U. S. ATOMIC ENERGY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION
(Nine Mile Point Nuclear Station Unit No. 1)

NOTICE OF ISSUANCE ~~OF A~~ FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Facility Operating License No. DPR-63 to Niagara Mohawk Power Corporation (the licensee) authorizing operation of the Nine Mile Point Nuclear Station Unit No. 1 at steady state reactor core power levels not in excess of 1850 megawatts (thermal), in accordance with the provisions of the license and the Technical Specifications. The Nine Mile Point Nuclear Station Unit No. 1 is a boiling light water reactor located on the Nine Mile Point site on the southeast shore of Lake Ontario in Oswego County, New York.

The Nine Mile Point Nuclear Station Unit No. 1 has been operated since August 22, 1969, under Provisional Operating License No. DPR-17. Facility Operating License No. DPR-63 supersedes Provisional Operating License No. DPR-17 in its entirety.

Notice of Proposed Issuance of Full-Term Operating License was published in the FEDERAL REGISTER on December 5, 1972 (37 F.R. 25870). The full-term operating license was not issued previously, pending review of the environmental considerations required by the September 9, 1971 revision of Appendix D to 10 CFR Part 50. The Regulatory staff has completed its review and the Final Environmental Statement was issued in January, 1974 (notice of which was published in the FEDERAL REGISTER on January 25, 1974 (39 F.R. 3309).

The application for the full-term operating license complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license.

The license is effective as of its date of issuance and shall expire on April 11, 2005.

For further information concerning this action, see (1) the licensee's application for a full-term operating license notarized June 27, 1972, accompanied by the licensee's Environmental Report, (2) Amendment Nos. 1, 2 and 3 to the application for the full-term operating license dated November 21, 1973, and March 14 and April 24, 1974 respectively, (3) applications for amendments to license notarized September 26 and November 18, 1974, (4) the Commission's Draft Environmental Statement dated June 3, 1973, (5) the Final Environmental Statement dated January 21, 1974, (6) Facility Operating License No. DPR-63, complete with Technical Specifications (Appendices A and B), (7) the related Safety Evaluation prepared by the Directorate of Licensing dated July 3, 1974, (8) the report of the Advisory Committee on Reactor Safeguards dated September 10, 1974, and (9) Supplement 1 to the Safety Evaluation prepared by the Directorate of Licensing dated November 15, 1974, which are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W. Washington, D. C., and at the Oswego City Library at 120 East Second Street, Oswego, New York.

A copy of items (5), (6), (7), (8) and (9) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 26th day of December, 1974.

FOR THE ATOMIC ENERGY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Directorate of Licensing

RADIOLOGICAL TECHNICAL SPECIFICATIONS

APPENDIX A

TO

FACILITY OPERATING LICENSE NO. DPR-63

FOR THE

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT 1

DOCKET NO. 50-220

DECEMBER 26, 1974

OCT 1 1975

FOREWORD

These revised specifications supersede in their entirety the previous technical specifications and are issued as Appendix A to Full-Term Operating License DPR-63 issued to Niagara Mohawk Power Corporation by the Atomic Energy Comission. The Environmental Technical Specifications are issued as Appendix B to License DPR-63.

NINE MILE POINT NUCLEAR STATION
UNIT 1 - TECHNICAL SPECIFICATIONS
CONTENTS

* to be submitted at a future date

Amendment No. ~~66~~, 71

1.0 DEFINITIONS

1.1 Reactor Operating Conditions

The various reactor operating conditions are defined below. Individual technical specifications amplify these definitions when appropriate.

a. Shutdown Condition - Cold

(1) The reactor mode switch is in the shutdown position or refuel position. *
(2) No core alterations leading to an addition of reactivity are being performed.
(3) Reactor coolant temperature is less than or equal to 212F.

b. Shutdown Condition - Hot

(1) The reactor mode switch is in the shutdown position. **
(2) No core alterations leading to an addition of reactivity are being performed.
(3) Reactor coolant temperature is greater than 212F.

c. Refueling Condition

(1) The reactor mode switch is in the refuel position.
(2) The reactor coolant temperature is less than 212F.
(3) Fuel may be loaded or unloaded.
(4) No more than one operable control rod may be withdrawn.

d. Power Operating Condition

(1) Reactor mode switch is in startup or run position.
(2) Reactor is critical or criticality is possible due to control rod withdrawal.

e. Major Maintenance Condition

(1) No fuel is in the reactor.

* The reactor mode switch may be placed in the startup position to perform the shutdown margin demonstration. See Special Test Exception 3.7.1 .

** The reactor mode switch may be placed in the refuel position to perform reactor coolant system pressure testing, control rod scram time testing and scram recovery operations.

f. Peaking Factor

The ratio of the fuel rod heat flux to the heat flux of an average rod in an identical geometry bundle operating at the average core power.

g. Total Peaking Factor

The Total Peaking Factor (TPF) is the highest product of radial, axial, and local peaking factors simultaneously operative at any segment of fuel rod.

h. Critical Power

That assembly power which causes some point in the assembly to experience 5
transition boiling.

i. Critical Power Ratio (CPR)

The ratio of critical power to the bundle power at the reactor condition of interest.

j. Minimum Critical Power Ratio (MCPR)

The minimum in-core critical power ratio.

Operable

A system, subsystem, train, component or device shall be operable when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, except as noted in 3.0, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

1.3 Operating

Operating means that a system or component is performing its required functions in its required manner.

1.4 Protective Instrumentation Logic Definitions

a. Instrument Channel

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

b. Trip System

A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

1.5 Sensor Check

A sensor check is a qualitative determination of acceptable operability by observation of sensor behavior during operation. This determinatin shall include, where possible, comparison of the sensor with other independent sensors measuring the same variable.

1.6 Instrument Channel Test

Instrument channel test means injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action.

1.7 Instrument Channel Calibration

Instrument channel calibration means adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip.

1.8 Major Refueling Outage

For the purpose of designating frequency of testing and surveillance, a major refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the end of the previous refueling outage, the test or surveillance need not be performed until the next regularly scheduled outage.

1.9 Operating Cycle

An operating cycle is that portion of Station operation between reactor startups following each major refueling outage.

1.10 Test Intervals

The test intervals specified are only valid during periods of power operation and do not apply in the event of extended Station shutdown.

1.11 Primary Containment Integrity

Primary containment integrity means that the drywell and absorption chamber are closed and all of the following conditions are satisfied:

a. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed.

b. At least one door in the airlock is closed and sealed.

c. All automatic containment isolation valves are operable or are secured in the closed position.

d. All blind flanges and manways are closed.

1.12 Reactor Building Integrity

Reactor Building Integrity means that the reactor building is closed and the following conditions are met:

a. At least one door at each access opening is closed.

b. The standby gas treatment system is operable.

c. All Reactor Building ventilation system automatic isolation valves are operable or are secured in the closed position.

1.13 Core Alteration

A core alteration is the addition, removal, relocation, or other manual movement of fuel or controls in the reactor core. Control rod movement with the control rod drive hydraulic system is not considered to be a core alteration.

1.14 Rated Flux

Rated flux is the neutron flux that corresponds to a steady-state power level of 1850 thermal megawatts. The use of the term 100 percent also refers to the 1850 thermal megawatt power level.

1.15 Surveillance

Surveillance means that process whereby systems and components which are essential to plant nuclear safety during all modes of operation or which are necessary to prevent or mitigate the consequences of incidents are checked, tested, calibrated and/or inspected, as warranted, to verify performance and availability at optimum intervals.

1.16 Fire Suppression Water System

A Fire Suppression Water System shall consist of: a water supply system, fixed extinguishing systems of both automatic sprinklers and sprays, and manual fire fighting equipment consisting of standpipe risers with hose connections and hose reels.

1.17 Fire Watch Patrol

At least each hour an area with inoperable Fire Protection Equipment shall be inspected for abnormal conditions.

1.18 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.22 Process Control Program (PCP)

The process control program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of radioactive wastes, based on demonstrated processing of actual or simulated wet or liquid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, and Federal and State regulations and other requirements governing the transport and disposal of radioactive waste.

1.23 Purge - Purging

Purge or purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. The purge is completed when the oxygen concentration exceeds 19.5 percent.

1.24 Site Boundary

The site boundary shall be that line around the Nine Mile Point Nuclear Station beyond which the land is neither owned, leased, nor otherwise controlled by Niagara Mohawk Power Corporation or the New York Power Authority.

1.25 Solidification

Solidification shall be the conversion of wet or liquid waste into a form that meets shipping and burial ground requirements.

1.26 Source Check

A source check shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

1.27 Unrestricted Area

The unrestricted area shall be any area at or beyond the site boundary access that is not controlled by Niagara Mohawk Power Corporation or the New York Power Authority for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes. That area outside the restricted area (10 CFR 20.3(a)(14)) but within the site boundary will be controlled by the owner as required.

1.28 .tilation Exhaust Treatment System

A ventilation exhaust treatment system is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

1.29 Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

1.30 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.31 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1f. Plant operation within these operating limits is addressed in individual specifications.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits on the important thermal-hydraulic variables to assure the integrity of the fuel cladding.

Specification:

a. When the reactor pressure is greater than 800 psia and the core flow is greater than 10%, the existence of a Minimum Critical Power Ratio (MCPR) less than the Safety Limit Critical Power Ratio (SLCPR) (Reference 12) shall constitute violation of the fuel cladding integrity safety limit.

b. When the reactor pressure is less than or equal to 800 psia or core flow is less than 10% of rated, the core power shall not exceed 25% of rated thermal power.

2.1.2 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings on automatic protective devices related to variables on which the fuel loading safety limits have been placed.

Objective:

To provide automatic corrective action to prevent exceeding the fuel cladding safety limits.

Specification:

Fuel cladding limiting safety system settings shall be as follows:

a. The flow biased APRM scram trip settings shall be less than or equal to that shown in Figure 2.1.1.

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux.

c. The reactor high pressure scram trip setting shall be $\leq$ 1080 psig.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

c. The neutron flux shall not exceed its scram setting for longer than 1.5 seconds as indicated by the process computer. When the process computer is out of service, a safety limit violation shall be assumed if the neutron flux exceeds the scram setting and control rod scram does not occur.

To ensure that the Safety Limit established in Specifications 2.1.1a and 2.1.1b is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

d. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be more than 6 feet, 3 inches (-10 inches indicator scale) below minimum normal water level (Elevation 302'9") except as specifed in "e" below.

e. For the purpose of performing major maintenance (not to exceed 12 weeks in duration) on the reactor vessel; the reactor water level may be lowered 9' below the minimum normal water level (Elevation 302'9"). Whenever the reactor water level is to be lowered below the low-low-low level setpoint redundant instrumentation will be provided to monitor the reactor water level.

d. The reactor water low level scram trip setting shall be no lower than -12 inches (53 inches indicator scale) relative to the minimum normal water level (302'9").

e. The reactor water low-low level setting for core spray initiation shall be no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9").

f. The flow biased APRM rod block trip settings shall be less than or equal to that shown in Figure 2.1.1.

SAFETY LIMIT

Written procedures will be developed and followed whenever the reactor water level is lowered below the low-low level set point. (5 feet below minimum normal water level) The procedures will define the valves that will be used to lower the vessel water level. All other valves that have the potential of lowering the vessel water level will be identified by valve number in the procedures and these valves will be red tagged to preclude their operation during the major maintenance with the water level below the low-low level set point.

In addition to the Facility Staff requirements given in Specification 6.2.2.b, there shall be another control room operator present in the control room with no other duties than to monitor the reactor vessel water level.

scram, for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine generator are bypassed.

The operator will set the trip setting at 1000 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1000 psig indicated set point due to the deviations discussed above.

d. A reactor water low level scram trip setting -12 inches (53 inches indicator scale) relative to the minimum normal water level (Elevation 302' 9") will assure that power production will be terminated with adequate coolant remaining in the core. The analysis of the feedwater pump loss in the Technical Supplement to Petition to Increase Power Level, dated April 1970, has demonstrated that approximately 4 feet of water remains above the core following the low level scram.

The operator will set the low level trip setting no lower than -12 inches relative to the lowest normal operating level. However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

e. A reactor water low-low level signal -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302' 9") will assure that core cooling will continue even if level is dropping. Core spray cooling will adequately cool the core, as discussed in LCO 3.1.4.

The operator will set the low-low level core spray initiation point at no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302' 9"). However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

f. Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the SLCPR. This rod block trip setting, which is automatically varied with recirculation flow rate; prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during

steady-state operation is at 110% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the design peaking factor, thus, preserving the APRM rod block safety margin.

g-h. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line isolation on reactor low pressure and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at $\leq$10% valve closure, there is no increase in neutron flux and peak pressure in the vessel dome is limited to 1141 psig. (8, 9, 10).

The operator will set the pressure trip at greater than or equal to 850 psig and the isolation valve stem position scram setting at less than or equal to 10% of valve stem position from full open. However, the actual pressure set point can be as much as 15.8 psi lower than the indicated 850 psig and the valve position set point can be as much as 2.5% of stem position greater. These allowable deviations are due to instrument error, operator setting error and drift with time.

In addition to the above mentioned LS^3, other reactor protection system devices (LCO 3.6.2) serve as a secondary backup to the LS^3 chosen. These are as follows:

High fission product activity released from the core is sensed in the main steam lines by the high radiation main steam line monitors. These monitors provide a backup scram signal and also close the main steam line isolation valves.

BASES FOR 2.1.2 FUEL CLADDING - LS[3]

The scram dump volume high level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharged from the control rod drive hydraulic system as a result of a reactor scram (Section X-C.2.10).*

i. The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to the worst case transient of a load rejection and subsequent failure of the bypass. In fact, analysis[9,10] shows that heat flux does not increase from its initial value at all because of the fast action of the load rejection scram; thus, no significant change in MCPR occurs. 5

j. The turbine stop valve closure scram is provided for the same reasons as discussed in i above. With a scram setting of $\leq$ 10% valve closure, the resultant transients are nearly the same as for those described in i above; and, thus, adequate margin exists.

*FSAR

REFERENCES FOR BASES 2.1.1 AND 2.1.2 FUEL CLADDING

(1) General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO-10958 and NEDE-10958.

(2) Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, February 1973.

(3) FSAR, Volume II, Appendix E.

(4) FSAR, Second Supplement.

(5) FSAR, Volume II, Appendix E.

(6) FSAR, Second Supplement.

(7) Letters, Peter A. Morris, Director of Reactor Licensing, USAEC, to John E. Logan, Vice-President, Jersey Central Power and Light Company, dated November 22, 1967 and January 9, 1968.

(8) Technical Supplement to Petition to Increase Power Level, dated April 1970.

(9) Letter, T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972.

(10) Letter, Philip D. Raymond, Niagara Mohawk Power Corporation, to A. Giambusso, USAEC, dated October 15, 1973.

(11) Nine Mile Point Nuclear Power Station Unit 1 Load Line Limit Analysis, NEDO 24012, May, 1977.

(12) Licensing Topical Report "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

(13) Nine Mile Point Nuclear Power Station Unit 1, Extended Load Line Limit Analysis, License Amendment Submittal(Cycle 6), NEDO-24185, April 1979.

(14) General Electric SIL 299 "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation."

(15) Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-B, Amendment 10."

SAFETY LIMIT

2.2.1 REACTOR COOLANT SYSTEM

Applicability:

Applies to the limit on reactor coolant system pressure.

Objective:

To define those values of process variables which shall assure the integrity of the reactor coolant system to prevent an uncontrolled release of radioactivity.

Specification:

system pressure shall not exceed [illegible] psig at any time with fuel in the vessel.

LIMITING SAFETY SYSTEM SETTING

2.2.2 REACTOR COOLANT SYSTEM

a. The settings on the safety valves of the pressure vessel shall be as shown below. The allowable initial set point error on each setting will be ± 1 percent.

Set Point (Psig)	Number of Safety Valves
1218	4
1227	3
1236	3
1245	3
[illegible]	[illegible]

b. The reactor high-pressure scram trip setting shall be ≤ 1080 psig.

c. The flow biased APRM scram trip settings shall be as shown in Figure 2.1.1.

BASES FOR 2.2.1 REACTOR COOLANT SYSTEM SAFETY LIMIT

The pressure safety limit of 1375 psig was derived from the design pressures and applicable codes for the reactor pressure vessel and the reactor coolant system piping. (ASME Boiler and Pressure Vessel Code Section I applies to the reactor pressure vessel and ASA Piping Code, Section B31.1 applies to the coolant system piping.) The ASME Code permits pressure transients up to 10 percent over design pressure (110% x 1250 = 1375 psig) and the ASA Code permits pressure transients up to 15 percent over the design pressure (115% x 1200 = 1380 psig).

Data presented in Volume IV, Section I-B* includes the design analyses which were performed to demonstrate that the reactor pressure vessel would meet the applicable code requirements. As a part of these analyses, both design and non-design events (Tables 7 and 8) were postulated to evaluate their strain effect to the vessel. Among the non-design events, a postulated over-pressure of 3750 psig was expected to result in vessel destruction. Comparable data concerning the piping system is not available, however, ASA Code (B31.1) indicates a margin of safety factor, code allowable (10,800 psi at 600F) vs. yield strength (75,000 psi), of 6.8 for the process piping system while the margin of safety factor (15,000 psi vs. 60,000 psi) for the high pressure feedwater system is 4. Additional data in Supplement 2, Table IV-1* indicates a calculated feedwater valve burst pressure of 13,000 psi based upon a yield strength of 36,000 psi.

Based upon the available data and for safety valve sizing calculations, 1375 psig was selected as a safety limit for the reactor coolant system. The maximum pressure of the critical hydro test of the unfueled system was selected as 1800 psig, while the normal system operating pressure will be 1030 psig.

*FSAR

a. The range of set points for a safety valve acutation is selected in accordance with code requirements. A safety valve capability study presented in the Technical Supplement to Petition to Increase Power Level using the stated LS^3 values has demonstrated the maximum pressures occurring at the bottom of the reactor vessel and the bottom of the recirculation piping are 1303 psig and 1315 psig, respectively, some 72 psig below the 1375 psig safety limit. This analysis has assumed the highly improbable event of reactor isolation occurring without scram, in spite of separate and redundant scram signals such that the power output reached 167 percent of rated (1850 Mwt).

In addition to the safety valves, the solenoid-actuated relief valves are used to prevent safety valve lift during rapid reactor isolation at power coupled with failure of the bypass system. Any five of these valves opening at 1090 psig to 1100 psig will keep the maximum vessel pressure below the lowest safety valve setting, as demonstrated in Appendix E-I.3.11 (p. E-35).* (The Technical Supplement to Petition to Increase Power Level, and letter from T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972.) Subsequently, six valves were provided due to the blowdown requirements, following a small line break. The capacity of a solenoid-actuated relief valve is about the same as a safety valve. Therefore, even without scram any combination of 16 safety valves and solenoid-actuated valves will limit the pressure below the safety limit following the worst isolation situation.

b. The reactor high pressure scram setting is relied upon to terminate rapid pressure transients if other scrams, which would normally occur first, fail to function. As demonstrated in Appendix E-I of the FSAR and the Technical Supplement to Petition to Increase Power Level, Page II-12, the reactor high pressure scram is a backup to the neutron flux scram, generator load rejection scram, and main steam isolation-valve closure scram for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine-generator are bypassed.

The operator will set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above.

*FSAR

c. As shown in Appendix E-I.3.8 and 3.11,* rapid Station transients due to isolation valve or turbine trip valve closures result in coincident high-flux and high-pressure transients. Therefore, the APRM trip, although primarily intended for core protection, also serves as backup protection for pressure transients.

Although the operator will set the scram setting at less than or equal to that shown in Figure 2.1.1 the actual neutron flux setting can be as much as 2.7 percent of rated neutron flux above the line. This includes the errors discussed above. The flow bias could vary as much as one percent of rated recirculation flow above or below the indicated point.

In addition to the above-mentioned LS^3, other reactor protection system devices (LCO 3.6.2) serve as secondary backup to the LS^3 chosen. These are as follows:

The primary containment high-pressure scram serves as backup to high reactor pressure scram in the event of lifting of the safety valves. As discussed in Vol. I, VIII, 2.0.c (p. VIII-9)* a pressure in excess of 3.5 psig due to steam leakage or blowdown to the drywell will trip a scram well before the core is uncovered.

A low condenser vacuum situation will result in loss of the main reactor heat sink, causing an increase in reactor pressure. The scram feature provided, therefore, anticipates the reactor high-pressure scram. A loss of main condenser vacuum in analyzed in Appendix E-I.3.17.*

The scram dump volume high-level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharge from the control-rod-drive hydraulic system as a result of a reactor scram (Section X-C.2.10).*

In the event of main-steam-line isolation valve closure, reactor pressure will increase. A reactor scram is, therefore, provided on main-steam-line isolation valve position and anticipates the high reactor pressure scram trip.

*FSAR

3.0 LIMITING CONDITIONS FOR OPERATION

3.0.1 OPERABILITY REQUIREMENTS

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are operable, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in a condition stated in the individual specification.

In the event a Limiting Condition for Operation and/or associated surveillance requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in a condition consistent with the individual specification unless corrective measures are completed that permit operation under the permissible surveillance requirements for the specified time interval as measured from initial discovery or until the reactor is placed in an operational condition in which the specification is not applicable.

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 SURVEILLANCE INTERVALS

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

BASES

Specification 4.0.1 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a 24 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.1 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

3.1.0 FUEL CLADDING

A) GENERAL APPLICABILITY

Applies to the power level regulation, control rod system, liquid poison system, emergency cooling system, and core spray system. LCO's for the minimum allowable circuits corresponding to the LS_3 settings are included in the Reactor Protection System LCO (3.6.2).

B) GENERAL OBJECTIVE

LIMITING CONDITIONS FOR OPERATION - To define the lowest functional capability or performance level of the systems and associated components which will assure the integrity of the fuel cladding as a barrier against the release of radioactivity.

SURVEILLANCE REQUIREMENTS - To define the tests or inspections required to assure the functional capability or performance level of the required systems or components.

Amendment No. 121
JAN 1 1 1991

LIMITING CONDITION FOR OPERATION

3.1.1 CONTROL ROD SYSTEM

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the capability of the control rod system to control core reactivity.

Specification:

a. Reactivity Limitations

(1) Reactivity margin - core loading

The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest control rod in its full-out position and all other operable rods fully inserted.

SURVEILLANCE REQUIREMENT

4.1.1 CONTROL ROD SYSTEM

Applicability:

Applies to the periodic testing requirements for the control rod system.

Objective:

To specify the tests or inspections required to assure the capability of the control rod system to control core reactivity.

Specification:

The control rod system surveillance shall be performed as indicated below.

a. Reactivity Limitations

(1) Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25 percent Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn

(2) Reactivity margin - stuck control rods

Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. Inoperable control rods shall be valved out of service, in such positions that Specification 3.1.1 a(1) is met. In no case shall the number of non-fully inserted rods valved out of service be greater than six during power operation. If this specification is not met, the reactor shall be placed in the cold shutdown condition. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

b. Control Rod Withdrawal

(1) The control rod shall be coupled to its drive or completely inserted and valved out of service When removing a control rod drive for inspection, this requirement does not apply as long as the

and all other operable rods fully inserted.

(2) Reactivity margin - stuck control rods

Each partially or fully withdrawn control rod shall be exercised at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

b. Control Rod Withdrawal

(1) The coupling integrity shall be verified for each withdrawn control rod by either:

(a) Observing the drive does not go to the overtravel position, or

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

reactor is in a shutdown or refueling condition.

(2) The control rod drive housing support system shall be in place during power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.1.1a(1) is met.

(3)(a) Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 delta k supercritical.

(b) A discernible response of the nuclear instrumentation.

(2) The control rod drive housing support system shall be inspected after reassembly.

(3)(a) To consider the rod worth minimizer operable, the following steps must be performed:

(i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

(ii) The rod worth minimizer computer on-line diagnostic test shall be successfully completed.

(iii) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.
(b) Whenever the reactor is in the startup or run mode below 20% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable, except as noted in 4.1.1.b(3)(a)(iv), or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. The second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods. If the rod worth minimizer fails prior to the complete withdrawal of the first twelve rods, then the withdrawn rods shall be inserted in the reverse order in which they were withdrawn. A second independent operator or engineer shall verify that the operator at the reactor controls is following the control rod program in reverse order.	(b) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 20% rated thermal power and a second independent operator or engineer is being used he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.
(4) Control rods shall not be withdrawn for approach to criticality unless at least three source range channels have an observed count rate equal to or greater than three counts per second.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.
(b) Whenever the reactor is in the startup or run mode below 20% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable, except as noted in 4.1.1.b(3)(a)(iv), or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. The second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods.	(b) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 20% rated thermal power and a second independent operator or engineer is being used he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.
(4) Control rods shall not be withdrawn for approach to criticality unless at least three source range channels have an observed count rate equal to or greater than three counts per second.	

LIMITING CONDITION FOR OPERATION

(3) Control rods with longer scram insertion time will be permitted provided that no other control rod in a nine-rod square array around this rod has a:

(a) Scram insertion time greater than the maximum allowed,

(b) Malfunctioned accumulator,

(c) Valved out of service in a non-fully inserted position.

d. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be out of service provided that no other control rod in a nine-rod square array around this rod has a:

(1) Malfunctioned accumulator,

(2) Valved out of service in a non-fully inserted position,

(3) Scram insertion greater than maximum permissible insertion time.

SURVEILLANCE REQUIREMENT

(3) Following any outage not initiated by a reactor scram, eight rods shall be scram tested with reactor pressure above 800 psig. The same criteria of 4.1.1.c(2) shall apply.

d. Control Rod Accumulators

Once a shift check the status of the accumulator pressure and level alarms in the control room.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

If a control rod with a malfunctioned accumulator is inserted "full-in" and valved out of service, it shall not be considered to have a malfunctioned accumulator.

e. Scram Discharge Volume

With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours.

With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore at least one vent and one drain valve to OPERABLE status within 8 hours.

e. Scram Discharge Volume (SDV)

Scram Discharge Volume Vent and Drain Valves shall be demonstrated OPERABLE during Power Operations by:

1. At least once per month verifying each valve to be open;*

2. At least once per quarter cycling each valve through at least one complete cycle of full travel; and

The Scram Discharge Volume Drain and Vent valves shall be demonstrated OPERABLE at least once per Operating Cycle by verifying that:

1. Valves close within 10 seconds after receipt of a signal for control rods to scram;

2. Valves open when the scram signal is reset;

3. Level instrumentation response proves that no blockage in the system exists.

* These valves may be closed intermittently for testing under administrative controls.

LIMITING CONDITION FOR OPERATION

f. If specification 3.1.1.a through e, above, are not met, the reactor shall be placed in the hot shutdown condition within ten hours except as noted in 3.1.1.a(2).

g. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded, the reactor shall be brought to the cold, shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and the appropriate corrective action has been completed.

SURVEILLANCE REQUIREMENT

g. Reactivity Anomalies

The observed control rod inventory shall be compared with a normalized computed prediction of the control rod inventory during startup, following refueling or major core alteration.
These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.
At specific power operating conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison will be made every equivalent full power month.

a. Reactivity Limitations

(1) Reactivity margin - core loading

The core reactivity limitation is a restriction to be applied to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading or reloading and must be such that it will apply to the entire subsequent fuel cycle. It is sufficient that the core in its maximum reactivity condition be subcritical with the control rod of highest worth fully withdrawn and all other rods fully inserted. In order to implement this requirement, it will be required that the amount of shutdown margin will be at least R + 0.25 percent Δk in the cold, xenon-free condition. In this generalized expression the value of R is the difference between the calculated value of core reactivity anytime later in the cycle where it may be greater than at the beginning. R must be a positive quantity or zero. A core which contains temporary control curtains or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum, and then decreases thereafter.

The 0.25 percent Δk in the expression R + 0.25 percent Δk is provided as a finite, demonstrable, subcriticality margin. For the first fuel cycle, core reactivity is calculated never to be greater than the beginning-of-life value; hence, R = 0. The new value of R must be determined for each fuel cycle.

(2) Reactivity margin - stuck control rods

The specified limits provide sufficient scram capability to accommodate failure to scram of any one operable rod. This failure is in addition to any inoperable rods that exist in the core, provided that those inoperable rods met the core reactivity Specification 3.1.1 a(1).

Control rods which cannot be moved with control rod drive pressure are indicative of an abnormal operating condition on the affected rods and are, therefore, considered to be inoperable. Inoperable rods are valved out of service to fix their position in the core and assure predictable behavior. If the rod is fully inserted and then valved out of service, it is in a safe position of

maximum contribution to shutdown reactivity. If it is valved out of service in a non-fully inserted position, that position is required to be consistent with the shutdown reactivity limitation stated in Specification 3.1.1 a(1), which assures the core can be shut down at all times with control rods.

The allowable inoperable rod patterns will be determined using informaiton obtained in the startup test program supplemented by calculations. During initial startup, the reactivity condition of the as-built core will be determined. Also, sub-critical patterns of widely separated withdrawn control rods will be observed in the control rod sequences being used. The observations, together with calculated strengths of the strongest control rods in these patterns will comprise a set of allowable separations of malfunctioning rods. During the fuel cycle, similar observations made during any cold shutdown can be used to update and/or increase the allowable patterns.

The number of rods permitted to be valved out of service could be many more than the six allowed by the specification, particularly late in the operating cycle; however, the occurrence of more than six could be indicative of a generic problem and the reactor will be shut down. Placing the reactor in the shutdown condition inserts the control rods and accomplishes the objective of the specifications on control rod operability. This operation is normally expected to be accomplished within ten hours. The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system. Experience with this control rod drive system has indicated that weekly tests are adequate, and that rods which move by drive pressure will scram when required as the pressure applied is much higher.

Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

b. Control Rod Withdrawal

(1) Control rod dropout accidents as discussed in Appendix E* can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides an indirect verification that the rod is coupled to its drive. Details of the control rod drive coupling are given in Section IV.B.6.1.*

(2) The rod housing support is provided to prevent control rod ejection accidents. Its design is discussed in Section VII-E.* Procedural control shall assure that the housing supports are in place for all control rods.

(3) Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta k supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.[3] These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow the sequences is backed up by the operation of the RWM. This 0.013 delta k limit, together with the integral rod velocity limiters and the action of the control rod drive system, limits potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in reference 1.

Recent improvements in analytical capability have allowed more refined analysis of the control rod drop accident. These techniques have been described in a topical report, two supplements and letters to the AEC.[1][2][3][4][5]. By using the analytical models described in these reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 20% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy content to less than 280 cal/gm. Above 20% power, even multiple operator errors cannot result in a peak fuel enthalpy content of 280 cal/gm should a postulated control rod drop accident occur.

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specificed 0.013 delta k limit on in-sequence control rod or control rod segment worths. The allowable boundary conditions used in the analysis are quantified in references (4) and (5). Each core reload will be analyzed to show conformance to the limiting parameters.

*FSAR

a. A startup inter-assembly local power peaking factor of 1.30 or less.(6)

b. An end of cycle delayed neutron fraction of 0.005.

c. A beginning of life Doppler reactivity feedback.

d. The Technical Specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft/sec).

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth of in-sequence rods or rod segments will be substantially less than 0.013 delta k. Further, the addition of 0.013 delta k worth of reactivity as a result of a rod drop in conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit. However, the 0.013 delta k limit is applied in order to allow room for future reload changes and ease of verification without repetitive Tehcnical Specification changes.

Should a control rod drop accident result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in offsite doses greater than previously reported in the FSAR, but still well below the guideline values of 10 CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate, which has nearly the same consequences as for the 7 x 7 fuel case because of the operating rod power rod differences.

The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is out of service when required, a second independent operator or engineer can manually fulfill the operator-follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 20% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup.

(4) The source range monitor (SRM) system performs no automatic safety function. It does provide the operator with a visual indication of neutron level which is needed for knowledgeable and efficient reactor startup at low neutron levels. The results of reactivity accients are functions of the initial neutron flux. The requirement of at least 3 cps assures that any transient at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to critical using homogeneous patterns of scattered control rods. A minimum of three operable SRM's is required as an added conservation.

c. Scram Insertion Times

The revised scram insertion times have been established as the limiting condition for operation since the postulated rod drop analysis and associated maximum in-sequence control rod worth are based on the revised scram insertion times. The specified times are based on design requirements for control rod scram at reactor pressures above 950 psig. For reactor pressures above 800 psig and below 950 psig the measured scram times may be longer. The analysis discussed in the next paragraph is still valid since the use of the revised scram insertion times would result in greater margins to safety valves lifting.

The insertion times previously selected were based on the large number of actual scrams of prototype control rod drive mechanisms as discussed in Section IV-B.6.3.* Rapid control rod insertion following a demand to scram will terminate Station transients before any possibility of damage to the core is approached. The primary consideration in setting scram time is to permit rapid termination of steam generation following an isolation transient (i.e., main-steam-line closure or turbine trip without bypass) such that operation of solenoid-actuated relief valves will prevent the safety valves from lifting. Analyses presented in Appendix E-I*, the Second Supplement and the Technical Supplement to Petition to Increase Power Level were based on times which are slower than the proposed revised times.

The scram times generated at each refueling outage when compared to previous scram times demonstrate that the control rod drive scram function has not deteriorated.

d. Control Rod Accumulators

The basis for this specification was not described in the FSAR and, therefore, is presented in its entirety. Requiring no more than one malfunctioned accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core problems of a cold, clean core. The worst one in a nine-rod withdrawal sequence resulted in a $k_{eff} < 1.0$--other repeating rod sequence with more rods withdrawn resulted in $k_{eff} > 1.0$. At reactor pressures in excess of 800 psig, even those control rods with malfunctioned accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with malfunctioned accumulators will be spaced in a one-in-nine array rather than grouped together.

e. Scram Discharge Volume

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram, isolate the reactor coolant system from the containment when required, and to comply with the requirements of the NRC Confirmatory letter of June 24, 1983. The fill/drain test was determined to be an acceptable alternative to a reactor scram test at approximately 50% ROD DENSITY. Performance of a water fill/drain test during cold shutdown will verify that the Scram Discharge Volume is OPERABLE and instrument lines are not plugged. The volume comparison test of water drained equal water used to fill will demonstrate that there is no blockage in the system. By comparing the response of the individual instrument lines during the drain test, partial or complete blockage in one line can be detected.

The SDV Instrumentation/valve response surveillance test will be satisfied anytime a scram occurs (less than or equal to 50% rod density) or by the fill/drain test not to exceed an operating cycle.

*FSAR

f. Reactivity Anomalies

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary controls is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behaviour in the excess reactivity may be detected by comparison of actual rod inventory at any base equilibrium core state to predicted rod inventory at that state. Equilibrium xenon, samarium and power distribution are considered in establishing the steady-state base condition to minimize any source of error. During an initial period, (on the order of 1000 MWD/T core average exposure following core reloading or modification) rod inventory predictions can be normalized to actual rod patterns to eliminate calculational uncertainties. Experience with other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

(1) Paone, C.J., Stirn, R.C., and Wooley, J.A., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

(2) Stirn, R.C., Paone, C.J., and Young, R.M., "Rod Drop Accident Analysis for Large BWR's" Supplement 1 - NEDO-10527, July 1972.

(3) Stirn, R.C., Paone, C.J., and Haun, J.M., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1973.

(4) Report entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3," transmitted by letter from L.O. Mayer (NSP) to J.F. O'Leary (USAEC) dated October 4, 1973.

(5) Letter, R.R. Schneider, Niagara Mohawk Power Corporation to A. Giambusso, USAEC, dated November 15, 1973.

(6) To include the power spike effect caused by gaps between fuel pellets.

LIMITING CONDITION FOR OPERATION

3.1.2 LIQUID POISON SYSTEM

Applicability:

Applies to the operating status of the liquid poison system.

Objective:

To assure the capability of the liquid poison system to function as an independent reactivity control mechanism.

Specification:

a. During periods when fuel is in the reactor and the reactor is not shutdown by the control rods, the liquid poison system shall be operable except as specified in 3.1.2.b.

b. If a redundant component becomes inoperable, Specification 3.1.2.a. shall be considered fulfilled, provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

SURVEILLANCE REQUIREMENT

4.1.2 LIQUID POISON SYSTEM

Applicability:

Applies to the periodic testing requirements for the liquid poison system.

Objective:

To specify the tests required to assure the capability of the liquid poison system for controlling core reactivity.

Specification:

The liquid poison system surveillance shall be performed as indicated below:

a. Overall System Test:

(1) At least once during each operating cycle -

Manually initiate the system from the control room. Demineralized water shall be pumped to the reactor vessel to verify minimum flow rates and demonstrate that valves and nozzles are not clogged.

LIMITING CONDITION FOR OPERATION

c. The liquid poison tank shall contain a minimum of 1185 gallons of boron bearing solution. The solution shall have a sufficient concentration of sodium pentaborate enriched with Boron-10 isotope to satisfy the equivalency equation.

$$\frac{C}{13\%\ \text{wt}} \times \frac{628300}{M} \times \frac{Q}{86\ \text{GPM}} \times \frac{E}{19.8\%\ \text{Atom}} \geq 1$$

Where: C = Sodium Pentaborate Solution Concentration (Wt %)
M = Mass of Water in Reactor Vessel and Recirculation piping at Hot Rated Conditions (501500 lb)
Q = Liquid Poison Pump Flow Rate (30 GPM nominal)
E = Boron-10 Enrichment (Atom %)

d. The liquid poison solution temperature shall not be less than the temperature presented in Figure 3.1.2.b.

e. If Specifications "a" through "d" are not met, initiate normal orderly shutdown within one hour.

SURVEILLANCE REQUIREMENT

Remove the squibs from the valves and verify that no deterioration has occurred by actual field firing of the removed squibs. In addition, field fire one squib from the batch of replacements.

Disassemble and inspect the squib-operated valves to verify that valve deterioration has not occurred.

(2) At least once per month -

Demineralized water shall be recycled to the test tank. Pump discharge pressure and minimum flow rate shall be verified.

b. Boron Solution Checks:

(1) At least once per month -

Boron concentration shall be determined.

(2) At least once per day -

Solution volume shall be checked. In addition, the sodium pentaborate concentration shall be determined and conformance with the requirements of the equivalency equation shall be checked any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.1.2.b.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

(3) At least once per day-

The solution temperature shall be checked.

(4) At least once per operating cycle

Verify enrichment by analysis.

c. Surveillance with Inoperable Components

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

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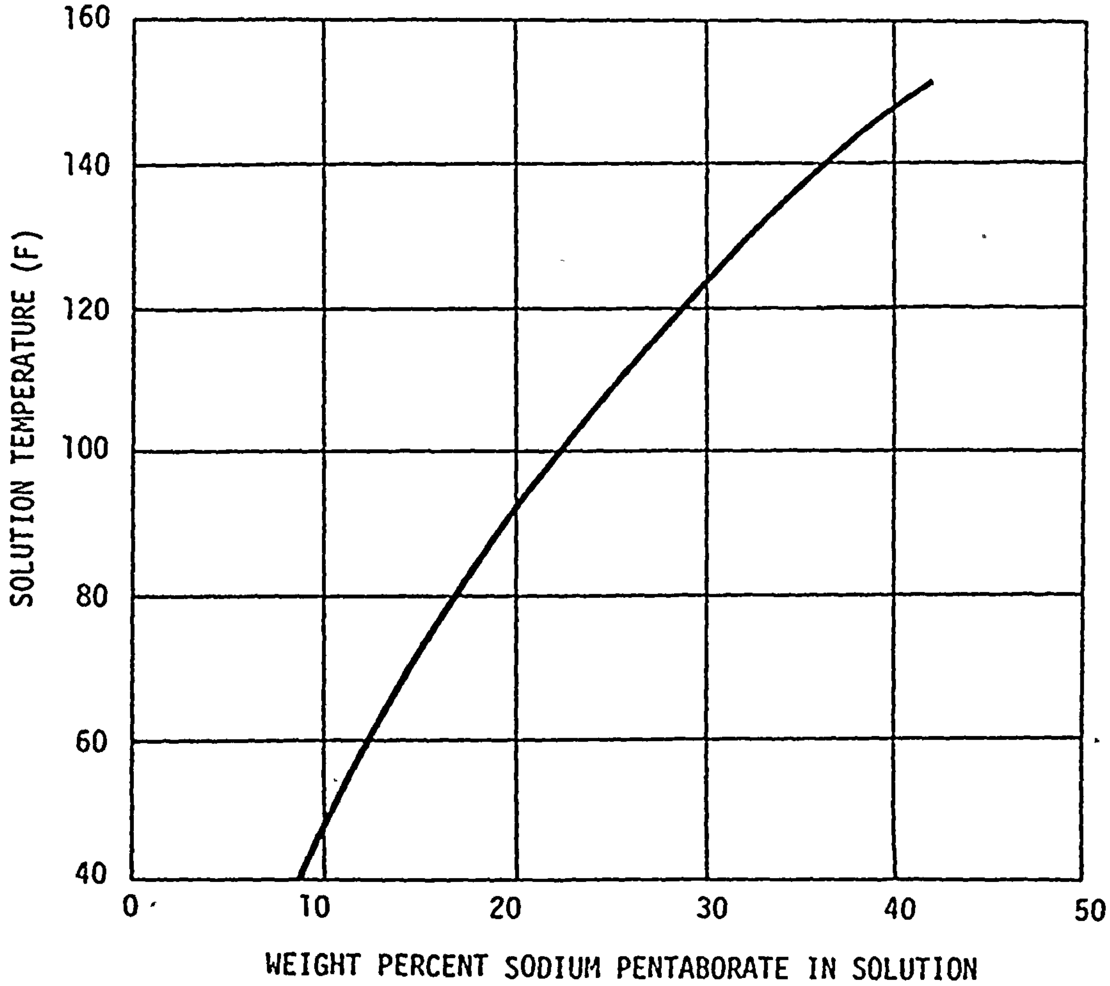
Figure 3.1.2b
MINIMUM ALLOWABLE SOLUTION TEMPERATURE



The liquid poison system (Section VII-C*) acting alone does not prevent fuel clad damage for any conceivable type of Station transient. This system provides a backup to permit reactor shutdown in the event of a massive failure of the control rods to insert.

The liquid poison system is designed to provide the capability to bring the reactor from full design rating (1850 thermal megawatts) to a cold, xenon free shutdown condition assuming none of the control rods can be inserted. A concentration of 120 ppm of boron-10 (the boron isotope with a high neutron cross section) in the reactor coolant will bring the reactor from full design rating (1850 thermal megawatts) to greater than 3 percent delta k subcritical (0.97 k_{eff}) considering the combined effects of the control rods, coolant voids, temperature change, fuel doppler, xenon, and samarium.

In order to provide good mixing, the injection time has to be greater than 17 minutes(2). The rate of boron-10 injection must also be sufficient to achieve hot shutdown during ATWS events.

The liquid poison storage tank minimum volume assures that the above requirements for boron solution insertion are met with one 30 gpm liquid poison pump. The quantity of Boron-10 isotope required to be stored in solution includes an additional 25 percent margin beyond the amount needed to shutdown the reactor to allow for any unexpected non-uniform mixing. The relationship between sodium pentaborate concentration and sodium pentaborate Boron-10 enrichment must satisfy the equivalency equation: (1)

$$\frac{C}{13\%\ \text{wt}} \times \frac{628300}{M} \times \frac{Q}{86\ \text{GPM}} \times \frac{E}{19.8\%\ \text{Atom}} \geq 1$$

Where: C = Sodium Pentaborate Solution Concentration (Wt %)
M = Mass of Water in Reactor Vessel and Recirculation piping at Hot Rated Conditions (501500 lb)
Q = Liquid Poison Pump Flow Rate (30 GPM nominal)
E = Boron-10 Enrichment (Atom %)

The tank volume requirements include consideration for 197 gallons of solution which is contained below the point where the pump takes suction from the tank and therefore cannot be inserted into the reactor.

The solution saturation temperature varies with the concentration of sodium pentaborate. Figure 3.1.2.b. includes a 5F margin above the saturation temperature to guard against precipitation. Temperature and liquid level alarms for the system are annunciated in the Control Room.

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(1) GE Topical Report NEDE-31096-P-A, "Anticipated Transients Without Scram. Response to ATWS Rule 10CFR50.62."
(2) GE Report NEDC-30921, "Assessment of ATWS Compliance Alternatives."

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

The system test specified demonstrates component response such as pump starting upon manual system initiation and is similar to the operating requirement under accident conditions. The only difference is that demineralized water rather than the boron solution will be pumped to the reactor vessel. The test interval between operating cycles results in a system failure probability of 1.1×10^{-6} (Fifth Supplement, p. 115)* and is consistent with practical considerations.

Pump operability will be demonstrated on a more frequent basis. A continuity check of the firing circuit on the explosive valves is provided by pilot lights in the control room. Tank level and temperature alarms are provided to alert the operator of off-normal conditions.

The functional test and other surveillance on components, along with the monitoring instrumentation, gives a high reliability for liquid poison system operability.

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LIMITING CONDITION FOR OPERATION

3.1.3 <u>EMERGENCY COOLING SYSTEM</u>

<u>Applicability</u>:

Applies to the operating status of the emergency cooling system.

<u>Objective</u>:

To assure the capability of the emergency cooling system to cool the reactor coolant in the event the normal reactor heat sink is not available.

<u>Specification</u>:

a. During power operating conditions and whenever the reactor coolant temperature is greater than 212°F except for hydrostatic testing with the reactor not critical, both emergency cooling systems shall be operable except as specified in 3.1.3.b and c.

b. During the remainder of Cycle 8 with one emergency cooling system inoperable, Specification 3.1.3a shall be considered fulfilled, provided the additional surveillance required in 4.1.3.f is performed.

c. During Cycle 9 and subsequent cycles, if one emergency cooling system becomes inoperable, Specification 3.1.3.a shall be considered fulfilled, provided that the inoperable system is returned to an operable condition within 7 days and the additional surveillance required in 4.1.3.f is performed.

SURVEILLANCE REQUIREMENT

4.1.3 <u>EMERGENCY COOLING SYSTEM</u>

<u>Applicability</u>:

Applies to periodic testing requirements for the emergency cooling system.

<u>Objective</u>:

To assure the capability of the emergency cooling system for cooling of the reactor coolant.

<u>Specification</u>:

The emergency cooling system surveillance shall be performed as indicated below:

a. <u>At least once every five years</u> -

The system heat removal capability shall be determined.

b. <u>At least once daily</u> -

The shell side water level and makeup tank water level shall be checked.

c. <u>At least once per month</u> -

The makeup tank level control valve shall be manually opened and closed.

LIMITING CONDITION FOR OPERATION

d. Make up water shall be available from the two gravity feed makeup Water tanks.

e. During Power Operating Conditions, each emergency cooling system high point vent to torus shall be operable.

1. With a vent path for one emergency cooling system inoperable, restore the vent path to an operable condition within 30 days.

2. With vent paths for both emergency cooling systems inoperable, restore one vent path to an operable condition with 14 days and both vent paths within 30 days.

f. If Specification 3.1.3.a, b, c, d or e are not met, a normal orderly shutdown shall be initiated within one hour, and the reactor shall be in the cold shutdown conditions within ten hours.

SURVEILLANCE REQUIREMENT

d. At least once each shift -

The area temperature shall be checked.

e. During each major refueling outage -

Automatic actuation and functional system testing shall be performed during each major refueling outage and whenever major repairs are completed on the system.

Each emergency cooling vent path shall be demonstrated operable by cycling each power-operated valve (05-01R, 05-11, 05-12, 05-04R, 05-05 and 05-07) in the vent path through one complete cycle of full travel and verifying that all manual valves are in the oper position.

f. Surveillance with an Inoperable System

During Cycle 8 with one of the emergency cooling systems inoperable, the level control valve and motor operated isolation valve in the operable system shall be demonstrated to be operable weekly.

During Cycle 9 and subsequent cycles, when one of the emergency cooling systems is inoperable, the level control valve and the motor-operated isolation valve in the operable system shall be demonstrated to be operable immediately and daily thereafter.

The turbine main condenser is normally available. The emergency cooling system (Section V-E*) is provided as a redundant backup for core decay heat removal following reactor isolation and scram. One emergency condenser system has a heat removal capacity at normal pressure of 19.0×10^7 Btu/hr, which is approximately three percent of maximum reactor steam flow. This capacity is sufficient to handle the decay heat production at 100 seconds following a scram. If only one of the emergency cooling systems is available, 2000 pounds of water will be lost from the reactor vessel through the relief valves in the 100 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory of about 450,000 pounds (Section V-E.3.1*).

The required heat removal capability is based on the data of Table V-1* adjusted to normal operating pressures. The only difference is manual system initiation rather than automatic initiation.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1080 psig sustained for 10 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips (Appendix E-I.3.13*). Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser.** To assist in depressurization for small line breaks the system is initiated on low-low reactor water level five feet (5 inches indicator scale)
1 below the minimum normal water level (Elevation 302'9") sustained for 10 seconds. The timers for initiation of the emergency condensers will be set at 10 seconds delay based on the analysis (Appendix E-I.3.13*) although they can be set anywhere between 10 and 15 seconds.

The initial water volume in each emergency condenser is 21,360 ±1500 gallons which keeps the level within ±6 inches of the normal water level. About 72,000 gallons are available from the two gravity feed condensate storage tanks. To assure this gallonage, a level check shall be done at least once per day.

This is sufficient to provide about eight hours of continuous system operation. This time is sufficient to restore additional heat sinks or pump makeup water from the two-200,000 gallon condensate storage tanks. The fire protection is also available as a makeup water supply.

* FSAR
** Technical Supplement to Petition to Increase Power Level

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

The system heat removal capability shall be determined at five-year intervals. This is based primarily on the low corrosion characteristics of the stainless steel tubing. During normal plant operation the water level will be observed at least once daily on emergency condensers and makeup water tanks. High and low water level alarms are also provided on the above pieces of equipment. The test frequency selected for level checks and valve operation is to assure the reliability of the system to operate when required.

The emergency cooling system is provided with high point vents to exhaust noncondensible gases that could inhibit natural circulation cooling. Valve redundancy in the vent path serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path. The function, capabilities and testing requirements of the emergency cooling vent paths are consistent with the requirements of item II.B.1 of NUREG 0737, "Clarification of TMI Action Plan Requirement," November 1980.

3.1.4 CORE SPRAY SYSTEM

Applicability:

Applies to the operating status of the core spray systems.

Objective:

To assure the capability of the core spray systems to cool reactor fuel in the event of a loss-of-coolant accident.

Specification:

a. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, each of the two core spray systems shall be operable except as specified in Specifications b and c below.

b. If a redundant component of a core spray system becomes inoperable, that system shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

c. If a redundant component in each of the core spray systems becomes inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

4.1.4 CORE SPRAY SYSTEM

Applicability:

Applies to the periodic testing requirements for the core spray systems.

Objective:

To verify the operability of the core spray systems.

Specification:

The core spray system surveillance shall be performed as indicated below.

a. At each major refueling outage automatic actuation of each subsystem in each core spray system shall be demonstrated.

b. At least once per quarter pump operability shall be checked.

c. At least once per quarter the operability of power-operated valves required for proper system operation shall be checked.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
d. If Specifications a, b and c are not met, a normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours.	d. Core spray header ΔP instrumentation check — Once/day calibrate — Once/3 months test — Once/3 months
e. During reactor operation, except during core spray system surveillance testing, core spray isolation valves 40-02 and 40-12 shall be in the open position and the associated valve motor starter circuit breakers for these valves shall be locked in the off position. In addition, redundant valve position indication shall be available in the control room.	e. <u>Surveillance with Inoperable Components</u> When a component becomes inoperable its redundant component or system shall be demonstrated to be operable immediately and daily thereafter.
f. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, two core spray subsystems shall be operable except as specified in g and h below.	f. With a core spray subsystem suction from the CST, CST level shall be checked once per day.
g. If one of the above required subsystems becomes inoperable, restore at least two subsystems to an operable status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.	g. At least once per month when the reactor coolant temperature is greater than 212°F, verify that the piping system between valves 40-03, 13 and 40-01, 09, 10, 11 is filled with water.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
h. If both of the above required subsystems become inoperable, suspend core alterations and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem to operable status within 4 hours or establish secondary containment integrity within the next 12 hours. i. With the downcomers in the suppression chamber having less than 3 ft. submergence, two core spray subsystems and the associated raw water pumps shall be operable with the core spray suction from the condensate storage tanks (CST), and the CST inventory shall not be less than 300,000 gallons.	

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The core spray system consists of two automatically actuated, independent systems capable of cooling reactor fuel for a range of loss-of-coolant accidents. Each of the two independent systems consists of 2 subsystems having one pump set of a core spray pump and core spray topping pump. Both systems (at least one subsystem in each system) are required to operate to limit peak clad temperatures below 2200°F (10 CFR 50 Appendix K model) for the worst case line break (recirculation line break at the point where the emergency condenser return line connects to the recirculation loop). When a component/subsystem is in a LCO state, additional surveillance requirements are imposed for the redundant component/subsystem. Consequently, application of the single failure criteria to the redundant component/subsystem is not a design requirement during the LCO period.

Allowable outage time is specified to account for redundant components that become inoperable.

Both core spray systems contain redundant supply pump sets and blocking valves. Operation of one pump set and blocking valve is sufficient to establish required delivery rate and flow path. Therefore, even with the loss of one of the redundant components, the system is still capable of performing its intended function. If a redundant component is found to have failed, corrective maintenance will begin promptly. Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages in excess of those specified. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

In determining the operability of a core spray system the required performance capability of its various components shall be considered. For example:

1. Periodic tests will demonstrate that adequate core cooling is provided to satisfy the core spray flow requirements used in the 10CFR 50 Appendix K analysis.

2. The pump shall be capable of automatic initiation from a low-low water level signal in the reactor vessel or a high containment pressure signal. The blocking valves shall be capable of automatically opening from either a low-low water signal or high containment pressure signal simultaneous with low reactor pressure permissive signal. (Section VII)*

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Instrumentation has been installed to monitor the integrity of the core spray piping within the reactor pressure vessel.

The testing specified for each major refueling outage will demonstrate component response upon automatic system initiation. For example, pump set starting (low-low level or high drywell pressure) and valve opening (low-low level or high drywell pressure and low reactor pressure) must function, under simulated conditions, in the same manner as the systems are required to operate under actual conditions. The only differences will be that demineralized water rather than suppression chamber water will be pumped to the reactor vessel and the reactor will be at atmospheric pressure. The core spray systems are designed such that demineralized water is available to the suction of one set of pumps in each system. (Section VII-Figure VII-1)*

The system test interval between operating cycles results in a system failure probability of 1.1×10^{-6} (Fifth Supplement, page 115) and is consistent with practical considerations. The more frequent component testing results in a more reliable system.

At quarterly intervals, startup of core spray pumps will demonstrate pump starting and operability. No flow will take place to the reactor vessel due to the lack of a low-pressure permissive signal required for opening of the blocking valves. A flow restricting device has been provided in the test loop which will create a low pressure loss for testing of the system. In addition, the normally closed power operated blocking valves will be manually opened and re-closed to demonstrate operability.

The intent of Specification 3.1.4i is to allow core spray operability at the time that the suppression chamber is dewatered which will allow normal refueling activities to be performed. With a core spray pump taking suction from the CST, sufficient time is available to manually initiate one of the two raw water pumps that provide an alternate core spray supply using lake water. Both raw water pumps shall be operable in the event the suppression chamber was dewatered.

*FSAR

Based on the limited time involved in performance of the concurrent refueling maintenance tasks, procedural controls to minimize the potential and duration of leakage and available coolant makeup (CST) provides adequate protection against drainage of the vessel while the suppression chamber is drained.

Specification 3.1.4e establishes provisions to eliminate a potential single failure mode of core spray isolation valves 40-02 and 40-12. These provisions are necessary to ensure that the core spray system safety function is single failure proof. During system testing, when the isolation valve(s) are required to be in the closed condition, automatic opening signals to the valve(s) are operable if the core spray system safety function is required.

In the cold shutdown and refuel conditions, the potential for a LOCA due to a line break is much less than during operation. In addition, the potential consequences of the LOCA on the fuel and containment is less due to the lower reactor coolant temperature and pressures. Therefore, one subsystem of a core spray system is sufficient to provide adequate cooling for the fuel during the cold shutdown or refueling conditions. Therefore, requiring two core spray subsystems to be operable in the cold shutdown and refuel conditions provides sufficient redundancy.

LIMITING CONDITION FOR OPERATION

3.1.5 SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)

Applicability:

Applies to the operational status of the solenoid-actuated relief valves.

Objective:

To assure the capability of the solenoid-actuated pressure relief valves to provide a means of depressurizing the reactor in the event of a small line break to allow full flow of the core spray system.

Specification:

a. During power operating condition whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, all six solenoid-actuated pressure relief valves shall be operable.

b. If specification 3.1.5a above is not met, the reactor coolant pressure and the reactor coolant temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.

SURVEILLANCE REQUIREMENT

4.1.5 SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)

Applicability:

Applies to the periodic testing requirements for the solenoid-actuated pressure relief valves.

Objective:

To assure the operability of the solenoid-actuated pressure relief valves to perform their intended functions.

Specification:

The solenoid-actuated pressure relief valve surveillance shall be performed as indicated below.

a. At least once during each operating cycle with the reactor at pressure, each valve shall be manually opened until acoustic monitors or thermocouples downstream of the valve indicate that the valve has opened and steam is flowing from the valve.

b. At least once during each operating cycle, automatic initiation shall be demonstrated.

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Pressure Blowdown

In the event of a small line break, substantial coolant loss could occur from the reactor vessel while it was still at relatively high pressures. A pressure blowdown system is provided which in conjunction with the core spray system will prevent significant fuel damage for all sized line breaks (Appendix E-11.2.0*).

Operation of three solenoid-actuated pressure relief valves is sufficient to depressurize the primary system to 110 psig which will permit full flow of the core spray system within required time limits (Appendix E-11.2*). Requiring all six of the relief valves to be operable, therefore, provides twice the minimum number required. Prior to or following refueling at low reactor pressure, each valve will be manually opened to verify valve operability. The malfunction analysis (Section II.XV, "Technical Supplement to Petition to Increase Power Level", dated April 1970) demonstrates that no serious consequences result if one valve fails to close since the resulting blowdown is well within design limits.

In the event of a small line break, considerable time is available for the operator to permit core spray operation by manually depressurizing the vessel using the solenoid-actuated valves. However, to ensure that the depressurization will be accomplished, automatic features are provided. The relief valves shall be capable of automatic initiation from simultaneous low-low-low water level (6 feet, 3 inches below miniumum normal water level at Elevation 302'-9", -10 inches indicator scale) and high containment pressure (3.5 psig). The system response to small breaks requiring depressurization is discussed in Section VII-A.3.3* and the time available to take operator action is summarized in Table VII-1*. Additional information is included in the answers to Questions III-1 and III-5 of the First Supplement.

Steam from the reactor vessel is discharged to the suppression chamber during valve testing. Conducting the tests with the reactor at nominal operating pressure is appropriate because 1) adequate redundant safety systems are provided to ensure adequate core cooling in the event of a small break loss of feedwater, and multiple relief valve failures, 2) dynamic loads and suppression pool heatups associated with high pressure testing are within allowable limits, and 3) testing at nominal operating pressures enhances plant safety and availability by assuring the relief valves can operate under normal operating conditions.

The test interval of once per operating cycle results in a system failure probability of 7.0×10^{-7} (Fifth Supplement, p. 115)* and is consistent with practical consideration.

* FSAR

LIMITING CONDITION FOR OPERATION

3.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

Applicability:

Applies to the operational status of the control rod drive pump coolant injection system.

Objective:

To assure the capability of the control rod drive pump coolant injection system to:

Provide core cooling in the event of a small line break, and

Provide coolant makeup in the event of reactor coolant leakage (see LCO 3.2.5).

Specification:

a. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212F, the control rod drive pump coolant injection system shall be operable except as specified in "b" below.

SURVEILLANCE REQUIREMENT

4.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the control rod drive pump coolant injection system.

Objective:

To assure the capability of the control rod drive pump coolant injection system in performing its intended functions.

Specification:

The control rod drive pump coolant injection system surveillance shall be performed as indicated below.

a. At least once per operating cycle -

Automatic starting of each pump shall be demonstrated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. If a redundant component becomes inoperable, the control rod drive pump coolant injection system shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.	b. <u>At least once per quarter</u> - Pump flow rate shall be determined.
c. If Specifications "a" or "b" above are not met, the reactor coolant temperature shall be reduced to 212F or less within ten hours.	c. <u>Surveillance with Inoperable Components</u> When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

BASES FOR 3.1.6 AND 4.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

The high pressure coolant injection capability of the control rod drive pumps is used to provide high pressure makeup for the specified leakage of 25 gpm (see LCO 3.2.5) and to provide core cooling in the case of a small line break. Each pump can supply 50 gpm water makeup to the reactor vessel.

One pump will normally be operating. Electric power for this system is normally available from the reserve transformer. Automatic initiation is provided to start each pump on its respective diesel generator in case offsite power is lost.

The system minimum delivery rate of 50 gpm within 60 seconds of receipt of signal will assure that automatic pressure blowdown is not actuated for the specified leakage rate of 25 gpm.

The 60-second delay in pump starting is acceptable since at least 15 minutes are available before the triple low reactor water level signals the automatic pressure blowdown to start. This analysis was based on the following assumptions; no makeup to the reactor vessel, a 50 gpm (two times allowable) leak rate exists, and the emergency condensers over-perform by 10 percent.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

The testing specified during an operating cycle will demonstrate component response upon automatic system initiation in the same manner that the system will operate if required. The testing interval results in a calculated failure probability of 1.1×10^{-6} for a control rod drive pump (Fifth Supplement), and is compatible with practical considerations. Continual monitoring of pump performance is provided since one pump is normally operating and instrumentation and alarms monitor operation of flow and pressure regulation (Section X).*

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3.1.7 FUEL RODS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value provided in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until APLHGR at all nodes is within the prescribed limits.

4.1.7 FUEL RODS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. Linear Heat Generation Rate (LHGR) During power operation, the Linear Heat Generation Rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 KW/FT. If at any time during power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded at any location, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR at all locations is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until LHGR at all locations is within the prescribed limits.	b. Linear Heat Generation Rate (LHGR) The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. Minimum Critical Power Ratio (MCPR)

During power operation, the MCPR for all fuel at rated power and flow shall be within the limit provided in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the above limit is no longer met, action shall be initiated within 15 minutes to restore operation to within the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit.

For core flows other than rated, the MCPR limit shall be the limit identified above times K_f where K_f is provided in the Core Operating Limits Report.

d. Power Flow Relationship During Operation

The power/flow relationship shall not exceed the limiting values shown in Figure 3.1.7.aa.

c. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at >25% rated thermal power.

d. Power Flow Relationship

Compliance with the power flow relationship in Section 3.1.7.d shall be determined daily during reactor operation.

e. Partial Loop Operation

Under partial loop operation, surveillance requirements 4.1.7,a,b,c and d above are applicable.

If at any time during power operation, it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated with 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until the power/flow relationship is within the prescribed limits.

e. Partial Loop Operation

During power operation, partial loop operation is permitted provided the following conditions are met.

When operating with four recirculation loops in operation and the remaining loop unisolated, the reactor may operate at 100 percent of full licensed power level in accordance with Figure 3.1.7aa and an APLHGR not to exceed the applicable limiting values provided in the Core Operating Limits Report for the fuel type.

When operating with four recirculation loops in operation and one loop isolated, the reactor may operate at 100 percent of full licensed power in accordance with Figure 3.1.7aa and an APLHGR not to exceed the applicable limiting values provided in the Core Operating Limits Report for the fuel type, provided the following conditions are met for the isolated loop.

1. Suction valve, discharge valve and discharge bypass valve in the isolated loop shall be in the closed position and the associated motor breakers shall be locked in the open position.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
2. Associated pump motor circuit breaker shall be opened and the breaker removed. If these conditions are not met, core power shall be restricted to 90.5 percent of full licensed power. When operating with three recirculation loops in operation and the two remaining loops isolated or unisolated, the reactor may operate at 90% of full licensed power in accordance with Figure 3.1.7aa and an APLHGR not to exceed the applicable limiting values provided in the Core Operating Limits Report for the fuel type. During 3 loop operation, the limiting MCPR shall be adjusted as described in the Core Operating Limits Report. Power operation in not permitted with less than three recirculation loops in operation. If at any time during power operation, it is determined by normal surveillance that the limiting value for APLHGR under one and two isolated loop operation is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is not returned to within the prescribed limits for one and two isolated loop operation within two (2) hours, reactor power reduction shall be initiated at a rate not less than 10 percent per hour until APLHGR at all nodes is within the prescribed limits.	

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f. Recirculation Loops

During all operating conditions with irradiated fuel in the reactor vessel, at least two (2) recirculation loop suction valves and their associated discharge valves will be in the full open position except when the reactor vessel is flooded to a level above the main steam nozzles or when the steam separators and dryer are removed.

g. Reporting Requirements

If any of the limiting values identified in Specification 3.1.7.a, b, c, d, and e are exceeded, a Reportable Occurrence Report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this Specification.

h. Operations Beyond the End-of-Cycle (Coastdown)

For coastdown operations beyond the End-of-Cycle (i.e., when the core reactivity has decreased such that full power cannot be maintained by further control rod withdrawal), steady state thermal power shall be limited to forty (40) percent minimum. Increasing core power level via reduced feedwater heating, once operation in the coastdown mode has begun, is not allowed.

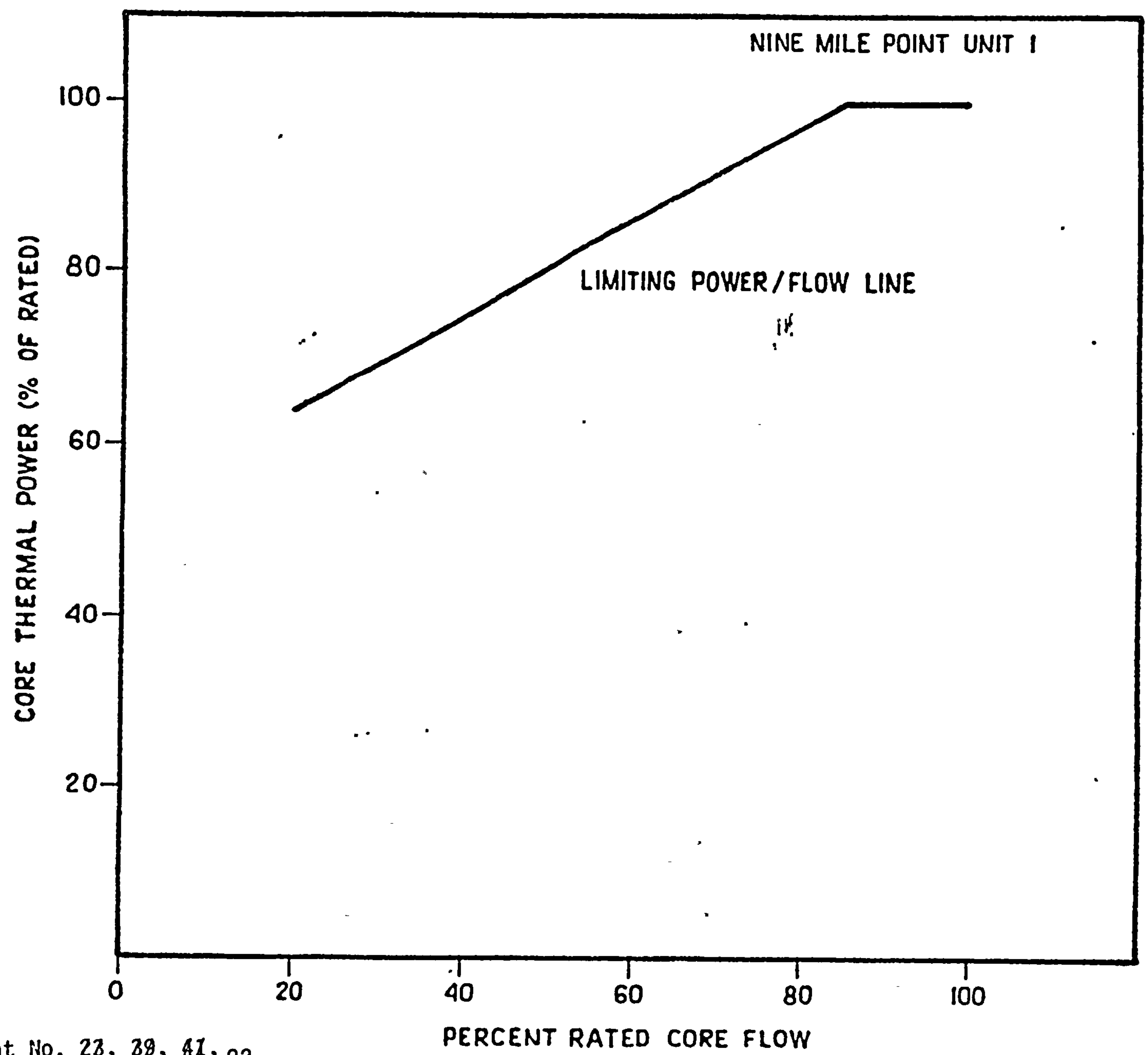

Figure 3.1.7.aa LIMITING POWER FLOW LINE

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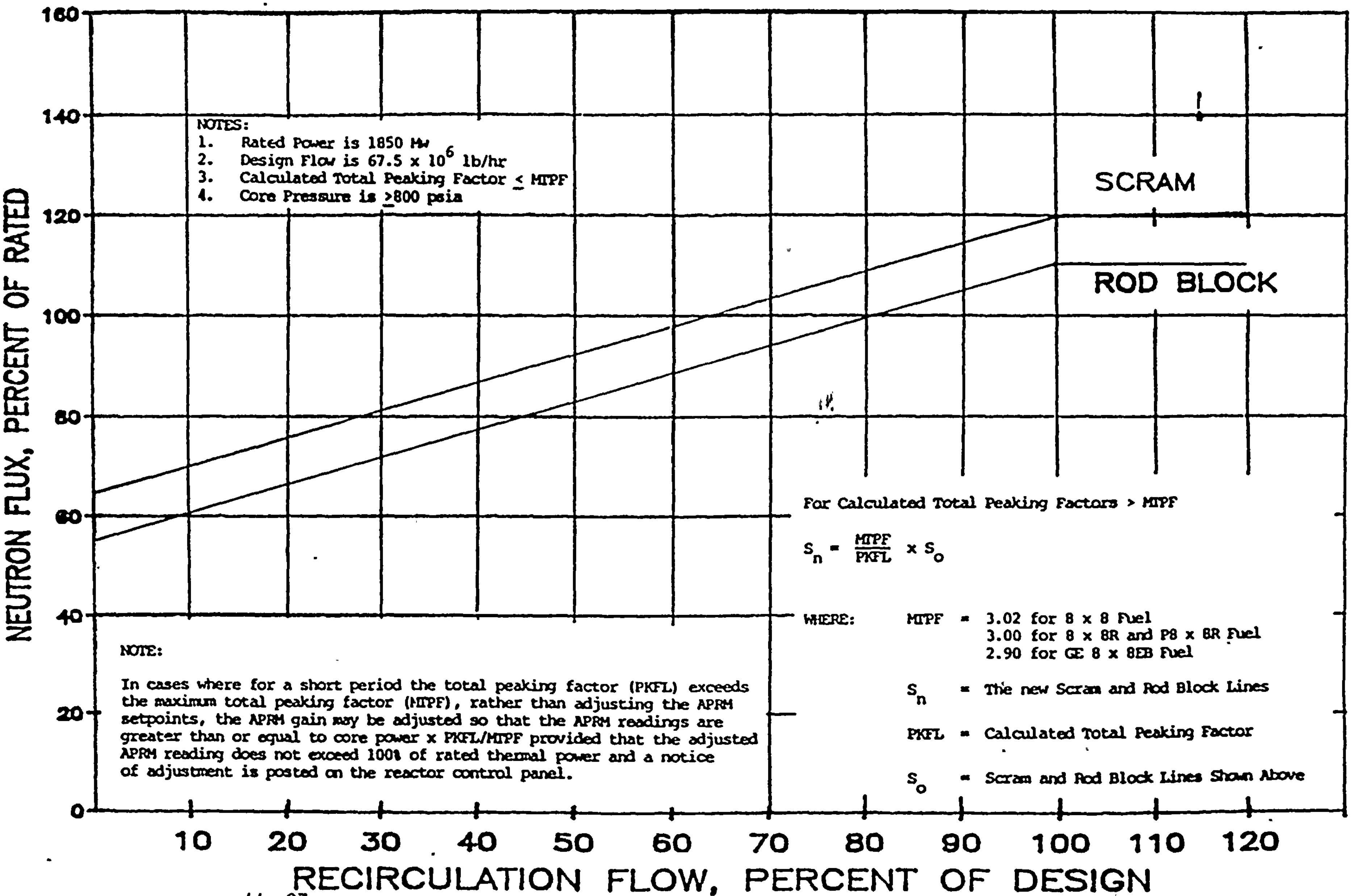
SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	g. The reactor low pressure setting for main-steam-line isolation valve closure shall be $\geq$ 850 psig when the reactor mode switch is in the run position. h. The main-steam-line isolation valve closure scram setting shall be $\leq$ 10 percent of valve closure (stem position) from full open. i. The generator load rejection scram shall be initiated by the signal for turbine control valve fast closure due to a loss of oil pressure to the acceleration relay any time the turbine first stage steam pressure is above a value corresponding to 833 Mwt, i.e., 45 percent of 1850 Mwt. j. The turbine stop valve closure scram shall be initiated at $\leq$ 10 percent of valve closure setting (Stem position) from full open whenever the turbine first stage steam pressure is above a value corresponding to 833 Mwt, i.e., 45 percent of 1850 Mwt.

FIGURE 2.1.1
FLOW BIASED SCRAM AND APRM ROD BLOCK



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The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the Minimum Critical Power Ratio (MCPR) is no less than the Safety Limit Critical Power Ratio (SLCPR) (Reference 12). The SLCPR represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, at reactor pressure > 800 psia and core flow > 10% of rated the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the Critical Power Ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective set points via the instrumented variables, by the nominal expected flow control line. The SLCPR has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the SLCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in References 1 and 12.

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of the SLCPR would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit SLCPR operation is constrained to a maximum LHGR of 13.4 kW/ft for 8x8, 8x8R, P8x8R and GE8x8EB fuel (Reference 15). At 100% power, this limit is reached with a Maximum Total Peaking Factor (MTPF) of 3.02 for 8x8 fuel, 3.00 for 8x8R and P8x8R fuel, and 2.90 for GE8x8EB fuel. During steady-state operation where the total peaking factor is above 2.90, the equation in Figure 2.1.1 will be used to adjust the flow biased scram and APRM rod block set points.

At pressure equal to or below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all core flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi.

Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28×10^3 lb/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28×10^3 lb/hr

is approximately 3.35 MWt. With the design peaking factor, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail.[3,4] In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked periodically to assume adequate insertion times. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of the SLCPR is not exceeded. Thus, use of a 1.5-second limit provides additional margin.

The process computer has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram set point is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 2.1.1.c will be relied on to determine if a safety limit has been violated.

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During periods when the reactor is shut down, consideration must also be given to water level requirements, due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds of the core height.

The lowest point at which the reactor water level can normally be monitored is approximately 7 feet 11 inches below minimum normal water level or 4 feet 8 inches above the top of the active fuel. This is the location of the reactor vessel tap for the low-low-low water level instrumentation. The actual low-low-low water level trip point is 6 feet 3 inches (-10 inches indicator scale) below minimum normal water level (Elevation 302'-9"). The 20 inch difference resulted from an evaluation of the recommendations contained in General Electric Service Information Letter 299 "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation." The low-low-low water level trip point was raised 20 inches to conservatively account for possible differences in actual to indicated water level due to potentially high drywell temperatures. The safety limit has been established here to provide a point which can be monitored and also can provide adequate margin. However, for performing major maintenance as specified in Specification 2.1.1.e, redundant instrumentation will be provided for monitoring reactor water level below the low-low-low water level set point. (For example, by installing temporary instrument lines and reference points to redundant level transmitters so that the reactor water level may be monitored over the required range.) In addition written procedures, which identify all the valves which have the potential of lowering the water level inadvertently, are established to prevent their operation during the major maintenance which requires the water level to be below the low-low level set point.

The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a safety limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a safety limit provided scram signals are operable is supported by the extensive plant safety analysis.

The abnormal operational transients applicable to operation of the plant have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1850 MWt. The analyses were based upon plant operation in accordance with the operating map given in Reference 11. In addition, 1850 MWt is the licensed maximum power level, and represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 2.

The absolute value of the void reactivity coefficient used in the analysis is conservatively esti-
mated to be about 25% greater than the nominal maximum value expected to occur during the core 5
lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

a. The Average Power Range Monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission

chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses (5,6,8,9,10,11,13) demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage.

However, in response to expressed beliefs (7) that variation of APRM flux scram with recirculation flow is a prudent measure to assure safe plant operation during the design confirmation phase of plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Figure 2.1.1 when the maximum total peaking factor is greater than the limiting total peaking factor.

b. Normal operation of the automatic recirculation pump control will be in excess of 30% rated flow; therefore, little operation below 30% flow is anticipated. For operation in the startup mode while the reactor is at low pressure, the IRM scram setting is 12% of rated neutron flux. Although the operator will set the IRM scram trip at 12% of rated neutron flux or less, the actual scram setting can be as much as 2.5% of rated neutron flux greater. This includes the margins discussed above. This provides adequate margin between the setpoint and the safety limit at 25% of rated power. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power flow condition. Effects of increasing pressure at zero or low

void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit.

Procedural controls will assure that the IRM scram is maintained up to 20% flow. This is accomplished by keeping the reactor mode switch in the startup position until 20% flow is exceeded and the APRM's are on scale. Then the reactor mode switch may be switched to the run mode, thereby switching scram protection from the IRM to the APRM system.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining a limit above the SLCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

c. As demonstrated in Appendix E-I* and the Technical Supplement to Petition to Increase Power Level, the reactor high pressure scram is a backup to the neutron flux scram, turbine stop valve closure scram, generator load rejection scram, and main steam isolation valve closure

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Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature and the peak local cladding oxidation following the postulated design basis loss-of-coolant accident will not exceed the limits specified in 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50, Appendix K limit. The limiting value for APLHGR is provided in the Core Operating Limits Report. The APLHGR curves in the Core Operating Limits Report are based on calculations using the models described in References 13, 15 and 16.

The Reference 13 and 15 LOCA analyses are sensitive to minimum critical power ratio (MCPR). In the Reference 15, analysis a MCPR value of 1.30 was assumed. If future transient analyses should yield a MCPR limit below this value, the Reference 15 LOCA analysis MCPR value would become limiting. The current MCPR limit is provided in the Core Operating Limits Report. For fuel bundles analyzed with the Reference 13 LOCA methodology, assume MCPR values of 1.30 and 1.36 for five recirculation loop and less than five loop operation respectively.

Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated (Reference 12). The LHGR shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at a minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal-hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing

of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

MCPR limits during operation at other than rated conditions are provided in the Core Operating Limits Report. For the case of automatic flow control, the K_f factor is determined such that any automatic increase in power (due to flow control) will always result in arriving at the nominal required MCPR at 100% power. For manual flow control, the K_f is determined such that an inadvertent increase in core flow (i.e., operator error or recirculation pump speed controller failure) would result in arriving at the 99.9% limit MCPR when core flow reaches the maximum possible core flow corresponding to a particular setting of the recirculation pump MG set scoop tube maximum speed control limiting set screws. These screws are to be calibrated and set to a particular value and whenever the plant is operating in manual flow control the K_f defined by that setting of the screws is to be used in the determination of required MCPR. This will assure that the reduction in MCPR associated with an inadvertent flow increase always satisfies the 99.9% requirement. Irrespective of the scoop tube setting, the required MCPR is never allowed to be less than the nominal MCPR (i.e., K_f is never less than unity).

Power/Flow Relationship

The power/flow curve is the locus of critical power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis (7, 8, 12, 14) justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

Partial Loop Operation

The requirements of Specification 3.1.7e for partial loop operation in which the idle loop is isolated, precludes the inadvertent startup of a recirculation pump with a cold leg. However, if these conditions cannot be met, power level is restricted to 90.5 percent power based on current transient analysis (Reference 9). For three loop operation, power level is restricted to 90 percent power based on the Reference 13 and 15 LOCA analyses.

The results of the ECCS calculation are affected by one or more recirculation loops being unisolated and out of service. This is due to the fact that credit is taken for extended nucleate boiling caused by flow coastdown in the unbroken loops. The reduced core flow coastdown following the break results in higher peak clad temperature due to an earlier boiling transition time. The results of the ECCS calculations are also affected by one or more recirculation loops being isolated and out of service. The mass of water in the isolated loops unavailable during blowdown results in an earlier uncovery time for the hot node. This results is an increase in the peak clad temperature.

For fuel bundles analyzed with the methodology used in Reference 13, MAPLHGR shall be reduced as required in the Core Operating Limits Report for 4 and 3 loop operation. For fuel bundles analyzed with the methodology used in References 15 and 16, MAPLHGR shall be reduced as required in the Core Operating Limits Report for both 4 and 3 loop operation.

Partial loop operation and its effect on lower plenum flow distribution is summarized in Reference 11. Since the lower plenum hydraulic design in a non-jet pump reactor is virtually identical to a jet pump reactor, application of these results is justified. Additionally, non-jet pump plants contain a cylindrical baffle plate which surrounds the guide tubes and distributes the impinging water jet and forces flow in a circumferential direction around the outside of the baffle.

Recirculation Loops

Requiring the suction and discharge for at least two (2) recirculation loops to be fully open assures that an adequate flow path exists from the annular region between the pressure vessel wall and the core shroud, to the core region. This provides for communication between those areas, thus assuring that reactor water level instrument readings are indicative of the water level in the core region.

When the reactor vessel is flooded to the level of the main steam line nozzle, communication between the core region and annulus exists above the core to ensure that indicative water level monitoring in the core region exists. When the steam separators and dryer are removed, safety limit 2.1.1d and e requires water level to be higher than 9 feet below minimum normal water level (Elevation 302'9"). This level is above the core shroud elevation which would ensure communication between the core region and annulus thus ensuring indicative water level monitoring in the core region. Therefore, maintaining a recirculation loop in the full open position in these two instances are not necessary to ensure indicative water level monitoring.

Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values of MAPLHGR, LHGR , MCPR, or Power/Flow Ratio. It is a requirement, as stated in Specifications 3.1.7a, b, c, and d that if at any time during power operation, it is determined that the limiting values for MAPLHGR, LHGR, MCPR, or Power/Flow Ratio are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving operation beyond a specified limit shall be reported as a Reportable Occurrence. If the specified corrective action described in the LCO's was taken, a thirty-day written report is acceptable.

Operations Beyond the End-of-Cycle (Coastdown)

The General Electric generic BWR analysis of coastdown operation (Reference 17) concludes that operation beyond the end-of-cycle (coastdown) is acceptable. Amendment No. 7 to GESTAR (Reference 18) concludes that the analysis conservatively bounds coastdown operation to forty (40) percent power. The margin to all safety limits analyzed increased linearly as the power decreased.

References (1) thru (6) intentionally deleted.

(7) "Nine Mile Point Nuclear Power Station Unit 1, Load Line Limit Analysis," NEDO-24012.

(8) Licensing Topical Report GE Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-A, August 1978.

(9) Final Safety Analysis Report, Nine Mile Point Nuclear Station, Niagara Mohawk Power Corporation, June 1967.

(10) NRC Safety Evaluation, Amendment No. 24 to DPR-63 contained in letter from G. Lear, NRC, to D. P. Dise dated May 15, 1978.

(11) "Core Flow Distribution in a GE Boiling Water Reactor as Measured in Quad Cities Unit 1," NEDO-10722A.

(12) Nine Mile Point Nuclear Power Station Unit 1, Extended Load Line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(13) Loss-of-Coolant Accident Analysis Report for Nine Mile Point Unit 1 Nuclear Power Station, NEDO-24348, Aug. 1981.

(14) GE Boiling Water Reactor Extended Load Line Limit Analysis for Nine Mile Point Unit 1 Cycle 9, NEDC-31126, February 1986.

(15) Nine Mile Point Unit 1, Loss-of-Coolant Accident Analysis, NEDC-31446P, June 1987.

(16) Supplement 1 to Nine Mile Point Generating Station Unit 1 SAFER/CORECOOL/GESTR-LOCA Analysis Report NEDC-31446P-1, Class III, September 1987.

(17) Communication: R. E. Engel (GE) to T. A. Ippolito (NRC) - "End-of-Cycle Coastdown Analyzed with ODYN/TASC", dated September 1, 1981.

(18) Amendment No. 7 to GESTAR, NEDE-24011-P-A-7-US, dated August 1985.

LIMITING CONDITION FOR OPERATION

3.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the operational status of the high pressure coolant injection system.

Objective:

To assure the capability of the high pressure coolant injection system to cool reactor fuel in the event of a loss-of-coolant accident.

Specification:

a. During the power operating condition* whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, the high pressure coolant injection system shall be operable except as specified in Specification "b" below.

b. If a redundant component of the high pressure coolant injection system becomes inoperable the high pressure coolant injection shall be considered operable provided that the component is returned to an operable condition within 15 days and the additional surveillance required is performed.

* One Feedwater Pump blocking valve in one HPCI pump train may be closed during reactor startup when core power is equal to or less than 25% of rated thermal power.

SURVEILLANCE REQUIREMENT

4.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the high pressure coolant injection system.

Objective:

To verify the operability of the high pressure coolant injection system.

Specification:

The high pressure coolant injection surveillance shall be performed as indicated below:

a. At least once per operating cycle -

Automatic start-up of the high pressure coolant injection system shall be demonstrated.

b. At least once per quarter -

Pump operability shall be determined.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specification "a" and "b" are not met, a normal orderly shutdown shall be initiated within one hour and reactor coolant pressure and temperature shall be reduced to less than 110 psig and saturation temperature within 24 hours.	c. <u>Surveillance with Inoperable Component</u> When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

The High Pressure Coolant Injection System (HPCI) is provided to ensure adequate core cooling in the unlikely event of small reactor coolant line break. The HPCI System is available for line breaks which exceed the capability of the Control Rod Drive pumps and which are not large enough to allow fast enough depressurization for core spray to be effective.

One set of high pressure coolant injection pumps consists of a condensate pump, a feedwater booster pump and a motor driven feedwater pump. One set of pumps is capable of delivering 3,420 gpm to the reactor vessel at reactor pressure. The performance capability of HPCI alone and in conjunction with other systems to provide adequate core cooling for a spectrum of line breaks is discussed in the Fifth Supplement of the FSAR.

In determining the operability of the HPCI System, the required performance capability of various components shall be considered.

a. The HPCI System shall be capable of meeting at least 3,420 gpm flow at normal reactor operating pressure.

b. The motor driven feedwater pump shall be capable of automatic initiation upon receipt of either an automatic turbine trip signal or reactor low-water-level signal.

c. The Condenser hotwell level shall not be less than 57 inches (75,000 gallons).

d. The Condensate storage tanks inventory shall not be less than 105,000 gallons.

e. The motor-driven feedwater pump will automatically trip if reactor high water level is sustained for ten seconds and the associated pump downstream flow control valve is not closed.

During reactor startup and shutdown, only the condensate and feedwater booster pumps are in operation at reactor pressures below approximately 400 psig. The feedwater pump is in standby. However, if the HPCI initiation signal occurs, the feedwater pump would automatically start. Calculations show that the condensate and feedwater booster pump alone are capable of providing 3,420 gpm at a reactor pressure of approximately 270 psig.

The capability of the condensate, feedwater booster and motor driven feedwater pumps will be demonstrated by their operation as part of the feedwater supply during normal station operation. Stand-by pumps will be placed in service at least quarterly to supply feedwater during station operation. An automatic system initiation test will be performed at least once per operating cycle. This will involve automatic starting of the motor driven feedwater pumps and flow to the reactor vessel.

BASES FOR 3.1.8 AND 4.1.8 HIGH PRESSURE COOLANT INJECTION

During reactor startup with periods of low reactor water feed demand, one feedwater train is operated with a blocking valve closed downstream of the main flow control valve when core power is less than or equal to 25% of rated thermal power. This allows the low flow control valve to control the reactor water flow during the startup period when feedwater flow demand is low. Use of the low flow control valve provides more uniform feedwater flow which reduces thermal cycling at the reactor pressure vessel feedwater nozzles and in the feedwater piping as well as eliminating a severe service condition in the main flow control valves during reactor startup. Under low feedwater flow conditions, the main flow control valves also experience high pressure drops and fluid velocities which shorten the valve's life and can cause plant transients due to control valve failure. Reactor startup with one HPCI train available is acceptable since LOCA makeup requirements are reduced during startup because of lower reactor pressure, less decay heat, and lower reactor power than assumed in LOCA analyses performed to Appendix K 10 CFR 50 requirements. The other feedwater train (other HPCI loop) with its blocking valve open would remain capable of supplying 3,800 gpm of feedwater upon automatic HPCI initiation at all reactor pressures.

3.2.0 REACTOR COOLANT SYSTEM

A) GENERAL APPLICABILITY

Applies to the operating conditions of the reactor coolant system and its associated systems and components.

B) GENERAL OBJECTIVE

LIMITING CONDITIONS FOR OPERATION - To define the lowest functional capability or performance level of the systems which will assure the integrity of the reactor coolant system as a barrier against the uncontrolled release of radioactivity.

SURVEILLANCE REQUIREMENTS - To define the tests or inspections required to assure the functional capability or performance level of the above.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.2.1 <u>REACTOR VESSEL HEATUP AND COOLDOWN RATES</u> <u>Applicability</u>: Applies to the reactor vessel heating or cooling rate. <u>Objective</u>: To assure that thermal stress resulting from reactor heatup and cooldown are within allowable code limits. <u>Specification</u>: During the startup and shutdown operations of the reactor, the reactor vessel temperature shall not be increased more than 100 F in any one hour period nor decreased more than 100 F in any one hour period.	

BASES FOR 3.2.1 REACTOR VESSEL HEATUP AND COOLDOWN

Design calculations reported in Volume I, Section V-A,4.0 (p. V-6)* have demonstrated that the heatup and cooldown rate of 100F/hr considered in the fatigue analysis will result in stresses well within code limits. A series of calculations have demonstrated that various extreme heatup and cooldown transients result in thermal strains well within the ASME Code limits stated in Volume I, Section V-C,3.0 (p. V-19)*. Cooldown incidents include: failure of the pressure regulator leading to a cooldown of 215F in 5.5 minutes (Appendix E-I,3.15 (p.E-45))*, inadvertent opening of a single solenoid-actuated pressure relief valve leading to a cooldown of 1050F/hr sustained for 10 minutes (Vol.I, Section V-B,1.3 (p.V-11))*, and finally, opening all six of the solenoid-actuated relief valves leads to a cooldown of 250F in 7.5 minutes (Volume IV, Section I-B)*. Reactor vessel heatup of 300F/hr (Volume IV, Section I-B)* also demonstrates stresses well within the code requirements. In view of the reported results, the specified heatup and cooldown rates are believed to be conservative.

*FSAR

LIMITING CONDITION FOR OPERATION

3.2.2 MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION

Applicability:

Applies to the minimum vessel temperature required for vessel pressurization.

Objective:

To assure that no substantial pressure is imposed on the reactor vessel unless its temperature is considerably above its Nil Ductility Transition Temperature (NDTT).

Specification:

a. During reactor vessel heatup and cooldown when the reactor is not critical, the reactor vessel temperature and pressure shall satisfy the requirements of Figure 3.2.2.a.

b. During reactor vessel heatup and cooldown when the reactor is critical, the reactor vessel temperature and pressure shall satisfy the requirements of Figure 3.2.2.b. except when performing low power physics testing with the vessel head removed at power levels not to exceed 5 mw(t).

SURVEILLANCE REQUIREMENT

4.2.2 MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION

Applicability:

Applies to the required vessel temperature for pressurization.

Objective:

To assure that the vessel is not subjected to any substantial pressure unless its temperature is greater than its Nil Ductility Transition Temperature (NDTT).

Specification:

a. Reactor vessel temperature and pressure shall be monitored and controlled to assure that the pressure and temperature limits are met.

b. Vessel material and surveillance samples located within the core region to permit periodic monitoring of exposure and material properties shall be inspected on the following schedule:

First capsule - one fourth service life
Second capsule - three fourth service life
Third capsule - standby

In the event the surveillance specimens at one quarter of the vessels service life indicate a shift of reference temperature greater than predicted the schedule shall be revised as follows:

Second capsule - one half service life
Third capsule - standby

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. During hydrostatic testing, the reactor vessel temperature and pressure shall satisfy the requirements of Figure 3.2.2.c. if the core is not critical. d. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head are equal to or greater than 100F.	

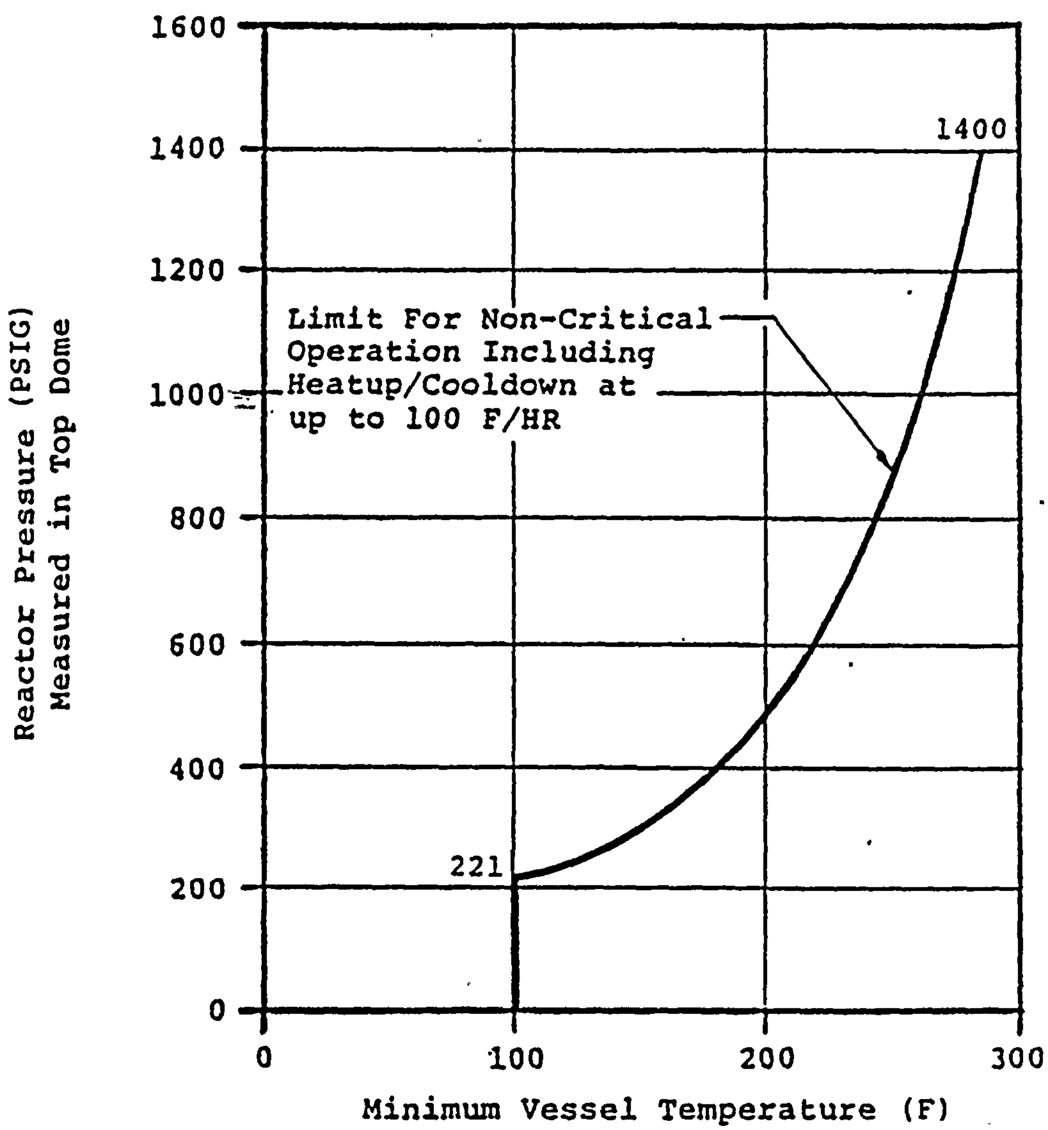

FIGURE 3.2.2.a

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING HEATUP OR COOLDOWN (REACTOR NOT CRITICAL) (HEATING OR COOLING RATE ≤ 100 F/HR) FOR UP TO THIRTEEN EFFECTIVE FULL POWER YEARS OF CORE OPERATION

LIMIT FOR NON-CRITICAL OPERATION
INCLUDING HEAT-UP/COOLDOWN AT
UP TO 100F/HR

PRESSURE (psig)	TEMPERATURE (F)
221	100
300	148
350	167
400	182
450	194
500	204
550	213
600	221
650	228
700	235
750	241
800	247
850	252
900	256
950	261
1000	265
1050	269
1100	272
1150	276
1200	279
1300	285
1400	291

TABLE 3.2.2.a

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING
HEAT-UP OR COOLDOWN (REACTOR NOT CRITICAL)
(HEATING OR COOLING RATE 100F/HR)
FOR UP TO THIRTEEN EFFECTIVE FULL
POWER YEARS OF CORE OPERATION

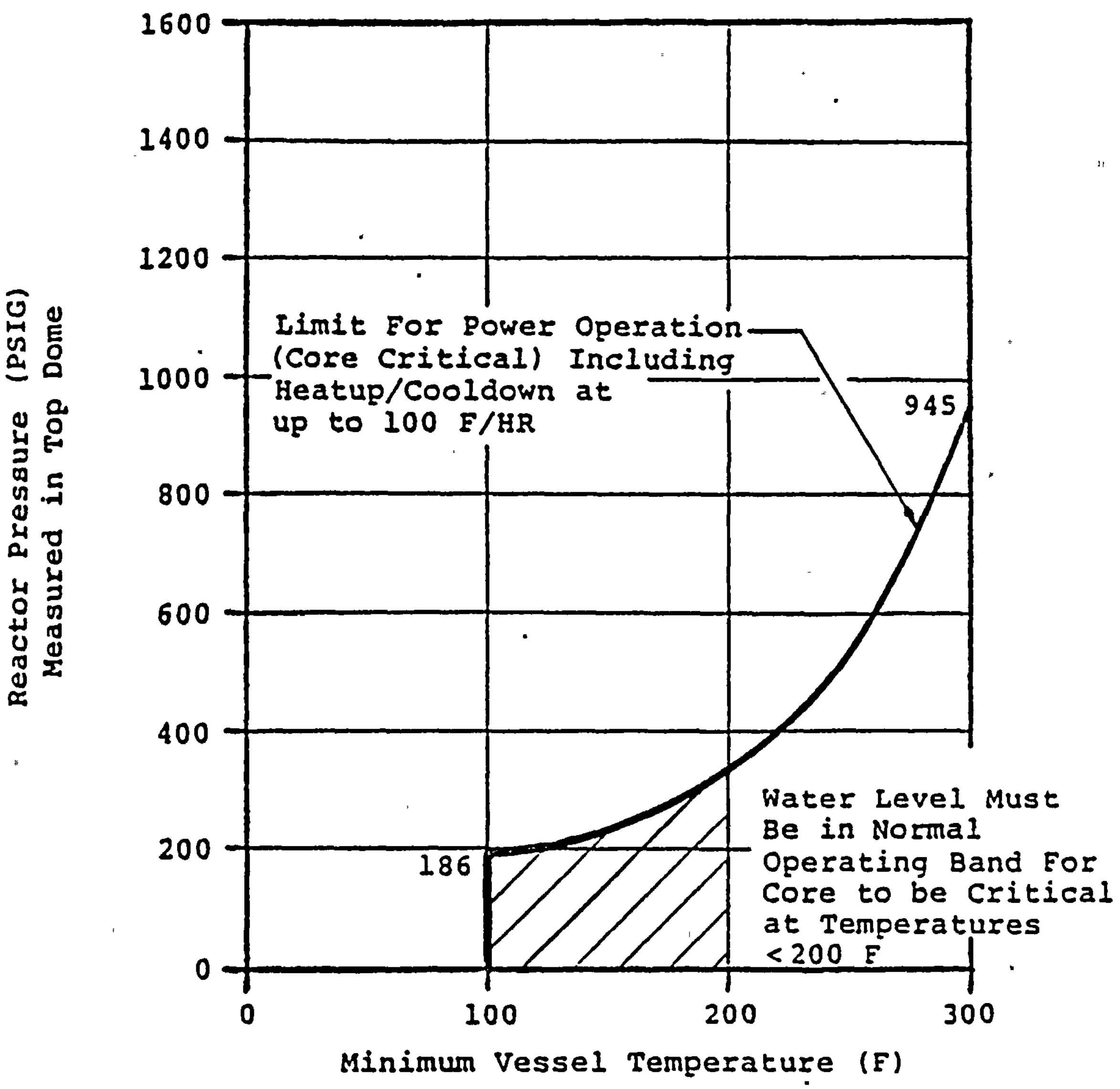

FIGURE 3.2.2.b

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING HEATUP OR COOLDOWN (REACTOR CRITICAL) (HEATING OR COOLING RATE ≤ 100 F/HR) FOR UP TO THIRTEEN EFFECTIVE FULL POWER YEARS OF CORE OPERATION

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LIMIT FOR POWER OPERATION
(CORE CRITICAL) INCLUDING HEAT-UP/
COOLDOWN AT UP TO 100F/HR

PRESSURE (psig)	TEMPERATURE (F)
186	100
250	162
300	188
350	207
400	222
450	234
500	244
550	253
600	261
650	269
700	275
750	281
800	287
850	292
900	296
950	301
1000	305
1050	308
1100	312
1150	316
1200	319
1300	325
1400	331

TABLE 3.2.2.b

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING
HEAT-UP OR COOLDOWN (REACTOR CRITICAL)
(HEATING OR COOLING RATE 100F/HR)
FOR UP TO THIRTEEN EFFECTIVE FULL
POWER YEARS OF CORE OPERATION

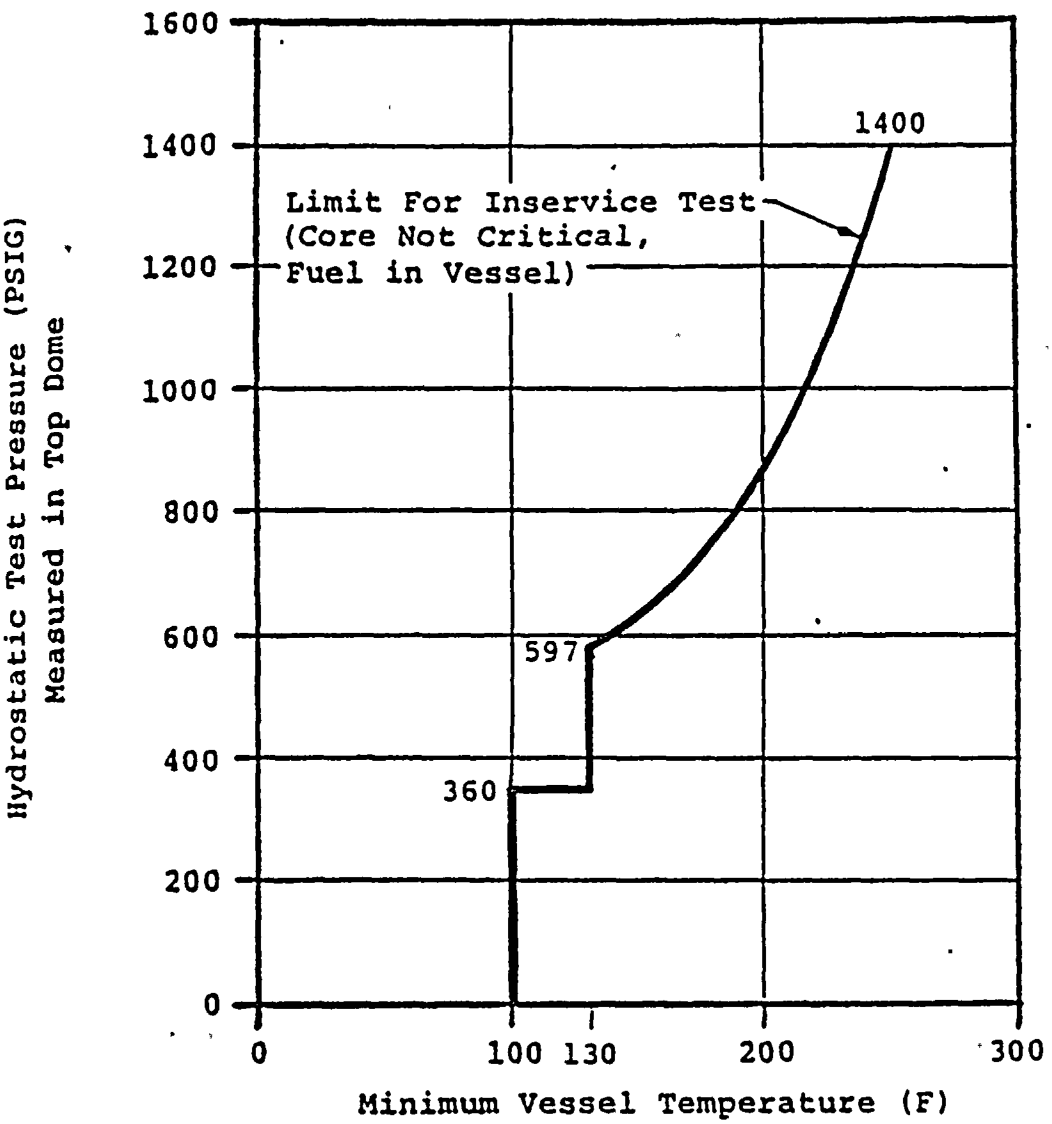

FIGURE 3.2.2.c

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING HYDROSTATIC TESTING (REACTOR NOT CRITICAL) FOR UP TO THIRTEEN EFFECTIVE FULL POWER YEARS OF CORE OPERATION

LIMIT FOR IN-SERVICE TEST
(CORE NOT CRITICAL, FUEL IN VESSEL)

PRESSURE (psig)	TEMPERATURE (F)
360	100-130
597	130
700	164
800	186
900	203
1000	216
1050	222
1100	228
1150	233
1200	237
1300	245
1400	253

TABLE 3.2.2.c

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING
HYDROSTATIC TESTING (REACTOR NOT CRITICAL)
FOR UP TO THIRTEEN EFFECTIVE FULL
POWER YEARS OF CORE OPERATION

BASES FOR 3.2.2 AND 4.2.2 MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION

Figures 3.2.2.a and 3.2.2.b are plots of pressure versus temperature for a heat-up and cool down rate of 100F/hr. maximum. (Specification 3.2.1). Figure 3.2.2.c is a plot of pressure versus temperature for hydrostatic testing. These curves are based on calculations of stress intensity factors according to Appendix G of Section III of the ASME Boiler and Pressure Vessel Code 1980 Edition with Winter 1982 Addenda. In addition, temperature shifts due to integrated neutron flux at thirteen effective full power years of operation were incorporated into the figures. These shifts were calculated from the formula presented in Regulatory Guide 1.99, proposed Revision 2. These curves are applicable to the beltline region at low and elevated temperatures and the vessel flange at intermediate temperatures. Reactor vessel flange/reactor head flange boltup is governed by other criteria as stated in Specification 3.2.2.d. The pressure readings on the figures have been adjusted to reflect the calculated elevation head difference between the pressure sensing instrument locations and the pressure sensitive area of the core beltline region.

The reactor head flange and vessel flange in combination with the double "O" ring type seal are designed to provide a leak-tight seal when bolted together. When the head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flanges. Both the head and vessel flanges have a NDT temperature of 40F and they are not subject to any appreciable neutron radiation exposure. Therefore, the minimum vessel flange and head flange temperature for bolting is established as 40 + 60F or 100F.

Figures 3.2.2.a., 3.2.2.b. and 3.2.2.c. have incorporated a temperature shift due to the calculated integrated neutron flux. The integrated neutron flux at the vessel wall is calculated from core physics data and has been measured using flux monitors installed inside the vessel. The curves are applicable for up to thirteen effective full power years of operation.

Vessel material surveillance samples are located within the core region to permit periodic monitoring of exposure and material properties relative to control samples. The material sample program conforms with ASTM E185-66 except for the material withdrawal schedule which is specified in Specification 4.2.2.b.

LIMITING CONDITION FOR OPERATION

3.2.3 COOLANT CHEMISTRY

Applicability:

Applies to the reactor coolant system chemical requirements.

Objective:

To assure the chemical purity of the reactor coolant water.

Specification:

a. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.2.3c:

Conductivity	2 μmho/cm
Chloride ion	0.1 ppm

b. The reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour except as specified in 3.2.3c:

Conductivity	5 μmho/cm
Chloride ion	0.2 ppm

SURVEILLANCE REQUIREMENT

4.2.3 COOLANT CHEMISTRY

Applicability:

Applies to the periodic testing requirements of the reactor coolant chemistry.

Objective:

To determine the chemical purity of the reactor coolant water.

Specification:

Samples shall be taken and analyzed for conductivity and chloride ion content at least 3 times per week with a maximum time of 96 hours between samples. In addition, if the conductivity becomes abnormal (other than short term spikes) as indicated by the continuous conductivity monitor, samples shall be taken and analyzed within 8 hours and daily thereafter until conductivity returns to normal levels

When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken and analyzed for conductivity and chloride ion content at least once per 8 hours.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. The limits specified in 3.2.3a and 3.2.3b may be exceeded for a period of time not to exceed 24 hours. In no case shall (1) the conductivity exceed a maximum limit of 10 μmho/cm, or (2) the chloride ion concentration exceed a maximum limit of 0.5 ppm. d. If Specifications 3.2.3.a, b, and c are not met, normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours. e. If the continuous conductivity monitor is inoperable for more than 7 days the reactor shall be placed in the cold shutdown condition within 24 hours.	

BASES FOR 3.2.3 AND 4.2.3 COOLANT CHEMISTRY

Materials in the primary system are primarily 304 stainless steel and the Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. When the steaming rate is less than 100,000 pounds per hour, a more restrictive limit of 0.1 ppm has been established. At steaming rates of at least 100,000 pounds per hour, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

A short term spike is defined as a rise in conductivity such as that which could arise from injection of additional feedwater flow for a duration of approximately 30 minutes in time.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt, e.g., Na_2SO_4, which would not have an affect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include, operation of the reactor clean-up system, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the clean-up system to re-establish the purity of the reactor coolant. During start-up periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho (other than short term spikes), samples will be taken to assure that the chloride concentration is less than 0.1 ppm.

BASES FOR 3.2.3 AND 4.2.3 COOLANT CHEMISTRY

The conductivity at the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. However, if the conductivity changes significantly, chloride measurements will be made to assure that the chloride limits of Specification 3.2.3 are not exceeded.

LIMITING CONDITION FOR OPERATION

3.2.4 REACTOR COOLANT ACTIVITY

Applicability:

Applies to the limits on reactor coolant activity at all operating conditions.

Objective:

To assure that in the event of a reactor coolant system line break outside the drywell permissible doses are not exceeded.

Specification:

a. The reactor coolant system radioactivity concentration in water shall not exceed 25 microcuries of total iodine per gram of water.

b. If Specification 3.2.4 a, above, cannot be met after a routine surveillance check, the reactor shall be placed in the cold shutdown condition within ten hours.

SURVEILLANCE REQUIREMENT

4.2.4 REACTOR COOLANT ACTIVITY

Applicability:

Applies to the periodic testing requirements of the reactor coolant activity.

Objective:

To assure that limits on coolant activity are not exceeded.

Specification:

a. Samples shall be taken at least every 96 hours and analyzed for gross gamma activity.

b. Isotopic analyses of samples shall be made at least once per month.

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BASES FOR 3.2.4 AND 4.2.4 REACTOR COOLANT ACTIVITY

The primary coolant radioactivity concentration limit of 25 μCi total iodine per gram of water was calculated based on a steamline break accident which is isolated in 10.5 seconds. For this accident analysis, all the iodine in the mass of coolant released in this time period is assumed to be released to the atmosphere at the top of the turbine building (30 meters). By limiting the thyroid dose at the site boundary to a maximum of 30 Rem, the iodine concentration in the primary coolant is back-calculated assuming fumigation meteorology, Pasquill Type F at 1 m/sec. The iodine concentration in the primary coolant resulting from this analysis is 25 μCi/gm.

A radioactivity concentration limit of 25 μCi/g total iodine could only be reached if the gaseous effluents were near the limit based on the assumed effluent isotopic content (Table A-12 of the FSAR) and the fact that the primary coolant cleanup systems were inoperative. When the cleanup system is operating, it is expected that the primary coolant radioactivity would be about 12 μCi/g total iodine. The concentrations expected during operations with a gaseous effluent of about 0.1μCi/sec would be about 1.5 μCi/g total iodine.

The reactor water sample will be used to assure that the limit of Specification 3.2.4 is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, as discussed in the bases for Specification 3.6.2, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

3.2.5 REACTOR COOLANT SYSTEM LEAKAGE

Applicability:

Applies to the limits on reactor coolant system leakage rate and leakage detection systems.

Objective:

To assure that the makeup capability provided by the control rod drive pump is not exceeded.

Specification:

a. Any time irradiated fuel is in the reactor vessel and the reactor temperature is above 212°F, reactor coolant leakage into the primary containment shall be limited to:

   1. Five gallons per minute unidentified leakage.

   2. A two gallon per minute increase in unidentified leakage within any period of 24 hours or less.

   3. Twenty-five gallons per minute total leakage (identified plus unidentified) averaged over any 24 hour period.

4.2.5 REACTOR COOLANT SYSTEM LEAKAGE

Applicability:

Applies to the monitoring of reactor coolant system leakage.

Objective:

To determine the reactor coolant system leakage rate and assure that the leakage limits are not exceeded.

Specification:

a. A check of the reactor coolant leakage shall be made every four hours.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, at least one of the leakage measurement channels associated with each sump (one for the drywell floor drain and one for the equipment drain) shall be operable. If conditions a or b cannot be met, the reactor will be placed in the cold shutdown condition within 24 hours.	b. The following surveillance shall be performed on each leakage detection system: 1. An instrument calibration once each refueling outage. 2. An instrument functional test once every three months.

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.2.5 on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.2.5, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage of the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the plant should be shut down to allow further investigation and corrective action.

Inspection and corrective action is initiated when unidentified leakage increases at a rate in excess of 2 gpm, within a 24 hour period or less. This minimizes the possibility of excessive propagation of intergranular stress corrosion cracking.

A total leakage of 25 gpm is well within the capacity of the control rod drive system makeup capability (page III-7 of the First Supplement).* As discussed in 3.1.6 above, for leakages within this makeup capability, the core will remain covered and automatic pressure blowdown will not be actuated.

The primary means of determining the reactor coolant leakage rate is by monitoring the rate of rise in the levels of the drywell floor and equipment drain lines. Checks will be made every four hours to verify that no alarms have been actuated due to high leakage. For sump inflows of one gpm, changes on the order of 0.2 gpm can be detected within 40 minutes. At inflows between one and five gpm, changes on the order of 0.5 gpm can be detected in eight minutes.

*FSAR

Leakage is detected by having all unidentified leakage routed to the drywell floor drain tank and identified leakage routed directly to the drywell equipment drain tanks. Identified leakage includes such items as recirculation pump seal leakage and recirculation pump suction and discharge valve packing leakoff.

Another method will monitor the time required to fill the tanks between two accurately determined levels. When the level in the tank reaches the low-level switch setting, a timer will start and operate for a preset time interval. If the timer resets before the high-level switch setting is reached indicating a leakage rate within allowable limits, no action will result, and the system resets for the next filling and timing cycle. If the leakage is high enough to cause the level to reach the high level switch setting before the timer resets automatically, an alarm is actuated indicating leak rate above the predetermined limit (First and Fifth Supplements).*

Additional information is available to the operator which can be used for the shift leakage check if the drywell sumps level alarms are out of service. The integrated flow pumped from the sumps to the waste disposal system can be checked.

Qualitative information is also available to the operator in the form of indication of drywell atmospheric conditions. Continuous leakage from the primary coolant system would cause an increase in drywell temperature. Any leakage in excess of 15 gpm of steam would cause a continuing increase in drywell pressure with resulting scram (First Supplement).*

Either the rate of rise leak detection system, the timer leak detection system or the integrated flow can be utilized to satisfy Specification 3.2.5.b.

*FSAR

LIMITING CONDITION FOR OPERATION

3.2.6 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To assure the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

a. Inservice Inspection

1. To be considered operable, Quality Group A, B and C components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2 and 3 components, respectively, except where relief has been granted by the Commission pursuant to 10CFR50, Section 50.55a(g)(6)(i).

SURVEILLANCE REQUIREMENT

4.2.6 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to periodic inspection and testing of components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To verify the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

a. Inservice Inspection

1. Inservice inspection of Quality Group A, B and C components shall be performed in accordance with the requirements for ASME Code Class 1, 2 and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10CFR part 50 Section 50.55a(g)(6)(i).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

2. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel and sample expansion included in this generic letter.

b. Inservice Testing

1. To be considered operable, Quality Group A, B and C pumps and valves, shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2 and 3 components, respectively, except where relief has been granted by the Commission pursuant to 10CFR50, Section 50.55a(g)(6)(i).

c. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

b. Inservice Testing

1. Inservice testing of Quality Group A, B and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2 and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10CFR Part 50, Section 50.55a(g)(6)(i).

BASES FOR 3.2.6 AND 4.2.6 INSERVICE INSPECTION AND TESTING

The inservice inspection and testing programs[1][2] for the Nine Mile Point Unit 1 plant conform to the requirements of 10CFR50, Section 50.55a(g). Where practical, the inspection of components, pumps and valves classified into NRC Quality Groups A, B and C conforms to the requirements of ASME Code Class 1, 2 and 3 components, pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code. If a Code required inspection is impractical for the Nine Mile Point Unit 1 facility, a request for relief from that requirement is submitted to the Commission in accordance with 10CFR50, Section 50.55a(g)(6)(i).

Request for relief from the requirements of Section XI of the ASME Code and applicable Addenda will be submitted to the Commission prior to the beginning of each 10-year inspection interval if they are known to be required at the time. Requests for relief which are identified during the course of inspection will be submitted quarterly throughout the inspection interval.

The inservice inspection program for piping conforms to the staff positions on schedules, methods, personnel and sample expansion contained in Generic Letter 88-01.[3] It is performed in order to detect and survey intergranular stress corrosion cracking of BWR austenitic stainless steel piping that is four inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation. Inspections shall be performed by individuals qualified to: (A) the ASME Boiler and Pressure Vessel Code, Section XI, and (B) Ultrasonic Testing Operator Training for the Detection of Intergranular Stress Corrosion Cracking developed by the EPRI Non-Destructive Examination Center. As an alternate, Niagara Mohawk may use other qualification programs approved by the NRC.

References

(1) Letter from the Nuclear Regulatory Commission (D. B. Vassallo) to Niagara Mohawk Power Corporation (G. K. Rhode), dated September 19, 1983.

(2) Letter from Niagara Mohawk Power Corporation (D. P. Dise) to the Nuclear Regulatory Commission (T. A. Ippolito), dated August 7, 1981.

(3) Generic Letter 88-01 endorses NUREG 0313 Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," dated January 1988.

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LIMITING CONDITION FOR OPERATION

3.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines connected to the reactor coolant system.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system.

Specification:

a. During power operating conditions whenever the reactor head is on, all reactor coolant system isolation valves on lines connected to the reactor coolant system shall be operable except as specified in "b" below.

b. In the event any isolation valve becomes inoperable the system shall be considered operable provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

SURVEILLANCE REQUIREMENT

4.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the periodic testing requirement for the reactor coolant system isolation valves.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system.

Specification:

The reactor coolant system isolation valves surveillance shall be performed as indicated below (see Table 3.2.7).

a. At least once per operating cycle the operable automatically initiated power-operated isolation valves shall be tested for automatic initiation and closure times.

b. At least once per quarter all normally open power-operated isolation valves (except the feedwater and main-steam-line power-operated isolation valves) shall be fully closed and reopened.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specifications 3.2.7a and b above are not met, initiate normal orderly shutdown within one hour and have reactor in the cold shutdown condition within ten hours.	c. <u>At least once per quarter</u> the feedwater and main-steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.
	d. <u>At least once per plant cold shutdown</u> the feedwater and main steam line power-operated isolation valves shall be fully closed and reopened, unless this test has been performed within the previous 92 days.

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JUL 3 1990

LIMITING CONDITIONS FOR OPERATION
Table 3.2.7

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power	Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Main Steam (Two Lines)	1	Inside	Open	AC Motor	10	Close	Reactor water level low-low, or main steam line high radiation, or main steam line high flow, or low condenser vacuum, or high temperature in the steam tunnel
	1	Outside	Open	Pn/DC Solenoid	10	Close	
Emergency Cooling Steam Line Drain to Main Steam (Two Lines)	2	Outside	Open	Pn/DC Solenoid	10	Close	
Emergency Cooling High Point Vent to Main Steam (One Line)	2	Outside	Open	Pn/DC Solenoid	10	Close	
Feedwater (Two Lines)	1	Outside	Open	AC Motor	60	-	-
	1	Outside	-	Self Act. Ck.	--	-	-
Emergency Cooling							
Steam Leaving Reactor (Two Lines)	1	Outside	Open	AC Motor	38	Close	High system flow
	1	Outside	Open	DC Motor	38	Close	
Condenser Return to Reactor (Two Lines)	1	Inside	-	Self Act. Ck.	--	-	
	1	Outside	Closed	Pn/DC Solenoid	60	Close	

NOTES:

(1) Pn - Pneumatically operated.

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LIMITING CONDITIONS FOR OPERATION
Table 3.2.7 (Continued)

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power	Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Reactor Cleanup							
Water Leaving Reactor (One Line)	1	Inside	Open	AC Motor	18	Close	Reactor water level low-low, or high area temperature, liquid poison initiation or high system pressure, or low system flow, or high system temperature
	1	Outside	Open	DC Motor	18	Close	
Water Return to Reactor (One Line)	1	Inside	Open	AC Motor	18	Close	
	1	Outside	-	Self Act. Ck.	--	-	
Shutdown Cooling							
Water Leaving Reactor (One Line)	1	Inside	Closed	AC Motor	40	Close	Reactor water level low-low, or high area temperature
	1	Outside	Closed	DC Motor	40	Close	
Water Return to Reactor (One Line)	1	Inside	Closed	AC Motor	40	Close	
	1	Outside	-	Self Act. Ck.	--	-	

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LIMITING CONDITIONS FOR OPERATION
Table 3.2.7 (Continued)

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Reactor Head Spray (One Line)	1	Inside	-	Self Act. Ck.	--	-	--
	1	Outside	Closed	R.M.P.O.	30	-	--
Liquid Poison (One Line)	1	Inside	-	Self Act. Ck.	--	-	--
	1	Outside	-	Self Act. Ck.	--	-	--
Control Rod Drive Hydraulic (One Line)	1	Inside	-	Self Act. Ck.	--	-	--
	1	Outside	-	Self Act. Ck.	--	-	--
Scram Discharge System Vent** (One Line)	2	Outside	Open	A.I.A.O.	10	Close	Automatic or manual reactor scram.
Scram Discharge System Drain** (One Line)	2	Outside	Open	A.I.A.O.	10	Close	
Core Spray High Point Vent (Two Lines)	1	Inside	Closed	AC Motor	30	Close	Reactor water level low-low or high drywell pressure.
	1	Outside	Closed	Air/DC Solenoid	30	Close	

* R.M.P.O. - Remote Manual Power Operated.
A.I.A.O. - Automatically Initiated Air Operated.

** See 3.1.1e for LCO requirements.

BASES FOR 3.2.7 AND 4.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Double isolation valves are provided in lines which connect to the reactor coolant system to assure isolation and minimize reactor coolant loss in the event of a line rupture. The specified valve requirements assure that isolation is already accomplished with one valve shut or provide redundancy in an open line with two operative valves. Except where check valves are used as one or both of a set of double isolation valves, the isolation valves shall be capable of automatic initiation and the closure times presented in Table 3.2.7. These closure times were selected to minimize coolant losses in the event of the specific line rupturing. Using the longest closure time on the main-steam-line valves following a main-steam-line break (Appendix E-II.1.0*), the core is still covered by the time the valves close. Following a specific system line break, the cleanup and shutdown cooling closing times will upon initiation from a low-low level signal limit coolant loss such that the core is not uncovered. Feedwater flow would quickly restore coolant levels to prevent clad damage. Closure times are discussed in Section VI-C.1.0*.

The valve operability test intervals are based on periods not likely to significantly affect operations, and are consistent with testing of other systems. Results obtained during closure testing are not expected to differ appreciably from closure times under accident conditions as in most cases, flow helps to seal the valve.

The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} (Fifth Supplement, p. 115)* that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

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LIMITING CONDITION FOR OPERATION

3.2.7.1 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Applicability:

Applies to the operating status of isolation valves for systems connected to the primary coolant system.

Objective:

To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

Specification:

a. The integrity of all pressure isolation valves listed in Table 3.2.7.1 shall be demonstrated. Valve leakage shall not exceed the amounts indicated.

b. If Specification a cannot be met, an orderly shutdown shall be initiated within 1 hour and the reactor shall be in the cold shutdown condition within 10 hours.

SURVEILLANCE REQUIREMENT

4.2.7.1 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Applicability:

Applies to the periodic testing of primary coolant system pressure isolation valves.

Objective

To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

Specification:

a. Periodic leakage testing[a] on each valve listed in Table 3.2.7.1 shall be accomplished prior to exceeding 2% power while in the power operating condition every time the plant is placed in a cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

[a] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 3.2.7.1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[a] Allowable Leakage
1. Core Spray System	40-03	$\leq$5.0 gpm
	40-13	$\leq$5.0 gpm
2. Condensate Supply to Core Spray (Keep Fill System)	40-20	$\leq$5.0 gpm
	40-21	$\leq$5.0 gpm
	40-22	$\leq$5.0 gpm
	40-23	$\leq$5.0 gpm

Footnote:

(a)

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable
2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amoun that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. Leakage rates greater than 5.0 gpm are considered unacceptable.
5. Test differential pressure shall not be less than 150 psid.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES	4.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES
Applicability:	Applicability:
Applies to the operational status of the safety valves.	Applies to the periodic testing requirements for the safety valves.
Objective:	Objective:
To assure the capability of the safety valves to limit reactor overpressure below the safety limit in the event of rapid reactor isolation and failure of all pressure relieving devices.	To assure the capability of the safety valves to limit reactor overpressure to below the safety limit.
Specification:	Specification:
a. During power operating conditions and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than saturation temperature all sixteen of the safety valves shall be operable.	At least once during each operating cycle at least eight of the sixteen safety valves shall be removed, tested for set point and partial lift, and then returned to operation or replaced.
b. If specification 3.2.8a is not met, the reactor coolant pressure and temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.	

BASES FOR 3.2.8 AND 4.2.8 PRESSURE RELIEF SYSTEM-SAFETY VALVES

The required number of operable safety valves is based on the analysis presented in Appendix E-I.3.7* which assumed reactor isolation with no scram. Operation of all 16 safety valves will limit reactor pressure below the safety limit of 1375 psig. Partial redundancy is provided by the solenoid-actuated pressure relief valves as the relieving capacity of each of these valves is approximately the same as a safety valve, as discussed in 2.2.2 above.

The safety valve testing and intervals between tests are based on manufacturer's recommendations and past experience with spring actuated safety valves.

*FSAR

3.2.9 PRESSURE RELIEF SYSTEMS - SOLENOID-ACTUATED PRESSURE RELIEF VALVES (OVERPRESSURIZATION)

Applicability:

Applies to the operational status of the solenoid-actuated pressure relief valves.

Objective:

To assure the capability of the solenoid-actuated pressure relief valves to limit reactor overpressure below the lowest safety valve setpoint in the event of rapid reactor isolation.

Specification:

a. During the power operating condition and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than saturation, five of the six solenoid-actuated pressure relief valves shall be operable.

b. If Specification 3.2.9a is not met, the reactor coolant pressure and temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.

4.2.9 PRESSURE RELIEF SYSTEMS - SOLENOID-ACTUATED PRESSURE RELIEF VALVES (OVERPRESSURIZATION)

Applicability:

Applies to the periodic testing requirements for the solenoid-actuated pressure relief valves.

Objective:

To assure the operability of the solenoid-actuated pressure relief valves to limit reactor overpressure in the event of rapid reactor isolation.

Specification:

The solenoid-actuated pressure relief valve surveillance shall be performed as indicated below.

a. The setpoints of the six relief valves shall be as follows:

No. of Valves	Setpoint
2	$\leq$ 1090 psig
2	$\leq$ 1095 psig
2	$\leq$ 1100 psig

4.2.9 PRESSURE RELIEF SYSTEMS - SOLENOID-ACTUATED PRESSURE RELIEF VALVES (OVERPRESSURIZATION)

Specification: (Continued)

b. At least once during each operating cycle with the reactor at pressure, each valve shall be manually opened until acoustic monitors or thermocouples downstream of the valve indicate that the valve has opened and steam is flowing from the valve.

c. At least once during each operating cycle, relief valve setpoints shall be verified.

BASES FOR 3.2.9 AND 4.2.9 PRESSURE RELIEF SYSTEM - SOLENOID ACTUATED PRESSURE RELIEF VALVES

As discussed in 2.2.2 and 3.2.8 above, the solenoid-actuated pressure relief valves are used to avoid actuation of the safety valves. The set points of the six relief valves are staggered. Two valves are set at 1090 psig, two are set at 1095 psig, and two are set at 1100 psig. The operator will endeavor to place the set-point at these figures. However, a set-point error for each valve can be as much as ±12 psig.

Six valves are provided for the automatic depressurization function, as described in 3.1.5. However, only five valves are required to prevent actuation of the safety valves, as discussed in the Technical Supplement to Petition to Increase Power Level, Section II.XV, letter, T. J. Brosnan to Peter A. Morris dated February 28, 1972, and letter, Philip D. Raymond to A. Giambusso, dated October 15, 1973.

The basis for the surveillance requirement is given in 4.1.5.

3.3.0 PRIMARY CONTAINMENT

APPLICABILITY

Applies to the operating status of the primary containment systems.

OBJECTIVE

To assure the integrity of the primary containment systems.

SPECIFICATION

Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 215F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mwt.

LIMITING CONDITION FOR OPERATION

3.3.1 OXYGEN CONCENTRATION

Applicability:

Applies to the limit on oxygen concentration within the primary containment system.

Objective:

To assure that in the event of a loss-of-coolant accident any hydrogen generation will not result in a combustible mixture within the primary containment system.

Specification:

a. The primary containment atmosphere shall be reduced to less than four percent by volume oxygen concentration with nitrogen gas whenever the reactor coolant pressure is greater than 110 psig and the reactor is in the power operating condition, except as specified in "b" below.

SURVEILLANCE REQUIREMENT

4.3.1 OXYGEN CONCENTRATION

Applicability:

Applies to the periodic testing requirem for the primary containment system oxygen concentration.

Objective:

To assure that the oxygen concentration within the primary containment system is within required limits.

Specification:

At least once a week oxygen concentration shall be determined.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. Within the 24-hour period subsequent to the reactor being placed in the run mode for the power operating condition, the containment atmosphere oxygen concentration shall be reduced to less than four percent by volume, and maintained in this condition. Deinerting may commence 24 hours prior to a major refueling outage or other scheduled shutdown. c. If Specifications "a" or "b" above are not met, the reactor coolant pressure shall be reduced to 110 psig or less within ten hours.	

The four percent by volume oxygen concentration eliminates the possibility of hydrogen combustion following a loss-of-coolant accident (Section VII-G.2.0 and Appendix E-11.5.2)*. The only way that significant quantities of hydrogen could be generated would be if all core spray systems failed to sufficiently cool the core. As discussed in Section VII-A.2.0 and illustrated in Figure VII-2,* the core spray system is capable of design flow of 3400 gpm at a reactor pressure of 113 psig. In addition to hydrogen generated by metal-water reaction, significant quantities can be generated by radiolysis. (Technical Specification to Petition for Conversion from Provisional Operating License to Full Term Operating License).

At reactor pressures of 110 psig or less, the reactor will have been shutdown for more than an hour and the decay heat will be at sufficiently low values so that fuel rods will be completely wetted by core spray. The fuel clad temperatures would not exceed the core spray water saturation temperature of about 344F.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be reasonable to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase the oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week, the oxygen concentration will be determined as added assurance that Specification 3.3.1 is being met.

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LIMITING CONDITION FOR OPERATION

3.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the interrelated parameters of pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the peak suppression chamber pressure does not exceed design values in the event of a loss-of-coolant accident.

Specification:

a. The downcomers in the suppression chamber shall have a minimum submergence of three feet and a maximum submergence of four and one half feet whenever the reactor coolant system temperature is above 215F.

b. During normal power operation, the combination of primary containment pressure and suppression chamber bulk pool temperature shall be within the shaded area of

SURVEILLANCE REQUIREMENT

4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the periodic testing of the pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the pressure suppression system pressure and suppression chamber water temperature and level are within required limits.

Specification:

a. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

b. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(1) Figure 3.3.2a when downcomer submergence is greater than or equal to 4 feet, or (2) Figure 3.3.2b when downcomer submergence is greater than or equal to 3 feet but less than 4 feet. If these temperatures are exceeded, pool cooling shall be initiated immediately.	c. Whenever heat from relief valve operation is being added to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.
c. If Specifications a and b above are not met within 24 hours, the reactor shall be shut down using normal shutdown procedures.	d. Whenever operation of a relief valve is indicated and the bulk suppression pool temperature reaches 160F or above while the reactor primary coolant system pressure is greater than 200 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.
d. During testing of relief valves which add heat to the torus pool, bulk pool temperature shall not exceed 10F above normal power operation limit specified in b above. In connection with such testing, the pool temperature must be reduced within 24 hours to below the normal power operation limit specified in b above.	e. Whenever there is indication of relief valve operation with the local temperature of the suppression pool reaching 200F or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.
e. The reactor shall be scrammed from any operating condition when the suppression pool bulk temperature reaches 110F. Operation shall not be resumed until the pool temperature is reduced to below the normal power operation limit specified in b above.	
f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool bulk temperature reaches 120F.	

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FIGURE 3.3.2 a

ALLOWABLE PRESSURE SUPPRESSION SYSTEM

4 FOOT DOWNCOMER SUBMERGENCE

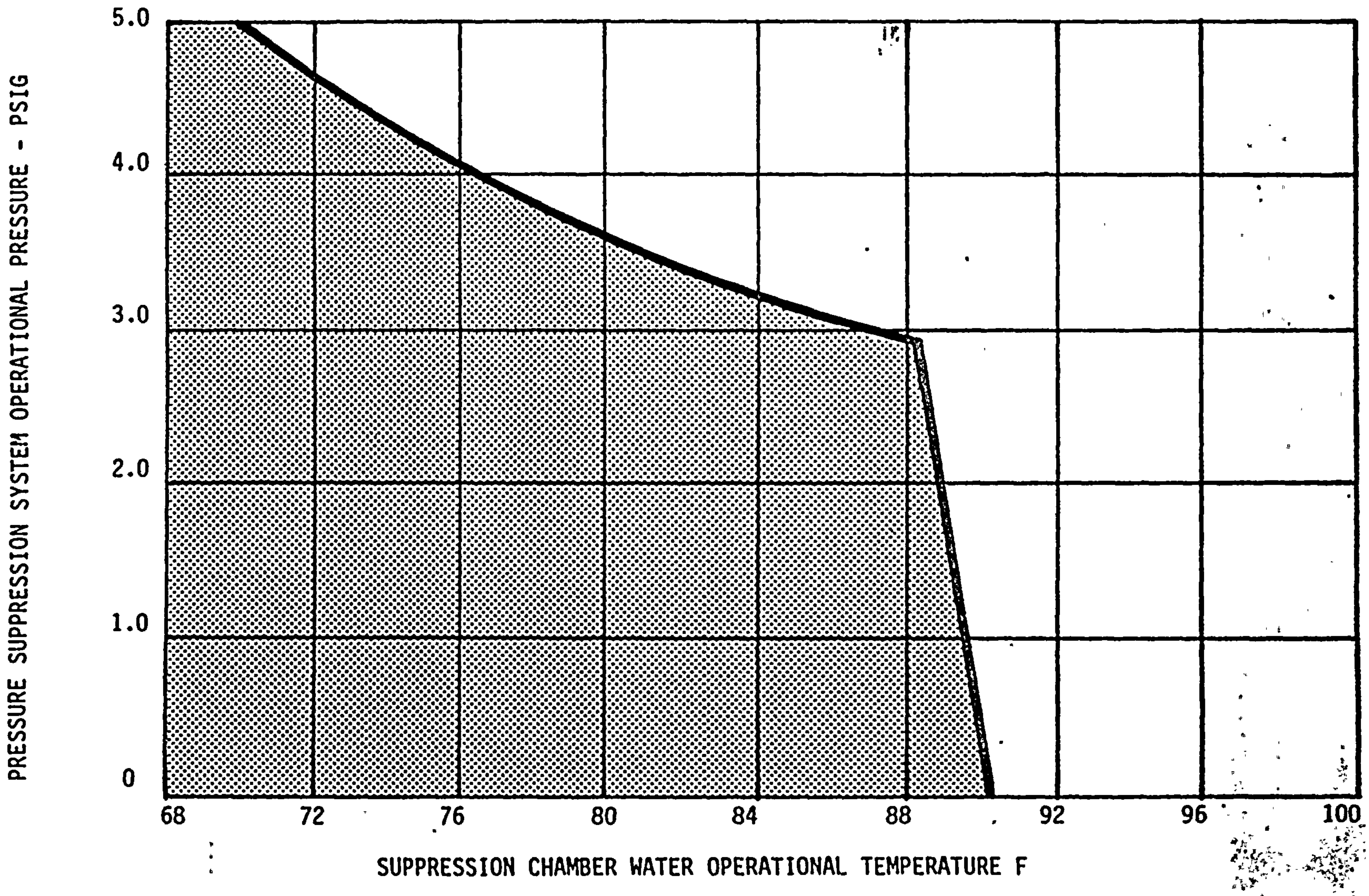
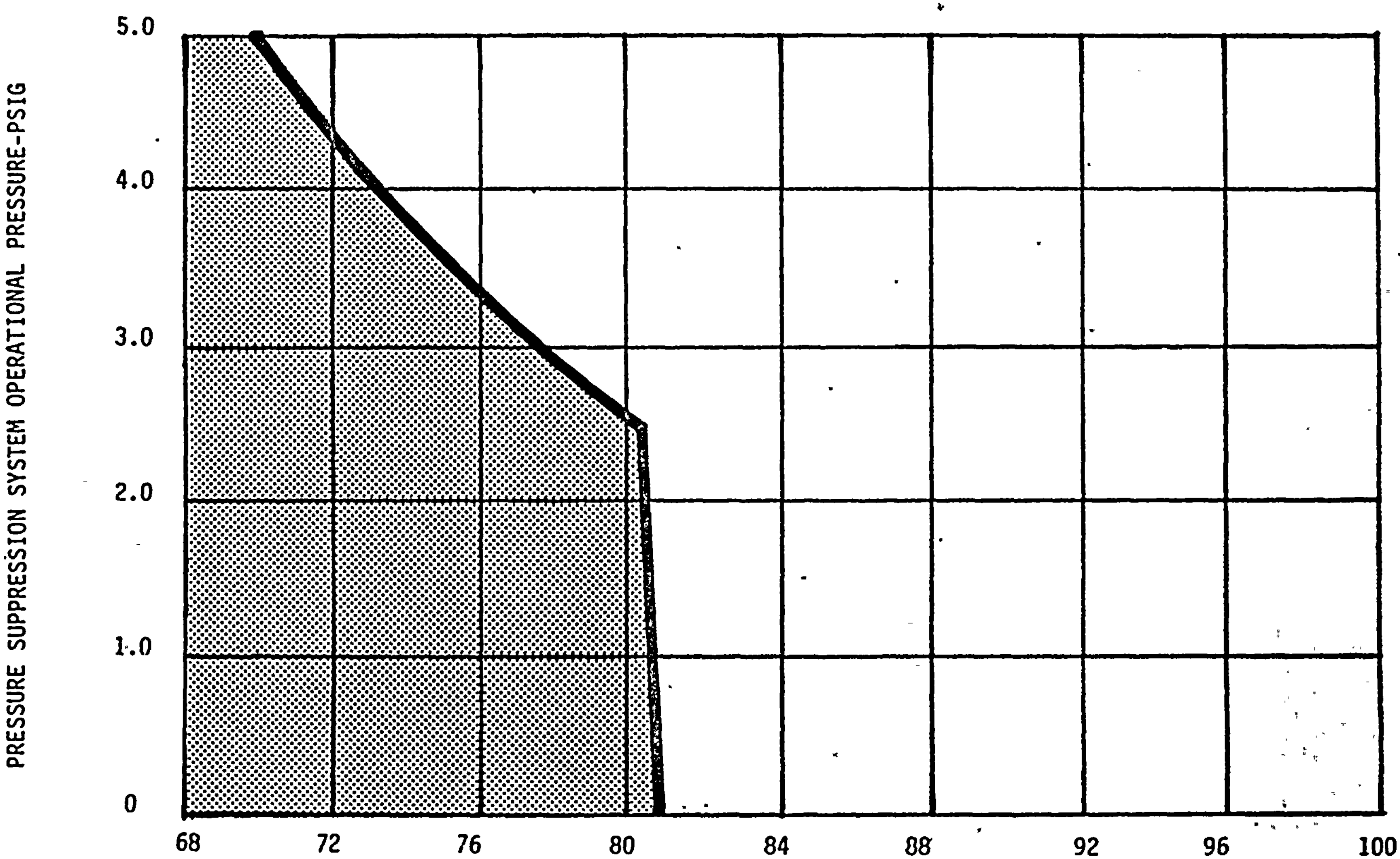


FIGURE 3.3.2 b

ALLOWABLE PRESSURE SUPPRESSION SYSTEM

3 FOOT DOWNCOMER SUBMERGENCE



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BASES FOR 3.3.2 AND 4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

The values specified for suppression chamber water temperature, maximum downcomer submergence, and system pressures are based on the effect these parameters have on the short-term post-accident system pressure following a loss-of-coolant accident. The combinations shown on Figures 3.3.2 a and b and the water level required are based on maintaining the post-accident pressure below the design value of 35 psig and the maximum suppression chamber water temperature below 140F in the containment design basis loss-of-coolant accident (Appendix E-11.2.2.3).*

The calculational basis for the pressure suppression system initial conditions, Figures 3.3.2 a and b are presented in the Fifth Supplement.*

The three foot minimum and the four and one-half foot maximum submergence are a result of the Mark I Containment Long Term Program.

The 215F limit for the reactor is specified, since below this temperature the containment can tolerate a blowdown without exceeding the 35 psig design pressure of the suppression chamber without condensation.

Actually, for reactor temperatures up to 312F the containment can tolerate a blowdown without exceeding the 35 psig design pressure of the suppression chamber, without condensation.

Some experimental data suggests that excessive steam condensing loads might be encountered if the bulk temperature of the suppression pool exceeds 160F during any period of relief valve operation with sonic conditions at the discharge exit. This can result in local pool temperatures in the vicinity of the quencher of 200F. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event of a relief valve inadvertently opens or sticks open. As a minimum, this action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings

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BASES FOR 3.3.2 AND 4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected 2
to be the points of highest stress.

Continuous monitoring of suppression chamber water level and temperature and pressure suppression system pressure is provided in the control room. Alarms for these parameters are also provided in the control room.

To determine the status of the pressure suppression system, inspections of the suppression chamber interior surfaces at each major refueling outage with water at its normal elevation will be made. This will assure that gross defects are not developing.

LIMITING CONDITION FOR OPERATION

3.3.3 LEAKAGE RATE

Applicability:

Applies to the allowable leakage rate of the primary containment system.

Objective:

To assure the capability of the containment in limiting radiation exposure to the public from exceeding values specified in 10 CFR 100 in the event of a loss-of-coolant accident accompanied by significant fuel cladding failure and hydrogen generation from a metal-water reaction.

Specification:

Whenever the reactor coolant system temperature is above 215 F the primary containment leakage rate shall be within the limits of 4.3.3.b.

SURVEILLANCE REQUIREMENT

4.3.3 LEAKAGE RATE

Applicability:

Applies to the primary containment system leakage rate.

Objective:

To verify that the leakage from the primary containment system is maintained within specified values.

Specification:

a. Integrated Primary Containment Leakage Rate Test

(1) Integrated leak rate tests shall be performed prior to initial Station operation at the test pressure of 35 psig (P_p) and the test pressure (P_t) of 22 psig to obtain the respective measured leak rates L_m (35) and L_m (22).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(2) Subsequent leakage rate tests shall be performed without preliminary leak detection surveys or leak repairs immediately prior to or during the test, at an initial pressure of approximately 22 psig.
	(3) Leak repairs, if necessary to permit integrated leakage rate testing, shall be preceded by local leakage measurements. The leakage rate difference, prior to and after repair when corrected to P_t shall be added to the final integrated leakage rate result.
	(4) Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.
	(5) After the containment test conditions have stabilized, the test duration shall not be less than eight hours for integrated leak rate measurements. The test shall be extended for sufficient duration, to verify by a supplemental test method the accuracy of the integrated leak rate test results.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

b. Acceptance Criteria

(1) The maximum allowable leakage rate L_p shall not exceed 1.5 weight percent of the contained air per 24 hours at the test pressure of 35 psig (P_p).

(2) The allowable test leak rate L_t (22) shall not exceed the value established as follows:

$$L_t\ (22) = 1.5\ L_m\ (22)/L_m\ (35)$$

(3) The allowable operational leak rate, L_{to} (22) which shall be met prior to resumption of power operation following a test (either as measured or following repairs and retest) shall not exceed 0.75 L_t (22).

c. Corrective Action

If leak repairs are necessary to meet the allowable operational leak rate, the integrated leak rate test need not be repeated provided local leakage measurements are conducted, and the leak rate differences prior to and after repairs, when corrected to P_t and deducted from the integrated leak rate measurement, yield a leakage rate value not in excess of the allowable operational leak rate L_{to} (22).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	d. <u>Frequency</u> Three integrated leak rate tests shall be performed at approximately equal intervals during each 10-year service period with the third test in each ten-year interval corresponding with the ten-year scheduled in-service inspection shutdown. e. <u>Local Leak Rate Tests</u> (1) Primary containment testable penetrations and isolation valves shall be tested at a pressure of 35 psig each major refueling outage except bolted double-gasketed seals shall be tested whenever the seal is closed after being opened, and at least at each refueling outage. (2) Personnel air lock door seals shall be tested once within 24 hours after opening when the reactor is in a power operating condition, at a pressure of 10 psig and the leak rate extrapolated to 35 psig. Air lock seals shall also be leak rate tested at a pressure of 35 psig at the beginning of each operating cycle. An additional 35 psig leak rate test shall be performed near the middle of the operating cycle should a shutdown requiring de-inerting arise. If the above shutdown does not occur or is not anticipated, the air lock seals will be

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

tested at 10 psig. In each test the leak rate corrected to 35 psig shall not exceed 5 percent L_a.

(3) Containment components not included in (1) and (2) which required leak repairs following any integrated leakage rates in order to meet the allowable leakage rate unit L_t shall be subjected to local leak tests at a pressure of 35 psig at each refueling outage.

(4) The main steam line isolation valves are to be tested at a pressure of 35 psig during each refueling outage.

f. <u>Corrective Action</u>

(1) If the total leakage rates listed below as adjusted to a test pressure of 22 psig are exceeded, repairs and retests shall be performed to correct the condition.

(a) double-gasketed seals

$10\%\ L_{to}\ (22)$

(b) (i) testable penetrations and isolation valves

$30\%\ L_{to}\ (22)$

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(ii) any one penetration or isolation valve 5% L_{to} (22) (c) primary containment air purge penetrations and reactor building to torus vacuum relief valves 50% L_{to} (22) g. Continuous Leak Rate Monitor (1) When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. (2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical. h. Inspection The accessible interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration.

BASES FOR 3.3.3 AND 4.3.3 LEAKAGE RATE

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be 35 psig which would rapidly reduce to 22 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 22 psig is calculated to be about 10 seconds. Following the pipe break, the suppression chamber pressure rises to 22 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay.[1]

The design pressures of the drywell and absorption chamber are 62 psig and 35 psig, respectively.[2] The design leak rate is 0.5%/day at a pressure of 35 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated primary containment pressure response discussed above and the suppression chamber design pressure; primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and a suppression chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.9%/day at 35 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90 percent for halogens, 95 percent for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 6.0 rem and the maximum total thyroid dose is about 150 rem at the site boundary considering fumigation conditions over an exposure duration of two hours. The resultant doses would occur for the duration of the accident at the low population distance of 4 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency (Specification 4.4.4) are conservative and provide margin between expected offsite doses and 10CFR100 guideline limits.

BASES FOR 3.3.3 AND 4.3.3 LEAKAGE RATE

The maximum allowable test leak rate as specified in 4.3.3.b is 1.5%/day at a pressure of 35 psig. This value for the test condition was derived from the maximum allowable accident leak rate of about 1.9%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air depleted of oxygen whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing.[3]

Although the dose calculations suggest that the allowable test leak rate could be allowed to increase to about 3.0%/day before the guideline thyroid dose limit given in 10CFR100 would be exceeded, establishing the limit at 1.5%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The operational limit is derived by multiplying the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the AEC guide for developing leak rate testing and surveillance of reactor containment vessels.[4]

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. If the leakage rates of the double-gasketed seal penetrations, testable penetration isolation valves, containment air purge inlets and outlets and the vacuum relief valves are at the maximum specified, they will total 90 percent of the allowed leak rate.[2] Hence, 10 percent margin is left for leakage through walls and untested components.

BASES FOR 3.3.3 AND 4.3.3 LEAKAGE RATE

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

(1) Appendix E, FSAR

(2) Volume 1, Section VI, FSAR

(3) TID-20583, Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations.

(4) 10CFR50 Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors".

LIMITING CONDITION FOR OPERATION

3.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines open to the free space of the primary containment.

Objective:

To assure that potential leakage paths from the primary containment in the event of a loss-of-coolant accident are minimized.

Specification:

a. Whenever the reactor coolant system temperature is greater than 215F, all containment isolation valves on lines open to the free space of the primary containment shall be operable except as specified in 3.3.4b below.

b. In the event any isloation valve becomes inoperable the system shall be considered operable provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

SURVEILLANCE REQUIREMENT

4.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the periodic testing requirements of the primary containment isolation valve system.

Objective:

To assure the operability of the primary containment isolation valves to limit potential leakage paths from the containment in the event of a loss-of-coolant accident.

Specification:

The primary containment isolation valves surveillance shall be performed as indicated (see Table 3.3.4)

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure times.

b. At least once per quarter all normally open power operated isolation valves shall be fully closed and reopened.

LIMITING CONDITION FOR OPERATION

c. If Specifications 3.3.4 a and b are not met, the reactor coolant system temperature shall be reduced to a value less than 215F within ten hours.

SURVEILLANCE REQUIREMENT

c. At least once per operating cycle, each instrument-line flow check valve will be tested for operability.

LIMITING CONDITION FOR OPERATION

Table 3.3.4

PRIMARY CONTAINMENT ISOLATION VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Drywell Vent & Purge							
N_2 Connection (One Line)	1 1	Outside Outside	Closed (a) Closed (a)	Air/D.C. Sol. A.C. Motor	60 60	Close Close	Reactor water level low-low or drywell high pressure
Air Connection (One Line)	1 1	Outside Outside	Closed (a) Closed (a)	Air/D.C. Sol. A.C. Motor	60 60	Close Close	
Suppression Chamber Vent & Purge							
N_2 Connection (One Line)	1 1	Outside Outside	Closed (a) Closed (a)	Air/D.C. Sol. A.C. Motor	60 60	Close Close	Reactor water level low-low or drywell high pressure
Air Connection (One Line)	1 1	Outside Outside	Closed (a) Closed (a)	Air/D.C. Sol. A.C. Motor	60 60	Close Close	
Drywell N_2 Makeup (One Line)	2	Outside	Closed (b)	Air/D.C. Sol.	60	Close	Reactor water level low-low or drywell high pressure
Suppression Chamber N_2 Makeup (One Line)	2	Outside	Closed (b)	Air/D.C. Sol.	60	Close	Reactor water level low-low or drywell high pressure
Drywell Equipment Drain Line (One Line)	1 1	Inside Outside	Open Open	A.C. Motor Air/D.C. Sol.	60 60	Close Close	Reactor water level low-low or drywell high pressure
Floor Drain Line (One Line)	1 1	Inside Outside	Open Open	A.C. Motor Air/D.C. Sol.	60 60	Close Close	
Suppression Chamber Water Makeup (One Line)	1 1	Outside Outside	Closed (b) -	A.C. Motor Self Act. Ck.	60 --	- -	Remote manual -
Vacuum Relief Atmosphere to Pressure Suppression System (Three Lines)	1 1	Outside Outside	Closed -	A.C. Motor Self Act. Ck.	5 --	Open -	Negative pressure relative to atmosphere -
Reactor Cleanup System Relief Valve Discharge (One Line to Suppression Chamber)	2	Outside	-	Self Act. Ck.	--	-	-

LIMITING CONDITIONS FOR OPERATION

Table 3.3.4 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
O_2 Sampling							
Drywell (Three Lines)	2	Outside	Open	D.C. Sol.	60	Close (b)	Reactor Water Level Low-Low or High Drywell Pressure
Suppression Chamber (One Line)	2	Outside	Open	D.C. Sol.	60	Close (b)	

NOTES: (a) These valves may be open for containment fill with nitrogen.

(b) These valves will periodically be opened for sampling or nitrogen makeup.

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Table 3.3.4 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Core Spray (c)							
Pump Suction (Four Lines from Suppression Chamber)	1	Outside	Open	AC Motor	90	-	Remote manual
Pump Discharge (Two Test Lines to Suppression Chamber)	1	Outside	Closed	AC Motor	90	Close	Reactor water level low-low
LINES WITH A CLOSED LOOP INSIDE CONTAINMENT VESSELS							
Recir. Pump Cooling Water Supply (c)							
Supply line	1	Outside	Open	Self Act. Ck.	--	-	--
Return line	1	Outside	Open	DC Motor	30	-	Remote manual
Drywell Cooler Water Supply (c)							
Supply line	1	Outside	Open	Self Act. Ck.	--	-	--
Return line	1	Outside	Open	DC Motor	30	-	Remote manual
LINES WITH A CLOSED LOOP OUTSIDE CONTAINMENT VESSELS							
Containment Spray (c)							
Drywell & Suppression Chamber Common Supply (four lines)	1	Outside	Open	Air/DC Sol.	60	-	Remote manual
Drywell Branch (four lines)	1	Outside	--	Self Act. Ck.	--	-	--
Suppression Chamber Branch (One Branch for Each System)	2*	Outside	--	Self Act. Ck.	--	-	--
Pump Suction From Suppression Chamber (four lines)	1	Outside	Open	AC Motor	70	-	Remote manual

* One valve in each separate line and one valve in each common line.
(c) These are classified as not-testable valves and penetrations.

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Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Except where check valves are used as one or both of a set of double isolation valves, the isolation closure times are presented in Table 3.3.4. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section VI-C.* For allowable leakage rate specification, see Section 3.3.3 above.

As illustrated in Figure E-34 of Appendix E* fuel rod perforation does not occur until about 150 seconds following the loss-of-coolant accident. A required closing time of 60 seconds for all primary containment isolation valves will be adequate to prevent fission product release through lines connecting to the primary containment.

For reactor coolant system temperatures less than 312F, the containment could not become pressurized due to a loss-of-coolant accident. The 215F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels.

The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate (Fifth Supplement, p. 115).* More frequent testing for valve operability results in a more reliable system.

In addition to routine surveillance as outlined in First Addendum to Technical Supplement to Petition to increase Power Level, each instrument-line flow check valve will be tested for operability. All instruments on a given line will be isolated at each instrument. The line will be purged by isolating the flow check valve, opening the bypass valves, and opening the drain valve to the equipment drain tank. When purging is sufficient to clear the line of non-condensibles and crud the flow-check valve will be cut into service and the bypass valve closed. The main valve will again be opened and the flow-check valve allowed to close. The flow-check valve will be reset by closing the drain valve and opening the bypass valve depressurizing part of the system. Instruments will be cut into service after closing the bypass valve. Repressurizing of the individual instruments assures that flow-check valves have reset to the open position.

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BASES 3.3.5 AND 4.3.5 ACCESS CONTROL

Access to the containment during operation is expected to be infrequent. However, each door of the two double-doored access locks is designed to withstand 62 psig drywell pressure. It is, therefore, possible to open one door at a time and still maintain containment integrity. Access door design is discussed in Section VI-A 2.2 of the FSAR.

The equipment hatch and drywell head and other flanged openings are provided with double "O" rings and must be secure in order to maintain the integrity of the primary containment system. Maintaining the pressure suppression system integrity when above the stated pressure and temperature will ensure that a reactor coolant system rupture will not result in an overpressurization of the reactor building.

LIMITING CONDITION FOR OPERATION

3.3.6 VACUUM RELIEF

Applicability:

Applies to the operational status of the primary containment vacuum relief system.

Objective:

To assure the capability of the vacuum relief system in the event of a loss-of-coolant accident to:

a. Equalize pressures between the drywell and suppression chamber, and

b. Maintain containment pressure above the vacuum design values of the drywell and suppression chamber.

Specification:

a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated above. Suppression chamber - drywell vacuum breakers shall be considered operable if:

(1) The valve is demonstrated to open fully with the applied force at all valve positions

SURVEILLANCE REQUIREMENT

4.3.6 VACUUM RELIEF

Applicability:

Applies to the periodic testing of the vacuum relief system.

Objective:

To assure the operability of the containment vacuum relief system to perform its intended functions.

Specification:

a. Periodic Operability Tests

Once each month and following any release of energy to the suppression chamber, each suppression chamber - drywell vacuum breaker shall be exercised. Operability of valves, position switches, and position indicators and alarms shall be verified monthly and following any maintenance on the valves and associated equipment.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk. (2) The valve can be closed by gravity, when released after being opened by remote or manual means, to within not greater than the equivalent of 0.06 inch at the bottom of the disk. (3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 0.06 inch at the bottom of the disk. b. Any drywell-suppression chamber vacuum breaker may be non-fully closed as indicated by the position indication and alarm systems provided that drywell to suppression chamber differential pressure decay rate is demonstrated to be not greater than 25% of the differential pressure decay rate for all vacuum breakers open the equivalent of 0.06 inch at the bottom of the disk.	b. <u>Refueling Outage Tests</u> (1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open. (2) All suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (3) Once each operating cycle, each vacuum breaker valve shall be visually inspected to ensure proper maintenance and operation. (4) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of the leakage rate through a 1-inch orifice. c. <u>Pressure Suppression Chamber - Reactor Building Vacuum Breakers</u> (1) The pressure suppression chamber-reactor building vacuum breaker

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. When it is determined that one or more vacuum breaker valves are not fully closed as indicated by the position indication system at a time when such closure is required, the apparently malfunctioning vacuum breaker valve shall be exercised and pressure tested as specified in 3.3.6 b immediately and every 15 days thereafter until appropriate repairs have been completed.

d. One drywell-suppression chamber vacuum breaker may be secured in the closed position.

e. If Specifications 3.3.6 a, b, c, or d cannot be met, the situation shall be corrected within 24 hours or the reactor shall be placed in a cold shutdown condition within 24 hours.

f. <u>Pressure Suppression Chamber - Reactor Building Vacuum Breakers</u>

(1) The three pressure suppression chamber reactor building vacuum breaker systems shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates

systems and associated instrumentation, including set point, shall be checked for proper operation every three months.

(2) During each refueling outage, each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.3.6.f(1) and each vacuum breaker shall be inspected and verified to meet design requirement.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
the pressure suppression chamber-reactor building air-operated vacuum breakers shall be ≤ 0.5 psid. The self-actuating vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk. (2) From and after the date that one of the pressure suppression chamber-reactor building vacuum breaker systems is made or found inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven (7) days unless such vacuum breaker system is sooner made operable, provided that the procedure does not violate containment integrity.	

Four vacuum relief valves are provided between the drywell and suppression chamber (Section VI-A.1.5 and 2.6*). Each valve is capable of opening on a differential pressure of 0.25 $\pm$ 0.10 psi. The operation of any one valve will prevent damage to the drywell under the accident conditions expected following the loss-of-coolant accident due to a recirculation line break. As discussed in Section VI-F,* one valve operation will limit maximum pressure differential between the two chambers to approximately 3 psi, well below the maximum allowable pressure differential of 8.94 psi.

At a coolant temperature of 215F, the steam generated during a loss-of-coolant accident would not be sufficient to purge the drywell or suppression chamber.

Three sets of vacuum relief valves are provided between the primary containment and atmosphere (Section VI-A.1.5 and 2.6*). Each valve is capable of opening on a differential pressure of 0.25 $\pm$ 0.10 psi. As discussed in Section VI-A.2.6,* operation of all three relief valve sets will prevent containment pressure from dropping below the vacuum ratings of the drywell and the suppression chamber. The selection of these valves is based on the conservative assumption that the ventilation valves on the suppression chamber were left open during a postulated loss-of-coolant accident, permitting the pressure suppression system to blow down to atmospheric pressure. Closure of the ventilation valves followed by startup of the containment spray and core spray pumps leads to a rapid condensation of the steam in the drywell and a consequent drop in pressure below atmospheric. Normally, the ventilation valves are locked closed and there is little likelihood of this series of events occurring. Subsequent calculations showed that with only two valve sets operating, the worst vacuum in the suppression chamber is -3.0 psig. At this pressure a safety factor of about 1.70 still exists to incipient buckling.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 15 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

Using an analysis which is the same as used in the Fifth Supplement (page 115)* results in a failure probability of 1.8×10^{-9} for the drywell to suppression chamber valves and a failure probability of 9.5×10^{-5} for the valves between the containment and the atmosphere.

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Each drywell-suppression chamber vacuum breaker is equipped with two independent switches to indicate the opening of the valve disk. Redundant control room alarms are provided to permit detection of any drywell-suppression chamber vacuum breaker opening in excess of the described allowable limits. The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.053 square feet.

The limit on each individual valve will be set such that with all valves at their limit, the maximum value of cumulative leakage will not exceed the maximum allowable. The value will be at approximately 0.06 inch of disk travel off its seat and will be alarmed in the control room.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and between the suppression chamber and reactor building so that the structural integrity of the containment is maintained.

The vacuum relief system from the pressure suppression chamber to the reactor building consists of three vacuum relief breakers (3 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 1 psig; the external pressure is 2 psig.

The leak rate testing program is based on AEC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression chamber-reactor building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. Therefore, a testing frequency of three months for operability is considered justified for this equipment. Inspections and calibrations are performed during the refueling outages, this frequency is based on equipment quality, experience, and engineering judgment.

During each refueling outage, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by approximately 1 psi with respect to the suppression pool pressure and then held constant. The subsequent suppression chamber transient will be monitored with a sufficiently sensitive pressure instrument. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure, it would indicate existence of a significant leakage path which will be identified and eliminated before further drywell vacuum breaker testing.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.7 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the operating status of the containment spray system.

Objective:

To assure the capability of the containment spray system to limit containment pressure and temperature in the event of a loss-of-coolant accident.

Specification:

a. During all reactor operating conditions whenever reactor coolant temperature is greater than 215 F and fuel is in the reactor vessel; each of the two containment spray systems and the associated raw water cooling systems shall be operable except as specified in 3.3.7.b.

b. If a redundant component of a containment spray system becomes inoperable, Specification 3.3.7.a shall be considered fulfilled, provided that the component is returned to an operable condition within 15 days and that the additional surveillance required is performed.

4.3.7 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the testing of the containment spray system.

Objective:

To verify the operability of the containment spray system.

Specification:

The containment spray system surveillance shall be performed as indicated below:

a. Containment Spray Pumps

(1) At least once per operating cycle, automatic startup of the containment spray pump shall be demonstrated.

(2) At least once per quarter, pump operability shall be checked.

b. Nozzles

At least once per operating cycle, an air test shall be performed on the spray headers and nozzles.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If a redundant component in each of the containment spray systems or their associated raw water systems become inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and that the additional surveillance required is performed.	c. Raw Water Cooling Pumps At least once per quarter manual startup and operability of the raw water cooling pumps shall be demonstrated.
d. If a containment spray system or its associated raw water system becomes inoperable and all the components are operable in the other systems, the reactor may remain in operation for a period not to exceed 7 days.	d. Surveillance with Inoperable Components When a component or system becomes inoperable its redundant component or system shall be demonstrated to be operable immediately and daily thereafter.
e. If Specifications "a" or "b" are not met, shutdown shall begin within one hour and the reactor coolant shall be below 215F within ten hours. If both containment spray systems become inoperable the reactor shall be in the cold shutdown condition within ten hours and no work (except as specified in "f" below) shall be performed on the reactor which could result in lowering the reactor water level to more than six feet, three inches (-10 inches indicator scale) below minimum normal water level (Elevation 302' 9").	e. Surveillance during control rod drive maintenance which is simultaneous with the suppression chamber unwatered shall include at least hourly checks that the conditions listed in 3.3.7.f are met.

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For reactor coolant temperatures less than 215 F not enough steam is generated during a loss-of-coolant accident to pressurize the containment. In fact, for coolant temperatures up to 312 F, the resultant loss-of coolant accident pressure would not exceed the design pressure of 35 psig.

Operation of only one containment spray pump is sufficient to provide the required containment spray flow. The specified flow of 3000 gpm (approximately 95 percent to the drywell and the balance to the suppression chamber) is sufficient to remove post accident core energy released including a substantial chemical reaction involving hydrogen generation and will also limit pressure and temperature rises in the pressure suppression system to below design values (Appendix E-II 2.2.3 p.E-78 and the Fifth Supplement).* Each containment spray system is considered operable when both pumps are capable of delivering at least 3000 gpm at a pump developed head of 375 feet of water at 60 F. Requiring both pumps in both systems operable (400 percent redundancy) will assure the availability of the containment spray system.

Allowable outages are specified to account for components that become inoperable in both systems and for more than one component in a system.

The corresponding raw water cooling system is designed to maintain containment spray water temperature no
greater than 140 F under the most limiting operating conditions. The containment spray raw water cooling
system is considered operable when the flow rate is not less than 3000 gpm and the pressure on the raw water
1 side of the containment spray heat exchangers is not less than 160 psig. The higher pressure on the raw water
side will assure that any leakage is into the containment spray system.

Electrical power for all system components is normally available from the reserve transformer. Upon loss of this service the pumping requirement will be supplied from the diesel generator. At least one diesel generator shall always be available to provide backup electrical power for one containment spray system, corresponding raw water cooling system and associated electronic equipment required for automatic system initiation.

Automatic initiation of the containment spray system assures that the containment will not be overpressurized due to hydrogen generation. This automatic feature would only be required if all core spray system malfunctioned and significant metal-water reaction occurred. For the normal operation condition of 90F suppression chamber water and two psig containment pressure, containment spray actuation would not be necessary for about 15 minutes. Raw water cooling affects the temperature of the spray water and the

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suppression chamber pool. Taking into account the reduced steam condensation capability and increased suppression chamber vapor pressure, the raw water cooling would not be required for more than 20 minutes for initial suppression chamber temperatures up to 110F. This assumes that all core spray systems fail. Therefore, manual initiation of the raw water system is acceptable.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 15 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

In conjunction with containment spray pump operation during each operating cycle, the raw water pumps and associated cooling system performance will be observed. The containment spray system shall be capable of automatic initiation from simultaneous low-low reactor water level and high containment pressure. The associated raw water cooling system shall be capable of manual actuation. Operation of the containment spray system involves spraying water into the atmosphere of the containment. Therefore, periodic system tests are not practical. Instead separate testing of automatic containment spray pump startup will be performed during each operating cycle. During pump operation, water will be recycled to the suppression chamber. Also, air tests to verify that the drywell and torus spray nozzles and associated piping are free from obstructions will be performed each operating cycle. Design features are discussed in Volume I, Section VII-B.2.0 (page VII-19*). The valves in the containment spray system are normally open and are not required to operate when the system is called upon to operate.

The test interval between operating cycle results in a system failure probability of 1.1×10^{-6} (Fifth Supplement, page 115*) and is consistent with practical considerations. Pump operability will be demonstrated on a more frequent basis and will provide a more reliable system.

The intent of Specification 3.3.7f is to allow control rod drive maintenance and instrument replacement at the time that the suppression chamber is unwatered and to perform normal fuel movement activities in the refuel mode with an unwatered suppression chamber.

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3.4.0 REACTOR BUILDING

APPLICABILITY

Applies to the operating status of the reactor building.

OBJECTIVE

To assure the integrity of the reactor building.

SPECIFICATION

Reactor building integrity must be in effect in the refueling and power operating conditions and also whenever irradiated fuel or the irradiated fuel cask is being handled in the reactor building.

3.4.1 LEAKAGE RATE

Applicability:

Applies to the leakage rate of the secondary containment.

Objective:

To specify the requirements necessary to limit exfiltration of fission products released to the secondary containment as a result of an accident.

Specification:

Whenever the reactor is in the refueling or power operating condition, the reactor building leakage rate as determined by Specification 4.4.1 shall not exceed 2000 cfm. If this cannot be met after a routine surveillance check, then the actions listed below shall be taken:

a. Suspend immediately irradiated fuel handling, fuel pool and reactor cavity activities, and irradiated fuel cask handling operations in the reactor building.

b. Restore the reactor building leakage rates to within specified limits within 4 hours or initiate normal orderly shutdown and be in a cold shutdown condition within 10 hours.

4.4.1 LEAKAGE RATE

Applicability:

Applies to the periodic testing requirements of the secondary containment leakage rate.

Objective:

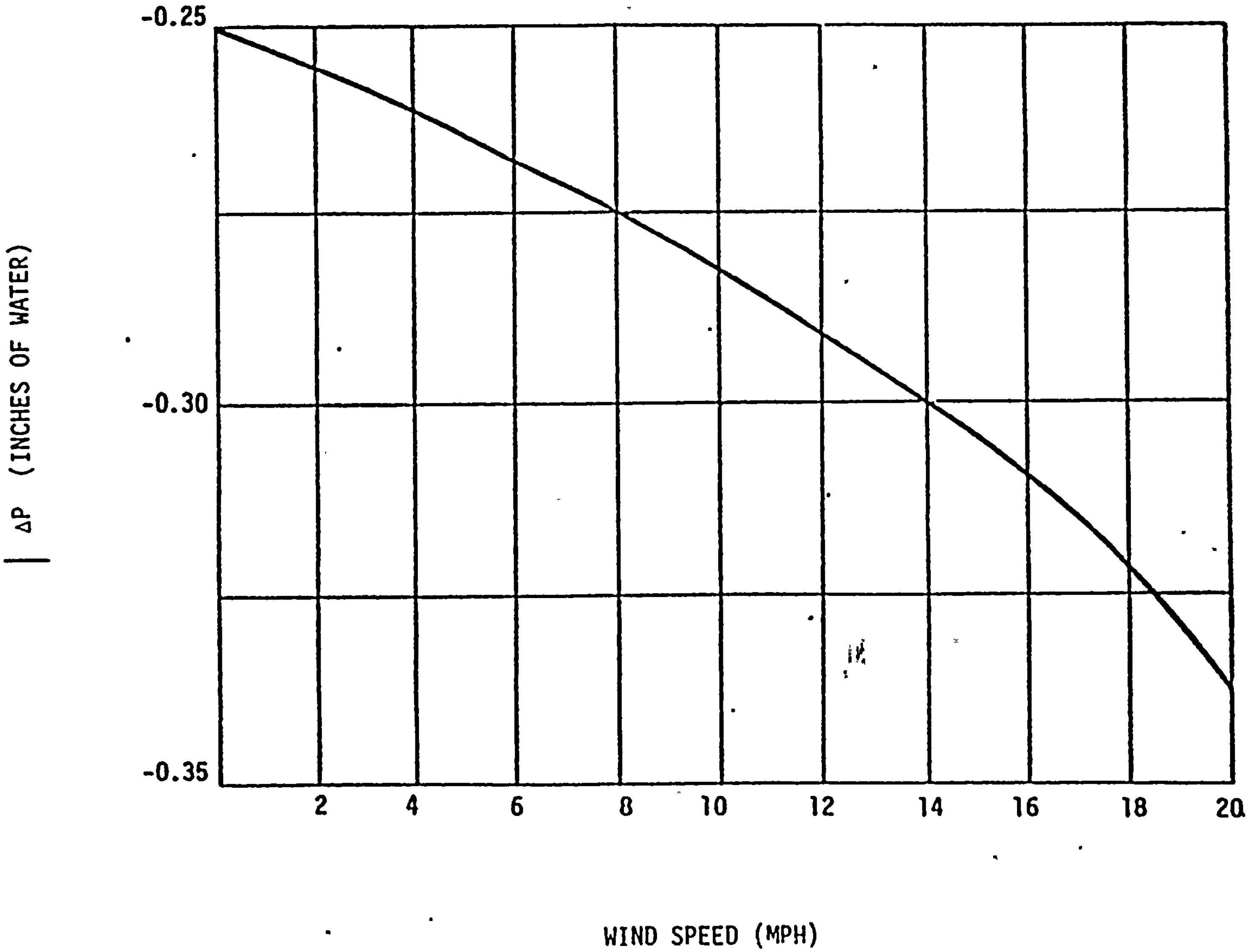
To assure the capability of the secondary containment to maintain leakage within allowable limits.

Specification:

Once during each operating cycle - isolate the reactor building and start emergency ventilation system fan to demonstrate negative pressure in the building relative to external static pressure. The fan flow rate shall be varied so that the building internal differential pressure is at least as negative as that on Figure 3.4.1 for the wind speed at which the test is conducted. The fan flow rate represents the reactor building leakage referenced to zero mph with building internal pressure at least 0.25 inch of water less than atmospheric pressure. The test shall be done at wind speeds less than 20 miles per hour.

FIGURE 3.4.1

REACTOR BUILDING DIFFERENTIAL PRESSURE



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BASES FOR 3.4.1 AND 4.4.1 LEAKAGE RATE

In the answers to Questions II-3 and IV-5 of the Second Supplement and also in the Fifth Supplement,* the relationships among wind speed direction, pressure distribution outside the building, building internal pressure, and reactor building leakage are discussed. The curve of pressure in Figure 3.4.1 represents the wind direction which results in the least building leakage. It is assumed that when the test is performed, the wind direction is that which gives the least leakage.

If the wind direction was not from the direction which gave the least reactor building leakage, building internal pressure would not be as negative as Figure 3.4.1 indicates. Therefore, to reduce pressure, the fan flow rate would have to be increased. This erroneously indicates that reactor building leakage is greater than if wind direction were accounted for. If wind direction were accounted for, another pressure curve could be used which was less negative. This would mean that less fan flow (or measured leakage) would be required to establish building pressure. However, for simplicity it is assumed that the test is conducted during conditions leading to the least leakage while the accident is assumed to occur during conditions leading to the greatest reactor building leakage.

As discussed in the Second Supplement and Fifth Supplement, the pressure for Figure 3.4.1 is independent of the reactor building leakage rate referenced to zero mph wind speed at a negative differential pressure of 0.25 inch of water. Regardless of the leakage rate at these design conditions, the pressure versus wind speed relationship remains unchanged for any given wind direction.

By requiring the reactor building pressure to remain within the limits presented in Figure 3.4.1 and a reactor building leakage rate of less than 2000 cfm, exfiltration would be prevented. This would assure that the leakage from the primary containment is directed through the filter system and discharged from the 350-foot stack. Secondary Containment may be broken at one penetration for up to four hours to allow work to proceed on modifications which will enhance the overall safety of the plant. While the secondary containment is open, administrative controls will be in place to assure integrity can be restored immediately, if necessary. Typically, existing penetrations will be opened in the Reactor Building walls to allow for cable installation for various plant modifications.

Preoperational reactor building capability tests shall be conducted after isolating the reactor building and placing either branch of the emergency ventilation system in operation. The tests shall be performed under a number of different environmental wind conditions, i.e. wind speed and direction.

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LIMITING CONDITION FOR OPERATION

3.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES

Applicability:

Applies to the operational status of the reactor building isolation valves.

Objective:

To assure that fission products released to the secondary containment are discharged to the environment in a controlled manner using the emergency ventilation system.

Specification:

a. The normal Ventilation System isolation valves shall be operable whenever the reactor is in the refueling or power operating conditions, and whenever irradiated fuel or the irradiated fuel cask is being handled in the reactor building.

b. If Specification 3.4.2a is not met, the reactor shall be in the cold shutdown condition within ten hours and handling of irradiated fuel cask shall cease.

SURVEILLANCE REQUIREMENT

4.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES

Applicability:

Applies to the periodic testing requirements of the reactor building isolation valves.

Objective:

To assure the operability of the reactor buidling isolation valves.

Specification:

At least once per operating cycle, automatic initiation of valves shall be checked.

BASES FOR 3.4.2 AND 4.4.2 REACTOR BUILDING INTEGRITY ISOLATION VALVES

Isolation of the reactor building occurs automatically upon high radiation of the normal building exhaust ducts or from high radiation at the refueling platform (See 3.6.2). Isolation will assure that any fission products entering the reactor building will be routed to the emergency ventilation system prior to discharge to the environment (Section VII-H.3.0 of the FSAR).

LIMITING CONDITION FOR OPERATION

3.4.3 ACCESS CONTROL

Applicability:

Applies to the access control to the reactor building.

Objective:

To specify the requirements necessary to assure the integrity of the secondary containment system.

Specification:

a. Only one door in each of the double-doored access ways shall be opened at one time.

b. Only one door or closeup of the railroad bay shall be opened at one time.

c. The core spray and containment spray pump compartments' doors shall be closed at all times except during passage in order to consider the core spray system and the containment spray system operable.

SURVEILLANCE REQUIREMENT

4.4.3 ACCESS CONTROL

Applicability:

Applies to the periodic checking of the condition of portions of the reactor building.

Objective:

To assure that pump compartments are properly closed at all times.

Specification:

a. The core and containment spray pump compartments shall be checked once per week and after each entry.

The reactor building serves as a secondary containment during normal Station operations and as a primary containment during refueling and other periods when the pressure suppression system is open. Maintaining the building integrity and an operative emergency ventilation system for the conditions listed will ensure that any fission products inadvertently released to the reactor building will be routed through the emergency ventilation system to the stack. The worst such incident is due to dropping a fuel assembly on the core during refueling. The consequences of this are discussed in Appendix E-II.3.0 of the FSAR.

As discussed in Section VI-F* all access openings of the reactor building have as a minimum two doors in series. Appropriate local alarms and control room indicators are provided to always insure that reactor building integrity is maintained.

Maintaining closed doors on the pump compartments ensures that suction to the core and containment spray pumps is not lost in case of a gross leak from the suppression chamber.

*FSAR

3.4.4 EMERGENCY VENTILATION SYSTEM

Applicability:

Applies to the operating status of the emergency ventilation system.

Objective:

To assure the capability of the emergency ventilation system to minimize the release of radioactivity to the environment in the event of an incident within the primary containment or reactor building.

Specification:

a. Except as specified in Specification 3.4.4e below, both circuits of the emergency ventilation system and the diesel generators required for operation of such circuits shall be operable at all times when secondary containment integrity is required.

b. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1980.

4.4.4 EMERGENCY VENTILATION SYSTEM

Applicability:

Applies to the testing of the emergency ventilation system.

Objective:

To assure the operability of the emergency ventilation system.

Specification:

Emergency ventilation system surveillance shall be performed as indicated below:

a. At least once per operating cycle, not to exceed 24 months, the following conditions shall be demonstrated:

(1) Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system rated flow rate ($\pm$ 10%).

(2) Operability of inlet heater at rated power when tested in accordance with ANSI N.510-1980.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal when tested in accordance with ANSI N.510-1980 at 80°C and 95% R.H.

d. Fans shall be shown to operate within ± 10% design flow.

e. From and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other emergency ventilation circuit shall be operable.

f. If these conditions cannot be met, within 36 hours, the reactor shall be placed in a condition for which the emergency ventilation system is not required.

b. The tests and sample analysis of Specification 3.4.4b, c and d shall be performed at least once per operating cycle or once every 24 months, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release in any ventilation zone communicating with the system.

c. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

e. Each circuit shall be operated with the inlet heater on at least 10 hours every month.

f. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at and in conformance with each test performed for compliance with Specification 4.4.4b and Specification 3.4.4b.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	g. At least once per operating cycle, not to exceed 24 months, automatic initiation of each branch of the emergency ventilation system shall be demonstrated.
	h. At least once per operating cycle, not to exceed 24 months, manual operability of the bypass valve for filter cooling shall be demonstrated.
	i. When one circuit of the emergency ventilation system becomes inoperable all active components in the other emergency ventilation circuit shall be demonstrated to be operable within 2 hours and daily thereafter.

The emergency ventilation system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both emergency ventilation system fans are designed to automatically start upon high radiation in the reactor building ventilation duct or at the refueling platform and to maintain the reactor building pressure to the design negative pressure so as to minimize in-leakage. Should one system fail to start, the redundant system is designed to start automatically. Each of the two fans has 100 percent capacity.

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to th environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent remomval of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Only one of the two emergency ventilation systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is brought to a condition where the emergency ventilation system is not required.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability and pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified in Table 5-1 of ANSI 509-1980.

The replacement charcoal for the adsorber tray removed for the test should meet the same adsorbent quality. Any HEPA filters found defective shall be replaced with filters qualified pursuant to ANSI 509-1980.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the inlet heater will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repairs and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one emergency ventilation system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation may continue during this period of time.

3.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the operating status of the control room air treatment system.

Objective:

To assure the capability of the control room air treatment system to minimize the amount of radioactivity or other gases entering the control room in the event of an incident.

Specification:

a. Except as specified in Specification 3.4.5e below, the control room air treatment system and the diesel generators required for operation of this system shall be operable at all times when reactor building integrity is required.

b. The results of the in-place cold DOP and halogenated hydrocarbon test design flows on HEPA filters and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1980.

4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the testing of the control room air treatment system.

Objective:

To assure the operability of the control room air treatment system.

Specification:

a. At least once per operating cycle, or once every 24 months, whichever occurs first, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate ($\pm$10%).

b. The tests and sample analysis of Specification 3.4.5b, c and d shall be performed at least once per operating cycle or once every 24 months, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release in any ventilation zone communicating with the system.

LIMITING CONDITION FOR OPERATION

Specification:

c. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodine removal when tested in accordance with ANSI N.510-1980 at 80°C and 95% R.H.

d. Fans shall be shown to operate within ±10% design flow.

e. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor operation or refueling operations is permissible only during the succeeding seven days unless the system is sooner made operable.

f. If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours for reactor operations and refueling operations shall be terminated within 2 hours.

SURVEILLANCE REQUIREMENT

Specification:

c. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.

e. The system shall be operated at least 10 hours every month.

f. At least once per operating cycle, not to exceed 24 months, automatic initiation of the control room air treatment system shall be demonstrated.

g. At least once per operating cycle, not to exceed 24 months, the control room air treatment system shall be shown to maintain a positive pressure within the control room of greater than one sixteenth of an inch (water) relative to areas adjacent to the control room.

BASES FOR 3.4.5 AND 4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

The control room air treatment system is designed to filter the control room atmosphere for intake air. A roughing filter is used for recirculation flow during normal control room air treatment operation. The control room air treatment system is designed to automatically start upon receipt of a high radiation signal from one of the two radiation monitors located on the ventilation intake and to maintain the control room pressure to the design positive pressure (one-sixteenth inch water) so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorber The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filter and charcoal adsorbers are as specified, adequate radiation protection will be provided such that resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be operable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the makeup system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 36 hours or refueling operations are terminated.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability. In addition, air intake radiation monitors will be calibrated and functionally tested each operating cycle, not to exceed 24 months, to verify system performance.

The frequency of tests and sample analysis are necessary to show the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 5-1 of ANSI 509-1980. The replacement charcoal for the adsorber tray removed for the test should

meet the same adsorbent quality. Any HEPA filters found defective shall be replaced with filters qualified pursuant to ANSI 509-1980.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

3.5.0 SHUTDOWN AND REFUELING

A) GENERAL APPLICABILITY

Applies to the neutron instrumentation systems required during shutdown and refueling operations.

B) GENERAL OBJECTIVE

LIMITING CONDITIONS FOR OPERATION - To define the lowest functional capability or performance level of equipment required during shutdown and refueling operations.

SURVEILLANCE REQUIREMENTS - To define the test or inspections required to assure the functional capability or performance level of the above items.

LIMITING CONDITON FOR OPERATION

3.5.1 SOURCE RANGE MONITORS

Applicability:

Applies to the operating status of the source range monitors.

Objective:

To assure the capability of the source range monitors to provide neutron flux indication required for reactor shutdown and startup and refueling operations.

Specification:

Whenever the reactor is in the shutdown, refueling or power operating conditions (unless the IRM's or APRM's are on scale) or whenever core alterations are being made at least three SRM channels will be operable except as noted in Specification 3.5.3. To be considered operable, the following conditions must be satisfied:

a. Inserted to normal operating level and available for monitoring the core. May be withdrawn as long as a minimum count rate of 100 cps is maintained.

SURVEILLANCE REQUIREMENT

4.5.1 SOURCE RANGE MONITORS

Applicability:

Applies to the periodic testing of the source range monitors.

Objective:

To assure the operability of the source range monitors to monitor low-level neutron flux.

Specification:

The source range monitoring system surveillance will be performed as indicated below.

During each operating cycle - check in-core to out-of-core signal ratio and minimum count rate.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. A 3/1 in-core to out-of-core signal ratio and a minimum count rate of 3 cps at a k_{eff} equivalent to the initial clean core with all rods and poison control curtains inserted. c. If following a routine surveillance check "a" or "b" is not met, the reactor shall be in the cold shutdown condition within ten hours.	

BASES FOR 3.5.1 AND 4.5.1 SOURCE RANGE MONITORS

The SRM's are provided to monitor the core during periods of Station shutdown and to guide the operator during refueling operations and Station startup. Requiring three operative SRM's will ensure adequate coverage for all possible critical configurations produced by fuel loading or dispersed withdrawals of control rods during Station startup. Allowing withdrawal of the SRM while maintaining a high count rate will extend the operating range of the SRM's. Evaluation of the SRM operation is presented in Section VIII-C.1.2.1 of the FSAR.

LIMITING CONDITION FOR OPERATION

3.5.2 REFUELING PLATFORM INTERLOCK

Applicability:

Applies to the refueling platform on interlocks.

Objective:

To assure that a loaded refueling platform hoist is never over the core when one or more control rods are withdrawn.

Specification:

During the refueling condition with the mode switch in the "refuel" position the following interlocks must be operative:

a. Control rod withdrawal block with a fuel assembly on the hoist over the reactor core.

b. With a control rod withdrawn from the core the refuel platform, if loaded with a fuel assembly, is blocked from travelling over the core.

c. If the interlocks for either "a" or "b" or both are not operable, double procedural control will be used to ensure that "a" and "b" are met.

SURVEILLANCE REQUIREMENT

4.5.2 REFUELING PLATFORM INTERLOCK

Applicability:

Applies to the periodic testing requirements for the refueling platform interlocks.

Objective:

To assure the operability of the refueling platform interlock.

Specification:

The refueling platform interlocks shall be tested prior to any fuel handling with the head off the reactor vessel, at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

BASES FOR 3.5.2 AND 4.5.2 REFUELING PLATFORM INTERLOCK

The control rod withdrawal block and refueling platform travel blocks are provided to back up normal procedural controls to prevent inadvertent large reactivity additions to the core. These interlocks are provided even though no more than one control rod can be removed from the core at a time during refueling with the mode switch in the "refuel" position. Even in the fresh fully loaded core if a new assembly is dropped into a vacant position adjacent to the withdrawn rod, no excursion would result. This is discussed in detail in Appendix E-II.3.0 of the FSAR.

There are normally two Station personnel directly involved in refueling the reactor, one in the control room and one at the platform. If the interlocks are inoperable, an additional person will check that "a" and "b" are not violated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.5.3 EXTENDED CORE AND CONTROL ROD DRIVE MAINTENANCE

Applicability:

Applies to core reactivity limitations during major core alterations.

Objective:

To assure that inadvertent criticality does not result when control rods are being removed from the core.

Specification:

Whenever, the reactor is in the refueling condition, control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

4.5.3 EXTENDED CORE AND CONTROL ROD DRIVE MAINTENANCE

Applicability:

Applies to monitoring during major core alterations.

Objective:

To assure that inadvertent withdrawal of an incorrect control rod does not occur.

Specification:

Whenever the reactor is in the refuel mode and rod block interlocks are being bypassed for core unloading, one licensed operator and a member of the reactor analysis staff will verify that all the fuel from the cell has been removed before the corresponding control rod is withdrawn.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

a. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock input signal from a withdrawn control rod may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable, except those necessary to pull the next control rods.

b. During core alterations two SRM's shall be operable, one in and one adjacent to any core quadrant where fuel or control rods are being moved. Operable SRM's shall have a minimum of 3 counts per second except as specified in d and e below.

c. The SRM's shall be inserted to the normal operating level. Use of special movable dunking type detectors during major core alterations is permissible as long as detector is connected into the normal SRM circuit.

d. Prior to spiral unloading, the SRM's shall have an initial count rate of 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps.

e. During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these two assemblies have been loaded, the 3 cps requirement is not

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The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as removal of temporary control curtains, control rod drive maintenance, in-service inspection requirements, examination of the core support plate, etc. When the refueling interlock input signal from a withdrawn control rod is bypassed, administrative controls will be in effect to prohibit fuel from being loaded into that control cell.

These operations are performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with the control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's, one in and one adjacent to any core quadrant where fuel or control rods are being moved, assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored.

A spiral unloading pattern is one by which the fuel in the outermost cells (four fuel bundles surrounding a control blade) is removed first. Unloading continues by removing the remaining outermost fuel by cell. The last cell removed will be adjacent to a SRM. Spiral reloading is the reverse of unloading. Spiral unloading and reloading will preclude the creation of flux traps (moderator filled or partially filled cells surrounded on all sides by fuel).

During spiral unloading, the SRM's shall have an initial count rate of 3 cps with all rods fully inserted. The count rate will diminish during fuel removal. After all the fuel is removed from a cell, the refueling interlock will be bypassed on the corresponding control rod. Prior to withdrawal of that rod, one licensed operator and a member of the reactor analysis staff will verify that the interlock bypassed is on the correct control rod. Once the control rod is withdrawn, it will be valved out of service.

Under this special condition of complete spiral core unloading, it is expected that the count rate of the SRM's will drop below 3 cps before all of the fuel is unloaded. Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRM's will no longer be required. Requiring the SRM's to be operational prior to fuel removal assures that the SRM's are operable and can be relied on even when the count rate may go below 3 cps.

During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these two assemblies have been loaded, the 3 cps requirement is not necessary.

3.6.0 GENERAL REACTOR PLANT

A) GENERAL APPLICABILITY

Applies to Station process effluents, reactor protection system and emergency power sources.

B) GENERAL OBJECTIVE

LIMITING CONDITIONS FOR OPERATION - To define the lowest functional capability or performance level of the equipment to assure overall Station safety.

SURVEILLANCE REQUIREMENTS - To define the test or inspection required to assure the functional capability or performance level of this equipment.

LIMITING CONDITION FOR OPERATION

3.6.1 STATION PROCESS EFFLUENTS

a. Effluent release limits are described in Specification 3.6.15.

b. The mechanical vacuum pump line shall be capable of automatic isolation by closure of the air-operated valve upstream of the pumps. The signal to initiate isolation shall be from high radioactivity (five times normal) in the mainsteam line.*

* Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection or within 12 hours of establishing reactor power levels below 20% rated power, while these functions are required to be operable. At reactor power levels below 20% rated power, hydrogen injection shall be terminated and the injection system secured.

SURVEILLANCE REQUIREMENT

4.6.1 STATION PROCESS EFFLUENTS

a. Monitoring the radioactive discharges from Nine Mile Point Unit 1 is described in Specification 4.6.15.

b. At least once during each operating cycle (prior to startup), verify automatic securing and isolation of the mechanical vacuum pump.

BASES FOR 3.6.1b AND 4.6.1b MECHANICAL VACUUM PUMP ISOLATION

The purpose of isolating the mechanical vacuum pump line is to limit release of activity from the main condenser during a control rod drop accident. During the accident, fission products would be transported from the reactor through the main-steam lines to the main condenser. The fission product radioactivity would be sensed by the main-steam line radioactivity monitors and initiate isolation.

LIMITING CONDITION FOR OPERATION

3.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs a safety function.

Objective:

To assure the operability of the instrumentation required for safe operation.

Specification:

a. The set points, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Tables 3.6.2a to 3.6.2l.

If the requirements of a table are not met, the actions listed below for the respective type of instrumentation shall be taken.

(1) Instrumentation that initiates scram - control rods shall be inserted, unless there is no fuel in the reactor vessel.

SURVEILLANCE REQUIREMENT

4.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs a safety function.

Objective:

To verify the operability of protective instrumentation.

Specification:

a. Sensors and instrument channels shall be checked, tested and calibrated at least as frequently as listed in Tables 4.6.2a to 4.6.2l.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
5 (2) Primary Coolant and Containment Isolation - Isolation valves shall be closed or the valves shall be considered inoperable and Specifications 3.2.7 and 3.3.4 shall be applied.	b. Each trip system shall be tested each time the respective instrument channel is tested.
5 (3) Emergency Cooling Initiation or Isolation - The emergency cooling system shall be considered inoperable and Specification 3.1.3 shall be applied.	c. At least daily during reactor power operation, the core power distribution shall be checked for Maximum Total Peaking Factor (MTPF) 5 and the flow-referenced APRM scram and rod block signals shall be adjusted, if necessary, as specified in Figure 2.1.1. 5
5 (4) Core Spray Initiation - The core spray system shall be considered inoperable and Specification 3.1.4 shall be applied.	
5 (5) Containment Spray Initiation - The containment spray system shall be considered inoperable and Specification 3.3.7 shall be applied.	
5 (6) Auto Depressurization Initiation - The auto depressurization system shall be considered inoperable and Specification 3.1.5 shall be applied.	
5 (7) Control Rod Withdrawal Block - No control rods shall be withdrawn.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

(8) Off-Gas and Vacuum Pump Isolation - The respective system shall be isolated or the instrument channel shall be considered inoperable and Specification 3.6.1 shall be applied.

(9) Diesel Generator Initiation - The diesel generator shall be considered inoperable and Specification 3.6.3 shall be applied.

(10) Emergency Ventilation Initiation - The emergency ventilation system shall be considered inoperable and Specification 3.4.4 shall be applied.

(11) High Pressure Coolant Injection Initiation - The high pressure coolant injection system shall be considered inoperable and Specification 3.1.8.c shall be applied.

(12) Control Room Ventilation - The control room ventilation system shall be considered inoperable and Specification 3.4.5 shall be applied.

b. During operation with a Maximum Total Peaking Factor (MTPF) greater than the design value, either:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(1) The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.2.a; or (2) The power distribution shall be changed such that the MTPF no longer exceeds the design value.	

Table 3.6.2a

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (o)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(1)	Manual Scram	2	1			x	x	x
(2)	High Reactor Pressure	2	2	≤ 1080 psig		x	x	x
(3)	High Drywell Pressure	2	2	≤ 3.5 psig		x	(a)	(a)
(4)	Low Reactor Water Level	2	2	≥ 53 inches (Indicator Scale)		x	x	x
(5)	High Water Level Scram Discharge Volume	2	2	≤ 45 gal.		(b)	x	x

Table 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (o)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(6) Main-Steam-Line Isolation Valve Position	2	4(h)	≤ 10 percent valve closure from full open		(c)	(c)	x
(7) High Radiation Main-Steam-Line	2	2	≤ 5 times normal background at rated power[n]		x	x	x
(8) Shutdown Position of Reactor Mode Switch	2	1	- -		(k)	x	x
(9) Neutron Flux							
(a) IRM							
(i)	2	3(d)	≤ 96 percent of full scale		(g)	(g)	(g)

Table 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (o)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(ii) Inoperative	2	3(d)	--		x	x	
(b) APRM							
(i) Upscale	2	3(e)	Figure 2.1.1		x	x	x
(ii) Inoperative	2	3(e)	--		x	x	x
(iii) Downscale	2	3(e)	≥ 5 percent of full scale		(g)	(g)	(g)
(10) Turbine Stop Valve Closure	2	4	≤ 10% valve closure				(i)
(11) Generator Load Rejection	2	2	(j)				(i)

TABLE 4.6.2a

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1) Manual Scram	None	Once per 3 months	None
(2) High Reactor Pressure	None	Once per month(1)	Once per 3 months(1)
(3) High Drywell Pressure	None	Once per month(1)	Once per 3 months(1)
(4) Low Reactor Water Level	Once/day	Once per month(1)	Once per 3 months(1)
(5) High Water Level Scram Discharge Volume	None	Once per month	Once per 3 months
(6) Main-Steam-Line Isolation Valve Position	None	Once per 3 months	Once per operating cycle
(7) High Radiation Main-Steam Line	Once/shift	Once per week	Once per 3 months

TABLE 4.6.2a (Cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(8) Shutdown Position of Reactor Mode Switch	None	Once during each major refueling outage	None
(9) Neutron Flux			
(a) IRM			
(i) Upscale	(f)	(f)	(f)
(ii) Inoperative	(f)	(f)	(f)
(b) APRM			
(i) Upscale	None	Once/week	Once/week(m), Once per 3 months
(ii) Inoperative	None	Once/week	None
(iii) Downscale	None	Once/week	Once/week(m), Once per 3 months
(10) Turbine Stop Valve Closure	None	Once per 3 months	Once per operating cycle
(11) Generator Load Rejection	None	Once per month	Once per 3 months

NOTES FOR TABLES 3.6.2a and 4.6.2a

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed in the refuel and shutdown positions of the reactor mode switch with a keylock switch.

(c) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi.

(d) No more than one of the four IRM inputs to each trip system shall be bypassed.

(e) No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed in order for the APRM to be considered operable. No more than one of the four APRM inputs to each trip system shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing.

(f) Calibrate prior to starting and normal shutdown and thereafter check once per shift and test once per week until no longer required.

(g) IRM's are bypassed when APRM's are onscale. APRM downscale is bypassed when IRM's are onscale.

(h) Each of the four isolation valves has two limit switches. Each limit switch provides input to one of two instrument channels in a single trip system.

(i) May be bypassed when reactor power level is below 45%.

(j) Trip upon loss of oil pressure to the acceleration relay.

(k) May be bypassed when placing the reactor mode switch in the SHUTDOWN position and all control rods are fully inserted.

(l) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2a, the primary sensor will be calibrated and tested once per operating cycle.

(m) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during reactor operation when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Figure 2.1.1 shall not be included in determining the absolute difference.

(n) Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection or within 12 hours of establishing reactor power levels below 20% rated power, while these functions are required to be operable. At reactor power levels below 20% rated power, hydrogen injection shall be terminated and the injection system secured.

(o) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same trip system is monitoring that parameter. This time interval is extended up to 5 hours for the High Radiation Main-Steam Line Instrument Channel Calibration surveillance.

Table 3.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (f)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
PRIMARY COOLANT ISOLATION							
(Main Steam, Cleanup, and Shutdown)							
(1) Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)			x	x
(2) Manual	2	1	- -	x	x	x	x
MAIN-STEAM-LINE ISOLATION							
(3) High Steam Flow Main-Steam Line	2	2	≤ 105 psid			x	x

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Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (f)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(4)	High Radiation Main Steam Line	2	2	≤ 5 times normal background at rated power (e)			x	x
(5)	Low Reactor Pressure	2	2	≥ 850 psig				x
(6)	Low-Low-Low Condenser Vacuum	2	2	≥ 7 in. mercury vacuum			(a)	x
(7)	High Temperature Main Steam Line Tunnel	2	2	≤ 200F			x	x

Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (f)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
CLEANUP SYSTEM ISOLATION							
(8) High Area Temperature	1	2	≤ 190	x	x	x	x
SHUTDOWN COOLING SYSTEM ISOLATION							
(9) High Area Temperature	1	1	≤ 170	x	x	x	x
CONTAINMENT ISOLATION							
(10) Low-Low Reactor Water	2	2	≥ 5 inches (Indicator Scale)	(c)		x	x

Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (f)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(11)	High Drywell Pressure	2	2	≤ 3.5 psig	(c)		(b)	(b)
(12)	Manual	2	1	- -	x	x	x	x

Table 4.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
PRIMARY COOLANT ISOLATION (Main Steam, Cleanup and Shutdown)			
(1) Low-Low Reactor Water Level	Once/day	Once per month(d)	Once per 3 months(d)
(2) Manual	--	Once during each major refueling outage	--
MAIN-STEAM-LINE ISOLATION			
(3) High Steam Flow Main-Steam Line	Once/day	Once per month(d)	Once per 3 months(d)
(4) High Radiation Main-Steam Line	Once/shift	Once/week	Once per 3 months
(5) Low Reactor Pressure	Once/day	Once per month(d)	Once per 3 months(d)

Table 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(6)	Low-Low-Low Condenser Vacuum	None	Once during each major refueling outage	Once during each major refueling outage
(7)	High Temperature Main-Steam-Line Tunnel	None	Once during each major refueling outage	Once during each major refueling outage
CLEANUP SYSTEM ISOLATION				
(8)	High Area Temperature	Once/week	Once during each major refueling outage	Once during each major refueling outage
SHUTDOWN COOLING SYSTEM ISOLATION				
(9)	High Area Temperature	Once/week	Once during each major refueling outage	Once during each major refueling outage

Table 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
CONTAINMENT ISOLATION			
(10) Low-Low Reactor Water Level	Once/day	Once per month[d]	Once per 3 months[d]
(11) High Drywell Pressure	Once/day	Once per month[d]	Once per 3 months[d]
(12) Manual	- -	Once during each operating cycle	- -

NOTES FOR TABLES 3.6.2b and 4.6.2b

(a) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi.

(b) May be bypassed when necessary for containment inerting.

(c) May be bypassed in the shutdown mode whenever the reactor coolant system temperature is less than 215°F.

(d) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2b, the primary sensor will be calibrated and tested once per operating cycle.

(e) Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection or within 12 hours of establishing reactor power levels below 20% rated power, while these functions are required to be operable. At reactor power levels below 20% rated power hydrogen injection shall be terminated and the injection system secured.

(f) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip system is monitoring that parameter. This time interval is extended up to 5 hours for the High Radiation Main-Steam Line Instrument Channel Calibration Surveillance.

Table 3.6.2c

INSTRUMENTATION THAT INITIATES OR ISOLATES EMERGENCY COOLING

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (d)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
EMERGENCY COOLING INITIATION							
(1) High-Reactor Pressure	2	2	≤ 1080 psig	(b)		x	x
(2) Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)	(b)		x	x
EMERGENCY COOLING ISOLATION (for each of two systems)							
(3) High Steam Flow Emergency Cooling System	2	2 (a)	≤ 11.5 psid			x	x

Table 4.6.2c

INSTRUMENTATION THAT INITIATES OR ISOLATES EMERGENCY COOLING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
EMERGENCY COOLING INITIATION			
(1) High Reactor Pressure	None	Once per month(c)	Once per 3 months(c)
(2) Low-Low Reactor Water Level	Once/day	Once per month(c)	Once per 3 months(c)
EMERGENCY COOLING ISOLATION (for each of two systems)			
(3) High Steam Flow Emergency Cooling System	None	Once per 3 months(c)	Once per 3 months(c)

NOTES FOR TABLES 3.6.2c AND 4.6.2c

(a) Each of two differential pressure switches provide inputs to one instrument channel in each trip system.

(b) May be bypassed in the cold shutdown condition.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2c, the primary sensor will be calibrated and tested once per operating cycle.

(d) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip system is monitoring that parameter.

Table 3.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY[e]

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System (g)	Setpoint	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
START CORE SPRAY PUMPS							
(1) High Drywell Pressure	2	2	≤ 3.5 psig	(d)	x	(a)	(a)
(2) Low-Low Reactor Water Level	2	2 [f]	≥ 5 inches (Indicator Scale)	(b)	x	x	x
OPEN CORE SPRAY DISCHARGE VALVES							
(3) Reactor Pressure and either (1) or (2) above.	2	2	≥ 365 psig	x	x	x	x

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Table 4.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
START CORE SPRAY PUMPS			
(1) High Drywell Pressure	Once/day	Once per month(c)	Once per 3 months(c)
(2) Low-Low Reactor Water Level	Once/day	Once per month(c)	Once per 3 months(c)
OPEN CORE SPRAY DISCHARGE VALVES			
(3) Reactor Pressure and either (1) or (2) above	None	Once per month(c)	Once per 3 months(c)

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NOTES FOR TABLES 3.6.2d AND 4.6.2d

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed when necessary for performing major maintenance as specified in Specification 2.1.1.e.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2d, the primary sensor will be calibrated and tested once per operating cycle.

(d) May be bypassed when necessary for integrated leak rate testing.

(e) The instrumentation that initiates the Core Spray System is not required to be operable, if there is no fuel in the reactor vessel.

(f) One instrument channel in each trip system may be bypassed in the cold shutdown and refuel conditions during the Spring 1986 refueling outage to perform the emergency condenser piping replacement.

(g) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip system is monitoring that parameter.

Table 3.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (c)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(1)a. High Drywell Pressure	2	2	≤ 3.5 psig	(a)		x	x
and							
b. Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)	(a)		x	x

Table 4.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)a. High Drywell Pressure	Once/day	Once per month(b)	Once per 3 months(b)
b. Low-Low Reactor Water Level	Once/day	Once per month(b)	Once per 3 months(b)

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NOTES FOR TABLES 3.6.2e AND 4.6.2e

(a) May be bypassed in the shutdown mode whenever the reactor coolant temperature is less than 215°F.

(b) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2e, the primary sensor will be calibrated and tested once per operating cycle.

(c) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip system is monitoring that parameter.

Table 3.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System (d)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable		
INITIATION						
(1)a. Low-Low-Low Reactor Water Level	2 (a)	2 (a)	≥ -10 inches * (Indicator Scale)	(b)	(b)	x
and						
b. High Drywell Pressure	2 (a)	2 (a)	≤ 3.5 psig	(b)	(b)	x

* greater than (≥) means less negative

Table 4.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
INITIATION			
(1) a. Low-Low-Low Reactor Water	None	Once per month(c)	Once per 3 months(c)
and			
b. High Drywell Pressure	Once/day	Once per month(c)	Once per 3 months(c)

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NOTES FOR TABLES 3.6.2f AND 4.6.2f

(a) Both instrument channels in either trip system are required to be energized to initiate auto depressurization. One trip system is powered from power board 102 and the other trip system from power board 103.

(b) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2f, the primary sensor will be calibrated and tested once per operating cycle.

(d) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip system is monitoring that parameter.

Table 3.6.2g

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(1) SRM							
a. Detector not in Startup Position	2	2 (a), (e)	- -		X	X	
b. Inoperative	2	2 (a)	- -		X	X	
c. Upscale	2	2 (a)	$\leq 10^5$ counts/sec		X	X	
(2) IRM							
a. Detector not in Startup Position	2	3 (b)	- -		X	X	
b. Inoperative	2	3 (b)	- -		X	X	

TABLE 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
c. Downscale	2	3 (b)	≥ 5 percent of full scale for each scale		X	X	
d. Upscale	2	3 (b)	≤ 88 percent of full scale for each scale		X	X	
(3) APRM							
a. Inoperative	2 (h)	3 (c)	--		X	X	X
b. Upscale (Biased by Recirculation Flow)	2 (h)	3 (c)	Figure 2.1.1(h)		X	X	X
c. Downscale	2 (h)	3 (c)	≥ 2 percent of full scale		(d)	(d)	X

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Table 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable				
				Shutdown	Refuel	Startup	Run	
(4) Recirculation Flow								
a. Comparator Off Normal	2	1	--		X	X	X	
b. Flow Unit Inoperative	2	1	--		X	X	X	
c. Flow Unit Upscale	2	1	Figure 2.1.1		X	X	X	5
(5) Refuel Platform and Hoists	2 (f)	1	--		X			
(6) Mode Switch in Shutdown	1	1	--	X				

Table 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(7) Mode Switch in Refuel (Blocks withdrawal of more than 1 rod)	1	1	- -		X		
(8) Scram Dump Volume Water Level Scram Bypass	2	1	- -	X	X		

Table 4.6.2g

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	SRM			
	a. Detector Not in Startup Position	N/A	(g)	N/A
	b. Inoperative	N/A	(g)	N/A
	c. Upscale	N/A	(g)	(g)
(2)	IRM			
	a. Detector not in Startup Position	N/A	(g)	N/A
	b. Inoperative	N/A	(g)	N/A
	c. Downscale	N/A	(g)	(g)
	d. Upscale	N/A	(g)	(g)

TABLE 4.6.2g (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(3) APRM			
a. Inoperative	None	Once per month	None
b. Upscale (Biased by Recirculation Flow)	None	Once per month	Once per 3 months
c. Downscale	None	Once per month	Once per 3 months
(4) Recirculation Flow			
a. Comparator Off Normal	None	Once per month	Once per month
b. Flow Unit Inoperative	None	Once per month	Once per month
c. Flow Unit Upscale	None	Once per month	Once per month

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Table 4.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(5)	Refuel Platform and Hoists	- -	(see 4.5.2)	- -
(6)	Mode Switch in Shutdown	- -	Once during each major refueling outage	- -
(7)	Mode Switch in Refuel (Blocks withdrawal of more than 1 rod)	- -	Once during each major refueling outage	- -
(8)	Scram Dump Volume Water Level Scram Bypass	- -	Once during each major refueling outage	- -

NOTES FOR TABLES 3.6.2g and 4.6.2g

(a) No more than one of the four SRM inputs to the single trip system shall be bypassed.

(b) No more than one of the four IRM inputs to each instrument channel shall be bypassed. These signals may be bypassed when the APRM's are onscale.

(c) No more than one of the four APRM inputs to each instrument channel shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to only one APRM shall be bypassed in order for the APRM to be considered operable. In the Run mode of operation, bypass of two chambers from one radial core location in any one APRM shall cause that APRM to be considered inoperative. A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing. If one APRM in a quadrant is bypassed and meets all requirements for operability with the exception of the requirement of at least one operable chamber at each radial location, it may be returned to service and the other APRM in that quadrant may be removed from service for test and/or calibration only if no control rod is withdrawn during the calibration and/or test.

(d) May be bypassed in the startup and refuel positions of the reactor mode switch when the IRM's are onscale.

(e) This function may be bypassed when the count rate is $\geq$ 100 cps.

(f) One sensor provides input to each of two instrument channels. Each instrument channel is in a separate trip system.

(g) Calibrate and/or test prior to startup and normal shutdown. Thereafter test once per week until no longer required.

(h) The actuation of either or both trip systems will result in a rod block.

Table 3.6.2h

VACUUM PUMP ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System (b)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable		
MECHANICAL VACUUM PUMP						
High Radiation Main Steam Line	2	2	$\leq$ 5 times normal[a] background	x	x	x

TABLE 4.6.2h

VACUUM PUMP ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
MECHANICAL VACUUM PUMP			
High Radiation Main Steam Line	Once/shift	Once per week	Once per 3 months

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NOTES FOR TABLES 3.6.2h and 4.6.2h

(a) Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection or within 12 hours of establishing reactor power levels below 20% rated power, while these functions are required to be operable. At reactor power levels below 20% rated power hydrogen injection shall be terminated and the injection system secured.

(b) A channel may be placed in an operable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter. This time interval is extended up to 5 hours for the High Radiation Main-Steam Line Instrument Channel Calibration surveillance.

Table 3.6.2i

DIESEL GENERATOR INITIATION

Limiting Condition for Operation

Parameter	Total No. of Channels	Channels[1] to Trip	Minimum Channels Operable (c)	Reactor Mode Switch Position in Which Function Must be Operable: Shutdown	Refuel	Startup	Run
Loss of Power							
a. 4.16kV PB 102/103 Emergency Bus Undervoltage (Loss of Voltage)	3 per Bus	2 per Bus	2 per Bus	x	x	x	x
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	3 per Bus	2 per Bus	2 per Bus	x	x	x	x

(1) If one out of three channels becomes inoperable, the inoperable channel will be placed in the trip condition.

Table 3.6.2i (continued)

DIESEL GENERATOR INITIATION

Limiting Condition for Operation

Parameter	Set Point (Inverse Time Undervoltage Relays)	
Loss of Power	Relay Dropout	Operating Time[a]
a. 4.16kV PB 102/103 Emergency Bus Undervolt (Loss of Voltage)	≥3200 volts	0 volts ≤ 3.2 seconds
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	≥3600 volts	3580 volts 18.5 ± 3 seconds

(a) The operating time indicated in the table is the time required for the relay to operate its contacts when the voltage is suddenly decreased from operating voltage level values to the voltage level listed in the table above.

Table 4.6.2i

DIESEL GENERATOR INITIATION

Surveillance Requirements

Parameter	Sensor Check	Instrument[a] Channel Test	Instrument[b] Channel Calibration
Loss of Power			
a. 4.16kV PB 102/103 Emergency Bus Undervoltage (Loss of Voltage)	NA	Once per month	Once per refueling cycle
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	NA	Once per month	Once per refueling cycle

(a) The instrument channel test demonstrate the operability of the instrument channel by simulating an undervoltage condition to verify that the tripping logic functions properly.

(b) The instrument channel calibration will demonstrate the operability of the instrument channel by simulating an undervoltage condition to verify that the tripping logic functions properly. In addition, a sensor calibration will be performed to verify the set points listed in Table 3.6.2.i.

(c) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

Table 3.6.2j

EMERGENCY VENTILATION INITIATION

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(1)	High Radiation Reactor Building Ventilation Duct	1	2	≤ 5mr/hr	X	X	X	X
(2)	High Radiation Refueling Platform	1	1	≤ 1000mr/hr	(a)	(a)	(a)	(a)

Table 4.6.2j

EMERGENCY VENTILATION INITIATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	High Radiation Reactor Building Ventilation Duct	Once/shift	Once during each operating cycle	Once per quarter
(2)	High Radiation Refueling Platform	(b)	(c)	Once per quarter

NOTES FOR TABLES 3.6.2j AND 4.6.2j

(a) This function shall be operable any time that irradiated fuel or the irradiated fuel cask is being handled in the reactor building.

(b) Once per shift whenever this function is required to be operable.

(c) Immediately prior to when function is required and once per week thereafter until function is no longer required.

Table 3.6.2k

HIGH PRESSURE COOLANT INJECTION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System (c)	Set-Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(1) Low Reactor Water Level	2	2	$\geq$ 53 inches (Indicator Scale)	(a)		(a)	x
(2) Automatic Turbine Trip	1	1	- -	(a)		(a)	x

Table 4.6.2k

HIGH PRESSURE COOLANT INJECTION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low Reactor Water Level	Once per day	Once per month[(b)]	Once per 3 months[(b)]
(2)	Automatic Turbine Trip	None	Once during each operating cycle	None

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NOTES FOR TABLES 3.6.2k AND 4.6.2k

(a) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(b) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2k, the primary sensor will be calibrated and tested once per operating cycle.

(c) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip system is monitoring that parameter.

Table 3.6.21

CONTROL ROOM AIR TREATMENT SYSTEM INITIATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(1) High Radiation Ventilation Intake	1	1	$\leq$ 1000 CPM		x	x	x

Table 4.6.21

CONTROL ROOM AIR TREATMENT SYSTEM INITIATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1) High Radiation Ventilation Intake	Once/shift	Once per quarter	Once each operating cycle not to exceed 24 months

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The reactor protection system automatically initiates a reactor scram to prevent exceeding established limits. In addition, other protective instrumentation is provided to initiate action which mitigates the consequences of accidents or terminates operator error.

The reactor protection system is a dual channel type (Table 3.6.2.a). Each trip system except the manual scram has two independent instrument channels. Operation of either channel will trip the trip system, i.e., the trip logic of the channel is one-out-of-two. A simultaneous trip of both trip systems will cause a reactor scram, i.e., the tripping logic of the trip systems is two-out-of-two. The tripping logic of the total system is referred to as one-out-of-two taken twice. This system will accommodate any single failure and still perform its intended function and in addition, provide protection against spurious scrams. The reliability of the dual channel system or probability that it will perform its intended function is less than that of a one-out-of-two system and somewhat greater than that of a two-out-of-three system (Section VIII-A.1.0 of the FSAR).

The instrumentation used to initiate action other than scram is generally similar to the reactor protection system. There are usually two trip systems required or available for each function. There are usually two instrument channels for each trip system. Either channel can trip the trip system but both trip systems are required to initiate the respective action. Where only one trip system is provided only one instrument channel is required to trip the trip system. All instrument channels except those for automatic depressurization are normally energized. De-energizing causes a trip. Power to the trip systems for each function is from reactor protection system buses 11 and 12.

The signals for initiating automatic blowdown and rod block differ from other initiating signals in that only one of the two trip systems is required to start blowdown or initiate rod block. Both instrument channels in the trip system must trip to initiate automatic blowdown. This difference is due to the requirement that automatic depressurization be prevented unless A.C. power is available to the emergency core cooling systems. The instrument channels in the trip system for automatic depressurization are normally de-energized. In order to cause a trip both instrument channels must be energized. Power to energize the instrument channels is from power boards 102 and 103. If A.C. power is lost to one power board, one trip system becomes inoperable

but the other trip system remains operable and capable of initiating automatic blowdown. If both power boards have lost A.C. power neither trip system can be energized and automatic blowdown is prevented. Only one instrument channel is required to initiate rod block.

Each reactor operating condition has a related reactor mode switch position for the safety system. The instrumentation system operability for each mode switch position is based on the requirements of the related safety system. For example, the specific high drywell pressure trip systems must be tripped or operable any time core spray, containment spray, automatic depressurization or containment isolation functions are required.

In instrumentation systems where two trip systems are required to initiate action, either both trip systems are operable or one is tripped. Having one trip system already tripped does not decrease the reliability in terms of initiating the desired action. However, the probability of spurious actuation is increased. Certain instrument channels or sensor inputs to instrument channels may be bypassed without affecting safe operation. The basis for allowing bypassing of the specified SRM's, IRM's, LPRM's and APRM's is discussed in Volume I (Section VII-C.1.2)*. The high area temperature isolation function for the cleanup system has one trip system. There are three instrument channels; each has four sensor inputs. Only two instrument channels are required since the area covered by any one sensor is also covered by a sensor in one of the other two instrument channels. The shutdown system also has one trip system for high area temperature isolation. However, since the area of concern is much smaller, only one instrument channel is provided. Four sensors provide input to the channel. Since the area covered is relatively small only three of the four sensors are required to be operable in order to assure isolation when needed.

Manual initiation is available for scram, reactor isolation and containment isolation. In order to manually initiate other systems, each pump and each valve is independently initiated from the control room. Containment spray raw water cooling is not automatically initiated. Manual initiation of each pump is required as discussed in 3.3.7 above.

*FSAR; Letter, R.R. Schneider to A. Giambusso, dated November 15, 1973

a. The set points included in the tables are those used in the transient analysis and the accident analysis. The high flow set point for the main steam line is 105 psi differential. This represents a flow of approximately 4.4×10^6 lb/hr. The high flow set point for the emergency cooling system supply line is $\leq$ 11.5 psi differential. This represents a flow of approximately 9.8×10^5 lb/hr at rated conditions.

Normal background for the main steam line radiation monitors is defined as the radiation level which exists in the vicinity of main steam lines after 1 hour or more of sustained full rated power. The dose rate at the monitor due to activity from the control rod drop accident of Appendix E or from gross failure of one rod with complete fission product release from the rod would exceed the normal background at the monitor. The automatic initiation signals for the emergency cooling systems have to be sustained for more than 10 seconds to cause opening of the return valves. If the signals last for less than 10 seconds, the emergency cooling system operating will not be automatically initiated.

The high level in the scram discharge volume is provided to assure that there is still sufficient free volume in the discharge system to receive the control rod drives discharge. Following a scram, bypassing is permitted to allow draining of the discharge volume and resetting of the reactor protection system relays. Since all control rods are completely inserted following a scram and since the bypass of this particular scram initiates a control rod block, it is permissible to bypass this scram function. The scram trip associated with the shutdown position of the mode switch can be reset after 10 seconds.

The condenser low vacuum, low-low vacuum and the main steam line isolation valve position signals are bypassed in the startup and refuel positions of the reactor mode switch when the reactor pressure is less than 600 psig. These are bypassed to allow warmup of the main steam lines and a heat sink during startup.

The set points on the generator load rejection and turbine stop valve closure scram trips are set to anticipate and minimize the consequences of turbine trip with failure of the turbine bypass system as described in the bases for Specification 2.1.2. Since the severity of the transients is dependent on the reactor operating power level, bypassing of the scrams below the specified power level is permissible.

Although the operator will set the setpoints at the values indicated in Tables 3.6.2.a-1, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations include inherent instrument error, operator setting error and drift of the set point. These errors are compensated for in the transient analyses by conservatism in the controlling parameter assumptions as discussed in the bases for Specification 2.1.2. The deviations associated with the set points for the safety systems used to mitigate accidents have negligible effect on the initiation of these systems. These safety systems have initiation times which are orders of magnitude greater than the difference in time between reaching the nominal set point and the worst set point due to error. The maximum allowable set point deviations are listed below:

Neutron Flux
APRM, +2.7% of rated neutron flux
IRM, +2.5% of rated neutron flux

Recirculation Flow, ±1% of rated recirculation flow

Reactor Pressure, ±15.8 psig

Containment Pressure, ±0.053 psig

Reactor Water Level, ±2.6 inches of water

Main Steam Line Isolation Valve Position, ±2.5% of stem position

Scram Discharge Volume, +0 and -1 gallon

Condenser Low Vacuum, ±0.5 inches of mercury

High Flow-Main Steam Line, ±1 psid

High Flow-Emergency Cooling Line, ±1 psid

High Area Temperature-Main Steam Line, ±10F

High Area Temperature-Clean-up and Shutdown, ±6F

High Radiation-Main Steam Line, +100% and -50% of set point value

High Radiation-Emergency Cooling System Vent, +100% and -50% of set point

High Radiation-Reactor Building Vent, +100% and -50% of set point

High Radiation-Refueling Platform, +100% and -50% of set point

High Radiation-Offgas Line, ±50% of set point, (Appendix D)*

The test intervals for the trip systems result to calculated failure probabilities $\leq 10^{-4}$ which corresponds to the proposed IEEE Criteria for System Failure Probability. (IEEE SG-3, Information Docket #1 - Protection System Reliability, April 24, 1968).

The test intervals for the trip systems result in calculated failure probabilities ranging from 6.7×10^{-7} to 1.76×10^{-10} (Fifth Supplement, p. 115).* The more frequent sensor checks result in even less probability that the particular system will fail. Because of local high radiation, testing instrumentation in the area of the main steam line isolation valves can only be done during periods of Station shutdown. These functions include high area temperature isolation, high radiation isolation and isolation valve position scram.

Testing of the scram associated with the shutdown position of the mode switch can be done only during periods of Station shutdown since it always involves a scram.

*FSAR

b. The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR is maintained greater than the SLCPR. The trip logic for this function is 1 out of n; e.g., any trip on one of the eight APRM's, eight IRM's or four SRM's will result in a rod block. The minimum instrument channel requirements provide sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the rod block may be reduced by one for a short period of time to allow maintenance, testing, or calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block trip is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the SLCPR.

The APRM rod block also provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked before the MCPR reaches the SLCPR, thus allowing adequate margin. Below ~60% power the worst case withdrawal of a single control rod results in a MCPR > SLCPR without rod block action, thus below this level it is not required.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the SLCPR.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and the control rod motion is prevented. The downscale rod blocks are set at 5 percent of full scale for IRM and 2 percent of full scale for APRM (APRM signal is generated by averaging the output signals from eight LPRM flux monitors).

LIMITING CONDITION FOR OPERATION

3.6.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the operational status of the emergency power sources.

Objective:

To assure the capability of the emergency power sources to provide the power required for emergency equipment in the event of a loss-of-coolant accident.

Specification:

a. For all reactor operating conditions except cold shutdown, there shall normally be available two 115 kv external lines, two diesel generator power systems and two battery systems, except as further specified in "b," "c," "d," "e," and "h" below.

b. One 115 kv external line may be de-energized provided two diesel-generator power systems are operable. If a 115 kv external line is de-energized, that line shall be returned to service within 7 days.

SURVEILLANCE REQUIREMENT

4.6.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the periodic testing requirements for the emergency power sources.

Objective:

To assure the operability of the emergency power sources to provide emergency power required in the event of a loss-of-coolant accident.

Specification:

The emergency power systems surveillance will be performed as indicated below. In addition, components on which maintenance has been performed will be tested.

a. During each major refueling outage - test for automatic startup and pickup of load required for a loss-of-coolant accident.

b. Monthly - manual start and operation at rated load shall be performed for a minimum time of one hour. Determine the specific gravity of each cell. Determine the battery voltage.

LIMITING CONDITION FOR OPERATION

c. One diesel-generator power system may be inoperable provided two 115 kv external lines are energized. If a diesel-generator power system becomes inoperable, it shall be returned to an operable condition within seven days. In addition, if a diesel-generator power system becomes inoperable coincident with a 115 kv line de-energized, that diesel-generator power system shall be returned to an operable condition within 24 hours.

d. If a reserve power transformer becomes inoperable, it shall be returned to service within seven days.

e. For all reactor operating conditions except startup and cold shutdown, the following limiting conditions shall be in effect:

  (1) One operable diesel-generator power system and one energized 115 kv external line shall be available. If this condition is not met, normal orderly shutdown will be initiated within one hour and the reactor will be in the cold shutdown condition within ten hours,

SURVEILLANCE REQUIREMENT

c. Weekly - determine the cell voltage and specific gravity of the pilot cells of each battery.

d. Surveillance for startup with an inoperable diesel-generator - prior to startup the operable diesel-generator shall be tested for automatic startup and pickup of the load required for a loss-of-coolant accident.

e. Surveillance for operation with an inoperable diesel-generator - the operable diesel-generator shall be manually started and operated at rated load for a minimum time of one hour immediately and once per week thereafter.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(2) If no 115 kv external line is available, both diesel-generator power systems shall be operable with one diesel-generator running. If no 115 kv external line is available after 24 hours, normal orderly shutdown will be initiated within one hour and the reactor will be in the cold shutdown condition within ten hours. f. For all reactor operating conditions except cold shutdown, there shall be a minimum of two day's fuel supply onsite for one diesel-generator or normal orderly shutdown will be initiated within one hour and the reactor will be in the cold shutdown condition within ten hours. g. When operating with only one diesel-generator, all emergency equipment aligned to the operable diesel-generator shall have no inoperable components. h. If a battery system becomes inoperable that system shall be returned to service within 24 hours.	

BASES FOR 3.6.3 AND 4.6.3 EMERGENCY POWER SOURCES

Other than the Station turbine generator, the Station is supplied by four independent sources of a-c power; two 115 kv transmission lines, and two diesel-generators. Any one of the required power sources will provide the power required for the worst loss-of-coolant accident. The required loads of 2500 kva and 2750 kva for the loss-of-coolant are calculated in detail in the First Supplement to the FSAR. This loading is greater than that required during a Station shutdown condition. The monthly test run paralleled with the system is based on the manufacturer's recommendation for these units in this type of service. The testing during operating cycle will simulate the accident conditions under which operation of the diesel-generators is required. A detailed tabulation of the equipment comprising the maximum diesel-generator load is given in the answer to Question V-10 of the First Supplement to the FSAR.

As mentioned above, a single diesel-generator is capable of providing the required power to equipment following a major accident. Two fuel oil storage tanks are provided with piping interties to permit supplying either diesel-generator. A two-day supply will provide adequate time to arrange for fuel makeup if needed. The full capacity of both tanks will hold a four-day supply.

It has been demonstrated in Appendix E-I.3.21* that even with complete d-c loss the reactor can be safely isolated and the emergency cooling system will be operative with makeup water to the emergency cooling system shells maintained manually. Having at least one d-c battery available will permit: automatic makeup to the shells rather than manual, closing of the d-c actuated isolation valve on all lines from the primary system and the suppression chamber, maintenance of electrical switching functions in the Station and providing emergency lighting and communications power.

A battery system shall have a minimum of 106 volts at the battery terminals to be considered operable.

*FSAR

3.6.4 Shock Suppressors (Snubbers)

Applicability

Applies to the operational status of shock suppressors (snubbers).

Objective

To assure the capability of the snubbers to:

Prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, and

Allow normal thermal motion during startup and shutdown.

4.6.4 Shock Suppressors (Snubbers)

Applicability

Applies to the periodic testing requirement for shock suppressors (snubbers).

Objective

To assure the operability of the snubbers to perform their intended functions.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

Specification

a. During all reactor operating conditions, except cold shutdown, snubbers shall be operable on those systems required to be operable during that particular operating condition except as noted in 3.6.4.b, c and d below.

Snubbers excluded from this inspection program are those installed on nonsafety- related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

b. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to the operable status or perform an engineering evaluation to determine that the components supported by the snubber(s) were not adversely affected by the inoperability of the snubber(s), i.e. the snubber(s) is (are) not required for system operability.

c. If after 72 hours the actions as described in Section 3.6.4 b have not been completed, the supported system shall be declared inoperable and the appropriate action statement for that system will be followed.

Specification

The following surveillance requirements apply to snubbers. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

a. Visual Inspection

(i) Visual Inspection Frequency

Snubbers shall be visually inspected in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	Refueling period
1	12 months $\pm$ 25%
2	6 months $\mp$ 25%
3,4	124 days $\mp$ 25%
5,6,7	62 days $\mp$ 25%
8 or more	31 days $\mp$ 25%

The required inspection interval shall not be lengthened more than one step at a time.

d. If the actions described in 3.6.4.b or c resulted in replacement or restoration to the operable status of the effected snubber(s), perform an engineering evaluation to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber.

Snubbers may be categorized into two types (mechanical and hydraulic). These may then be classified as "accessible" or "inaccessible" based on accessibility for inspection during operation. These four groups may be inspected independently according to the above schedule.

(ii) Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined operable for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined operable per Specification 4.6.4.b as applicable..

b. Functional Testing

(i) Functional Test Frequency

At least once each refueling cycle, 10% of the total of each type (mechanical or hydraulic, accessible or inaccessible) of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of 4.6.4b(ii) an additional 10% of that type of snubber shall be functionally tested.

(ii) Functional Test Acceptance Requirement

Hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity.

2. Freedom of movement exists in both tension and compression.

Mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

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BASES FOR 3.6.4 and 4.6.4 SHOCK SUPPRESSORS (SNUBBERS)

Snubbers are required to be operable to ensure that the structural integrity of the reactor coolant system and other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the number of observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Hydraulic or mechanical, accessible or inaccessible, snubbers may each be treated as a different entity for the above surveillance programs.

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3.6.5 Radioactive Material Sources

Applicability:

Applies to the limit on source leakage for sealed or start-up sources.

Objective:

To specify the requirements necessary to limit contamination from radioactive source materials.

Specification:

1. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

2. Results of required leak tests performed on sources, if the tests reveal the presence of 0.005 microcurie or more of removable contamination, shall be reported within 90 days.

4.6.5 Radioactive Material Sources

Applicability:

Applies to the periodic testing requirements for source leakage.

Objective:

To assure the capability of each source material container to limit leakage within allowable limits.

Specification:

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except start-up sources subject to core flux, containing radioactive material, other than hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

3.6.5 Radioactive Material Sources (Continued)

Specification: (Continued)

3. A complete inventory of radioactive by-product materials, exceeding the limits set forth in 10CFR 30.71, in sealed sources in possession shall be maintained current at all times.

4.6.5 Radioactive Material Sources (Continued)

Specification: (Continued)

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Start-up sources shall be leak tested within 31 days prior to being subjected to core flux and following any repair or maintenance.

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The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the probable leakage from the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Quantities of interest to this specification which are exempt from the leakage testing are consistent with the criteria of 10 CFR Parts 30.11-20 and 70.19. Leakage from sources excluded from the requirements of this specification is not likely to represent more than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

3.6.6 FIRE DETECTION

Applicability:

Applies to the operational status of the fire detection system.

Objective:

To assure the capability of fire detection instrumentation for each fire detection zone shown in Table 3.6.6a to provide fire detection.

Specification:

a. With the number of detectors OPERABLE less than the number required by Table 3.6.6a.

   1. Within one hour, estabish a fire watch patrol to inspect the zone with the inoperable detector(s); and

   2. Restore the inoperable detector(s) to OPERABLE status within 14 days

      OR

   3. Prepare and submit a report in accordance with 6.9.2.b.

4.6.6 FIRE DETECTION

Applicability:

Applies to the periodic surveillance of the fire detection system.

Objective:

To assure the operability of the fire detection instrumentation for each fire detection zone shown in Table 3.6.6a to provide fire detection.

Specification:

a. Each of the fire detectors shall be demonstrated OPERABLE:

   1. By performance of an instrument channel test at least once per six months for all detection devices.

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #1

	LOCATION	# DETECTORS	MINIMUM # OPERABLE
1	DA-2031 - Turb. Bldg. 250 North of Cable Spread Room	17	17
2	DA-2041N - Turb. Bldg. 250 Diesel Gen. 102	12	12
3	DA-2041S - Turb. Bldg. 250 Diesel Gen. 103	6	6
4	DA-2051E - Turb. Bldg. 250 South Side East	14	14
5	DA-2051W - Turb. Bldg. 250 South Side West	16	16
6	DA-2081S - Turb. Bldg. 261 East Corridor	41	41
7	DX-2141A - Turb. Bldg. 261 Diesel Gen. 102	3	3
8	DX-2141B - Turb. Bldg. 261 Diesel Gen. 102	3	3
9	D-2151 - Turb. Bldg. 261 D.G. 103 Cable Tray	3	2 (Note 1)
10	DX-2151A - Turb. Bldg. 261 Diesel Gen. 103	3	3
11	DX-2151B - Turb. Bldg. 261 Diesel Gen. 103	3	3
12	DA-2161E - Turb. Bldg. 261 South Side East	23	23
13	DA-2161M - Turb. Bldg. 261 P.B. 11 & 12 Area	22	22
14	DX-3011A - Turb. Bldg. 250 Cable Spreading Room	6	6
15	DX-3011B - Turb. Bldg. 250 Cable Spreading Room	6	6
16	D-3031PL - Turb. Bldg. 261 Aux. Control Room Panels	99	80 (Note 1)

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #1 (continued)

	LOCATION	# DETECTORS	# OPERABLE
17	DX-3031A - Turb. Bldg. 261 Aux. Control Room	16	16
18	DX-3031B - Turb. Bldg. 261 Aux. Control Room	16	16
19	D-8151 - South Yard Foam Room	2	1
20	DA-2141 - Turb. Bldg. 261 Diesel Gen. 102	4	4
21	DA-2151 - Turb. Bldg. 261 Diesel Gen. 103	4	4

Note 1: No two (2) adjacent detectors may be out of service simultaneously.

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #2

	LOCATION	# DETECTORS	MINIMUM # OPERABLE
22	DA-2022N - Turb. Bldg. 250 North Corner	20	20
23	DA-2022S - Turb. Bldg. 250 West Side	22	22
24	DA-2092E - Turb. Bldg. 261-277 Booster Pump Area	40	40
25	DA-2092W - Turb. Bldg. 261-277 Recirc. MG Set Area	36	36
26	DA-2162W - Turb. Bldg. 261 West Side South	33	33
27	DA-2092MG - Turb. Bldg. 261 Recirc. MG Sets	15	15

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #3

	LOCATION	# DETECTORS	MINIMUM # OPERABLE
28	DA-2013S - Turb. Bldg. 250 North Side East	11	11
29	DA-2013N - Turb. Bldg. 250 Cond. Stor. Tank	16	16
30	DA-2083M - Turb. Bldg. 261 Cool Water Pump Area	44	44
31	DA-2083N - Turb. Bldg. 261 Cond. Stor. Tank	23	23
32	DX-2113A - Turb. Bldg. 261 Power Board 103 Room	1	1
33	DX-2113B - Turb. Bldg. 261 Power Board 103 Room	1	1
34	DX-2123A - Turb. Bldg. 261 Power Board 102 Room	1	1
35	DX-2123B - Turb. Bldg. 261 Power Board 102 Room	1	1
36	D-5013 - Screen House 250-261 P.B. 176 Area	6	5
37	D-5023 - Screen House 243-256 South Side	17	14 (Note 1)

Note 1: No two (2) adjacent detectors may be out of service simultaneously.

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #4

	LOCATION	# DETECTORS	MINIMUM # OPERABLE
38	D-2224 - Turb. Bldg. 277 P.B. 101 Area	22	18 (Note 1)
39	D-2234 - Turb. Bldg. 277 South East Side	27	22 (Note 1)
40	D-3054 - Turb. Bldg. 277 Control Room	26	22 (Note 1)

Note 1: No two (2) adjacent detectors may be out of service simultaneously.

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #5

	LOCATION	# DETECTORS	MINIMUM # OPERABLE
41	D-2345 - Turb. Bldg. 305 Rx Bldg. Supply Fan Area	13	11 (Note 1)
42	D-2395 - Turb. Bldg. 300 Control Ventilation Area	7	6

Note 1: No two (2) adjacent detectors may be out of service simultaneously.

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #6

	LOCATION	# DETECTORS	MINIMUM # OPERABLE
43	D-4016 - Rx Bldg. 198 Northeast Corner	2	1
44	D-4026 - Rx Bldg. 198 Northwest Corner	2	1
45	D-4036 - Rx Bldg. 198 Southwest Corner	2	1
46	D-4046 - Rx Bldg. 198 Southeast Corner	2	1
47	DA-4076E - Rx Bldg. - 237 East Side	16	16
48	DA-4076W - Rx Bldg. - 237 West Side	21	21
49	D-4086 - Rx Bldg. Drywell	9	7 (Note 1) (Note 2)
50	DA-4116E - Rx Bldg. 261 East Side	11	11
51	DA-4116W - Rx Bldg. 261 West Side	21	21
52	D-4156 - Rx Bldg. 281 West Side	16.	13 (Note 1)
53	D-4166 - Rx Bldg. 281 East Side	16	13 (Note 1)

Note 1: No two (2) adjacent detectors may be out of service simultaneously.

Note 2: Detectors in service <u>only</u> when unit is shutdown and drywell is open for major maintenance.

TABLE 3.6.6a

FIRE DETECTORS PROTECTING SAFETY-RELATED EQUIPMENT

DETECTION ZONES

LOCAL FIRE ALARM CONTROL PANEL #7

	LOCATION	# DETECTORS	MINIMUM # OPERABLE
54	D-4197 - Rx Bldg. 298 North Side	9	7 (Note 1)
55	D-4207 - Rx Bldg. 298 South Side	7	6
56	DA-4237 - Rx Bldg. 318 Storage Area	30	30
57	D-4267 - Rx Bldg. 340	13	10 (Note 1)
58	DX-4217A - Rx Bldg. 298 Emerg. Cond. Vlv. Room	4	4
59	DX-4217B - Rx Bldg. 298 Emerg. Cond. Vlv. Room	4	4

Note 1: No two (2) adjoining detectors may be out of service simultaneously.

The basic function and capabilities of the system are to provide the means to detect fires with a visual indication of its location and an audible alarm at central points, and also to control certain ancillary actions, such as extinguishment and ventilation subsequent to the detection of fire.

The system is comprised of seven (7) Local Fire Alarm Control Panels (LFACP) located throughout the Reactor Building and Turbine Building, primarily in a central location to the zones of fire detection for which each panel serves. In addition there is a Main Fire Alarm Control Panel (MFACP-2), in the Control Room to which all seven (7) panels report, and their indications and control functions are duplicated.

Five types of detection instruments are employed in the system:

a) Ionization Smoke
b) Photoelectric Smoke
c) Infrared
d) Thermal
e) Thermistor Wire

The configuration of the fire detection instrument locations has been examined and found satisfactory to detect a fire with the minimum number of detectors operable as indicated in Table 3.6.6a.

3.6.7 FIRE SUPPRESSION

Applicability:

Applies to the operational status of the fire suppression system.

Objective:

To assure the capability of the fire suppression system to provide fire suppression in the event of a fire.

Specification:

a. The FIRE SUPPRESSION WATER SYSTEM shall be OPERABLE with;

   1. Two high pressure pumps each with a capacity of 2500 gal./min. with their discharge aligned to the fire suppression header.

   2. Automatic initiation logic for each fire pump.

b. With an inoperable redundant pump or water supply line inoperable, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a report in accordance with 6.9.2.b.

4.6.7 FIRE SUPPRESSION

Applicability:

Applies to the surveillance of the fire suppression system.

Objective:

To assure the operability of the fire suppression system to provide fire suppression in the event of a fire.

Specification:

a. The FIRE SUPPRESSION WATER SYSTEM shall be demonstrated OPERABLE:

   1. At least once per 31 days by starting each pump and operating it for 30 minutes on recirculation flow.

   2. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

   3. At least once per 12 months by cycling each manually-operable valve through one complete cycle.

   4. At least once per 6 months by a flush of the hydrants.

   5. At least once per operating cycle.

      (a) By performing a system automatic start on low header pressure.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.7 FIRE SUPPRESSION (Continued)

c. With no FIRE SUPPRESSION WATER SYSTEM operable, within 24 hours:

1. Establish a backup fire suppression system, and

2. Report to the NRC in accordance with 6.9.2.a.

d. The spray and sprinkler systems located in the following areas shall be OPERABLE:

1. Automatic water spray systems

(a) Reserve Transformer 101N

(b) Reserve Transformer 101S

4.6.7 FIRE SUPPRESSION (Continued)

(b) By verifying that each pump will develop a flow of at least 2500 gpm at a pump discharge of 115 psig.

(c) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

(d) Verifying that each automatic valve in the flow path actuates to its correct position.

6. At least once per 3 years by performing a flow test of the system in accordance with Chapter 8, Section 16 of the Fire Protection Handbook, 15th Edition, published by the National Fire Protection Association.

b. The fire pump diesel engine shall be demonstrated OPERABLE:

1. Daily by checking the starting air tank pressure

2. At least once per 31 days by verifying:

(a) That the fuel day storage tank contains at least 150 gallons of fuel.

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3.6.7 FIRE SUPPRESSION (Continued)

2. Automatic Sprinkler System for the Diesel Fire Pump Room in the Screen House.

3. Pre-Action Systems:

   (a) Rx Bldg., El. 237
   (b) Rx Bldg., El. 261
   (c) Rx Bldg., El. 318
   (d) Turb. Bldg., El. 250 South
   (e) Turb. Bldg., El. 250 West
   (f) Turb. Bldg., El. 250 North
   (g) Turb. Bldg., El. 250 East
   (h) Diesel Gen., El. 250
   (i) Cable Spreading Room
   (j) Turb. Bldg., El. 261 South
   (k) Turb. Bldg., El. 261 North
   (l) Turb. Bldg., El. 261 East
   (m) Turb. Bldg., El. 277 East
   (n) Turb. Bldg., El. 300 Storage Area

e. With a spray or sprinkler system inoperable, establish a fire watch patrol with backup fire suppression equipment for the unprotected area within one hour.

f. With a pre-action system inoperable, trip system wet or establish a fire watch patrol with backup fire suppression equipment for the unprotected area within one hour.

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4.6.7 FIRE SUPPRESSION (Continued)

(b) The fuel storage tank contains at least 1000 gallons of fuel.

(c) The fuel transfer pump starts and transfers fuel from the storage tank to the day tank.

(d) The diesel starts from ambient conditions and operates for greater than or equal to 30 minutes on recirculation flow.

(e) The method of starting the diesel fire engine will alternate between the normal air start method and the low air pressure start.

3. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 with respect to viscosity, water control and sediment.

4. At least once per six months by using the manual bypass of the solenoid on the starting air system.

3.6.7 FIRE SUPPRESSION (Continued)

g. Restore the system to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2.b.

4.6.7 FIRE SUPPRESSION (Continued)

5. At least once per 18 months, subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service, and verifying the diesel starts from ambient conditions on the auto-start signal and operates for greater than or equal to 30 minutes while loaded with the fire pump.

c. The spray systems shall be demonstrated to be OPERABLE:

1. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

2. At least once per year by cycling each manually operable valve through one complete cycle.

3. At least once per operating cycle.

(a) By performing a system functional test which includes simulated automatic actuation of the system and verifying that the automatic deluge valves in the flow path actuate to their correct positions.

(b) By visual inspection of spray headers to verify their integrity.

4.6.7 FIRE SUPPRESSION (Continued)

(c) By visual inspection of each nozzle to verify no blockage.

4. At least once per 3 years by performing an air or water flow test through each open head spray header and verifying each open head spray nozzle is unobstructed.

d. The sprinkler system shall be demonstrated to be OPERABLE:

1. At least once per operating cycle.

(a) By performing a system functional test which includes simulated automatic actuation of the system.

(b) By visual inspection of sprinkler headers to verify their integrity.

(c) By visual inspection of each nozzle to verify no blockage.

The fire water supply is provided by two vertical turbine fire pumps, one electric and a diesel-driven unit which are design rated at 2500 gpm at 125 psig pump discharge head. These pumps are located in the screen house and take suction from the station cooling water intake tunnel and have relief valves set at 140 psig.

The automatic initiation logic for each fire pump indicated in Specification 3.6.7.a.2 functions such that these pumps are automatically started sequentially upon a drop in discharge header pressure. Each pump can also be manually started. In addition, the diesel fire engine will be started on low air pressure at alternate testing intervals to verify the adequacy of the low air pressure start system. A bypass of the starting air solenoid valves is provided for additional assurance in starting the diesel fire engine.

The verification of the hydraulic performance of the fire suppression water system required once per 3 years in Surveillance Requirement 4.6.7.a.5 will be done by means of a measured hydrant flow test.

The redundant components in the fire water supply system are the fire pumps, which discharge to the same header. They are the only components addressed in Specification 3.6.7.b.

The backup water supply system referenced in Specification 3.6.7.c.1 is the Oswego City water system, which can be connected to the fire main if required.

The water spray systems provide fire protection for the safety-related reserve transformers 101N and 101S. Supply for these systems is provided by the fire line. The systems employ open nozzles and are conrolled by deluge valves. Valve actuation is by pneumatic type rate-of-rise devices installed over the protected equipment.

In addition to the automatic operation, systems may be tripped manually either at the deluge valves on elevation 250' or at remote cable pull stations on elevation 261'.

The fire control panel annunciator records system operation, low supervisory air pressure and valve closure.

In addition to the spray systems described above, a closed head wet pipe automatic sprinkler system is provided for the diesel fire pump room in the Screen House on Elevation 254. The sprinkler heads used have fusible elements rated at 165°F. The system has flow alarms connected to the fire control panel annunciator.

Fourteen pre-action type systems are used for various hazards throughout the plant. These systems employ closed heads, under an air pressure of 20 psig, and are controlled by a pre-action type valve.

Valve actuation is automatic by ionization type detectors installed over the protected equipment. In addition to the automatic operation, systems may be tripped manually either at the pre-action valve or from the Main Fire Panel in Control Room.

System operation, low supervisory air pressure and valve closure is monitored on both the Main Fire Control and Local Fire Panels.

3.6.8 CARBON DIOXIDE SUPPRESSION SYSTEM

Applicability:

Applies to the operational status of the carbon dioxide suppression system.

Objective:

To assure the capability of the carbon dioxide suppression system to provide fire suppression in the event of a fire.

Specification:

a. The CO_2 system, which supplies the Recirculation Pumps Motor-Generator Sets, Power Boards 102 and 103, Diesel Generators 102 and 103, Cable Room fire hazards, shall be OPERABLE with a minimum level of 40% of tank and a minimum pressure of 250 psig in the storage tank.

b. With one or more of the above required CO_2 systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged.

c. The Auxiliary Control Room CO_2 system shall be operated as a manual backup for the Halon System.

4.6.8 CARBON DIOXIDE SUPPRESSION SYSTEM

Applicability:

Applies to the periodic surveillance requirements of the carbon dioxide suppression system.

Objective:

To verify the operability of the carbon dioxide suppression system.

Specification:

a. The CO_2 system shall be demonstrated operable.

   1. At least once per 7 days by verifying the CO_2 storage tank level and pressure.

   2. At least once per 31 days by verifying that each valve, manual power operated or automatic, in the flow path is in its correct position.

   3. At least once every six months by verifying the system valves and associated ventilation dampers actuate automatically to a simulated actuation signal. A brief flow test shall be made to verify flow from each nozzle ("Puff Test").

3.6.8 CARBON DIOXIDE SUPPRESSION SYSTEM (Continued)

d. Restore the system to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2.b.

BASES FOR 3.6.8 AND 4.6.8 CARBON DIOXIDE SUPPRESSION SYSTEM

A low pressure carbon dioxide system is installed to serve seven different safety-related hazard points in the station indicated in Specification 3.6.8.a.

Supply is provided by a 10 ton tank of liquid carbon dioxide located on elevation 261 feet. The self-contained refrigeration unit maintains the liquid at $0^{o}F$ with a resultant pressure of 300 psig. Carbon dioxide to the individual hazards is controlled by a series of carbon dioxide operated, pilot type master valves at the tank. Each of these valves serve a group of hazard valves of similar construction located at the individual areas.

Fire extinguishment by carbon dioxide is either by total flooding or local application. In total flooding, sufficient CO_2 is injected into a closed room or space to inert the atmosphere and suppress combustion. Local application is employed for unenclosed areas and involves application of CO_2 on the equipment protected to extinguish the fire with additional discharge to permit cooling and inhibit reflash.

The automatically actuated CO_2 systems employ either thermostats set at $225^{o}F$ or smoke detectors to trip a timer located in the main cardox control cabinet. One or more sirens and a strobe light in the hazard area are initially operated for a pre-discharge period of 30 seconds to enable personnel to leave the area. The related master and hazard valves are then opened for a timed discharge period. Restoration of the CO_2 hazard area to service is accomplished manually by pushbutton at the fire control panel. Manual pushbutton stations are also located at the individual areas to initiate the cycle. The control switch for each area on the fire control panel has three positions and is normally set for "Automatic" operation. An "Alarm only" position permits greater safety when men are working in the hazard area and the 30 second delay may be insufficient.

A "Manual" position permits the operator to actuate the discharge cycle on his own initiative. An area pushbutton station will override the "Alarm only" setting on the Fire Control Panel. Due to the high rate of personnel access, and thus safety requirements, the Auxiliary Control Room CO_2 system is a manual system, used to backup a total flood automatic 6% Halon system.

All CO_2 systems except hose reels are provided with odorizing devices as a safety measure. A glass flask of wintergreen concentrate is inserted in a capped tee beyond each hazard valve. This flask ruptures upon operation of the hazard and must be replaced after each use.

In the event of total loss of D.C. control power to the CO_2 system, all master valves will open since their pilot valve solenoids are normally energized. The CO_2 system hazard valves remain closed since their pilot valve solenoids are normally de-energized. CO_2 can be discharged into an area by operating the manual lever provided in each pilot valve cabinet. This is a manual operation within predischarge alarm or timer.

BASES FOR 3.6.8 AND 4.6.8 CARBON DIOXIDE SUPPRESSION SYSTEM
(Continued)

The flow test ("Puff Test") of the CO_2 system is performed by closing the CO_2 tank valve. This allows only the CO_2 vapor in the line to be discharged to the various designated areas in the plant.

Carbon dioxide hose reels are provided at various points throughout the Turbine Building. These reels are provided with 150 feet of 1" high pressure hose with manual shutoff at the nozzle. Removal of the nozzle from its mounting bracket trips a switch which opens the master valve serving the hose reels. Carbon dioxide then flows to the nozzles of all hose reels. No odorant capsules are provided for hose reels. Certain hose stations are provided with timer operated bleeder valves to discharge vapor and speed arrival of liquid CO_2 at the hose station.

All system operations are monitored on the annunciator on the fire control panel.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.9 FIRE HOSE STATIONS

Applicability:

Applies to the operational status of the fire hose stations.

Objective:

To assure the capability of the fire hose stations to provide fire suppression in the event of a fire.

Specification:

a. The fire hose stations in the locations shown in Table 3.6.9a shall be operable.

b. With one or more of the fire hose stations shown in Table 3.6.9a inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an operable hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression, otherwise route the additional hose within 24 hours.

c. Restore the inoperable fire hose station(s) to operable status within 14 days or prepare and submit a report in accordance with 6.9.2.b.

4.6.9 FIRE HOSE STATIONS

Applicability:

Applies to the periodic surveillance of the fire hose stations.

Objective:

To assure the operability of the fire hose station to provide fire suppression in the event of a fire.

Specification:

a. Each fire hose station shown in Table 3.6.9a shall be verified to be OPERABLE:

   1. At least once per 31 days by visual inspection of the fire hose stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per operating cycle by:

   1. Visual inspection of the fire hose stations not accessible during plant operation to assure all required equipment is at the hose station.

   2. Removing the hose for inspection and re-racking.

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

4.6.9 FIRE HOSE STATIONS (Continued)

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve operability and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any hose station.

Table 3.6.9a

FIRE HOSE STATIONS

	Building	Elevation (feet)	Column	Station Number
1.	TURBINE	267	Aa-7	FS-144
2.	TURBINE	267	C -3	FS-132
3.	TURBINE	267	G -2	FS-128
4.	TURBINE	267	H -9	FS-123
5.	REACTOR	346	L -12	FS-112
6.	REACTOR	346	P -4	FS-106
7.	REACTOR	324	K -11	FS-111
8.	REACTOR	324	P -5	FS-105
9.	REACTOR	309	K -11	FS-110
10.	REACTOR	304	P -5	FS-104
11.	REACTOR	287	K -11	FS-109
12.	REACTOR	287	P -5	FS-103
13.	REACTOR	267	K -11	FS-108
14.	REACTOR	267	P -5	FS-102
15.	REACTOR	243	K -11	FS-107
16.	REACTOR	243	P -5	FS-101
17.	WASTE	267	H -19	FS-301
18.	WASTE	267	MB-19	FS-300
19.	WASTE	267	MB-16	FS-116
20.	TURBINE	375	H -8	FS-126
21.	TURBINE	357	H -10	FS-121
22.	TURBINE	311	H -9	FS-125
23.	TURBINE	311	G -2	FS-130
24.	TURBINE	311	C -3	FS-134
25.	TURBINE	297	H -9	FS-124
26.	TURBINE	297	G -2	FS-129
27.	TURBINE	267	F -15	FS-117
28.	DIESEL	267	C -18	FS-164
29.	DIESEL	267	Ba-17	FS-166

Table 3.6.9a

FIRE HOSE STATIONS
(Continued)

	Building	Elevation (feet)	Column	Station Number
30.	TURBINE	256	Aa-13	FS-152
31.	DIESEL	256	Aa-17	FS-163
32.	DIESEL	256	C -17	FS-165
33.	TURBINE	256	H -9	FS-122
34.	TURBINE	267	Aa-14	FS-156
35.	TURBINE	267	B -2	FS-139
36.	TURBINE	267	P -14	FS-114
37.	TURBINE	297	C -3	FS-133
38.	TURBINE	283	B -2	FS-140
39.	TURBINE	283	Aa-7	FS-145
40.	TURBINE	283	Aa-13	FS-153
41.	TURBINE	283	F -15	FS-118
42.	TURBINE	256	Aa-7	FS-143
43.	TURBINE	256	B -2	FS-138
44.	TURBINE	256	C -3	FS-131
45.	TURBINE	256	G -2	FS-127
46.	TURBINE	256	M -13	FS-115
47.	SCREEN	256	UV-16	FS-408
48.	TURBINE	277	H -9	FS-405
49.	TURBINE	256	F -16	FS-406
50.	REACTOR	267	Track Bay	FS-401
51.	REACTOR	267	M -12	FS-422
52.	REACTOR	243	P -9	FS-402
53.	REACTOR	243	Drywell Entrance	FS-403
54.	REACTOR	243	P-Q	FS-404
55.	SCREEN	262	R-14	FS-113
56.	TURBINE	306	Aa-13	FS-154

BASES FOR 3.6.9 AND 4.6 IRE HOSE STATIONS

Standpipe risers at various locations in the turbine, reactor, waste and diesel buildings serve hose connections. This equipment is located to permit hose stream coverage of safety-related equipment in the buildings. Each hose connection is equipped with 100 feet of 1 1/2 inch hose mounted on a reel.

All hand line nozzles are of the adjustable spray type which can be varied down to 10^{0} minimum spray pattern to render them safe for use on electrical equipment. Eight foot long applicator spray nozzles and foam induction nozzles with five gallon cans of foam solution are also provided for use on hose lines as required.

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LIMITING CONDITION FOR OPERATION

3.6.10 ADDITIONAL FIRE EQUIPMENT

3.6.10.1 FIRE BARRIER PENETRATIONS

Applicability:

Applies to the condition of the fire barrier penetrations, including cable penetration barriers, fire doors and fire dampers.

Objective:

To assure the capability of the fire barrier penetrations to perform their intended function.

Specification:

a. All fire barrier penetrations protecting safety related areas shall be intact.
b. With one or more of the above required fire barrier penetrations non-functional, within one hour establish a continuous fire watch on one side of the affected penetration, or
c. Verify the operability of fire detectors on one side of the non-functional fire barrier and establish a fire watch patrol.
d. Restore the non-functional fire barrier penetrations to functional status within 14 days or prepare and submit a report in accordance with 6.9.2.b.

SURVEILLANCE REQUIREMENT

4.6.10 ADDITIONAL FIRE EQUIPMENT

4.6.10.1 FIRE BARRIER PENETRATIONS

Applicability:

Applies to the periodic surveillance requirements for the fire barrier penetrations.

Objectives:

To verify the condition of the fire barrier penetrations.

Specification:

a. Fire barrier penetrations shall be verified to be functional by:

   1. A visual inspection at least once per opearting cycle.

   2. A visual inspection of a fire barrier penetration after repair or maintenance, prior to restoring the fire barrier penetration to functional status.

BASES FOR 3.6.10.1 AND 4.6.10.1 FIRE BARRIER PENETRATION FIRE SEALS

Cable penetrations of the primary containment (drywell and pressure suppression chamber), reactor building, auxiliary control room and the cable room have been designed to provide adequate fire stop and to prevent a fire from spreading through the penetration. Drywell and pressure suppression chamber penetrations are double-sealed, 12-inch pipes that are inerted with nitrogen. Reactor building penetrations consist of standard conduit (pipe) sleeves, which vary in diameter from 3/4" to 4" and which are sealed at both ends. The auxiliary control room and the cable room have formed pipe sleeves and cable tray penetrations. These sleeves and penetrations are sealed at the ends with rock-wool filler and externally applied fire-resistant material for fire proofing.

LIMITING CONDITION FOR OPERATION

3.6.10.2 HALON SUPPRESSION SYSTEM

Applicability:

Applies to the operational status of the Halon suppression system.

Objective:

To assure the capability of the Halon suppression system to provide fire suppression in the event of a fire.

Specification:

a. The Halon systems which supply the Auxiliary Control and Emergency Condenser I.V. Rooms shall be operable with the storage tanks having at least 95% of full charge weight (level) and 90% of full charge pressure.
b. With a Halon system inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment.
c. Restore the system to operable status within 14 days or prepare and submit a report in accordance with 6.9.2.b.

SURVEILLANCE REQUIREMENT

4.6.10.2 HALON SUPPRESSION SYSTEM

Applicability:

Applies to the periodic surveillance requirement of the Halon suppression system.

Objective:

To verify the operability of the Halon suppression system.

Specification:

a. Each of the required Halon systems shall be demonstrated operable:

   1. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

   2. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

   3. At least once per 18 months by:

      (a) Verifying the system and associated ventilation dampers and fire door release mechanisms actuate manually and automatically.

      (b) Performance of a flow test through headers and nozzles to assure no blockage.

BASES FOR 3.6.10.2 AND 4.6.10.2 HALON SUPPRESSION SYSTEM

The Halon 1301 fire protection systems are a gaseous fire suppressant system used in the Auxiliary Control and the Emergency Condenser Isolation Valve Rooms.

The Halon 1301 fire protection system is comprised of a fire detection system status monitoring network and a fire suppression system. The fire detection system's status monitoring network monitors the areas covered by the Halon 1301 systems for fire conditions and system abnormalities. The fire suppression system consists of storage tanks of Halon 1301 and a delivery system to route Halon 1301 to the affected area in the event of a fire.

Fire extinguishment by Halon 1301 is by total flooding. In total flooding, sufficient Halon is injected into the area to extinguish the fire.

Both Halon systems are provided with odorizing devices as a safety measure. A glass flask of wintergreen concentrate is inserted in a capped tee in the main line piping. This flask ruptures upon operation of the system and must be replaced after each operation.

A siren and strobe light in the protected area are initially operated for a pre-discharge period of 30 seconds to enable personnel to leave the area.

Both systems may be manually tripped, either from the Main Fire Control Panel or at the storage banks.

3.6.10.3 YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

Applicability:

Applies to the operational status of the Yard Fire Hydrants and Hose Houses.

Objectives:

To assure the capability of the yard fire hydrant to provide fire suppression in the event of a fire.

Specification:

a. The yard fire hydrants shown in Table 3.6.10.3a shall be operable.
b. With one or more of the yard fire hydrants or associated hydrant houses shown in Table 3.6.10.3a inoperable, route sufficient additional lengths of 2-1/2 inch diameter hose located in an adjacent operable hydrant hose house to provide service to the unprotected area(s) within one hour, if the inoperable fire hydrant is the primary means of fire suppression, otherwise, route an additional hose within 24 hours.
c. Restore the inoperable hydrant(s) and/or hose house(s) to operable status within 14 days or prepare and submit a report in accordance with 6.9.2.b.

4.6.10.3 YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

Applicability:

Applies to the periodic surveillance requirement of the yard fire hydrants and associated hose houses.

Objective:

To assure the operability of the yard fire hydrant to provide fire suppression in the event of a fire.

Specification:

a. Each of the yard fire hydrants and associated hose houses shown in Table 3.6.10.3a shall be demonstrated operable:

   1. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

   2. At least once per 6 months during March, April, May and during September, October and November by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

   3. At least once per 12 months by:

      (a) Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any yard fire hydrant.
      (b) Replacement of all degraded gaskets in couplings.

TABLE 3.6.10.3a

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

Hydrant Number	Location
3	El. 261' - West
4	El. 261' - Southwest

3.6.11 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs an accident monitoring function.

Objective:

To assure high reliability of the accident monitoring instrumentation.

Specification:

a. During the power operating condition, the accident monitoring instrumentation channels shown in Table 3.6.11-1 shall be operable except as specified in Table 3.6.11-2.

4.6.11 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs an accident monitoring function.

Objective:

To verify the operability of accident monitoring instrumentation.

Specification:

Instrument channels shall be tested and calibrated at least as frequently as listed in Table 4.6.11.

TABLE 3.6.11-1

ACCIDENT MONITORING INSTRUMENTATION

Parameters	Total Number of Channels	Minimum Number of Operable Sensors or Channels	Action (See Table 3.6.11-2)
1) Relief Valve Position Indication	2/Valve	1/Valve	1
2) Safety Valve Position Indication	2/Valve	1/Valve	1
3) Reactor Vessel Water Level	2	1	2
4) Drywell Pressure Monitor	2	1	4
5) Suppression Chamber Water Level	2	1	4
6) Containment Hydrogen Monitor	2	1	4
7) Containment High Range Radiation Monitor	2	1	3
8) Suppression Chamber Water Temperature	2	1	2

TABLE 3.6.11-2

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION - 1

a. With the number of OPERABLE accident monitoring instrumentation channels 1 less than the total number shown in Table 3.6.11-1, restore to an OPERABLE status during the next cold shutdown when there is access to the drywell.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the minimum number shown in Table 3.6.11-1, restore the inoperable channel to an OPERABLE status within 30 days or be in at least a HOT SHUTDOWN within the next 12 hours.

c. The total number of channels shown in Table 3.6.11-1 will be OPERABLE prior to the beginning of each cycle.

ACTION -2

a. With the number of OPERABLE accident monitoring instrumentation channels less than the total Number of Channels shown in Table 3.6.11-1, restore the inoperable channel(s) to OPERABLE status within seven days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the minimum Channels OPERABLE requirements of Table 3.6.11-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION - 3

a. With the number of OPERABLE channels less than the total Number of Channels shown in Table 3-6.11-1, prepare and submit a Special Report to the Commission within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. With the number of OPERABLE channels less than required by the minimum channels OPERABLE requirements, initiate the pre-planned alternate method of monitoring the appropriate parameter(s) within 72 hours, and :

   1) either restore the inoperable channel(s) to OPERABLE status within seven days of the event, or

   2) prepare and submit a Special Report to the Commission within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION - 4

a. With the number of OPERABLE channels less than the total Number of Channels shown in Table 3-6.11-1, prepare and submit a Special Report to the Commission within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. With the number of OPERABLE channels less than required by the minimum channels OPERABLE requirements, initiate the pre-planned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

   1) either restore the inoperable channel(s) to OPERABLE status within seven days of the event, or

   2) prepare and submit a Special Report to the Commission within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE system.

c. If the pre-planned alternate method of monitoring the appropriate parameter(s) is not available, either restore the inoperable channel(s) to OPERABLE status within seven days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.6.11

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENT

Parameter	Instrument Channel Test	Instrument Channel Calibration
(1) Relief valve position indicator (Primary - Acoustic)	Once per month	Once during each major refueling outage
Relief valve position indicator (Backup - Thermocouple)	Once per month	Once during each major refueling outage
(2) Safety valve position indicator (Primary - Acoustic)	Once per month	Once during each major refueling outage
Safety valve position indicator (Backup - Thermocouple)	Once per month	Once during each major refueling outage
(3) Reactor vessel water level	Once per month	Once during each major refueling outage
(4) Drywell Pressure Monitor	Once per month	Once during each major refueling outage
(5) Suppression Chamber Water Level Monitor	Once per month	Once during each major refueling outage
(6) Containment Hydrogen Monitor	Once per month	Once per quarter
(7) Containment High Range Radiation Monitor	Once per month	Once during each major refueling outage
(8) Suppression Chamber Water Temperature	Once per month	Once during each major refueling outage

BASES 3.6.11 AND 4.6.11 ACCIDENT MONITORING INSTRUMENTATION

Accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," and/or NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 and NUREG 0661, "Safety Evaluation Report Mark I Containment Long Term Program.".

The maximum allowable setpoint deviation for the Suppression Chamber Water Level instrumentation is ± 1.8 inches.

LIMITING CONDITION FOR OPERATION

3.6.12 REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET MONITORING

Applicability:

Applies to the operability of instrumentation that provides protection of Motor Generator sets and the maintenance bus that supplies power to the reactor protection system and reactor trip system.

Objective:

To assure the operability of the instrumentation required for safe operation of the Motor Generator sets and the maintenance bus that supplies power to the reactor protection system and reactor trip system.

Specification:

a. Except as specified in specifications b and c below, two protective relay systems shall be operable for each Motor Generator set and the maintenance bus.

SURVEILLANCE REQUIREMENT

4.6.12 REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET MONITORING

Applicability:

Applies to the surveillance of instrumentation that provides protection of the reactor protection Motor Generator sets and maintenance bus that supplies power to the reactor protection system and reactor trip system.

Objective:

To verify the operability of protection instrumentation on the Motor Generator sets and maintenance bus that supplies power to the reactor protection and reactor trip buses.

Specification:

a. At least once every six months Demonstrate operability of the overvoltage, undervoltage and under frequency protective instrumentation by performing an instrument channel test. This instrument channel test will consist of simulating abnormal Motor Generator Set conditions by applying from a test source, an overvoltage signal, an undervoltage signal and an underfrequency signal to verify that the tripping logic up to but not including the output contactors functions properly.

LIMITING CONDITION FOR OPERATION

3.6.12 REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET MONITORING (cont'd)

Specification: (cont'd)

b. With one protective relaying system inoperable, restore the inoperable system to an operable status within 72 hours or remove the Motor Generator set or maintenance bus from service.

c. With both protective relaying systems inoperable, restore at least one to an operable status within 30 minutes or remove the associated Motor Generator sets or maintenance bus from service.

SURVEILLANCE REQUIREMENT

4.6.12 REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET MONITORING (cont'd)

Specification:

b. At least once per refueling cycle Demonstrate operability of the overvoltage, undervoltage and underfrequency protective instrumentation by performing an instrument channel test. This instrument channel test will consist of simulating abnormal Motor Generator Set conditions by applying from a test source an overvoltage signal, an undervoltage signal and an underfrequency signal to verify that the tripping logic including the output contactors functions properly at least once. In addition, a sensor calibration will be performed to verify the following setpoints.

   i. Overvoltage $\leq$132 volts, $\leq$4 seconds
   ii. Undervoltage $\geq$108 volts, $\leq$4 seconds
   iii. Underfrequency $\geq$ 57 hertz, $\leq$2 seconds

BASES FOR 3.6.12 and 4.6.12 REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET MONITORING

To eliminate the potential for undetectable single component failure which could adversely affect the operability of the reactor protection system, protection relaying schemes installed on MG sets 131, 141, 162, 172 and maintenance bus 130, provide for overvoltage, undervoltage and underfrequency protection.

3.6.13 REMOTE SHUTDOWN PANELS

Applicability:

Applies to the operating status of the remote shutdown panels.

Objective:

To assure the capability of the remote shutdown panels to provide 1) initiation of the emergency condensers independent of the main/auxiliary control room 2) control of the motor-operated steam supply valves independent of the main/auxiliary control room and 3) parameter monitoring outside the control room.

Specification:

a. During power operation and whenever the reactor coolant temperature is greater than 212°F, at least one remote shutdown panel shall be operable.

4.6.13 REMOTE SHUTDOWN PANELS

Applicability:

Applies to the periodic testing requirements for the remote shutdown panels.

Objective:

To assure the capability of the remote shutdown panels to provide 1) initiation of the emergency condensers independent of the main/auxiliary control room 2) control of the motor-operated steam supply valves independent of the main/auxiliary control room and 3) parameter monitoring outside the control room.

Specification:

The remote shutdown panels surveillance shall be performed as indicated below:

a. Each remote shutdown panel monitoring instrumentation channel shall be demonstrated operable by performance of the operations and frequencies shown in Table 4.6.13-1.

b. During each major refueling outage

   1. Each remote shutdown panel shall be demonstrated to initiate the emergency condensers independent of the main/auxiliary control room.

3.6.13 REMOTE SHUTDOWN PANELS (Continued)

b. A remote shutdown panel shall be considered inoperable if either the emergency condenser condensate return valve control switch is inoperable, either motor-operated steam supply valve control switch is inoperable, or the number of operable instrumentation channels is less than that required by Table 3.6.13-1.

c. If Specification 3.6.13.a cannot be met, commence an orderly shutdown within 24 hours and be in cold shutdown within 36 hours.

4.6.13 REMOTE SHUTDOWN PANELS (Continued)

2. Each remote shutdown panel shall be demonstrated to open both the motor-operated steam valves.

TABLE 3.6-13-1

REMOTE SHUTDOWN PANEL MONITORING

Limiting Condition for Operation

INSTRUMENT	MINIMUM NUMBER OF OPERABLE CHANNELS
Reactor Pressure	1
Reactor Water Level	1
Reactor Water Temperature	1
Torus Water Temperature	1
Drywell Pressure	1
Emergency Condenser Water Level	1
Drywell Temperature	1
"All Rods In" Light	1

TABLE 4.6.13-1

REMOTE SHUTDOWN PANEL MONITORING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Calibration
Reactor Pressure	Once per day	Once per 3 months (a)
Reactor Water Level	Once per day	Once per 3 months (a)
Reactor Water Temperature	Once per day	Once per refueling cycle
Torus Water Temperature	Once per day	Once per refueling cycle
Drywell Pressure	Once per day	Once per 3 months (a)
Emergency Condenser Water Level	Once per day	Once per refueling cycle
Drywell Temperature	Once per day	Once per refueling cycle
"All Rods In" Light	Once per refueling cycle	N/A

(a) The indicator located at the remote shutdown panel will be calibrated at the frequency listed in Table 4.6.13-1. Calibration of the remaining channel instrumentation is provided by Specification 4.6.2.

The remote shutdown panels provide 1) manual initiation of the emergency condensers 2) manual control of the steam supply valves and 3) parameters monitoring independent of the main/auxiliary control room. Two panels are provided, each located in a separate fire area, for added redundancy. Both panels are also in separate fire areas from the main/auxiliary control room. One remote shutdown panel provides the necessary capabilities consistent with 10CFR50 Appendix R. Therefore, only one remote shutdown panel is required to be operable. The electrical design of the panels is such that no single fire can cause loss of both emergency condensers.

Each remote shutdown panel is provided with controls for one emergency condenser loop. The emergency condensers are designed such that automatic initiation is independently assured in the event of a fire 1) in the Reactor Building (principle relay logic located in the auxiliary control room or 2) in the main/auxiliary control room or Turbine Building (redundant relay logic located in the Reactor Building). Each remote shutdown panel also has controls to operate the two motor-operated steam supply valves on its respective emergency condenser loop. A key operated bypass switch is provided to override the automatic isolation signal to these valves. Once the bypass switch is activated, the steam supply valves can be manually controlled from the remote shutdown panels. Since automatic initiation of the emergency condenser is assured, the remote shutdown panels serve as additional manual controlling stations for the emergency condensers. In addition, certain parameters are monitored at each remote shutdown panel.

The remote shutdown panels are normally de-energized, except for the monitoring instrumentation, which is normally energized. To energize the remaining functions on a remote shutdown panel, a power switch located on each panel must be activated. Once the panels are completely energized, the emergency condenser condensate return valve and steam supply valve controls can be utilized.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION

Applicability:

Applies to the operability of plant instrumentation that monitors plant effluents.

Objective:

To assure the operability of instrumentation to monitor the release of radioactive plant effluents.

Specification:

a. Liquid Effluent

The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.6.14-1 shall be operable with their alarm setpoints set to ensure that the limits of Specification 3.6.15.a.1 are not exceeded. The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

With a radioactive liquid effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of 3.6.15.a.1 are met, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

4.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION

Applicability:

Applies to the surveillance of instrumentation that monitors plant effluents.

Objective:

To verify operation of monitoring instrumentation.

Specification:

a. Liquid Effluent

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable by performance of the sensor check, source check instrument channel calibration and channel test operations at the frequencies shown in Table 4.6.14-1.

Records - Auditable records shall be maintained, in accordance with procedures in the Offsite Dose Calculation Manual, of all radioactive liquid effluent monitoring instrumentation alarm setpoints. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.6.15.a.1 are met.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION (Cont) a. Liquid Effluent (Continued) With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the action shown in Table 3.6.14-1. Restore the instruments to OPERABLE status within 30 days, or outline in the next Semi-Annual Radioactive Effluent Release Report the cause of the inoperability and how the instruments were or will be restored to operable status.	

Table 3.6.14-1
RADIOACTIVE LIQUID EFFLUFNT MONITORING INSTRUMENTATION

Limiting Condition for Operation

Instrument	Minimum Channels Operable	Applicability
1. Gross Radioactivity Monitors [a]		
A. Liquid Radwaste Effluent Line	1[c]	At all times [b]
B. Service Water System Effluent Line	1[d]	At all times [i]
2. Flow Rate Measurement Devices		
A. Liquid Radwaste Effluent Line	1[e]	At all times
B. Discharge Canal	**	**
3. Tank Level Indicating Devices [g]		
A. Outside Liquid Radwaste Storage Tanks	1[f]	At all times

**Pumps curves or rated capacity will be utilized to estimate flow.

NOTES FOR TABLE 3.6.14-1

(a) Provide alarm, but do not provide automatic termination of release.

(b) An operator shall be present in the Radwaste Control Room at all times during a release.

(c) With the number of channels operable less than required by the minimum channels operable requirement, effluent releases may continue provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Specification 4.6.15.a, and

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving.

Otherwise suspend release of radioactive effluents via this pathway.

(d) With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gamma radioactivity at a lower limit of detection of at least $5x10^{-7}$ microcurie/ml.

(e) During discharge, with the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases.

(f) With the number of channels operable less than required by the minimum channels operable requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during liquid additions to the tank.

(g) Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents.

(h) deleted

(i) Monitoring will be conducted continuously by alternately sampling the reactor building and turbine building service water return lines for approximately 15-minute intervals.

Table 4.6.14-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirement

Instrument	Sensor Check	Source Check(f)	Channel Test	Channel Calibration
1. Gross Beta or Gamma Radioactivity Monitors				
a. Liquid Radwaste Effluent Line	Once/day*	Once/ discharge*	Once/3 months(a)*	Once/year (b)*
b. Service Water Effluent Line	Once/day	Once/month	Once/3 months(a)	Once/year(b)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	Once/day(c)	None	None	Once/year
b. Discharge Canal (d)	None	None	None	Once/year
3. Tank Level Indicating Devices (e)				
a. Outside Liquid Radwaste Storage Tanks	Once/day**	None	Once/3 months	Once/18 months

* Required prior to removal of blank flange in discharge line and until blank flange is replaced.
** During liquid addition to the tank.

NOTES FOR TABLE 4.6.14-1

(a) The channel test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrumentation indicates measured levels above the alarm setpoint.

2. Instrument indicates a downscale failure.

3. Instrument controls not set in operate mode.

(b) The channel calibration shall be performed using one or more reference standards certified by the National Bureau of Standards or using standards that are traceable to the National Bureau of Standards or using actual samples of liquid waste that have been analyzed on a system that has been calibrated with National Bureau of Standard traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement.

(c) Sensor check shall consist of verifying indication of flow during periods of release. Sensor check shall be made at least once per 24 hours on days on which continuous, periodic or batch releases are made.

(d) Pump performance curves or rated data may be used to estimate flow.

(e) Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents.

(f) Source check may consist of an installed check source, response to an external source, or (for liquid radwaste monitors) verification within 30 minutes of commencing discharge of monitor response to effluent.

3.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION (Cont)

b. Gaseous Process and Effluent

The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.6.14-2 shall be operable with their alarm setpoints set to ensure that the limits of Specification 3.6.15.b.1 are not exceeded. The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

With a radioactive gaseous process and effluent monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

With less than the minimum number of radioactive gaseous process and effluent monitoring instrumentation channels operable, take the action shown in Table 3.6.14-2. Restore the instruments to OPERABLE status within 30 days or outline in the next Semi-Annual Radioactive Effluent Release Report the cause of the inoperability and how the instruments were or will be restored to operable status.

4.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION (Cont)

b. Gaseous Process and Effluent

Each radioactive gaseous process and effluent monitoring instrumentation channel shall be demonstrated operable by performance of the sensor check, source check, instrument channel calibration and instrument channel test operations at the frequencies shown in Table 4.6.14-2.

Auditable records shall be maintained of the calculations made, in accordance with procedures in the Offsite Dose Calculation Manual, of radioactive gaseous process and effluent monitoring instrumentation alarm setpoints. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.6.15.b.1 are met.

ıable 3.6.14-2

RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

Limiting Condition for Operation

Instrument	Minimum Channels Operable	Applicability	Action
1. Stack Effluent Monitoring			
a. Noble Gas Activity Monitor	1	*	(a)
b. Iodine Sampler Cartridge	1	*	(b)
c. Particulate Sampler Filter	1	*	(b)
d. Sample Flow Rate Measuring Device	1	*	(c)
e. Stack Gas Flow Rate Measuring Device	1	*	(d)
2. Main Condenser Offgas Treatment Explosive Gas Monitoring System			
a. Hydrogen Monitor (f)	1	**	(e)

* At all times.
** During Offgas System Operation.

ıe 3.6.14-2
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION (Continued)

Instrument	Minimum Channels Operable	Applicability	Action
3. Condenser Air Ejector Radioactivity Monitor (Recombiner discharge or air ejector discharge)			
a. Noble Gas Activity	1	***	(g)
b. Offgas System Flow Rate Measuring Devices	1	***	(c)
c. Sampler Flow Rate Measuring Devices	1	***	(c)
4. Emergency Condenser System			
a. Noble Gas Activity Monitor	1 per vent	****	(h)

*** During operation of the main condenser air ejector
**** During reactor power operating condition

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NOTES FOR TABLE 3.6.14-2

(a) With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are taken once per 12 hours and these samples are analyzed for gross activity within 24 hours.

(b) With the number of channels operable less than required by the minimum channels operable requirements, effluent releases via this pathway may continue provided that samples are continuously collected with auxiliary sampling equipment starting within 8 hours of discovery in accordance with the requirements of Table 4.6.15-2.

(c) With the number of channels operable less than required by the minimum channels operable requirements, effluent releases via this pathway may continue provided the flow rate is estimated once per 8 hours.

(d) Stack gas flow rate may be estimated by exhaust fan operating configuration.

(e) With the number of channels operable less than required by the minimum channels operable requirement, operation of the main condenser offgas treatment system may continue provided gas samples are collected and analyzed once per 8 hours.

(f) One monitor on each recombiner. The system is designed to withstand the effects of a hydrogen explosion.

(g) With the number of channels operable less than required by the minimum channels operable requirement, gases from the main condenser offgas treatment system may be released provided:

1. Offgas grab samples are collected and analyzed once per 12 hours.
2. The stack monitor is operable.
3. Otherwise, be in at least hot shutdown within 12 hours.

(h) With the number of channels operable less than required by the minimum channels operable requirements, steam release via this pathway may commence or continue provided vent pipe radiation dose rates are monitored once per four hours.

Table 4.6.14-2
RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENT

Surveillance Requirements

Instrument	Sensor Check	Source Check	Channel Test	Channel Calibration
1. Stack Effluent Monitoring System				
a. Noble Gas Activity Monitor	Once/day (a)	Once/month	Once/3 months(g)	Once/year(b)
b. Iodine Sampler Cartridge	None	None	None	None
c. Particulate Sampler Filter	None	None	None	None
d. Sampler Flow Rate Measuring Device	Once/day (a)	None	None	Once/year
e. Stack Gas Flow Rate Measuring Device	Once/day	None	None	Once/year
2. Main Condenser Offgas Treatment System Explosive Gas Monitoring system (for system designed to withstand the effects of a hydrogen explosion)				
a. Hydrogen Monitor	Once/day (d)	None	Once/month	Once/3 months(e)
3. Condenser Air Ejector Radioactivity Monitor (Recombiner Discharge or Air Ejector Discharge)				
a. Noble Gas Activity Monitor	Once/day (f)	Once/month	Once/operating cycle(c)	Once/year(b)
b. Flow Rate Monitor	Once/day (f)	None	None	Once/year
c. Sampler Flow Rate Monitor	Once/day (f)	None	None	Once/year
4. Emergency Condenser System				
a. Noble Gas Activity Monitor	Once/day(h)	Once/month	Once/3 months(g)	Once/operating cycle(b)

NOTES FOR TABLE 4.6.14-2

(a) At all times.

(b) The channel calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards, standards that are traceable to the National Bureau of Standards or using actual samples of gaseous effluent that have been analyzed on a system that has been calibrated with National Bureau of Standards traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement.

(c) The channel function test shall demonstrate that control room alarm annunciation occurs if either of the following conditions exist:

1) Instrument indicates measured levels above the Hi or Hi Hi alarm setpoint.

2) Instrument indicates a downscale failure.

The channel function test shall also demonstrate that automatic isolation of this pathway occurs if either of the following conditions exist:

1) Instruments indicate two channels above Hi Hi alarm setpoint.

2) Instruments indicate one channel above Hi Hi alarm setpoint and one channel downscale.

(d) During main condenser offgas treatment system operation.

(e) The channel calibration shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen.

2. Four volume percent hydrogen, balance nitrogen.

(f) During operation of the main condenser air ejector.

(g) The channel test shall produce upscale and downscale annunciation.

(h) During reactor power operating condition

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BASES FOR RADIOACTIVE EFFLUENT INSTRUMENTATION 3.6.14 and 4.6.14

The radioactive liquid and gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid and gaseous effluents during actual or potential releases of liquid and gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to unrestricted areas.

3.6.15 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the radioactive effluents from the station.

Objective:

To assure that radioactive material is not released to the environment in any uncontrolled manner and is within the limits of 10CFR20 and 10CFR50 Appendix I.

Specification:

a. Liquid

(1) Concentration

The concentration of radioactive material released in liquid effluents to unrestricted areas shall be limited to the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} microcuries/ml total activity.

Should the concentration of radioactive material released in liquid effluents to unrestricted areas exceed the above limits, restore the concentration to within the above limits immediately.

4.6.15 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the periodic test and recording requirements of the station process effluents.

Objective:

To ascertain that radioactive effluents from the station are within allowable values of 10CFR20, Appendix B and 10CFR50, Appendix I.

Specification:

a. Liquid

(1) Concentration

Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.6.15-1.

The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the Offsite Dose Calculation Manual to assure that the concentrations at the point of release are maintained within the limits of Specification 3.6.15.a.(1)

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.15 RADIOACTIVE EFFLUENTS (Continued)	4.6.15 RADIOACTIVE EFFLUENTS (Continued)

3.6.15 <u>RADIOACTIVE EFFLUENTS</u> (Continued)

a. Liquid (Continued)

(2) Dose

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released, from each reactor unit, to unrestricted areas (see Figures 5.1-1) shall be limited:

(a) During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

(b) During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3 a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

4.6.15 <u>RADIOACTIVE EFFLUENTS</u> (Continued)

a. Liquid (Continued)

(2) Dose

Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual, prior to each release of a batch of liquid waste.

Table 4.6.15-1
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
Surveillance Requirement

Liquid Release Type	Minimum Sampling Frequency	Analysis Frequency	Type of Activity Analysis	Lower Limit[a] of Detection (LLD) (uCi/ml)
A. Batch Waste [b] Tanks	* Each Batch	* Each Batch	Principal Gamma [c] Emitters	5×10^{-7}
			I-131	1×10^{-6}
	* Each Batch[d]	* Each Batch [d]	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	* Each Batch	Monthly Composite[e]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	* Each Batch	Quarterly Composite[e]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Service Water System Effluent	Once/month[f]	Once/month[f]	Principal Gamma [c] Emitters	5×10^{-7}
			I-131	1×10^{-6}
			Dissolved and Entrained Gases	1×10^{-5}
			H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Once/quarter[f]	Once/quarter[f]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

* Completed prior to each release.

NOTES FOR TABLE 4.6.15-1

(a) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count above system background that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal. For a particular measurement system which may include radiochemical separation:

$$LLD = \frac{4.66\ S_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact for a particular measurement.

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NOTES FOR TABLE 4.6.15-1 (Continued)

(b) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semi-Annual Radioactive Effluent Release Report

(d) If more than one batch is released in a calendar month, only one batch need be sampled and analyzed during that month.

(e) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(f) If the alarm setpoint of the service water effluent monitor, as determined by the method presented in the Offsite Dose Calculation Manual, is exceeded, the frequency of sampling shall be increased to daily until the condition no longer exists. Frequency of analysis shall be increased to daily for principal gamma emitters (including dissolved and entrained gases) and an incident composite for H-3, gross alpha, Sr-89, Sr-90 and Fe-55.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.15 <u>RADIOACTIVE EFFLUENTS</u> (Continued) b. Gaseous (1) Dose Rate The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following: (a) For noble gases: Less than or equal to 500 mrems/year to the total body and less than or equal to 3000 mrems/year to the skin, and (b) For iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/year to any organ. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).	4.6.15 <u>RADIOACTIVE EFFLUENTS</u> (Continued) b. Gaseous (1) Dose Rate The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.6.15 in accordance with the methodology and parameters in the Offsite Dose Calculation Manual. The dose rate due to iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.6.15 in accordance with methodology and parameters in the Offsite Dose Calculation Manual by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.6.15-2.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.15 RADIOACTIVE EFFLUENTS (Continued)	4.6.15 RADIOACTIVE EFFLUENTS (Continued)
b. Gaseous (Continued)	b. Gaseous (Continued)
(2) Air Dose	(2) Air Dose
The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:	Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.
(a) During any calendar quarter: Less than or equal to 5 milliroentgen for gamma radiation and less than or equal to 10 mrads for beta radiation and,	
(b) During any calendar year: Less than or equal to 10 milliroentgen for gamma radiation and less than or equal to 20 mrads for beta radiation.	
With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous (Continued)

(3) Tritium, Iodines and Particulates

The dose to a member of the public from iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:

(a) During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

(b) During any calendar year: Less than or equal to 15 mrems to any organ.

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous (Continued)

(3) Tritium, Iodines and Particulates

Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days shall be determined monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.15 RADIOACTIVE EFFLUENTS (Continued) b. Gaseous (Continued) With the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	

Table 4.6.15-2
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Surveillance Requirements

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit[a] of Detection (LLD) (uCi/ml)
A. Containment Purge [b]	Each Purge	Prior to each release	Principal Gamma Emmiters [c]	1×10^{-4}
	Grab Sample	Each Purge	Principal Gamma Emitters [c]	1×10^{-4}
			H-3	1×10^{-6}
B. Stack	Once/Month [d]	Once/Month [d]	Principal Gamma Emitters [c]	1×10^{-4}
	Once/Month [h]	Once/Month	H-3	1×10^{-6}
C. Stack	Continuous [e]	Once/Week [f] Charcoal Sample	I-131	1×10^{-12}
	Continuous [e]	Once/Week [f] Particulate Sample	Principal Gamma Emitters [c]	1×10^{-11}
	Continuous [e]	Once/Month Composite Particulate Sample	Gross alpha Sr-89, Sr-90	1×10^{-11}
	Continuous [e]	Noble gas monitor	Noble Gases, Gross Gamma or Principal Gamma Emitters [c]	1×10^{-6}[g]

NOTES FOR TABLE 4.6.15-2

(a) The LLD is defined in notation (a) of Table 4.6.15-1.

(b) Purge is defined in Section 1.23.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, I-131 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semi-Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.

(d) Sampling and analysis shall also be performed following shutdown, startup or an increase on the recombiner discharge monitor of greater than 50 percent, factoring out increases due to changes in thermal power level or dilution flow; or when the stack release rate is in excess of 1000 uCi/second and steady-state gaseous release rate increases by 50 percent.

(e) The sample flow rate and the stack flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.6.15.b.(1).(b) and 3.6.15.b.(3).

(f) When the release rate is in excess of 1000 uCi/sec and steady state gaseous release rate increases by 50 percent. The iodine and particulate collection device shall be removed and analyzed to determine the changes in iodine-131 and particulate release rate. The analysis shall be done daily following each change until it is shown that a pattern exists which can be used to predict the release rate; after which it may revert to weekly sampling frequency. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

(g) When RAGEMS is inoperable the LLD for noble gas gross gamma analysis shall be 1×10^{-4}.

(h) Tritium grab samples shall be taken weekly from the station ventilation exhaust (stack) when fuel is offloaded until stable tritium release levels can be demonstrated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.15 RADIOACTIVE EFFLUENTS (Continued)	4.6.15 RADIOACTIVE EFFLUENTS (Continued)

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

c. Main Condenser

The gross radioactivity (beta and/or gamma) rate of noble gases measured at the recombiner discharge shall be limited to less than or equal to 500,000 uCi/sec. This limit can be raised to 1 Ci/sec. for a period not to exceed 60 days provided the offgas treatment system is in operation.

With the gross radioactivity (beta and/or gamma) rate of noble gases at the recombiner discharge exceeding the above limits, restore the gross radioactivity rate to within its limit within 72 hours or be in at least Hot Shutdown within the next 12 hours.

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

c. Main Condenser

The radioactivity rate of noble gases at the recombiner discharge shall be continuously monitored in accordance with Table 3.6.14-2.

The gross radioactivity (beta and/or gamma) rate of noble gases from the recombiner discharge shall be determined to be within the limits of Specification 3.6.15 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner discharge:

Monthly.

Within 4 hours following an increase on the recombiner discharge monitor of greater than 50%, factoring out increases due to changes in thermal power level and dilution flow changes.

LIMITING CONDITION FOR OPERATION

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

d. Uranium Fuel Cycle

The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.6.15.a.2(b), 3.6.15.b.2(b) and 3.6.15.b.3(b), calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above listed 40CFR190 limits have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the

SURVEILLANCE REQUIREMENT

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

d. Uranium Fuel Cycle

Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.6.15.a.(2), 4.6.15.b.(2) and 4.6.16.b.(3) and in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual. This requirement is applicable only under conditions set forth in Specification 3.6.15.d.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

d. Uranium Fuel Cycle (Continued)

calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40CFR 190. Submittal of the report is considered a timely request and a variance is granted until staff action on the request is complete.

LIQUID CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than the concentration levels specified in 10CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I, 10CFR Part 50, to a member of the public and (2) the limits of 10CFR Part 20.106 (e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its maximum permissable concentration in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40. 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

Liquid Dose

This specification is provided to implement the requirements of Section II.A, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Conditions for Operation expressed as quarter and annual limits are set at those values found in Section II.A. of Appendix I, in accordance with Section IV.A. The Limiting Conditions for Operation provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to unrestricted areas will be kept "as low as is reasonably achievable." There are no drinking water supplies that can be potentially affected by plant operations. The dose calculation methodology and parameters in the Offsite Dose Calculation Manual implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculation procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the Offsite Dose Calculation Manual for calculating the doses due to the actual release rates of radioactiave materials in liquid effluents are consistent with the methodology provided in Reguatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

Gaseous Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10CFR Part 20 to unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10CFR Part 20 (10CFR Part 20.106(b)). For members of the public who may at times be within the site boundary, the occupancy of that member of the public will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

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Dose - Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation expressed as quarter and annual limits are set at those values found in Section II.B of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV-A of Appendix I to assure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the Offsite Dose Calculation Manual for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977.

The Offsite Dose Calculation Manual equations provided to determine the air doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

Dose - Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Conditions for Operation expressed as quarter and annual limits are set at those valves found in Section II.C of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Offsite Dose Calculation Manual calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The Offsite Dose Calculation Manual calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man and 4) deposition on the ground with subsequent exposure of man.

Main Condenser

Restricting the gross radioactivity rate of noble gases from the main condenser provides assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a very small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The primary purpose of providing this specification is to limit buildup of fission product activity within the station systems which would result if high fuel leakage were to be permitted over extended periods.

BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

Total Dose - Uranium Fuel Cycle

This specification is provided to meet the dose limitations of 40CFR Part 190 that have been incorporated into 10CFR Part 20 by 46FR 182525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to a member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contribution from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been corrected), in accordance with the provisions of 40CFR Part 190.11 and 10CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in Specification 3.6.15.a.(1) and 3.6.15.b.(1). An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Applicability:

Applies to the operating status of the liquid, gaseous and solid effluent treatment systems.

Objective:

To assure operability of the liquid, gaseous and solid effluent treatment system.

Specification:

a. Liquid

The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge as necessary to meet the requirements of Specification 3.6.15.

b. Gaseous

The gaseous radwaste treatment system shall be operable. The gaseous radwaste treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge as necessary to meet the requirements of Specification 3.6.15.

With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to

4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Applicability:

Applies to the surveillance requirements for the liquid, gaseous and solid effluent treatment systems.

Objective:

To verify operability of the liquid, gaseous and solid effluent treatment system.

Specification:

a. Liquid

Doses due to liquid releases to unrestricted areas shall be projected prior to the release of each batch of liquid radioactive waste in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

b. Gaseous

Doses due to gaseous releases to areas at and beyond the site boundary shall be calculated monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

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LIMITING CONDITION FOR OPERATION

3.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS (Cont)

b. Gaseous (Continued)

Specification 6.9.3, Special Report that identifies the inoperable equipment and the reason for its inoperability, actions taken to restore the inoperable equipment to OPERABLE status, and a summary description of those actions taken to prevent a recurrence.

c. Solid

The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

With the provisions of the process control program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

SURVEILLANCE REQUIREMENT

4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS (Cont)

c. Solid

The process control program shall be used to verify the solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges and evaporator bottoms).

(1) If any test specimen fails to verify solidification, the solidification of the batch may then be resumed using the alternative solidification parameters determined by the process control program.

(2) If the initial test specimen from a batch of waste fails to verify solidification, the process control program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate solidification.

BASES FOR 3.6.16 AND 4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Liquid Radwaste Treatment System

The requirement that the appropriate portions of this system be used provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10CFR Part 50 and the design objective given in Section II.D of Appendix I to 10CFR Part 50.

Gaseous Radwaste Treatment System

The requirement that this system be used provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10CFR Part 50. Since the capability exists to operate within specification without use of the system, it is conceivable that due to unforeseen circumstances, limited operation without the system may be made sometime during the life of the plant.

Solid Radioactive Waste

This specification implements the requirements of 10CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10CFR part 50. The process parameters included in establishing the process control program may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constitutents and mixing and curing times.

3.6.17 EXPLOSIVE GAS MIXTURE

Applicability:

Applies to the operability of instrumentation to monitor hydrogen concentration in the main condenser off-gas treatment system.

Objective:

To assure the operability of the hydrogen monitoring instrumentation in the main condenser off-gas treatment system.

Specification:

The concentration of hydrogen in the main condenser off-gas treatment system shall be limited to 4 percent by volume.

If the concentration of hydrogen in the main condenser off-gas treatment system exceeds this limit, restore the concentration to within the limit within 48 hours.

4.6.17 EXPLOSIVE GAS MIXTURE

Applicability:

Applies to the surveillance of instrumentation that monitors hydrogen concentration in the main condenser off-gas treatment system.

Objective:

To verify operation of monitoring instrumentation.

Specification:

The concentration of hydrogen in the main condenser off-gas treatment system shall be determined to be within the above limits by continuously monitoring the waste gases in the main condenser off-gas treatment system in accordance with Table 3.6.14-2 of Specification 3.6.14.

BASES FOR 3.6.17 AND 4.6.17 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen. Automatic control features are included in the system to prevent the hydrogen concentration from reaching these flammability limits. Maintaining the concentration of hydrogen below flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10CFR Part 50.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.18 MARK I CONTAINMENT

Applicability:

Applies to the venting/purging of the Mark I Containment.

Objective:

To assure that the Mark I Containment is vented/purged so that the limits of specifications 3.15.b.1 and 3.6.15.b.3 are met.

Specification:

The Mark I Containment drywell shall be vented/purged through the Emergency Ventilation System unless Specification 3.6.15.b.1 and 3.6.15.b.3 can be met without use of the Emergency Ventilation System.

If these requirements are not satisfied, suspend all venting/purging of the drywell.

4.6.18 MARK I CONTAINMENT

Applicability:

Applies to the surveillance requirement for venting and purging of the Mark I Containment when required to be vented/purged through the Emergency Ventilation System.

Objective:

To verify that the Mark I Containment is vented through the Emergency Ventilation System when required.

Specification:

The containment drywell shall be determined to be aligned for venting/purging through the Emergency Ventilation System within four hours prior to start of and at least once per 12 hours during venting/purging of the drywell.

BASES FOR 3.6.18 AND 4.6.18 MARK I CONTAINMENT

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10CFR Part 20 for unrestricted areas.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.19 LIQUID WASTE HOLDUP TANKS*	4.6.19 LIQUID WASTE HOLDUP TANKS
Applicability:	Applicability:
Applies to the quantity of radioactive material that may be stored in an outdoor liquid waste holdup tank.	Applies to the surveillance requirements for outdoor liquid waste holdup tanks.
Objective:	Objective:
To assure that the quantity of radioactive material stored in outdoor holdup tanks does not exceed a specified level.	To verify the quantity of radioactive material stored in an outdoor liquid waste holdup tank.
Specification:	Specification:
The quantity of radioactive material contained in an outdoor liquid waste tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.	The quantity of radioactive material contained in each of the tanks listed in Specification 3.6.19 shall be determined to be within the limit of Specification 3.6.19 by analyzing a representative sample of the tank's contents at least weekly when radioactive materials are being added to the tank.
With the quantity of radioactive material in any such tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank. Within 48 hours reduce the tank contents to within the limit and describe the events leading to this condition in the next Semi-Annual Radioactive Effluent Release Report.	
*Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.	

BASES FOR 3.6.19 AND 4.6.19 LIQUID HOLDUP TANKS

This specification applies to any outdoor tank that is not surrounded by liners, dikes or walls capable of holding the tank contents and that does not have tank overflows and surrounding areas drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

3.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Applicability:

Applies to radiological samples of station environs.

Objective:

To evaluate the effects of station operations and radioactive effluent releases on the environs and to verify the effectiveness of the controls on radioactive material sources.

Specification:

The radiological environmental monitoring program shall be conducted as specified in Table 3.6.20-1.

With the radiological environmental monitoring program not being conducted as specified in Table 3.6.20-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

Deviations are permitted from the required sample schedule if samples are unobtainable due to hazardous conditions, seasonal unavailability, theft, uncooperative residents or to malfunction of automatic sampling equipment. In the event of the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

4.6.20 RADIOLOGICAL ENVIRONMENTAL MONITIORING PROGRAM

Applicability:

Applies to the periodic sampling and monitoring requirements of the radiological environmental monitoring program.

Objective:

To ascertain what effect station operations and radioactive effluent releases have had upon the environment.

Specification:

The radiological environmental monitoring samples shall be collected pursuant to Table 3.6.20-1 from the specific locations given in the table and figure(s) in the Offsite Dose Calculation Manual and shall be analyzed pursuant to the requirements of Table 3.6.20-1 and the detection capabilities required by Table 4.6.20-1.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (Continued)

Specification: (Continued)

With the level of radioactivity (as the result of plant effluents), in an environmental sampling medium exceeding the reporting levels of Table 6.9.3-1 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report pursuant to Specification 6.9.3. The Special Report shall identify the cause(s) for exceeding the limit(s) and define the corrective action(s) to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3). When more than one of the radionuclides in Table 6.9.3-1 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 6.9.3-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specification 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (Continued)

Specification: (Continued)

This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

With milk or fruit and/or vegetables no longer available at one or more of the sample locations specified in Table 3.6.20-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semi-Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the Offsite Dose Calculation Manual reflecting the new location(s).

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Table 3.6.20-1
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Limiting Condition for Operation

Exposure Pathway and/or Sample	Number of Samples[(a)] and Locations	Sampling and Collection Frequency (a)	Type of Analysis and Frequency
Radioiodine & Particulates	Samples from 5 locations: 1) 3 samples from off-site locations in different sectors of the highest calculated site average D/Q (based on all site licensed reactors) 2) 1 sample from the vicinity of an established year round community having the highest calculated site average D/Q (based on all site licensed reactors) 3) 1 sample from a control location 10-17 miles distant and in a least prevalent wind direction[(d)]	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent	Radioiodine Canisters analyze once/week for I-131. Particulate Samplers Gross beta radioactivity following filter change,[(b)] composite (by location) for gamma isotopic analysis[(c)] once per 3 months, (as a minimum)

Exposure Pathway and/or Sample	Number of Samples[(a)] and Locations	Sampling and Collection Frequency(a)	Type of Analysis and Frequency
Direct Radiation [(e)]	32 stations with two or more dosimeters to be placed as follows: an inner ring of stations in the general area of the site boundary and an outer ring in the 4 to 5 mile range from the site with a station in each land based sector.* The balance of the stations should be placed in special interest areas such as population centers, nearby residences, schools and in 2 or 3 areas to serve as control stations.	Once per 3 months	Gamma dose once per 3 months
WATERBORNE			
Surface [(f)]	1) 1 sample upstream 2) 1 sample from the site's downstream cooling water intake	Composite sample over 1 month period [(g)]	Gamma isotopic analysis[(c)] once/month. Composite for once per 3 months tritium analysis.
Sediment from Shoreline	1 sample from a downstream area with existing or potential recreational value	Twice per year	Gamma isotopic analysis[(c)]

* At this distance, 8 wind rose sectors are over Lake Ontario.

Exposure Pathway and/or Sample	Number of Samples(a) and Locations	Sampling and Collection Frequency (a)	Type of Analysis and Frequency
INGESTION			
Milk	1) Samples from milk sampling locations in 3 locations within 3.5 miles distance having the highest calculated site average D/Q. If there are none, then 1 sample from milking animals in each of 3 areas 3.5-5.0 miles distant having the highest calculated site average D/Q (based on all site licensed reactors)	Twice per month, April-December (samples will be collected in January-March if I-131 is detected in November and December of the preceding year)	Gamma isotopic(c) and I-131 analysis twice per month when animals are on pasture (April-December); once/month at other times (January-March) if required
	2) 1 sample from a milk sampling location at a control location (9-20 miles distant and in a least prevalent wind direction)(d)		
Fish	1) 1 sample each of two commercially or recreationally important species in the vicinity of a plant discharge area(h)	Twice per year	Gamma isotopic analysis(c) on edible portions twice per year
	2) 1 sample each of the same species from an area at least 5 miles distant from the site.(d)		

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency (a)	Type of Analysis and Frequency
Food Products	1) Samples of three different kinds of broad leaf vegetation (such as vegetables) grown nearest to each of two different off-site locations of highest calculated site average D/Q (based on all licensed site reactors) 2) One sample of each of the similar broad leaf vegetation grown at least 9.3-20 miles distant in a least prevalent wind direction	Once per year during harvest season	Gamma isotopic (c) analysis of edible portions (isotopic to include I-131 or a separate I-131 analysis may be performed) Once during the harvest season

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NOTES FOR TABLE 3.6.20-1

(a) It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and may be substituted. Actual locations (distance and directions) from the site shall be provided in the Annual Radiological Environmental Operating Report. Highest D/Q locations are based on historical meteorological data for all site licensed reactors.

(b) Particulate sample filters should be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If the gross beta activity in air is greater than 10 times a historical yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(c) Gamma isotopic analysis means the identification and quantification of gamma emitting radionuclides that may be attributable to the effluents from the facility.

(d) The purpose of these samples is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites, such as historical control locations which provide valid background data may be substituted.

(e) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purpose of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges shall not be used for measuring direct radiation.

(f) The "upstream sample" should be taken at a distance beyond significant influence of the dischage. The "downstream sample" should be taken in an area beyond but near the mixing zone, if possible.

(g) Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g. hourly) relative to the compositing period (e.g. monthly) in order to assure obtaining a representative sample.

(h) In the event commercial or recreational important species are not available as a result of three attempts, then other species may be utilized as available.

TABLE 4.6.20-1
DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a,b]
LOWER LIMIT OF DETECTION LLD (c)

Surveillance Requirement

Analysis	Water (c) (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	2000 *					
Mn-54	15		130			
Fe-59	30		260			
Co-58, Co-60	15		130			
Zn-65	30		260			
Zr-95, Nb-95	15					
I-131	1 **	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba/La-140	15			15		

* If no drinking water pathway exists, a value of 3000 pCi/liter may be used.

** If no drinking water pathway exists, a value of 15 pCi/liter may be used.

NOTES FOR TABLE 4.6.20-1

(a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.d.

(b) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in ANSI N.545 (1975), Section 4.3. Allowable exceptions to ANSI N.545 (1975), Section 4.3 are contained in the Nine Mile Point Unit 1 Offsite Dose Calculation Manual (ODCM).

(c) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ S_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute.

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, where applicable.

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period and time of counting

Typical values of E, V, Y and Δt should be used in the calculation.

NOTES FOR TABLE 4.6.20-1

It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact limit for the particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.d.

BASES 3.6.20 AND 4.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PLAN

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.6.20-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem 40, 586-93 (1968) and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.21 INTERLABORATORY COMPARISON PROGRAM

Applicability:

Applies to participation in an interlaboratory comparison program on environmental sample analysis.

Objective:

To ensure the accuracy of measurements of radioactive material in environmental samples.

Specification:

Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission. Participation in this program shall include media for which environmental samples are routinely collected and for which intercomparison samples are available.

With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

4.6.21 INTERLABORATORY COMPARISON PROGRAM

Applicability:

Applies to testing the validity of measurements on environmental samples.

Objective:

To verify the accuracy of measurements on radioactive material in environmental samples.

Specification:

The Interlaboratory Comparison Program shall be described in the Offsite Dose Calculation Manual. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report. Participants in the EPA Cross Check Program may provide the EPA program code designation in lieu of providing results.

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BASES FOR 3.6.21 AND 4.6.21 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring for the purposes of Section IV.B.2 of Appendix I to 10CFR Part 50.

LIMITING CONDITION FOR OPERATION

3.6.22 LAND USE CENSUS

Applicability:

Applies to the performance of a land use census in the vicinity of the Nine Mile Point Nuclear Facility.

Objective:

To determine the utilization of land within a distance of three miles from the Facility.

Specification:

A land use census shall be conducted and shall identify within a distance of three miles the location in each of the 16 meteorological sectors the nearest residence and within a distance of three miles the location in each of the 16 meteorological sectors of all milk animals. In lieu of a garden census, specifications for vegetation sampling in Table 3.6.20-1 shall be followed, including analysis of appropriate controls.

With a land use census identifying a milk animal location(s) that represents a calculated D/Q value greater than the D/Q value currently being used in specification 4.6.15.b.3, identify the new location(s) in the next Semi-Annual Radioactive Effluent Release Report.

SURVEILLANCE REQUIREMENT

4.6.22 LAND USE CENSUS

Applicability:

Applies to assuring that current land use is known.

Objective:

To verify the appropriateness of the environmental surveillance program.

Specification:

The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as conducting a door-to-door survey, aerial survey or consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.22 LAND USE CENSUS (Continued) Specification: (Continued) If the D/Q value at a new milk sampling location is significantly greater (50%) than the D/Q value at an existing milk sampling location, add the new location to the radiological environmental monitoring program within 30 days. The sampling location(s) excluding the control station location, having the lowest calculated D/Q may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.e identify the new location(s) in the next Semi-Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the Offsite Dose Calculation Manual reflecting the new location(s).	

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BASES FOR 3.6.20 AND 4.6:20 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best survey information such as from a door-to-door survey(s), from an aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10CFR Part 50.

In lieu of a garden census, the significance of the exposure via the garden pathway can be evaluated by the sampling of vegetation as specified in Table 3.6.20-1.

A milk sampling location, as defined in Section 1, requires that at least 10 milking cows are present at a designated milk sample location. It has been found from past experience, and as a result of conferring with local farmers, that a minimum of 10 milking cows is necessary to guarantee an adequate supply of milk twice per month for analytical purposes. Locations with less than 10 milking cows are usually utilized for breeding purposes which eliminates a stable supply of milk for samples as a result of suckling calves and periods when the adult animals are dry.

LIMITING CONDITIONS FOR OPERATION

3.7.1 Special Test Exception- Shutdown Margin Demonstrations

Applicability:

Applies to shutdown margin demonstration in the cold shutdown condition.

Objective:

To assure the capability of the control rod system to control core reactivity.

a. The reactor mode switch may be placed in the startup position to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

   (1) The source range monitors are operable in the noncoincident condition.

   (2) The rod worth minimizer is operable per Specification 3.1.1b(3)(b) and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

SURVEILLANCE REQUIREMENTS

4.7.1 Special Test Exception - Shutdown Margin Demonstrations

Applicability:

Applies to periodic inspections required to perform shutdown margin demonstrations in the cold shutdown condition.

Objective:

To specify the inspections required to perform the shutdown margin demonstration in the cold shutdown condition.

a. Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that:

   (1) The source range monitors are operable per Specification 3.5.1.

   (2) The rod worth minimizer is operable with the required program per Specification 3.1.1b(3)(b) or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown margin demonstration procedure.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
(3) The continuous rod withdrawal control shall not be used during out-of-sequence movement of the control rods. (4) No core alterations are in progress. b. With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the shutdown or refuel position.	(3) No core alterations are in progress.

BASES FOR 3.7.1 AND 4.7.1 SHUTDOWN MARGIN DEMONSTRATION

The shutdown margin demonstration has to be performed prior to power operation. However, the mode switch must be placed in the startup position to allow withdrawal of more than one control rod. Specifications 3.7.1 and 4.7.1 require certain restrictions in order to ensure that an inadvertent criticality does not occur while performing the shutdown margin demonstration.

The shutdown margin demonstration will be performed in the cold shutdown condition with the vessel head in place. The shutdown margin demonstration will be performed prior to the reactor coolant system pressure and control rod scram time tests following refueling outages when core alterations are performed. The shutdown margin demonstration is performed using the in-sequence non-critical method.

5.0 DESIGN FEATURES

5.1 Site

The Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant site comprising approximately 1500 acres, is located on the shores of Lake Ontario, about seven miles northeast of Oswego, New York. An exclusion distance of nearly 4000 feet is provided between the Station and the nearest site boundary to the west, a mile to the boundary on the east, and a mile and a half to the southern site boundary (as described in the Sixth Supplement of the FSAR).

Figure 5.1-1 is a Site Boundary Map of Nine Mile Point which allows the identification of gaseous and liquid waste release points. Figure 5.1-1 also defines the unrestricted area within the site boundary that is accessible (except for fenced areas) to member of the public.

5.2 Reactor

The reactor core consists of no more than 532 fuel assemblies containing enriched uranium dioxide pellets clad in Zircaloy-2. The core excess reactivity will be controlled by movable control rods and burnable poisons. The core will be cooled by circulation of water internally and external to the pressure vessel through recirculation loops.

5.3 Reactor Vessel

The pertinent features of the reactor vessel other than those referred to in the technical specifications are as follows:

Internal Height	63'-10"
Internal Diameter	17'- 9"
Vessel Design Lifetime	40 years
Materials of Construction Base Metal	SA302B
Clad	Weld Deposited 308L Electrode

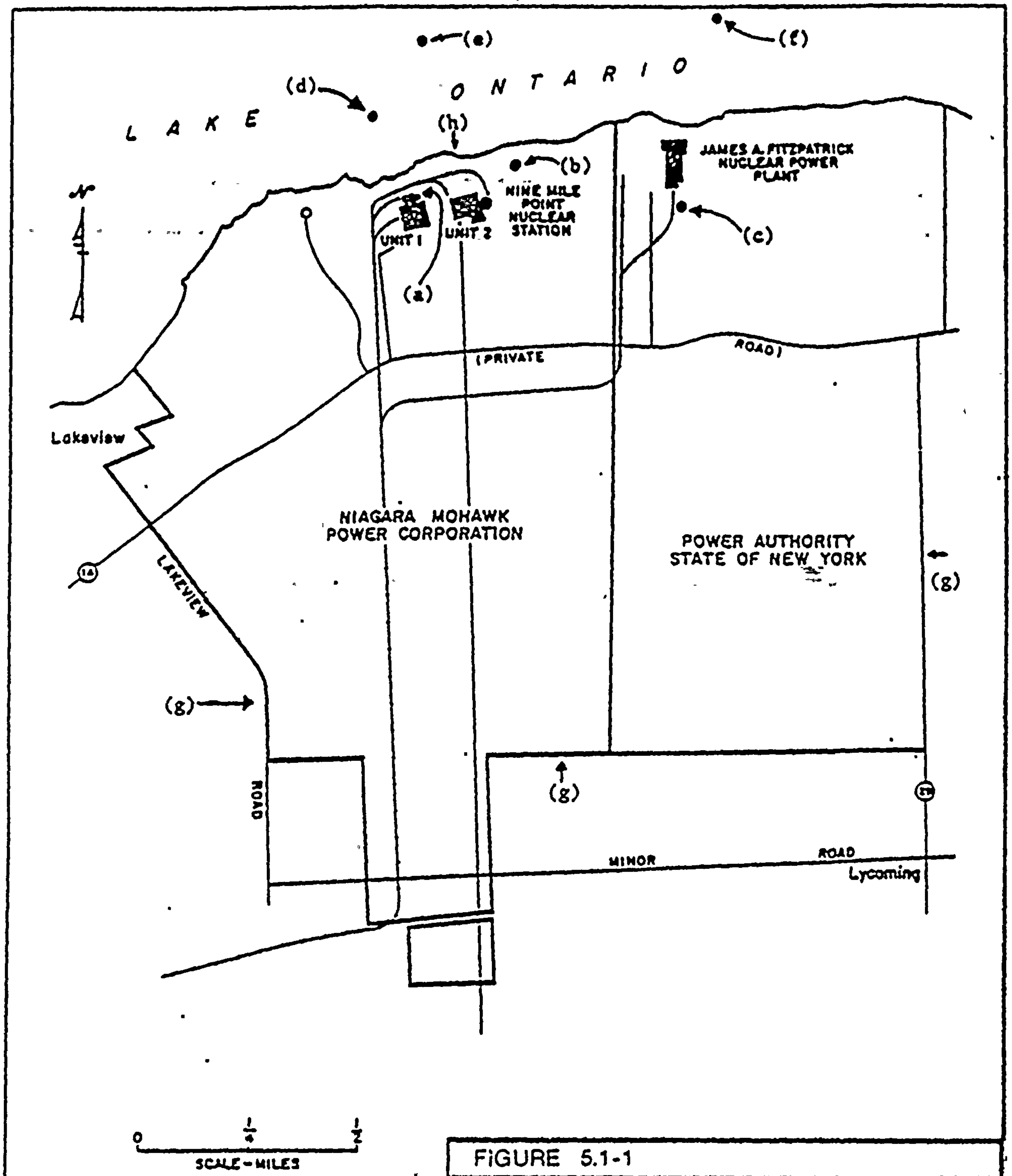

FIGURE 5.1-1

SITE BOUNDARIES

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 1

NOTES TO FIGURE 5.1-1

(a) NMP1 Stack (height is 350')

(b) NMP2 Stack (height is 430')

(c) JAFNPP Stack (height is 385')

(d) NMP1 Radioactive Liquid Discharge (Lake Ontario, bottom)

(e) NMP2 Radioactive Liquid Discharge (Lake Ontario, bottom)

(f) JAFNPP Radioactive Liquid Discharge (Lake Ontario, bottom)

(g) Site Boundary

(h) Lake Ontario Shoreline

Additional Information:

- NMP2 Reactor Building Vent is located 187 feet above ground level
- JAFNPP Reactor and Turbine Building Vents are located 173 feet above ground level
- JAFNPP Radwaste Building Vent is 112 feet above ground level

5.4 Containment

The containment system consists of a drywell, suppression chamber and a reactor building. The pressure suppression system consists of a drywell with a volume of approximately 243,000 cubic feet and an interconnected suppression chamber with a volume of 209,000 cubic feet. Of this total volume some 180,000 and 120,000 cubic feet of free space are available in the drywell and suppression chamber, respectively.

The pertinent design features not discussed elsewhere in the technical specifications are as follows:

	Drywell & Vents	Pressure Suppression Chamber
Internal Design Pressure	62 psig	35 psig
Internal Design Temperature	310F	205F
External Design Pressure	2 psig	1 psig
Material of Construction	A-201 and A-212 Grade "B" Firebox Steel made to A-300 requirements.	

For long-term post-accident recovery, the pressure suppression system is designed to permit flooding to a level at least six feet above the core.

The reactor building is designed for a maximum in-leakage rate of 100 percent per day at 0.25 inch of water internal vacuum and zero wind speed. Exterior loadings for wind, snow and ice meet all applicable codes. The roof and supporting structures are designed to withstand a loading of 40 psf of snow or ice. The walls and building structure are designed to withstand an external or internal loading of 40 psf which is approximately equivalent to that caused by a wind velocity of 125 mph 30 feet above the ground level. Pressure relief is provided to prevent damage to the superstructure due to the break of any primary system line in the reactor building. In this event, blowout panels will fail, relieving pressure in the event of a major line rupture.

5.5 Storage of Unirradiated and Spent Fuel

Unirradiated fuel assemblies will normally be stored in critically safe new fuel storage racks in the reactor building storage vault. Even when flooded with water, the resultant k_{eff} is less than 0.95. Fresh fuel may also be stored in shipping containers. The unirradiated fuel storage vault is designed and shall be maintained with a storage capacity limited to no more than 200 fuel assemblies.

The spent fuel storage facility is designed to maintain fuel in a geometry such that k_{eff} is less than 0.95 under conditions of optimum water moderation. The spent fuel storage facility is designed and shall be maintained with a storage capacity limited to no more than 2776 fuel assemblies. Fuel assemblies stored in the 1066 spent fuel storage locations of the non-poison flux trap design are limited to 15.6 grams (3.0 weight percent) of Uranium-235 per axial centimeters of assembly. Fuel assemblies stored in the 1,710 spent fuel storage positions of the poison type which use Boraflex as the neutron absorber are limited to 18.13 grams (3.75 weight percent) of Uranium-235 per axial centimeters of assembly.

Calculations for k_{eff} values have been based on methods approved by the Nuclear Regulatory Commission covering special arrays (10CFR70.55).

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 11 percent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Station Shift Supervisor - Nuclear (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Executive Vice President - Nuclear shall be re-issued to station personnel on an annual basis.

6.2 Organization

Onsite and Offsite Organization

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions or in equivalent forms of documentation. The organization charts shall be documented in the Final Safety Analysis Report, and the functional descriptions of departmental responsibilities and relationships and job descriptions for key personnel positions are documented in procedures.

b. The Executive Vice President - Nuclear shall have corporate responsibility for overall plant nuclear safety. The Executive Vice President - Nuclear shall take any measures needed to assure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The Plant Manager shall have responsibility for overall unit operation and shall have control over those resources necessary for safe operation and maintenance of the plant.

Onsite and Offsite Organization (Cont'd)

d. The persons responsible for the training, health physics and quality assurance functions may report to an appropriate manager onsite, but shall have direct access to responsible corporate management at a level where action appropriate to the mitigation of training, health physics and quality assurance concerns can be accomplished.

Facility Staff

6.2.2 The unit organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. During reactor operation, this licensed operator shall be present at the controls of the facility.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection* procedures shall be on site when fuel is in the reactor.

* The Radiation Protection qualified individual and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

Facility Staff (Cont'd)

e. A licensed Senior Reactor Operator shall be required in the Control Room during power operations, hot shutdown, and when the emergency plan is activated. This may be the Station Shift Supervisor-Nuclear or the Assistant Station Shift Supervisor-Nuclear or another Senior Reactor Operator during power operations or hot shutdown. When the emergency plan is activated during normal operations or hot shutdown, the Assistant Station Shift Supervisor-Nuclear becomes the Shift Technical Advisor and the Station Shift Supervisor-Nuclear is restricted to the control room until an additional licensed Senior Reactor Operator arrives.

f. A licensed Senior Reactor Operator shall be responsible for all movement of new and irradiated fuel within the site boundary. All core alterations shall be directly supervised by a licensed senior reactor operator who has no other concurrent responsibilities during this operation. A Licensed Operator will be required to manipulate the controls of all fuel handling equipment except movement of new fuel from receipt through dry storage. All fuel moves within the core shall be directly monitored by a member of the reactor analyst group.

g. A Fire Brigade of five (5) members* shall be maintained on site as defined by 5.1 at all times.

h. Administrative procedures shall be developed and implemented to limit the working hours of facility staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the facility is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications on a temporary basis, the following guidelines shall be followed:

* The Radiation Protection qualified Individual and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

Facility Staff (Cont'd)

1) An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7 day period (all excluding shift turnover time).

3) A break of at least 8-hours should be allowed between work periods (including shift turnover time).

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Vice President - Nuclear Generation or designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

i. The General Supervisor Operations, Supervisor Operations, Station Shift Supervisor Nuclear and Assistant Station Shift Supervisor Nuclear shall hold senior reactor operator licenses.

FIGURE DELETED

FIGURE DELETED

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION[1] [6]

License	Normal Operation	Shutdown Condition	Operation[3] W/O Process Computer	Reactor Startups
Senior Operator	1	1[5]	1	1
Operator	2	2[4]	2	3
Unlicensed[2]	2	1	3	2
Asst. Station Shift Supervisor (Shift Technical Advisor Function) (Senior Operator License)[7]	1	1[4]	1	1

Notes:

(1) At any one time, more licensed or unlicensed operating people could be present for maintenance, repairs, refuel outages, etc.

(2) Those operating personnel not holding an "Operator" or "Senior Operator" License.

(3) For operation longer than eight hours without process computer.

(4) Hot shutdown condition only. For cold shutdown and refueling conditions, only one senior operator and one operator are required to be on shift.

(5) An additional senior reactor operator who has no other concurrent responsibilities shall supervise all core alterations.

(6) The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(7) The Assistant Station Shift Supervisor performs the Shift Technical Advisor function when the emergency plan is activated during normal operations or hot shutdown and shall hold a senior reactor operator license. Normally, the Assistant Station Shift Supervisor is a combined Assistant Station Shift Supervisor/Shift Technical Advisor; however, there may be instances when a shift may be staffed by two Senior Reactor Operators plus a dedicated Shift Technical Advisor.

6.3 Facility Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Manager Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and the Shift Technical Advisor who shall have a bachelor's degree in a physical science or engineering or a professional engineer license issued by examination and shall have received specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 Training

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Manager Training and shall meet or exceed the recommendations and requirements of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager Training and Supervisor-Fire Protection, Nuclear and shall meet or exceed the requirements of Appendix R to 10CFR50.

6.5 Review and Audit

6.5.1 Station Operations Review Committee (SORC)

Function

6.5.1.1 The Station Operations Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

Composition

6.5.1.2 The SORC shall be composed of the:

Chairman: Plant Manager
Member: Manager Operations
Member: Manager Maintenance
Member: Manager Technical Support
Member: Manager Chemistry
Member: Manager Radiation Protection
Member: General Supervisor Instrument and Control Maintenance
Member: General Supervisor System Engineering
Member: General Supervisor Operations
Member: Supervisor Reactor Engineering

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SORC activities at any one time.

Meeting Frequency

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman.

Quorum

6.5.1.5 A quorum of the SORC shall consist of the Chairman or a designated alternate and five members including alternates.

Responsibilities

6.5.1.6 The SORC shall be responsible for:

a. Review of all REPORTABLE EVENTS.

b. Review of unit operations to detect potential safety hazards.

c. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the Safety Review and Audit Board.

d. Investigation of violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Generation and to the Safety Review and Audit Board.

Authority

6.5.1.7 The SORC shall:

a. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6 (a) through (d) above constitutes an unreviewed safety question.

b. Provide written notification within 24 hours to the Vice President - Nuclear Generation and the Safety Review and Audit Board of disagreement between the SORC and the Plant Manager; however, the Plant Manager shall have the responsibility for resolution of such disagreements pursuant to 6.1.1 above.

Records

6.5.1.8 The SORC shall maintain written minutes of each meeting and copies shall be provided to the Vice President - Nuclear Generation and the Safety Review and Audit Board.

6.5.2 Technical Review and Control

Activities

6.5.2.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto. Approval of procedures and programs and changes thereto and their safety evaluations, shall be controlled by administrative procedures.

6.5.2.2 Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Plant Manager.

Activities (Cont'd)

6.5.2.3 Proposed modifications to unit structures, systems and components that affect nuclear safety shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to structures, systems and components and the safety evaluations shall be approved prior to implementation by the Plant Manager; or the Manager Technical Support as previously designated by the Plant Manager.

6.5.2.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.2.1, 6.5.2.2 and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the Plant Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary such review shall be performed by the appropriate designated station review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications and their safety evaluations shall be reviewed by the Plant Manager, or the Manager Technical Support as previously designated by the Plant Manager.

6.5.2.6 The Plant Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President - Nuclear Generation.

6.5.2.7 The facility security program, and implementing procedures, shall be reviewed at least every 12 months. Recommended changes shall be approved by the Plant Manager and transmitted to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.

6.5.2.8 The facility emergency plan, and implementing procedures shall be reviewed at least every 12 months. Recommended changes shall be approved by the Plant Manager and transmitted to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.

6.5.2.9 The Plant Manager shall assure the performance of a review by a qualified individual/organization of changes to the Radiological Waste Treatment systems.

6.5.2.10 Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear Generation and to the Safety Review and Audit Board.

6.5.2.11 Review of changes to the Process Control Program and the Offsite Dose Calculation Manual. Approval of any changes shall be made by the Plant Manager or his designee before implementation of such changes.

6.5.2.12 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained. Copies shall be provided to the Vice President-Nuclear Generation and the Safety Review and Audit Board.

6.5.3 Safety Review and Audit Board (SRAB)

Function

6.5.3.1 The Safety Review and Audit Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices
i. (other appropriate fields associated with the unique characteristics of the nuclear power plant)

mposition

6.5.3.2 The Safety Review and Audit Board shall be composed of the:

Chairman:	Staff Engineer or Manager or Vice President
Member:	Plant Manager or Designee
Member:	Staff Engineer - Nuclear
Member:	Staff Engineer - Mechanical or Electrical
Member:	Consultant (See 6.5.3.4)

Alternates

6.5.3.3 Alternate members shall be appointed in writing by the SRAB Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SRAB activities at any one time.

Consultants

6.5.3.4 Consultants shall be utilized as determined by the SRAB Chairman to provide expert advice to the SRAB.

Meeting Frequency

6.5.3.5 The SRAB shall meet at least once per six months.

Quorum

6.5.3.6 The quorum of the SRAB necessary for the performance of the SRAB review and audit functions of these Technical Specifications shall consist of the Chairman or the Chairman's designated alternate and at least three SRAB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

6.5.3.7 The SRAB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or operating license.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the SORC.

Audits

6.5.3.8 Audits of facility activities shall be performed under the cognizance of the SRAB. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10CFR50, at least once per two years.

e. The Facility Emergency Plan and implementing procedures at least once every 12 months.

f. The Facility Security Plan and implementing procedures at least once every 12 months.

g. The Facility Fire Protection Program and implementing procedures at least once per two years.

h. Any other area of facility operation considered appropriate by the SRAB or the Vice President - Nuclear Generation.

i. The radiological environmental monitoring program and the results thereof at least once per 12 months.

j. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

k. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

Authority

6.5.3.9 The SRAB shall report to and advise the Executive Vice President - Nuclear on those areas of responsibility specified in Section 6.5.3.7 and 6.5.3.8.

Records

6.5.3.10 Records of SRAB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRAB meeting shall be prepared, approved and forwarded to the Executive Vice President - Nuclear within 30 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.7 e,f,g, and h above, shall be prepared, approved and forwarded to the Executive Vice President - Nuclear within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.8 above, shall be forwarded to the Executive Vice President - Nuclear within 90 days following completion of the review.

6.6 Reportable Occurrence Action

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Sections 50.72 and 50.73 to 10CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review submitted to the SRAB and the Vice President - Nuclear Generation.

6.7 Safety Limit Violation

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear Generation and the SRAB shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, within 30 days of the violation, and to the SRAB, and the Vice President - Nuclear Generation within 14 days.

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved by the Plant Manager or designee prior to implementation and periodically as set forth in administrative procedures.

6.8 Procedures (Continued)

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the Plant Manager or designee within 14 days of implementation.

6.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1 Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1 Routine Reports (Cont'd)

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Occupational Exposure Report. A tabulation shall be submitted on an annual basis which includes the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience including documentation of challenges to the safety relief valves or safety valves, shall be submitted on a monthly basis, which will include a narrative of operating experience, in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

1/ This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

6.9.1 Routine Reports (Continued)

d. Annual Radiological Environmental Operating Report*.

Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1, 1985.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls as appropriate, and with environmental surveillance reports from the previous 5 years, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.6.22.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.6.21; discussion of all deviations from the sampling schedule of Table 3.6.20-1; and discussion of all analyses in which the LLD required in Table 4.6.20-1 was not achievable.

* A single submittal may be made for a multiple unit station.

** One map shall cover stations near the site boundary; a second shall include the more distant stations.

6.9.1 Routine Reports (cont'd)

e. Semi-annual Radioactive Effluent Release Report **

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin on January 1, 1985.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of thepublic due to their activities inside the site boundary (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the Offsite Dose Calculation Manual.

* In lieu of submission with the Semi-annual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

** A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6.9.1 Routine Reports (cont'd)

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and,

f. Solidifcation agent or absorbent (e.g., cement)

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.6.20.

Changes to the Process Control Program (PCP) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes: and

c. Documentation of the fact that the change has been reviewed and found acceptable.

6.9.1 Routine Reports (cont'd)

Changes to the Offsite Dose Calculation Manual (ODCM): Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

f. CORE OPERATING LIMITS REPORT

.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.1.7.a and 3.1.7.e.

2) The K_f core flow adjustment factor for Specification 3.1.7.c.

3) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.1.7.c and 3.1.7.e.

and shall be documented in the CORE OPERATING LIMITS REPORT.

.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

1) NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL" (Latest approved revision).

2) NEDE-30966-P-A "SAFER MODEL FOR EVALUATION OF LOSS-OF-COOLANT ACCIDENTS FOR JET PUMP AND NON-JET PUMP PLANTS" (Latest Approved Revisions)

Vol I "SAFER LONG TERM INVENTORY MODEL FOR BWR LOSS-OF-COOLANT ACCIDENT ANALYSIS"

Vol II "SAFER APPLICATION METHODOLOGY FOR NON-JET PUMP PLANTS"

3) NEDO-20556-P-A "GENERAL ELECTRIC COMPANY ANALYTICAL MODEL FOR LOSS-OF-COOLANT ACCIDENT ANALYSIS IN ACCORDANCE WITH 10CFR50 APPENDIX K". (Latest approved revision)

.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.2 Fire Protection Program Reports

IN ACCORDANCE WITH 10CFR50.4 OHB 7/26/86

a. Submit a special report ~~to the appropriate regional office~~ as follows:

REGIONAL ADMINISTRATOR OHB 7/26/86

- Notify the ~~Director~~ of the appropriate Regional Office by telephone within 24 hours.
- Confirm by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and
- Follow-up in writing within 14 days after the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to an operable status.

IN ACCORDANCE WITH 10CFR50.4 OHB 7/26/86

b. Submit a special report ~~to the Director of the appropriate Regional Office~~ within 30 days following the event outlining the plans and procedures to be used to restore the inoperable equipment to an operable status.

6.9.3 Special Reports

Special reports shall be submitted in accordance with 10 CFR 50.4 Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Reactor Vessel Material Surveillance Specimen Examination, Specification 4.2.2(b) (12 months)

b. Safety Class 1 Inservice Inspection, Specification (See Table 4.2.6(a)) (Three months)

c. Safety Class 2 Inservice Inspections, Specification (See Table 4.2.6(b)) (Three months)

d. Safety Class 3 Inservice Inspections; Specification (See Table 4.2.6(c)) (Three months)

e. Primary Containment Leakage Testing, Specification 3.3.3 (Three months)

f. Secondary Containment Leakage Testing, Specification 3.4.1 (Three months)

g. Sealed Source Leakage In Excess Of Limits, Specification 3.6.5.2 (Three months)

h. Calculate Dose from Liquid Effluent in Excess of Limits, Specification 3.6.15.a(2)(b) (30 days from the end of the affected calendar quarter).

i. Calculate Air Dose from Noble Gases Effluent in Excess of Limits, Specification 3.6.15.b(2)(b) (30 days from the end of the affected calendar quarter).

j. Calculate Dose from I-131, H-3 and Radioactive Particulates with half lives greater than eight days in Excess of Limits, Specification 3.6.15.b(3)(b) (30 days from the end of the affected calendar quarter).

k. Calculated Doses from Uranium Fuel Cycle Source in Excess of Limits, Specification 3.6.15.d (30 days from the end of the affected calendar year).

l. Inoperable Gaseous Radwaste Treatment System, Specification 3.6.16.b (30 days from the event).

m. Environmental Radiological Reports. With the level of radioactivity (as the result of plant effluents) in an environmental sampling medium exceeding the reporting level of Table 6.9.3-1, when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within thirty (30) days from the end of the calendar quarter a special report identifying the cause(s) for exceeding the limits, and define the corrective action to be taken.

TABLE 6.9.3-1
REPORTING LEVEL FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	20,000 *				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-95, Nb-95	400				
I-131	2 **	0.9		3	100
Cs-134	30	10.0	1,000	60	1,000
Cs-137	50	20.0	2,000	70	2,000
Ba/La-140	200			300	

* For drinking water samples. This is a 40 CFR 141 value. If no drinking water pathway exists, a value of 30,000 pCi/liter may be used.

** If no drinking water pathway exists, a value of 20 pCi/liter may be used.

6.10 Record Retention

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. REPORTABLE EVENT REPORTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

6.10 Record Retention (Continued)

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the SRAB.

l. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and Quality Assurance records showing that these procedures were followed.

6.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 High Radiation Area

6.12.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10CFR20, each high radiation area normally accessible* by personnel in which the intensity of radiation is greater than 100 mrem/hr** but less than 1000 mrem/hr** shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit in accordance with site approved procedures. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by oneor more of the following:

6.12 High Radiation Area (Continued)

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Manager Radiation Protection or designate in the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1 areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem** shall be provided with locked doors*** to prevent unauthorized entry, and the hard keys or access provided by magnetic keycard shall be maintained under the administrative control of the Station Shift Supervisor or designate on duty and/or the Manager Radiation Protection or designate. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify in accordance with site approved procedures accordingly, the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote, such as use of closed circuit TV cameras, may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as the drywell, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

* by accessible passage and permanently fixed ladders
** measurement made at 18" from source of radioactivity
*** The requirement for locked doors to prevent unauthorized entry does not apply to areas which may temporarily exceed 1000 mrem/hr during the hydrogen water chemistry tests to be conducted during approximately a six-week period following startup from the spring 1986 refueling outage.

[illegible]

[illegible]

6.13 Fire Protection Inspection

6.13.1 An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

6.13.2 An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

6.14 Systems Integrity

Procedure shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.6.a of NUREG 0578.

6.15 Iodine Monitoring

Procedures shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.8.c of NUREG 0578.