste Treatment Demineralizer
5. Purification System
6. Feedwater Filters

Each monitor shall detect activity of liquid either flowing by or in storage. The liquid waste discharge, radioactivity concentration and flow rate to the environment shall be recorded in the control room. The liquid waste discharge monitor shall initiate closure of the liquid waste discharge and laundry waste discharge valves if limits are approached.

(f) Ventilation Monitors

- (1) The exhaust monitors shall detect activity in the reactor and fuel building exhaust ducts.
- (2) The turbine building ventilation exhaust monitor shall have two detectors, one which shall monitor activity in the turbine building exhaust duct and one which shall monitor activity in the flash tank vault exhaust duct.

6.3.2 Area Monitors

The location and number of area monitors shall be as follows:

<u>Location</u>	<u>Number</u>
Reactor Building	3
Fuel Handling Building	4
Control Room	1
Turbine Building	2
Hot Chemical Laboratory	1

Each unit shall contain an alarm bell, an alarm light, an indicating meter, and an integral detector. A remote panel for reactor building monitors shall be located outside the reactor building near the personnel access airlock.

At least one monitor shall be in service near the fuel storage area during fuel handling operations. A remote panel for the the fuel handling building monitors shall be located on the fuel handling building operating floor.

The area monitoring system shall normally be in operation; however, individual monitors may be taken out of service for maintenance and repairs. Spare parts shall be on hand to allow necessary repairs to be made promptly. When a required area monitor is not in operation, portable radiation detection instrumentation shall be provided and used by personnel in the area ordinarily monitored by the inoperative instrument.

6.3.3 Environmental and Health Physics Monitors

At least two environmental film monitoring stations shall be provided for determining the integrated gamma dose rate to the site environs during operation at stack release rates of up to 0.01 curies per second. Films with a minimum sensitivity of 20 mr shall be provided at each site monitoring station. These stations shall be located at the site boundary in the areas where the river enters and leaves the exclusion area.

Operation at stack release rates above 0.01 curies per second of noble gas shall not exceed eight hours without at least ten film monitoring stations in service which will be used to provide assurance that downwind doses received by persons living near the site do not exceed allowable limits. The film at each station shall be replaced and analyzed at least once per month.

All persons leaving radiation areas likely to become contaminated shall be monitored for radioactivity using a radiation detection probe.

The health physics monitors shall meet the requirements listed in Tables 3 and 4.

TABLE 3 - AIR SAMPLING EQUIPMENT

Type of Instrument	Number Available	Radiation Detected	Sensitivity Range	Minimum Calibration Frequency
Moving Filter Air Monitor	1	β, γ	10^{-2} uc/ml will give .2 cpm build-up at 1 cfm.	35 days
Fixed Filter	3	α, β, γ in counting instruments.	See sensitivity of counting instruments.	35 days

TABLE 4 - PORTABLE MONITORING INSTRUMENTS

Type of Instrument	Number Available	Radiation Detected	Sensitivity Range	Minimum Calibration Frequency
Low Range Cutie Pie	5	β, γ	0-2500 mr/hr	35 days
Low Range Integrating Cutie Pie	1	β, γ	0-2500 mr	35 days
High Range Cutie Pie	2	γ	0-250 R/hr	35 days
Survey Meter	2	α	2 to 2000 per $\text{cm}^2/\text{min.}$	35 days
Neutron Probe for Survey Meter	1	n	0-300 mrem/hr	35 days
Portable Geiger Survey	3	β, γ	0-20 mr/hr	35 days
GM Frisker	5	β, γ	50-50,000 cpm	35 days

6.4 Radioactive Waste Disposal Systems

6.4.1 Airborne Radioactive Wastes

Gases originating in the reactor and passing with the steam to the main condenser shall be removed from the condenser by air ejectors. These gases shall pass through approximately 75 feet of 6 inch hold-up pipe which shall provide approximately 3 min of holdup. The gases shall then pass through baffled holdup tanks of 270 ft³ volume to allow approximately 12 minutes additional decay time, and subsequently pass through a high efficiency filter (99.9% efficient, .3 micron particles). The gases shall be compressed in holdup tanks if additional decay time is desired. Two holdup tanks shall be provided each of which shall be sized to allow storage of the reactor off-gas for approximately 12 hours of plant operation. Off-gas may be passed directly to the stack from the holdup pipe for a short period of time to allow maintenance of equipment. The air ejector and the stack monitors shall all be in service when the holdup tanks are bypassed, and the gases shall pass through a high efficiency filter (99.9% efficient, .3 micron particles) and be diluted with ventilation air before being discharged.

Ventilation air from around the reactor shall pass directly to the inlet plenum of the induced draft fans for the stack. Each induced draft fan shall be designed to deliver 48,000 cfm. The stack shall be divided into two flow channels. The minimum stack flow during operation shall be 25,000 cfm. If the induced draft fans are lost from service for five minutes, the plant shall be shut down. The top of the stack shall be 107 feet above grade level. In addition, the radioactive gas disposal system shall have the following characteristics:

- (a) Noncondensable gases shall be removed from the turbine steam seals by the gland seal exhauster. The gases shall be discharged from the exhauster to the stack plenum.
- (b) All ventilation air from the reactor containment vessel and the turbine building shall be discharged through the stack.
- (c) All other potential sources of gaseous radioactive wastes, except ventilation air from the Hot Chemical Laboratory, shall be discharged to the stack. The concentration of radioisotopes at the point of discharge from the Hot Chemical Laboratory to the atmosphere shall not exceed the limits in Column 1, Table II, Appendix B of 10 CFR Part 20 when averaged over a year.

6.4.2

6.4.2 Solid Waste

Solid active waste shall be collected, packaged in suitable containers, and shipped offsite for disposal in accordance with 10 CFR 20.

6.4.3 Liquid Waste

Liquid waste shall be processed by any or all of the following: filtration, demineralization, sedimentation, storage for decay, and dilution. Liquid waste shall be diluted with river and cooling tower blowdown water. During those periods when water is discharged from the cooling tower to the waste discharge ditch the activity of the cooling tower water shall be checked weekly to verify that it does not contain significant activity of plant origin.

Tanks in the liquid waste system shall have the following capacities:

<u>Tanks</u>	<u>Approximate Gallons</u>
4 - Low Solids Holdup	3000 ea
2 - Waste Surge	500 ea
2 - Reclaimed Water	1500 ea
Neutralizing Holdup	300
Neutralizing Tank	1500
High Solids Holdup	2000
Concentrated Waste Storage	2000
2 - Spent Resin Storage	6000 ea
4 - Sumps	7200

6.4.4 Operating Requirements

- (a) The annual average stack release rate of radioactive isotopes, other than particulate matter and halogens with half-lives longer than eight days, shall not exceed 5×10^{11} cm³/sec times the MPC for individual isotopes and mixtures presented in Column 1, Table II, Appendix B of 10 CFR Part 20. The maximum annual average stack release rate for particulate matter and halogens with half-lives longer than eight days shall be 7×10^8 cm³/sec times the MPC for individual isotopes and mixtures presented in Column 1, Table II, Appendix B of 10 CFR Part 20. The instantaneous release rate limit for all radioactive isotopes shall be a factor of ten times the annual average release rate.

- (b) The liquid radioactive wastes may be released if the gross activity of plant origin in the effluent from the discharge ditch can be regulated so that it does not exceed, on the annual average, the values stated in Column 2, Table II, Appendix B of 10 CFR Part 20.

A current inventory of liquid and solid radioactive wastes stored on site shall be kept.

TABLE 5 - AREA MONITORS

<u>Area Monitors</u>		Location of Alarm	Alarm Setpoint	Frequency of Calibration	
Location	Activity Detected			Check	Range
Control Room	γ	Building entrance and local	None	35 days	3 decades
All other required monitors	γ	Control Room and Local	Twice back-ground or 10 mR/hr, whichever is greater	35 days	3 decades

TABLE 6 - VENTILATION AND MISCELLANEOUS MONITORS

Ventilation Monitors							
Location	Activity Detected	Location of Instrument		Automatic Action		Frequency of Calibration Check	In Service Requirement
		Indicator	Recorder	Setpoint	Action		
Fuel Building	β, γ	Fuel Building Entrance	Control Room	None	None	35 days	If irradiated fuel is stored in storage pool, this or storage pool area monitor or portable replacement is required.
Reactor Building	β, γ	Fuel Building Entrance	Control Room	10 times normal full power level or 10 mr/hr which ever is larger	Reactor Scram and Bldg. Vent isolation	35 days	May bypass for repair or replacement for up to six hours.
Turbine Building	β, γ	Fuel Bldg. Entrance	Control Room	10 times normal full power level or 10 mr/hr which ever is larger	Main Steam Isolation Scram	35 days	While out of service, operator shall actuate main steam isolation scram on initiation of gas holdup.
Miscellaneous Monitors							
Location	Activity Detected	Location of Instrument		Location of Alarm	Frequency of Calibration Check	In Service Requirement	
		Indicator	Recorder				
Laundry	B, γ	Local	None	Laundry Room	35 days	If out of service, determine background outside fan room daily if occupied.	
Stack	β, γ	Local	Control Room	Control Room and fuel handling	35 days	Continual samples shall be taken if gaseous release rate indicates that the particulate and I-131 release rate is .1 or more of the permissible release rate when temporarily out of service.	

TABLE 7 - PROCESS MONITORS

Process Monitors - Gaseous

Location	Detected	Location of Instruments		Location of Alarm	Frequency of Calibration Check	Range	In Service Requirement
		Indicator	Recorder				
Air ejector discharge	Y	Mezz Fuel Building	Control Room	Control Room and at inst.	35 days	3 decades	Either air ejector or main steam monitor must be in service.
Main Steam line	Y	Control Room	Control Room	Control Room	3 months	3 decades	Either main steam or ejector monitor must be in service.
Off-gas Discharge	Y	Fuel Building Entrance	Mezz Floor Building & Control Rm.	Control Room	35 days	5 decades	Not required if holding gases or stack gas monitor in service.
Stack Gas	B,Y	Fuel Building Entrance & Mezz Floor	Control Room	Control Room	35 days	5 decades	Not required if holding gases or off-gas discharge monitor in service.

Process Monitors - Liquid

Location	Activity Detected	Location of Instruments		Location of Alarm	Automatic Action Setpoint and Action	Frequency of Calibration Check	Range
		Indicator	Recorder				
Plant Discharge	Y	Waste Panel	Control Room	Waste Panel and Control Room	Close discharge valve at one decade above analysis value.	35 days	5 decac.
All other process monitors	Y	Waste Panel	Waste Panel by selection	Waste Panel	None	3 months	5 decades

7.0 OPERATING PROCEDURES

This section describes those plant operating procedures and procedural safeguards which have a potential effect on safety. Operating principles and procedures are presented for initial start-up of the plant, for normal and emergency operation of the plant, and for the initial phase of testing. These procedures shall not modify the express requirements of Parts 50 and 55 of the Commission Regulations.

7.1 GENERAL OPERATING PRINCIPLES

- a. Operation and control of the reactor and most of the process equipment shall be centralized in the control room.
- b. There shall be at least two operations personnel in the control room for startup and shutdown of the plant. There shall be at least one AEC licensed operator in the control room at all times, except when the reactor is in the cold shutdown condition. The minimum shift complement during reactor operations other than cold shutdown shall consist of a Shift Supervisor and two operations personnel. An AEC licensed senior operator shall be at the plant during power and refueling operations and other operations involving reactor criticality, operations which may adversely affect core reactivity, operation involving modifications of core components while fuel assemblies are in the reactor, and operations involving movement of fuel assemblies. Normally for startup and shutdown operations a senior operator shall be in the control room.
- c. Operators may perform certain operating functions that may effect the reactor outside of the control room, but only at the direction of or with prior knowledge of the licensed operator in the control room.
- d. Radiation monitoring by fixed or portable instrumentation shall be performed to establish radiation levels before initial entry into radiation zones.
- e. All personnel leaving radiation zones, and all equipment being removed from such zones, shall be surveyed to an extent adequate for control of contamination.
- f. Procedures for operation of the plant equipment, including loading, starting up, maintenance, testing, shutting down, unloading and other operations involving changes in reactivity or operation with the reactor critical, shall be in accordance with detailed written instructions. Written instructions pertaining to emergencies shall be available to all personnel in convenient places. Plant personnel shall be trained in and familiar with standard and emergency procedures which he is required to perform.
- g. In the event of any situation which may compromise the safety of continued operation, it shall be required procedure to shut the plant down and to take other planned emergency action to protect persons and property.

- h. Incidents and acts having a potential detrimental effect on nuclear safety shall be investigated to prevent recurrences. Initial review shall be by the Operations Committee. Further review shall be by the Safety Committee.
- i. The Plant Superintendent shall have the overall on-site responsibility for the plant. Technical support within the plant organization shall include personnel with training and experience in the areas of reactor engineering and operation; instrumentation, chemistry and radiation protection.

7.2 PROCEDURAL SAFEGUARDS

The following procedural safeguards have been established for the operating safety of the plant.

7.2.1 Security

Access to the restricted area shall be controlled to prevent entrance of unauthorized personnel. Visiting persons requesting admittance shall be required to identify themselves before entering the restricted area and shall be admitted only under criteria established by plant management.

7.2.2 Detailed Operating and Emergency Instructions

- a. Written instructions for normal and emergency operation (The Plant Operations Manual) shall be prepared, approved by the Operations Committee, and issued prior to startup of the plant. The Safety Committee shall review portions of these instructions which involve nuclear safety and portions which are appropriate to the Safety Committee's authorization of plant operations.

The above instructions shall be reviewed and approved by responsible persons on the Plant Operating staff and by appropriate representatives of the Company's General Office in Minneapolis. These instructions shall conform to the Technical Specifications. Copies of the Site Emergency Plan shall be kept in the Control Room, Information Center and the Company's General Office.

- b. The Plant Operations Manual shall include Radiation Control Procedures to cover aspects of the plant's radiation protection program.

7.2.3 Administrative Procedural Controls

The following controls shall be employed to promote safety for the plant.

- a. Training of the operating staff so that each employee is acquainted with his specified duties and responsibilities and the action to be taken in the event of off-standard conditions.

- b. Training of new personnel to a level consistent with their specific duties and responsibilities.
- c. Periodic management review for strict adherence to the normal and emergency procedures, the radiation control procedures, the operating limits and requirements for the plant, control of access to the plant, and the procedure for investigating and reporting unusual or unexpected incidents. This review shall be on an interval not to exceed 6 months and is the responsibility of the Safety Committee.

7.2.4 Operational Review Procedures

Day-to-day maintenance and operating experience and plant incidents shall be reviewed periodically by the Safety Committee. Recommendations with respect to both nuclear safety and equipment protection shall be made to Northern States Power Company management.

7.3 PREOPERATIONAL TESTING

A program of preoperational testing shall be conducted prior to the initial operation of the plant. It shall be the purpose of this program to demonstrate that the plant has been built in accordance with specifications and is ready for initial fuel loading and startup. This program shall include the following tests relating to the main steam system and reactor containment.

- a. The control rod system shall be tested to demonstrate that it functions properly. Such testing shall include determination of the scram times, normal withdrawal and insertion times.
- b. The boron injection system shall be tested with water to demonstrate that all valving and instrumentation function properly.
- c. The reactor protection system and all associated controls and instrumentation shall be tested to the maximum extent practical at this time. All such instrumentation shall be calibrated, when practical and it shall be established that the reactor nuclear instrumentation is responsive to source neutrons.
- d. The primary system shall be pressure tested and heated by means of an auxiliary boiler to demonstrate proper thermal expansion of components.
- e. The emergency condenser system shall be tested to demonstrate its proper operation by circulating heated reactor water through the emergency condenser.
- f. Pressure and leak rate tests of the reactor building shall be made.
- g. The containment spray system shall be tested to demonstrate its proper operation.

- h. The reactor clean-up demineralizer system shall be tested to check its ability to maintain the specified water quality.
- i. The radioactive waste disposal system shall be tested insofar as practical to establish that intended disposal of contamination can be accomplished.

7.4 INITIAL CORE LOADING AND CRITICAL TESTS

Detailed procedures and a description of tests to be conducted during the startup program shall be prepared and submitted to the Operations Committee for review. The procedures shall also be submitted to the Safety Committee for review. Following committee approvals, the Northern States Power Company personnel shall be responsible for the safe operation of the plant during the startup program and subsequent operation.

7.4.1 Basic Test Conditions

The loading and critical testing program shall begin after the special initial loading instrumentation and the necessary reactor equipment have been checked and found to be in a safe and operable condition.

At the start of loading, the reactor vessel water level shall be at least one foot below the fuel.

A 6 curie Pu-Be neutron source shall be provided to yield meaningful readings on neutron sensitive chambers. The neutron population shall be monitored during the loading.

The control rod scram circuit shall be operated by at least four neutron sensitive channels with single channel coincidence whose chambers shall be capable of seeing neutrons originating in the fuel.

These shall consist of:

- (a) Two compensated ion chambers connected to picosammeters having high flux scram trips in the regular safety circuit.
- (b) Two BF_3 proportional counters connected to log count rate meters and period meters. These channels shall be connected into the regular safety circuit.

This instrumentation shall be utilized throughout the low-level testing program until the operating source is installed at which time the normal out of vessel instrumentation shall be used.

Between fuel loading steps, sensors will be moved as necessary to accommodate the increases in core size. Such moves shall be limited to two sensors between loading steps. Only one sensor shall be moved at a time and the instrumentation channel shall be checked for proper operation prior to movement of the next sensor.

Each control rod, before it is encompassed by fuel, will be checked for proper functioning in all modes. Presence of poison in the control rod will be verified by the time it is encompassed by fuel.

7.4.1.1 Core Loading and Test Program - General Conditions

- (a) During each water fill after fuel has been added the control room shall be staffed by at least one Northern States Power Supervisor, a Reactor Operator, one Physicist or Nuclear Engineer, and one Allis-Chalmers Operations Engineer.
- (b) The Supervisor shall be in communication with the man in charge in the reactor building whenever fuel is being moved.
- (c) Should an extended interruption of the loading to the initial critical configuration occur, the fuel loaded to that time shall be removed.

When loading is resumed, the complete procedure shall be repeated.

- (d) Should a short interruption of loading to the initial critical configuration occur, multiplication of the previous step taken shall be verified before loading is continued.
- (e) During the low power test program foils may be irradiated to determine the power distribution for various core and control rod configurations.

7.4.2 Boiler Slab Core Loading and Criticality

The number and configuration of the boiler elements for this minimum mass criticality test are predicted by calculations and extrapolation of Pathfinder critical facility slab experiments.

With the core drained, the boiler boxes and control rods shall be loaded into the vessel. Nuclear instrumentation for information and protection shall be operable in accordance with 7.4.1. A 6-curie Pu-Be neutron source shall be located near the periphery of the superheater region. The boiler fuel elements shall be loaded into the dry core to form a slab array. The neutron population shall be monitored during the loading.

The reactor vessel head shall be installed and a control rod worth about $4\% \Delta k/k$ shall be cocked. The neutron multiplication shall be monitored as water is added to the vessel. At specified levels, water addition shall be stopped and count rates taken. The maximum rate of flow shall not exceed that of both seal water injection pumps (rated at 30 gpm each).

The calculated shutdown margin of this minimum critical mass assembly with all control rods full in shall be about $13\% \Delta k/k$. The slab core loading and water fill shall be performed as if

criticality is expected at any point even though the fully moderated assembly is expected to be at least 9% $\Delta k/k$ subcritical with a rod cocked (4% $\Delta k/k$).

Criticality shall not be achieved any time during the fuel loading or water fill. If criticality is predicted at any time, the core region shall be drained and fuel removed to reduce reactivity.

Criticality shall be approached by the withdrawal of boiler control rods. If criticality is not achieved, the core shall be drained and a single 2.2 w/o boiler element added to the slab. The core shall again be water filled and criticality approached by control rod withdrawal. These steps shall be repeated adding a single 2.2 w/o boiler element at each step until criticality is achieved. After initial criticality is achieved, the core shall be drained, and the test repeated adding additional boiler fuel elements to determine the incremental increase in core reactivity associated with additional single boiler elements.

7.4.3 Boiler Fuel Core Loading Criticality

With the core drained, poison shims worth approximately 4% $\Delta k/k$ shall be inserted. Boiler elements shall be added to complete the boiler core in a manner similar to that outlined in Section 7.4.2. The neutron population shall be monitored during this process.

The calculated shutdown margin of this assembly with all rods and 4% $\Delta k/k$ in poison shims inserted shall be approximately 10% $\Delta k/k$. Criticality shall not be achieved at any time during fuel addition or water fill. If criticality is predicted at any time during water fill, the core region shall be drained and additional poison shims added to reduce reactivity.

Criticality shall only be achieved by the withdrawal of control rods. After the reactor is critical at a low power, various reactivity data shall be taken and the shutdown margin determined.

7.5 POWER OPERATION TEST PROGRAM

The power operation test program, consisting of three phases, shall commence only after the initial loading and critical test program has been completed and the results of this program found to be satisfactory and shall include at least the following tests:

7.5.1 Phase 1, 200 KW (th) or Less

Phase 1 experiments shall be performed (not necessarily in the order listed) at low power to establish the reference core. Critical control rod configurations, core power distributions, and reactivity coefficients shall be measured. These experimental results shall be compared with calculations. These measurements and comparisons shall provide verification of the shutdown margin and the validity of the analytical model used for the design calculations of Pathfinder.

7.5.1.1 Superheater Fuel Loading and Full Core Criticality

The superheater fuel shall be loaded and the neutron population monitored during this loading.

The reactor vessel head shall be installed and control rods worth about $4\% \Delta k/k$ cocked. The neutron multiplication shall be monitored as water is added to the vessel. At specified levels, water addition shall be stopped and counts rates taken. The maximum rate of water additional shall not exceed 60 gpm. The superheater steam passages remain voided (most reactive configuration) during water addition. The calculated shutdown margin of this assembly with all rods and poison shims inserted shall be approximately $10\% \Delta k/k$. The superheater fuel loading and water fill shall be performed as though criticality were expected at any point, even though the moderated assembly is expected to be at least $6\% \Delta k/k$ subcritical with the cocked rod pattern ($4\% \Delta k/k$).

Criticality shall be achieved by the withdrawal of boiler control rods.

7.5.1.2 Establishment of the Reference Core

The reference core shall be defined as that full core which is subcritical by at least .003 with the most reactive control rod withdrawn and the superheater voided.

Poison shims shall be removed incrementally until this criterion is met, thereby establishing the reference core. If the most-reactive-rod-withdrawn criterion is met with substantial shutdown margin, positive shim boiler fuel elements (3.2 w/o U-235 enrichment) shall be available to increase the core excess reactivity for operational purposes.

7.5.1.3 Reference Core Cold Flooding Coefficient

The superheater steam passage shall be flooded with the reactor shutdown. The reactor shall be brought to criticality and the reactivity defect associated with the flooding the steam passages shall be evaluated by means of a calibrated control rod.

7.5.1.4 Flux Maps

At various poison concentrations, flux wires shall be loaded into the superheater and boiler. The boron concentration shall be varied to provide control rod configurations typical of those to be encountered during operation from reactor start-up to the control-rods-full-out condition. The data shall be analyzed for superheater-boiler power sharing, various power shapes, and nuclear instrumentation calibration.

Boron removal shall be checked by returning to the boron-free condition and comparing the critical rod height with that measured during the establishment of the reference core.

7.5.1.5 Insertion of Reactor Source and Refueling Test

Preparatory to further testing, the antimony-beryllium reactor source shall be loaded in the superheater and the Pu-Be source removed. The reactor vessel shall be completely filled, including the superheater steam passages, during this operation. Reference core criticality shall be repeated to correlate the regular instrumentation indication to known conditions.

Tests shall be run to determine the worth of 2.2 and 3.2% boiler elements in various locations and configurations. Subsequent refueling operations shall be consistent with these tests so that refueling will not be an unreviewed safety question.

7.5.1.6 Cold Core Pressurization

The system shall be gradually pressurized. Calibrated boiler control rods shall be used to determine the reactivity change resulting from the pressurization.

7.5.1.7 Temperature Coefficient

The temperature coefficient shall be measured from ambient to about 420F. The reactor may be pressurized with nitrogen gas to prevent boiling during this test. The reactor shall be taken critical and a slow heating rate established with the startup heater.

7.5.1.8 Hot Core Flooding Coefficient

The reactivity change associated with flooding the superheater steam passages with approximately 440F water shall be determined by means of calibrated control rods.

7.5.2 Phase II, 5 MW(th) or Less

The objective of the Phase II testing shall be to raise the reactor power level in a safe manner from essentially zero power to a level at which the onset of boiling in the flooded superheater is expected, and to determine the superheater radiative cooling ability.

7.5.2.1. Power Calibration

The power level shall be increased to 5 MW(th), and the nuclear instrumentation calibrated with the superheater flooded.

7.5.2.2 Superheater Radiative Cooling Ability

The purpose of this test shall be to determine the ability of the superheater fuel elements to dissipate reactor decay heat under the no-steam-flow condition by thermal radiation and conduction to the superheater moderator water.

With the superheater drained, the steam line isolation valves closed, and the boiler water temperature near its operating point, reactor power shall be slowly increased in steps. At each step the superheater fuel temperature shall be measured by thermocouples on special instrumented assemblies. The test shall be terminated at the power level at which superheater surface measurements are approaching a pre-determined limit of 1270 F.

7.5.2.3 Steam Flow to Superheater

With the reactor shut down, steam flow to the condenser shall be established by opening the bypass isolation valve. Superheater fuel and bulk steam temperatures and boiler coolant temperature and pressure shall be monitored during initial steam flow. Reactor power shall be increased to 5 MW(th) with the superheater power fraction suppressed by means of the rod program.

7.5.2.4 Reactor System Tests

The performance of the emergency condenser, other reactor protection, and appropriate process systems shall be evaluated at this power level.

7.5.3 Phase III, Full Power or Less

The objective of this phase shall be to reach full power in a safe manner. During Phase III, power shall be increased in about five steps, starting from some power near 5 MW(th) and going to full power. There shall be approximately a 20% power increment between steps. The following tests shall be performed at each step.

7.5.3.1. Power Calibration

At each power level heat balance calculations shall be performed to calibrate the nuclear instrumentation at steady-state conditions.

7.5.3.2 Radiation Testing

At three power levels radiation data shall be taken to verify shielding calculations.

7.5.3.3 Superheater Steam Operation

Information on the relation between steam flow and temperature and superheater fuel temperature shall be generated. The superheater performance evaluated in Phase III at each power step shall be a continuation of Phase II testing.

7.5.3.4 Fluid Dynamics Effects

The reactor response associated with changes in each of the following variables shall be determined: feedwater temperature, feedwater flow, recirculation flow, and reactor pressure.

7.5.3.5 Response to Runback

At each power level the response of the reactor to rod runback shall be determined.

7.5.3.6 Transfer Function

Transfer function measurements shall be made at various power levels to measure reactor stability margin. The oscillator rod shall be calibrated at several angular positions. The maximum peak-to-peak $\Delta k_{\text{eff}}/k_{\text{eff}}$ shall be less than 10 cents. Measurements shall be made of the zero power transfer function. The frequency range investigated shall extend from about 0.01 cycles/sec to about 12 cycles/sec. After extrapolation of the stability margin to full power, the oscillator rod shall be removed from the core.

7.5.3.7 Xenon Reactivity

The reactivity associated with transient xenon shall be determined at several power levels by operation until near-equilibrium poisoning is reached--then decreasing the reactor power substantially and following the resultant reactivity changes with rod movement.

7.5.3.8 Water Level Calibration

Two-phase interface level shall be determined at each power level by means of vessel detectors and correlated to the indicated water level. A calibrating curve of two-phase interface versus power level shall be established.

7.5.3.9 Steam Dryer Efficiency

At each power level samples of steam, before and after passing through the steam dryer, shall be taken and its moisture content determined. The efficiency of the steam dryer for various steam rates shall be determined.

7.6 NORMAL OPERATION

7.6.1. General

Detailed operating procedures for each normal mode of plant operation shall be prepared prior to operation. The following is an outline of the principal normal operation procedures having a potential effect on the safe operation of the plant.

7.6.2 Cold Start-Up After Extended Shutdown

A cold start-up shall occur each time the reactor is returned to service following an extended shutdown. The procedure for a normal cold start-up involving turbine operation shall be as follows:

- a. A start-up check list shall be followed prior to beginning the actual start-up so that applicable equipment and systems shall be in condition for start-up. Containment vessel integrity provisions shall be in effect.
- b. Each control rod shall be exercised and scrambled as a check of the control rod system and the reactor safety system. A coupling verification check shall be included prior to or during start-up, if the control rods have been unlatched during the outage.
- c. The start-up check list shall be reviewed and approved by the Shift Supervisor prior to start-up.
- d. Tests shall be performed to verify that the start-up channel rate is at least two counts per second due to neutrons. The Operations Committee shall review the start-up channel count rate and determine when these are to be performed.
- e. The reactor shall be brought critical by control rod withdrawal following a prescribed withdrawal pattern.
- f. The power shall be adjusted once criticality is reached to maintain a reactor vessel temperature rise rate not to exceed 200 F per hour.
- g. The turbine shaft sealing system shall be placed in service using steam from auxiliary supplies.
- h. The condenser shall be evacuated and the air ejector will be placed in service.
- i. Turbine heating shall be started during this operation sequence. After turbine heating is completed, and the reactor reaches rated pressure, the turbine shall be gradually brought up to speed.
- j. The mode of turbine control shall be transferred to the reactor pressure control system.
- k. The control rods shall be adjusted to provide the desired power level.

7.6.5 Hot Startup

Whenever the plant has been shutdown for a period of time with the reactor vessel and auxiliaries remaining pressurized, a hot startup procedure shall be followed to return the plant to service. This procedure will be essentially independent of the cause of shutdown assuming that the cause is recognized and any non-standard conditions have been corrected. The reactor instrumentation shall be reset and downscaled and a hot startup check list shall be completed prior to the withdrawal of control rods. The start-up shall then proceed in accordance with Paragraphs (d) through (k) of 7.6.2 of the normal cold start-up procedure outlined above.

7.6.4 Normal Power Operation

During normal power operation, the pressure control system shall maintain the reactor pressure at its rated value by operating the turbine admission valves. The turbine-generator load shall be established by the control rod positions. The principal function of the operating personnel during this period shall be as follows:

- a. The maintenance of a continuous watch in the control room for prompt attention to any annunciated alarms.
- b. The adjustment of the control rod pattern to accommodate changes in reactivity and to maintain the desired power distribution.
- c. The evaluation of abnormal conditions and the initiation of corrective action as required.

7.6.5 Extended Shutdown

An extended shutdown will be accomplished as follows:

- a. Reactor power shall be reduced by manipulation of the control rods, and the main generator load shall be decreased. The turbine-generator will be separated from the system.
- b. All control rods shall be inserted.
- c. The removal of reactor decay heat and the reduction of reactor pressure shall be accomplished by controlling reactor steam flow. The rate of cooling of the reactor vessel shall not be allowed to exceed 200 F per hour.
- d. The reactor shutdown cooling system shall be placed in operation after the superheater has been flooded. This system will complete the cooling of the reactor water.
- e. A minimum of one start-up channel and one power range channel shall be left in operation. All instrumentation pertaining to control of activity release shall be left in operation.

7.6.6 Short Duration Shutdown

A shutdown of short duration may be accomplished with maintaining system pressure. The turbine-generator shall be unloaded and separated from the system. Reactor decay heat removed shall be accommodated by system losses or bypassing steam to the main condenser.

7.7 REFUELING OPERATION

The refueling operation shall be conducted in accordance with the following basic principles:

- a. Written procedures shall be available prior to refueling.
- b. The insertion and removal of fuel assemblies shall be done through the top of the reactor vessel after opening reactor vessel head closures as appropriate. Water shielding shall be provided by flooding the reactor vessel and the shield pool. Fuel bundles shall be handled by means of a handling tool, transfer carriage, and crane.

Fuel movement shall follow the following sequence for each fuel assembly replaced:

1. Removal of selected assemblies from core and transfer to spent fuel storage.
2. Reshuffling of remaining assemblies in core as desired.
3. Insertion of new assemblies in the vacant positions.
4. Partial core criticality tests may be performed to establish the core reactivity condition.

Shutdown margin checks shall be as described in 5.9.2. Assembly replacement shall proceed as described above until the desired number of fuel assemblies have been changed.

- c. At least two start-up nuclear channels shall be in service and measuring neutron flux during all refueling operations.
- d. Written procedures shall be used for core alterations which are known to increase reactivity. Communications between the control room and the loading area shall exist during all core alterations.
- e. The liquid poison system shall be available and ready for use.
- f. Containment integrity provisions shall be in effect during actual refueling operations except for the fuel transfer valve.
- g. Unirradiated fuel shall normally be stored in air in a new fuel storage area.
- h. Irradiated fuel shall be stored as described in 4.9.9 (c).
- i. The minimum refueling crew during refueling operations shall be four men. There shall be a licensed operator in the control room at all times, and the Qualified Supervisor shall be in charge.

7.8 MAINTENANCE

The following basic principles shall guide the maintenance program at the plant:

- 7.8.1 Damaged or defective equipment shall be repaired or replaced.

- 7.8.2 Maintenance check lists shall be used wherever practical to assure that equipment is included in the systematic preventive maintenance program and to guard against error damage in carrying out the maintenance effort.
- 7.8.3 A system of equipment history records shall be kept in which will be recorded the extent of and type of repair, the regular preventive maintenance actions, as well as any non-routine maintenance which is required.
- 7.8.4 The preventive maintenance program shall include a schedule for exercising of normally idle components.
- 7.8.5 Instrumentation and control systems, especially the neutron power level instrumentation and the reactor safety system, can be tested periodically with the plant in operation, and certain portions of the systems can be replaced with spare units while the plant is in operation should it be necessary.
- 7.8.6 Radiological protection practices shall be observed in maintenance activities.
- 7.8.7 Control rods shall be inspected periodically to determine their applicability for continued operation. Enough fuel shall be removed so that the reactor is more subcritical with that rod out than it was prior to its removal for inspection.
- 7.8.8 It shall be permissible to remove a control rod drive from the core when the reactor is in the cold xenon-free condition. The core shutdown margin of $0.3\% \Delta k_{\text{eff}}/k_{\text{eff}}$ with the strongest rod out of the core and the superheater voided, shall be met. The rod power switch shall be locked in the off position and all associated equipment properly tagged. A spare control rod drive mechanism shall be used to replace the removed drive immediately upon removal of the defective drive.

8.0 RESEARCH AND DEVELOPMENT PROGRAM

During the initial start-up and during subsequent operation various R and D programs will be in progress. The Technical Specification limitations will apply with the following exceptions:

Four special fuel assemblies and the oscillator rod will replace one of the inner boiler control rods, quad box, four regular fuel assemblies and the normal control rod drive.

During the experiments with the slab array the maximum possible reactivity addition rate shall be as high as 40 cents/sec. During experiments with the boiler only the maximum possible reactivity addition rate shall be as high as 10 cents/sec.

While building the slab array and during part of "boiler only" experiments, two of the quad boxes may be removed to accommodate an in-vessel chamber which is part of the core monitoring system.

Up to 5 superheater fuel assemblies instrumented with thermocouples shall be in the core to determine relative metal temperatures.