UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NINE MILE POINT NUCLEAR STATION, LLC

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-63

1. The Nuclear Regulatory Commission (NRC or the Commission) having previously made the findings set forth in License No. DPR-63 issued on December 26, 1974, has now found that:

   A. The application for license, as amended, originally filed by the Niagara Mohawk Power Corporation as supplemented by Nine Mile Point Nuclear Station, LLC (NMP LLC)* complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Nine Mile Point Nuclear Station Unit No. 1 has been substantially completed in conformity with Construction Permit No. CPPR-16 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   C. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;

* By Order dated October 9, 2009, as superseded by Order dated October 30, 2009, the transfer of this license to Nine Mile Point Nuclear Station, LLC, was approved. By Order dated March 25, 2014, the transfer of the operating authority under this license to Exelon Generation Company, LLC was approved. By Order dated November 16, 2021, a transaction was approved that resulted in Exelon Generation Company, LLC being renamed Constellation Energy Generation, LLC. Unless otherwise noted, references to "the licensee" are to Constellation Energy Generation, LLC as the operating licensee.

D. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

E. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and the safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

F. Constellation Energy Generation, LLC and NMP LLC are technically and financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

G. Constellation Energy Generation, LLC as operator of the facility and NMP LLC** as owner of the facility have satisfied the applicable provisions of 10 CFR Part 140 "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations;

H. The issuance of this full-term renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the adverse environmental impacts of license renewal are not so great that preserving the option of license renewal would be unreasonable and the issuance of the full-term Renewed Facility Operating License No. DPR-63 (subject to the conditions for protection of the environment set forth herein) is in accordance with Appendix D, 10 CFR Part 50 of the Commission's regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70 including Section 30.33, 40.32, 70.23 and 70.31.

2. Renewed Facility Operating License No. DPR-63 is hereby issued to Constellation Energy Generation, LLC and Nine Mile Point Nuclear Station, LLC to read as follows:

A. This license applies to the Nine Mile Point Nuclear Station Unit No. 1, a single cycle, force circulation, boiling light water reactor, and associated equipment (the facility), owned by Nine Mile Point Nuclear Station, LLC. The facility is located on the Nine Mile Point site on the southeast shore of Lake Ontario in Oswego County, New York and is described in the "Final Safety Analysis Report" (with its Amendments Nos. 3 through 13 and its Supplements Nos. 1 through 10) and the "Environmental Report" (with its Supplements Nos. 1 through 3).

** Constellation Energy Generation, LLC is authorized to act for Nine Mile Point Nuclear Station, LLC and has exclusive responsibility and control over the physical possession, operation, and maintenance of the facility.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) NMP LLC to possess and (b) Constellation Energy Generation, LLC to possess, use, and operate the facility at the designated location in Oswego County, New York, in accordance with the procedures and limitations set forth in this amended license;

(2) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components.

(5) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I:

Part 20, Section 30.34 of Part 30; Section 40.41 of Part 40; Section 50.54 and 50.59 of Part 50; and Section 70.32 of Part 70. This renewed license is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect and is also subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1850 megawatts (thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, which is attached hereto, as revised through Amendment No. 255, is hereby incorporated into this license. Constellation Energy Generation, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Deleted

A. This license is subject to the following additional conditions:

(1) NMP LLC will complete construction of a new radwaste facility in conformance with the design defined and evaluated in the FES, to be operational no later than June 1976.

(2) Deleted by License Amendment No. 51

(3) Deleted by License Amendment No. 51

(4) Security, Training and Qualification and Safeguards Contingency Plans

Constellation Energy Generation, LLC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to the provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21 is entitled “Nine Mile Point Nuclear Station, LLC Physical Security, Safeguards Contingency, and Security Training and Qualification Plan, Revision 1,” and was submitted by letter dated April 26, 2006.

Constellation Energy Generation, LLC shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee’s CSP was approved by License Amendment No. 209 and modified by License Amendment No. 219. The licensee has obtained Commission authorization to use Section 161A preemption authority under 42 U.S.C 2201a for weapons at its facility.

(5) Paragraph 2.D(5) of the license has been combined with paragraph 2.D(4) as amended above into a single paragraph.

(6) Recirculation System Safe-end Replacement

The recirculation system and sate-end replacement program including the cutting and welding of the replacement components and the dose mitigation program (ALARA) is approved, subject to the following conditions:

a. NMP LLC shall complete the recirculation piping stress reanalysis prior to restart of Nine Mile Point Nuclear Power Station, Unit No. 1. The results of this analysis for selected representative portions of the recirculation system shall be submitted to the NRC prior to restart of the facility.

b. All fuel and control rods shall be removed from the reactor pressure vessel and stored in the spent fuel pool during the period that work on the safe-end and recirculation system replacement program is in progress.

c. Constellation Energy Generation, LLC shall update the collective occupational dose estimate weekly. If the updated estimate exceeds the 1908 person-rem estimate by more than 10%, the licensee shall provide a revised estimate, including the reasons for such changes, to the NRC within 15 days of determination.

d. Progress reports shall be provided at 90-day intervals from June 30, 1982 and due 30 days after close of the interval, with a final report within 60 days after completion of the repair. These reports will conclude:

(1) a summary of this occupational dose received to date by major task, and

(2) a comparison of estimated doses with the doses actually received.

(7) Fire Protection

Constellation Energy Generation, LLC shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated June 11, 2012, supplemented by letters dated February 27, March 27, April 30, and December 9, 2013; and January 22, March 14, April 15, May 9, and May 23, 2014 and as approved in the safety evaluation report dated June 30, 2014. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the

Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) <u>Risk-Informed Changes that May Be Made Without Prior NRC Approval</u>

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1x10-7/year (yr) for CDF and less than 1x10-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) <u>Other Changes that May Be Made Without Prior NRC Approval</u>

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is “adequate for the hazard.” Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- Fire Alarm and Detection Systems (Section 3.8);
- Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9);
- Gaseous Fire Suppression Systems (Section 3.10); and
- Passive Fire Protection Features (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee’s fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated June 30, 2014 to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) <u>Transition License Conditions</u>

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) below, risk-informed changes to the licensee’s fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in (2) above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, “Plant Modifications Committed,” of NMPNS letter dated May 9, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) prior to startup from the first refueling outage following issuance of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Table S-2, "Implementation Items," of NMPNS letter dated May 9, 2014, 180 days after issuance of the license amendment unless that date falls within a scheduled refueling outage, then the due date will be 60 days following startup from the scheduled refueling outage.

(8) Hot Process Pipe Penetrations

Hot Process Pipe Penetrations in the Emergency Condenser Steam Supply (2 each), Main Steam (2 each), Feedwater (2 each), Cleanup Suction (1 each), and Cleanup Return (1 each) piping systems have been identified as not fully in conformance with FSAR design criteria. This anomaly in design condition from the original design is approved for the duration of Cycle 8 or until March 31, 1986, whichever occurs first, subject to the following conditions:

(a) An unidentified leakage limit of a change of 1 gallon per minute in 24 hours to permit operation will be imposed by administrative control (Standing Order) at the facility for the interim period.

(b) NMP LLC shall restore the facility to a condition consistent with the FSAR or provide a change to the FSAR criteria for staff review and approval prior to restart from the forthcoming Cycle 8 outage.

(9) Deleted.

(10) Deleted.

(11) Deleted.

(12) Deleted.

(13) <u>Mitigation Strategy License Condition</u>

Constellation Energy Generation, LLC shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

a. Fire fighting response strategy with the following elements:

(1) Pre-defined coordinated fire response strategy and guidance
(2) Assessment of mutual aid fire fighting assets
(3) Designated staging areas for equipment and materials
(4) Command and control
(5) Training of response personnel

b. Operations to mitigate fuel damage considering the following:

(1) Protection and use of personnel assets
(2) Communications
(3) Minimizing fire spread
(4) Procedures for implementing integrated fire response strategy
(5) Identification of readily-available pre-staged equipment
(6) Training on integrated fire response strategy
(7) Spent fuel pool mitigation measures

c. Actions to minimize release to include consideration of:

(1) Water spray scrubbing
(2) Dose to onsite responders

(14) Constellation Energy Generation, LLC shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(15) Upon implementation of Amendment No. 195 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by TS 4.4.5.g, in accordance with TS 6.5.8.c.(i), the assessment of CRE habitability as required by Specification 6.5.8.c.(ii), and the measurement of CRE pressure as required by Specification 6.5.8.d, shall be considered met. Following implementation:

(a) The first performance of TS 4.4.5.g, in accordance with Specification 6.5.8.c.(i), shall be within the specified Frequency of 6 years plus the 18-month allowance of TS 4.0.2, as measured from February 19, 2004, the date of the most recent tracer gas test, as stated in the January 31, 2005 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 6.5.8.c.(ii), shall be within 3 years, plus the 9-month allowance of TS 4.0.2, as measured from February 19, 2004, the date of the most recent tracer gas test, as stated in the January 31, 2005 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 6.5.8.d, shall be within 24 months, plus the 182 days allowed by TS 4.0.2, as measured from March 1, 2007, the date of the most recent successful pressure measurement test, or within the next 182 days if not performed previously.

(16) Deleted.

(17) Constellation Energy Generation, LLC shall, no later than the date the closing of the transaction approved on November 16, 2021, occurs, enter into a Support Agreement of approximately $128 million with NMP LLC. NMP LLC shall not take any action to cause Constellation Energy Generation, LLC, or its successors and assigns, to void, cancel, or materially modify the Constellation Energy Generation, LLC Support Agreement or cause it to fail to perform, or impair its performance under the Constellation Energy Generation, LLC Support Agreement, without the prior written consent of the NRC. The Constellation Energy Generation, LLC Support Agreement may not be amended or modified without 30 days prior written notice to the Director of the Office of Nuclear Reactor Regulation or their designee. An executed copy of the Constellation Energy Generation, LLC Support Agreement shall be submitted to the NRC no later than 30 days after the completion of the proposed transaction. Constellation Energy Generation, LLC shall inform the NRC in writing no later than 14 days after any funds are provided to or for NMP LLC under the Constellation Energy Generation, LLC Support Agreement.

(18) Deleted.

(19) Within 14 days of the closing of the transaction approved on November 16, 2021, Constellation Energy Generation, LLC shall submit to the NRC the Nuclear Operating Services Agreement reflecting the terms set forth in the application dated February 25, 2021. Section 7.1 of the Nuclear Operating Services Agreement may not be modified in any material respect related to financial arrangements that would adversely impact the ability of the licensee to fund safety-related activities authorized by the license without the prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(20) Deleted.

(21) Deleted.

(22) Deleted.

(23) Deleted.

(24) Deleted.

(25) Constellation Energy Generation, LLC shall provide to the Director of the Office of Nuclear Reactor Regulation or the Director of the Office of Nuclear Material Safety and Safeguards, as applicable, a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from Constellation Energy Generation, LLC to its direct or indirect parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of Constellation Energy Generation, LLC's consolidated net utility plant, as recorded on Constellation Energy Generation, LLC's books of account.

(26) <u>Adoption of Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times-RITSTF Initiative 4b"</u>

Constellation Energy Generation, LLC is approved to implement TSTF-505, Revision 2, modifying the Technical Specification requirements related to Completion Times (CT) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). The methodology for using the new Risk-Informed Completion Time Program is described in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the NRC on May 17, 2007.

(27) Implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

Constellation Energy Generation, LLC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; high wind safe shutdown equipment list to evaluate high wind / tornado missile events; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Constellation Energy Generation, LLC's submittal letter dated, December 15, 2022, and all its subsequent associated supplements as specified in License Amendment No. 251 dated December 7, 2023.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

E. This license is effective as of the date of issuance and shall expire on August 22, 2029.

F. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, the licensee may make changes to the programs and activities described in the supplement without prior Commission approval, provided that the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. NMP LLC shall complete these activities in accordance with the schedule in Appendix A of NUREG-1900, "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2", dated September 2006, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

H. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Enclosure:

Appendix A – Technical Specifications

Date of Issuance: October 31, 2006

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR

NINE MILE PT. UNIT 1

50-220

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 212°F.

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 212°F.

c. Refueling Condition

(1) The reactor mode switch is in the refuel position.
(2) The reactor coolant temperature is less than 212°F.
(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. <u>Linear Heat Generation Rate (LHGR)</u>

The heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the fuel length.

g. <u>Average Planar Linear Heat Generation Rate (APLHGR)</u>

The Average Planar Linear Heat Generation Rate (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all fuel rods in the specified bundle at the specified height, divided by the number of fuel rods in the fuel bundle at that height.

h. <u>Critical Power</u>

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

i. <u>Critical Power Ratio (CPR)</u>

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

j. <u>Minimum Critical Power Ratio (MCPR)</u>

The minimum in-core critical power ratio.

k. <u>Fraction of Limiting Power Density (FLPD)</u>

The linear heat generation rate (LHGR) existing at a given location divided by the specified LHGR limit for that bundle type.

l. <u>Core Maximum Fraction of Limiting Power Density (CMFLPD)</u>

The highest value of the fraction of limiting power density which exists in the core.

1.2 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).

A verification of operability is an administrative check, by examination of appropriate plant records (logs, surveillance test records) to determine that a system, subsystem, train, component or device is not inoperable. Such verification does not preclude the demonstration (testing) of a given system, subsystem, train, component or device to determine operability.

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 determination 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. The channel test may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

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. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The channel calibration may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

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 <u>Reactor Building Integrity</u>

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, except when the access opening is being used for entry and exit.

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 <u>Core Alteration</u>

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 <u>Rated Flux</u>

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 <u>Surveillance</u>

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 <u>Dose Equivalent I-131</u>

Dose Equivalent I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be the Committed Effective Dose Equivalent dose conversion factors listed in Table 2.1 of Federal Guidance Report No. 11, EPA, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

1.17 <u>Recently Irradiated Fuel</u>

Recently irradiated fuel is fuel that has occupied part of a critical reactor core within the previous 24 hours.

1.18 <u>Pressure and Temperature Limits Report (PTLR)</u>

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.6.7.

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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 6.6.5. Plant operation within these operating limits is addressed in individual specifications.

1.32 Shutdown Margin (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:

a. The reactor is xenon free,

b. The moderator temperature is ≥ 68° F, corresponding to the most reactive state, and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

1.33 INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

1.34 Drain Time

The drain time is the time it would take for the water inventory in and above the Reactor Pressure Vessel (RPV) to drain to -10 inches indicator scale (74 inches above the top of the active fuel seated in the RPV) assuming:

a. The water inventory above -10 inches indicator scale is divided by the limiting drain rate;

b. The limiting drain rate is the larger of the drain rate through a single penetration flow path with the highest flow rate, or the sum of the drain rates through multiple penetration flow paths susceptible to a common Mode failure for all penetration flow paths below -10 inches indicator scale except:

   1. Penetration flow paths connected to an intact closed system, or isolate by manual or automatic valves that are closed and administrative controlled in the closed position, blank flanges, or other devices that prevent flow of reactor coolant through the penetration flow paths;
   2. Penetration flow paths capable of being isolated by valves that will close automatically without offsite power prior to the RPV water level being equal to -10 inches indicator scale when actuated by RPV water level isolation instrumentation; or
   3. Penetration flow paths with isolation devices that can be closed prior to the RPV water level being equal to the -10 inches indicator scale by a dedicated operator trained in the task, who is in continuous communication with the control room, is stationed at the controls, and is capable of closing the penetration flow path isolation device without offsite power.

c. The penetration flow paths required to be evaluated per paragraph b are assumed to open instantaneously and are not subsequently isolated, and no water is assumed to be subsequently added to the RPV water inventory;

d. No additional draining events occur; and

e. Realistic cross-sectional areas and drain rates are used.

A bounding drain time may be used in lieu of a calculated value.

SAFETY LIMIT

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 700 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 700 psia or core flow is less than 10% of rated, the core power shall not exceed 25% of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

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 and rod block trip settings shall be established according to the following relationships:

The minimum of:

For $W \geq 0\%$:

$S \leq (0.55W + 67\%)T$ with a maximum value of 122%

$S_{RB} \leq (0.55W + 62\%)T$ with a maximum value of 117%

SAFETY LIMIT

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 specified 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.

LIMITING SAFETY SYSTEM SETTING

AND:

<u>For 18% $\leq$ W $\leq$ 40%</u>:

$S \leq (1.287W + 20.83\%)$
$S_{RB} \leq (1.287W + 13.54\%)$

WHERE:

S or S_{RB} = The respective scram or rod block setpoint

W = Loop Recirculation Flow as a percentage of the loop recirculation flow which produces a rated core flow of 67.5 MLB/HR

T = FRTP/CMFLPD (T is applied only if less than or equal to 1.0)

FRTP = Fraction of Rated Thermal Power where Rated Thermal Power equals 1850 MW

CMFLPD = Core Maximum Fraction of Limiting Power Density

With CMFLPD greater than the FRTP for a short period of time, rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of rated thermal power and a notice of adjustment is posted on the reactor control panel.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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 operating during the major maintenance with the water level below the low-low level set point.

In addition to the requirement that at least one licensed Operator be in the control room when fuel is in the reactor, there shall be another control room operator present in the control room with no other duties than to monitor the reactor vessel water level.

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux for IRM range 9 or lower.

   The IRM scram trip setting shall not exceed 38.4% of rated neutron flux for IRM range 10.

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

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 reactor low pressure setting for main-steam-line isolation valve closure shall be ≥ 850 psig when the reactor mode switch is in the run position or the IRMs are on range 10.

g. The main-steam-line isolation valve closure scram setting shall be ≤ 10 percent of valve closure (stem position) from full open.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	h. 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. i. The turbine stop valve closure scram shall be initiated at ≤ 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.

BASES FOR 2.1.1 FUEL CLADDING - SAFETY LIMIT

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 operation (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 ensure that actual fuel operation is maintained within the assumptions of the fuel rod thermal-mechanical design and the safety analysis basis. At full power, this limit is the linear heat generation rate limit with overpower transients constrained by the unadjusted APRM scram and rod block. During steady-state operation at lower power levels, where the fraction of rated thermal power is less than the core maximum fraction of limiting power density, the APRM flow biased scram and rod block settings are adjusted by the equations in Specification 2.1.2a.

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.

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 Fuel Zone Water Level Monitoring System (FZWLMS) instrumentation has an indicated range which allows continuous indication of reactor water level from below the bottom of the active fuel to above the maximum normal water level. The reactor vessel tap for the low-low-low water level instrumentation is located approximately 7 feet 11 inches below the minimum normal water level or approximately 4 feet 6 inches above the top of the active fuel. The low-low-low water level trip point is 6 feet 3 inches (-10 inches indicator scale) below the minimum normal water level (Elevation 302'-9"). The 20 inch difference between the reactor vessel tap and the trip point 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.

BASES FOR 2.1.2 FUEL CLADDING - LIMITING SAFETY SYSTEM SETTING

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 estimated to be about 25% greater than the nominal maximum value expected to occur during the core 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, 18] demonstrate that with a 122% 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.

Also, a scram setting has been established to preclude thermal-hydraulic instabilities which could compromise fuel safety limits. Specifically, the scram setting will limit the oscillation magnitude at reactor trip, thereby limiting the associated CPR change, and in conjunction with MCPR operating limits, assure compliance with the MCPR safety limit.

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 FRTP and CMFLPD. The scram setting is adjusted in accordance with Specification 2.1.1a when the core maximum fraction of limiting power density exceeds the fraction of rated thermal power.

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 a 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 117% 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 core maximum fraction of limiting power density exceeds the fraction of rated thermal power, thus, preserving the APRM rod block safety margin.

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 (<800 psia), the IRM range 9 high flux [16, 17] scram setting is calibrated to correspond to 12% of rated neutron flux. The IRM range 9, 12% of rated neutron flux calibration is on a nominal basis, which provides adequate margin between the setpoint and the safety limit at 25% of rated power. The margin is also adequate to accommodate anticipated maneuvers associated with 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 for low flow condition. This is accomplished by keeping the IRMs on range 9 until 20% flow is exceeded and reactor pressure is >850 psig and that control rods shall not be withdrawn if recirculation flow is less than 30%. If the APRMs are onscale, then the reactor mode switch may be placed in run, thereby switching scram protection from the IRM to the APRM system. If the APRMs are not onscale, then operation with the mode switch in startup (including normal startup mode steam chest warming and bypass valve operation) may continue using IRM range 10, provided that the main turbine generator is not placed in operation.

To continue operation with the mode switch in startup beyond 12% of rated neutron flux, the IRMs must be transferred into range 10. The Reactor Protection System is designed such that reactor pressure must be above 850 psig to successfully transfer the IRMs into range 10, thus assuring added protection for the fuel cladding safety limit. The RPS design will cause the low reactor pressure main-steam-line isolation to be unbypassed when one IRM in trip system 11 and one IRM in trip system 12 are placed in range 10. Procedural controls assure that IRM range 9 is maintained on all IRM channels up to 850 psig reactor pressure. The IRM scram remains active until the mode switch is placed in the RUN position at which time the scram function is transferred to APRMs.

The adequacy of the IRM scram in range 10 (approximately 38.4% of rated neutron flux) was determined by comparing the scram level on the IRM range 10 to the minimum APRM scram level for transient protection. The APRM scram level for transient protection is defined by the Section 2.1.2a equation for $W \geq 0\%$. This equation results in a minimum APRM scram of 67% of rated power at zero recirculation flow. Therefore, startup mode transients (i.e., those not including turbine operation) requiring a scram based on a flux excursion will be terminated sooner with an IRM Range 10 scram than with an APRM scram.

Above the RWM low power setpoint of rated power, the ability of the IRMs to terminate a rod withdrawal transient is limited due to the number and location of IRM detectors. An evaluation was performed that showed by maintaining a minimum core flow of 20.25×10^6 lb/hr (30% rated flow) in range 10, a complete rod withdrawal initiated below 40% of rated power would not result in violating the fuel cladding safety limit. Normal operation of the automatic recirculation pump control will be in excess of 30% rated flow; therefore, little operation below 30% flow is anticipated. Therefore, IRM upscale rod block and scram in range 10 provide adequate protection against a rod withdrawal error transient.

The IRM Limiting Safety System Setting 2.1.2.b for IRM range 9 of <12% rated neutron flux and IRM range 10 of <38.4% of rated neutron flux are nominal trip setpoints as defined by GE Setpoint Methodology as outlined in NEDC-31336. The calibration of these Limiting Safety System Setting values is completed by adjusting IRM amplifier gain such that IRM indication is correlated to rated neutron flux. With the IRM indication correlated to neutron flux, the IRM upscale on range 9 corresponds to 12% and range 10 to 38.4% of rated neutron flux, respectively.

For IRM operation in range 9 or less, 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 UFSAR Section XV-A and B, 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 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.

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 UFSAR Section XV-B.3.13 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-g. 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 and the IRMs on range 9 or lower, 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 ≤10% valve closure, there is no increase in neutron flux and peak pressure if 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 Limiting Safety System Setting, the scram dump volume high level scram trip (LCO 3.6.2) serves as a secondary backup to the Limiting Safety System Setting chosen. This 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)*.

h. 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.

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

*UFSAR

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) UFSAR Section XV-A and B.

(4) UFSAR Section XV-A and B.

(5) UFSAR Section XV-A and B.

(6) UFSAR Section XV-A and B.

(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) UFSAR Section XV-A and B.

(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."

(16) GENE-909-16-0393, "IRM/APRM Overlap Analysis for Nine Mile Point Nuclear Station Unit One," Revision 1, dated April 14, 1993.

(17) GENE-909-39-1093, "IRM/APRM Overlap Improvement for Nine Mile Point Nuclear Station Unit One," dated March 8, 1994.

(18) GENE-C5100196-04, "APRM Flow-Biased Trip Setpoints Stability Long-Term Solution Option II," dated June 1997.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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:

The reactor vessel or reactor coolant system pressure shall not exceed 1375 psig at any time with fuel in the vessel.

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	3
1227	2
1236	2
1245	1
1254	1
	9

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

c. The flow biased APRM scram trip settings shall be in accordance with Specification 2.1.2a.

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 600°F) 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 actuation is selected in accordance with code requirements. The safety valves are sized according to the code for a condition of main steam isolation valve closure while operating at 1850 Mwt, followed by a reactor scram on high neutron flux. Under these conditions, a total of nine (9) safety valves are required to limit reactor pressure below the safety limit of 1375 psig.

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.

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 Sections XV-B.3.1 and 3.5*, 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.

For the APRM scram, the setpoint has been derived based on GE setpoint methodology as outlined in NEDC-31336, "GE Instrumentation Setpoint Methodology." In this methodology, the setpoint is defined as three values, Nominal Trip Setpoint, Allowable Value, and Analytical Limit. The operator will set the Nominal Trip Setpoint. The Allowable Value is listed in the Bases for Specifications 3.6.2 and 4.6.2. The analytical limit is listed in Specification 2.1.2a.

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 Limiting Safety System Setting, other reactor protection system devices (LCO 3.6.2) serve as secondary backup to the Limiting Safety System Setting 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 Section VIII-A.2.1*, 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 is analyzed in Section XV-B.3.1.8*.

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.

*UFSAR

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

3.0.1 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 LCO, 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 LCO 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 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.

3.0.2 through 3.0.7 – Reserved for Future Use

3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is Operable and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is Operable and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required Operable train or subsystem becomes inoperable while this specification is in use, it must be restored to Operable status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1 SRs shall be met during the applicable reactor operating or other specified conditions for individual LCOs, unless otherwise stated in the SR. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

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

4.0.3 If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable specification(s) must be entered.

When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable specification(s) must be entered.

Specifications 4.0.1 through 4.0.3 establish general requirements applicable to all specifications in Sections 4.1 through 4.7 and apply at all times, unless otherwise stated.

4.0.1 Specification 4.0.1 establishes the requirement that SRs must be met during the applicable reactor operating or other specified conditions for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification is to ensure that surveillances are performed to verify the operability of systems and components, and that variables are within specified limits. Failure to meet a surveillance within the specified frequency, in accordance with Specification 4.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire surveillance is performed within the specified frequency.

Systems and components are assumed to be operable when the associated SRs have been met. Nothing in this specification, however, is to be construed as implying that systems or components are operable when either:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the surveillance(s) are known to be not met between required surveillance performances.

Surveillances do not have to be performed when the unit is in a reactor operating or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a special test exception LCO are only applicable when the special test exception LCO is used as an allowable exception to the requirements of a specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given reactor operating or other specified condition.

Surveillances, including surveillances invoked by LCO actions, do not have to be performed on inoperable equipment because the applicable individual specifications define the remedial measures that apply. Surveillances have to be met and performed in accordance with Specification 4.0.2, prior to returning equipment to operable status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment operable. This includes ensuring applicable surveillances are not failed and their most recent performance is in accordance with Specification 4.0.2. Post maintenance testing may not be possible in the current reactor operating or other specified conditions in the LCO due to the necessary unit parameters not having been established. In these situations, the equipment may be considered operable provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a reactor operating or other specified condition where other necessary post maintenance tests can be completed.

4.0.2 Specification 4.0.2 establishes the limit for which the specified time interval for SRs may be extended. It permits an allowable extension of the 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 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs. 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.

4.0.3 Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time it is discovered that the surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified frequency was not met. This delay period permits the completion of a surveillance before complying with LCO actions or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to power operation, or in accordance with the 10 CFR 50 Appendix J Testing Program Plan, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

Specification 4.0.3 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of operating condition changes imposed by LCO actions.

Failure to comply with specified frequencies for surveillance requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. The risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable then is considered outside the specified limits and entry into the applicable LCO actions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and entry into the applicable LCO actions begin immediately upon failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the times allowed by LCO actions, restores compliance with Specification 4.0.1.

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. LCOs for the minimum allowable circuits corresponding to the Limiting Safety System Setting 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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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

(a) The Shutdown Margin (SDM) under all operational conditions shall be equal to or greater than:

0.38% $\Delta k/k$, with the highest worth control rod analytically determined, or

0.28% $\Delta k/k$, with the highest worth control rod determined by test.

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

The SDM shall be verified within limits:

(a) Prior to each in vessel fuel movement during the fuel loading sequence, and

(b) Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(b) If one or more control rods are determined to be inoperable as defined in Specification 3.1.1a(2) while in the power operating condition, then a determination of whether Specification 3.1.1a(1)(a) is met must be made within 6 hours. If a determination cannot be made within the specified time period, then assume Specification 3.1.1a(1)(a) is not met. (c) If Specification 3.1.1a(1)(a) is not met while in the power operating condition, restore compliance with Specification 3.1.1a(1)(a) within 6 hours or be in a shutdown condition within the following 10 hours. (d) If Specification 3.1.1a(1)(a) is not met while in the hot shutdown condition or the cold shutdown condition, then: Immediately initiate action to fully insert all insertable control rods, <u>and</u> Initiate action within 1 hour to restore secondary containment to operable status, <u>and</u> Initiate action within 1 hour to restore one emergency ventilation system to operable status, <u>and</u> Initiate action within 1 hour to restore isolation capability in each required	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

secondary containment penetration flow path not isolated.

(e) If Specification 3.1.1a(1)(a) is not met while in the refueling condition, then:

Immediately suspend core alterations, except for fuel assembly removal, and

Immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

(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 3.1.1a(1)(a) 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.

(2) Reactivity margin - stuck control rods

Each withdrawn control rod shall be exercised in accordance with the Surveillance Frequency Control Program after the control rod has been withdrawn and power level is greater than the low power set point of the RWM. Insert each withdrawn control rod at least one notch.

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.

LIMITING CONDITION FOR OPERATION

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 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)(a) 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 that 0.013 Δk supercritical.

SURVEILLANCE REQUIREMENT

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

(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.

LIMITING CONDITION FOR OPERATION

(b) Whenever the reactor is in the startup or run mode below 10% 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 as follows:

(i) If the rod worth minimizer becomes inoperable after the first 12 control rods have been withdrawn, continue startup provided that a second licensed operator or other qualified member of the technical staff verifies that the licensed operator at the reactor console is following the control rod program.

SURVEILLANCE REQUIREMENT

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

(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) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 10% rated thermal power and a second licensed operator or other qualified member of the technical staff is being used he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(ii) If the rod worth minimizer becomes inoperable before a startup is commenced or before the first 12 control rods have been withdrawn, continue startup provided that a startup with the rod worth minimizer inoperable has not been performed in the last calendar year, and provided that a second licensed operator or other qualified member of the technical staff verifies that the licensed operator at the reactor console is following the control rod program. (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

LIMITING CONDITION FOR OPERATION

c. Scram Insertion Times

(1) The average scram insertion time of all operable control rods, in the power operation condition, shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.00
90	5.00

(2) Except as noted in 3.1.1.c(3), the maximum insertion scram time, in the power operation condition, shall be no greater than:

% Inserted From Fully Withdrawn	Maximum Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.12
90	5.30

SURVEILLANCE REQUIREMENT

c. Scram Insertion Times

The maximum scram insertion time shall be demonstrated through measurement for*:

(1) All control rods prior to thermal power exceeding 40% power with reactor pressure above 800 psig, after each major refueling outage or after a reactor shutdown that is greater than 120 days.

(2) Specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods with reactor pressure above 800 psig.

(3) At least 20 control rods, on a rotating basis, on a frequency in accordance with the Surveillance Frequency Control Program, with reactor pressure above 800 psig.

* For single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(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.	d. Control Rod Accumulators In accordance with the Surveillance Frequency Control Program 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. In accordance with the Surveillance Frequency Control Program verifying each valve to be open;*

2. In accordance with the Surveillance Frequency Control Program 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 In accordance with the Surveillance Frequency Control Program 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	SURVEILLANCE REQUIREMENT
f. If specification 3.1.1.b through e, above, are not met, the reactor shall be placed in the hot shutdown condition within ten hours.	
g. Reactivity Anomalies The difference between a monitored and predicted core k_{eff} shall be within ± 1% Δk/k. 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.	g. Reactivity Anomalies The monitored core k_{eff} shall be compared with the predicted core k_{eff} 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 monitored core k_{eff} will be compared with the predicted core k_{eff}. This comparison will be made every equivalent full power month.

a. Reactivity Limitations

(1) Reactivity margin - core loading

The control rod drop accident analysis assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a control rod drop accident could exceed the fuel damage limits for the accident.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential control rod drop accidents involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

The SDM limits specified in Specification 3.1.1a(1)(a) account for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin must be added to the specified SDM limit to account for uncertainties in the calculation. To ensure adequate SDM, a design margin is included to account for uncertainties in the design calculations (Reference (8)).

The inability to meet the SDM limits during power operating conditions would most likely be due to withdrawn control rods that cannot be inserted. A reduced SDM is not considered an immediate threat to nuclear safety; therefore, time is allowed for analysis to ensure Specification 3.1.1a(1)(a) is met, and for repair before requiring the plant to undergo a transient to achieve a shutdown condition. The allowed completion times of 6 hours for analysis and an additional 6 hours for repair, if Specification 3.1.1a(1)(a) is not met, are considered reasonable while limiting the potential for further reductions in SDM or the occurrence of a transient.

If the SDM cannot be restored within the allowed time, a plant shutdown is required to minimize the potential for, and consequences of, an accident or malfunction of equipment important to safety. The allowed completion time of 10 hours is considered reasonable to achieve the shutdown condition from full power in an orderly manner and without challenging plant systems.

The inability to meet the SDM limits in the hot shutdown condition or the cold shutdown condition could be due to withdrawn control rods that cannot be inserted, discovery of errors in the SDM analysis, or discovery of errors in previous core alterations. The immediate action to fully insert all insertable control rods will result in the least reactive condition for the core and maximizes SDM. This action must continue until all insertable control rods are fully inserted. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is operable, at least one emergency ventilation system is operable, and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are operable, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the operability of the components. If, however, any required component is inoperable, then it must be restored to operable status. In this case, surveillances may need to be performed to restore the component to operable status. Actions must continue until all required components are operable.

The inability to meet the SDM limits in the refueling condition would most likely be due to fuel loading errors. The immediate action to suspend core alterations (e.g., fuel loading) prevents further reductions in SDM. Suspension of core alterations shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and is, therefore, allowed in order to recover SDM. Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted.

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial reactor operating condition, except the major maintenance condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC is required. For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods.

The frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During the refueling condition, adequate SDM is also required to ensure the reactor does not reach criticality during control rod withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload or reload sequences inherently satisfy the surveillance, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

(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 3.1.1a(1)(a).

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 3.1.1a(1)(a), which assures the core can be shut down at all times with control rods.

The allowable inoperable rod patterns will be determined using information 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*.

*FSAR

(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 Δk 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 Δk 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.

Improvements in analytical capability have allowed more refined analysis of the control rod drop accident [1][2][3][4][5][7]. By using the analytical models described in these references coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% 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 10% 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 specified 0.013 Δk 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 Δk. Further, the addition of 0.013 Δk 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 Δk limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical 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 10% 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 accidents are functions of the initial neutron flux. The requirement of at least 3 cps assures that any transient begins 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 SRMs 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-1*, 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 behavior 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 BWRs," 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.

(7) NRC Safety Evaluation, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," dated December 27, 1987.

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

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 and as a post-LOCA suppression pool pH control mechanism.

Specification:

a. During power operating conditions, and whenever the reactor coolant system temperature is greater than 212°F except for reactor vessel hydrostatic or leakage testing with the reactor not critical, 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 or in accordance with the Risk Informed Completion Time Program 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) In accordance with the Surveillance Frequency Control Program -

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	SURVEILLANCE REQUIREMENT

LIMITING CONDITION FOR OPERATION

c. The liquid poison tank shall contain a minimum of 1325 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) <u>In accordance with the Surveillance Frequency Control Program -</u>

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

b. <u>Boron Solution Checks</u>:

(1) <u>In accordance with the Surveillance Frequency Control Program -</u>

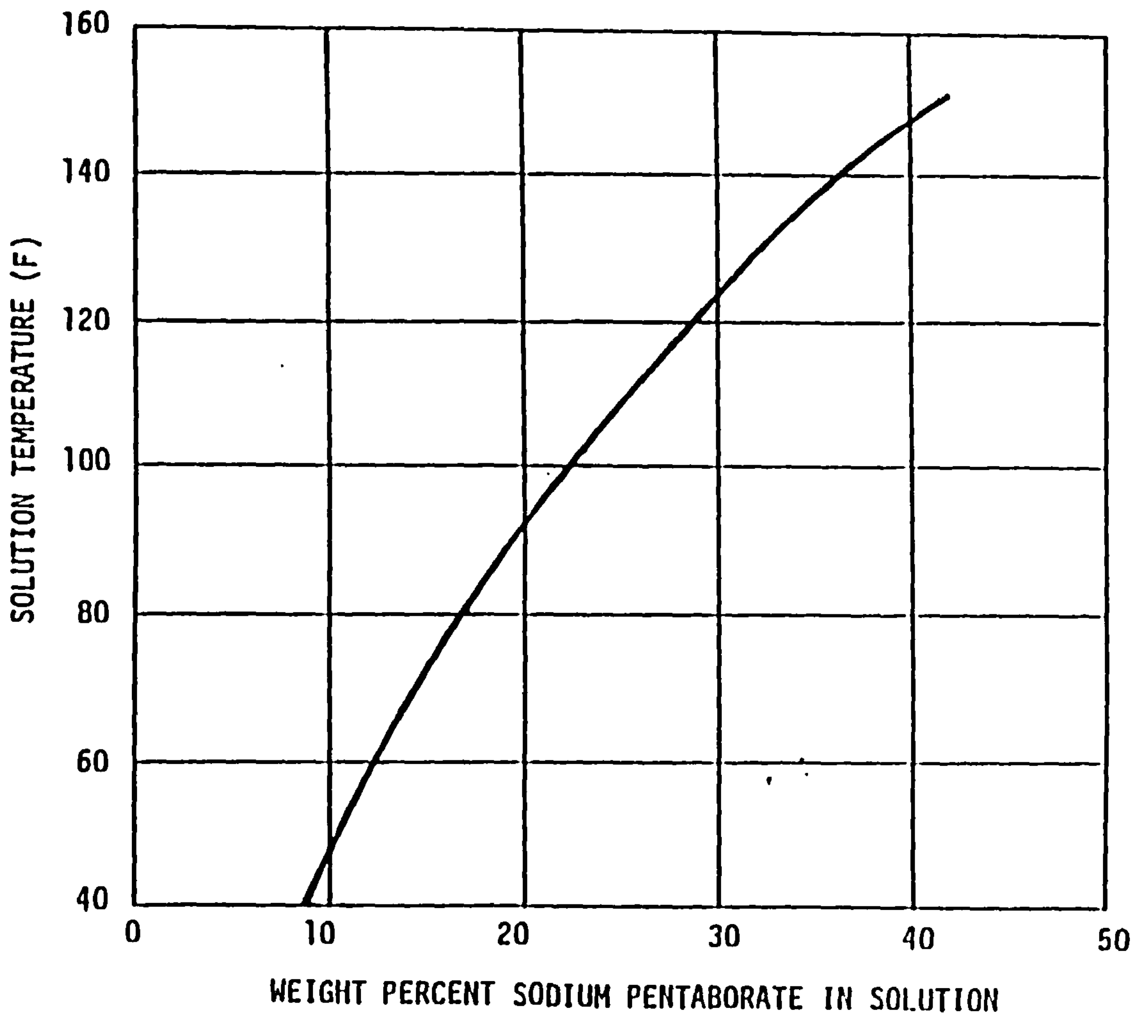
Boron concentration shall be determined.

(2) <u>In accordance with the Surveillance Frequency Control Program -</u>

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) In accordance with the Surveillance Frequency Control Program - The solution temperature shall be checked. (4) In accordance with the Surveillance Frequency Control Program - Verify enrichment by analysis. c. Surveillance with Inoperable Components When a component becomes inoperable its redundant component shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

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 109.8 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 Δk 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 5°F margin above the saturation temperature to guard against precipitation. Temperature and liquid level alarms for the system are annunciated in the Control Room.

*FSAR
(1) GE Topical Report NEDE-31096-P-A, "Anticipated Transients Without Scram. Response to ATWS Rule 10 CFR 50.62."
(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.

*FSAR

LIMITING CONDITION FOR OPERATION

3.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to the operating status of the emergency cooling system.

Objective:

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.

Specification:

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.

b. 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 or in accordance with the Risk Informed Completion Time Program and the additional surveillance required in 4.1.3.f is performed.

SURVEILLANCE REQUIREMENT

4.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to periodic testing requirements for the emergency cooling system.

Objective:

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

Specification:

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

a. In accordance with the Surveillance Frequency Control Program -

   The system heat removal capability shall be determined.

b. In accordance with the Surveillance Frequency Control Program -

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

LIMITING CONDITION FOR OPERATION

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

d. 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.

e. If Specification 3.1.3.a, b, c, or d 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

c. In accordance with the Surveillance Frequency Control Program -

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

d. In accordance with the Surveillance Frequency Control Program -

The area temperature shall be checked.

e. In accordance with the Surveillance Frequency Control Program -

Automatic actuation and functional system testing shall be performed in accordance with the Surveillance Frequency Control Program 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 open position.

f. Surveillance with an Inoperable System -

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 verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

BASES FOR 3.1.3 AND 4.1.3 EMERGENCY COOLING SYSTEM

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 12 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips (Section XV-B.3.15)*. 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) below the minimum normal water level (Elevation 302'9") sustained for 12 seconds. The timers for initiation of the emergency condensers will be set at 12 seconds delay based on the analysis (Section XV-B.3.15)*. For the MSIV closure analysis (Section XV-B.3.5)*, emergency condenser action is ignored.

The minimum water volume in each emergency condenser is 10,680 gallons and the maximum emergency condenser shell side water level is limited to the elevation of the shell overflow lines in each tank. 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.

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**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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.4 CORE SPRAY SYSTEM

Applicability:

Applies to the operating status of the core spray systems when in the Power Operating Condition or Shutdown Condition - Hot.

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 or in accordance with the Risk Informed Completion Time Program 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 or in accordance with the Risk Informed Completion Time Program 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. In accordance with the Surveillance Frequency Control Program automatic actuation of each subsystem in each core spray system shall be demonstrated.

b. In accordance with the Surveillance Frequency Control Program pump operability shall be checked.

c. In accordance with the Surveillance Frequency Control Program the operability of power-operated valves required for proper system operation shall be checked.

LIMITING CONDITION FOR OPERATION

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.

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.

f. (Deleted)

g. (Deleted)

SURVEILLANCE REQUIREMENT

d. (Deleted)

e. <u>Surveillance with Inoperable Components</u>

When a component becomes inoperable its redundant component or system shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

f. With a core spray subsystem suction from the CST, CST level shall be checked in accordance with the Surveillance Frequency Control Program.

g. In accordance with the Surveillance Frequency Control Program 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. (Deleted) i. With the downcomers in the suppression chamber having less than three and one half foot 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.	

BASES FOR 3.1.4 AND 4.1.4 CORE SPRAY SYSTEM

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 10 CFR 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.

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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. In accordance with the Surveillance Frequency Control Program, verify each valve actuator strokes when manually actuated.

b. In accordance with the Surveillance Frequency Control Program, automatic initiation shall be demonstrated.

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 minimum 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.

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.6 <u>CONTROL ROD DRIVE PUMP COOLANT INJECTION</u>

<u>Applicability</u>:

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

<u>Objective</u>:

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).

<u>Specification</u>:

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

4.1.6 <u>CONTROL ROD DRIVE PUMP COOLANT INJECTION</u>

<u>Applicability</u>:

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

<u>Objective</u>:

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

<u>Specification</u>:

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

a. <u>In accordance with the Surveillance Frequency Control Program -</u>

Automatic starting of each pump shall be demonstrated.

LIMITING CONDITION FOR OPERATION

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 or in accordance with the Risk Informed Completion Time Program and the additional surveillance required is performed.

c. If Specifications "a" or "b" above are not met, the reactor coolant temperature shall be reduced to 212°F or less within ten hours.

SURVEILLANCE REQUIREMENT

b. In accordance with the Surveillance Frequency Control Program -

Pump flow rate shall be determined.

c. Surveillance with Inoperable Components

When a component becomes inoperable, its redundant component shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program 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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LIMITING CONDITION FOR OPERATION

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 axial location and average planar exposure shall not exceed the limiting value provided in the Core Operating Limits Report. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown 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

SURVEILLANCE REQUIREMENT

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 axial location and average planar exposure shall be determined in accordance with the Surveillance Frequency Control Program during reactor operation at ≥25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION

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.

b. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR of any rod in any fuel assembly at any axial location shall not exceed the limiting value specified in the Core Operating Limits Report.

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.

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

SURVEILLANCE REQUIREMENT

b. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked in accordance with the Surveillance Frequency Control Program during reactor operation at ≥25% rated thermal power.

c. Minimum Critical Power Ratio (MCPR)

(1) MCPR shall be determined in accordance with the Surveillance Frequency Control Program during reactor power operation at >25% rated thermal power.

(2) MCPR operating limit shall be determined within 72 hours of completing scram time testing as required in Specification 4.1.1(c).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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 This power/flow relationship shall not exceed the limiting values shown in the Core Operating Limits Report. 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 within 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.	d. Power Flow Relationship Compliance with the power flow relationship in Section 3.1.7.d shall be determined in accordance with the Surveillance Frequency Control Program during reactor operation.
e. Partial Loop Operation During power operation, partial loop operation is permitted provided the following conditions are met.	e. Partial Loop Operation Under partial loop operation, surveillance requirements 4.1.7, a, b, c and d above are applicable.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
When operating with four recirculation loops in operation and the remaining loop unisolated, the reactor may operate at 100% of full licensed power level in accordance with the power/flow limits specified in the Core Operating Limits Report 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 the power/flow limits specified in the Core Operating Limits Report 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. 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.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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 the power/flow limits specified in the Core Operating Limits Report 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 is 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.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
i. Required Minimum Recirculation Flow Rate for Operation in IRM Range 10 During startup mode of operation in IRM range 10, a minimum recirculation flow rate of 30% of rated core flow is required. Control rods shall not be withdrawn if recirculation flow is less than 30% of rated.	

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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, an 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, an 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 in 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 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 is not necessary to ensure indicative water level monitoring.

Reporting Requirements

The LCOs 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 LCOs 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.

Required Minimum Recirculation Flow Rate for Operation in IRM Range 10

During power operation above the low power setpoint of 20% power and less than 40% power when in IRM range 10 with the mode switch in startup, the control rod withdrawal error analysis requires the minimum flow to be greater than 30% to ensure protection against the SLMCPR for control rod withdrawal error to the full out position. To ensure compliance with this analysis, the LCO prohibits control rod withdrawal in IRM range 10 if recirculation flow is less than 30%. This is procedurally controlled. This minimum flow restriction does not apply in the run mode.

REFERENCES FOR BASES 3.1.7 AND 4.1.7 FUEL RODS

References (1) through (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, August 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	SURVEILLANCE REQUIREMENT

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 or in accordance with the Risk Informed Completion Time Program 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.

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. In accordance with the Surveillance Frequency Control Program -

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

b. In accordance with the Surveillance Frequency Control Program -

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. Surveillance with Inoperable Components When a component becomes inoperable, its redundant component shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

BASES FOR 3.1.8 AND 4.1.8 HIGH PRESSURE COOLANT INJECTION

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 3,420 gpm of feedwater upon automatic HPCI initiation at all reactor pressure.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.9 Reactor Pressure Vessel (RPV) Water Inventory Control

Applicability:

Applies to the operating status of the core spray systems and Reactor Water Inventory Control when the reactor coolant temperature is less than or equal to 212°F.

Objective:

To assure the RPV water inventory is maintained -10 inches indicator scale.

Specification:

a. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, drain time of RPV water inventory to -10 inches indicator scale shall be ≥36 hours and one core spray subsystem shall be operable except as specified in Specifications b through f below.

b. If the required core spray subsystem becomes inoperable, the component shall be returned to an operable condition within 4 hours.

c. If Specifications a and b are not met, then immediately initiate action to establish a method of water injection capable of operating without offsite electrical power.

4.1.9 Reactor Pressure Vessel (RPV) Water Inventory Control

Applicability:

Applies to the periodic testing requirements for the core spray system and RPV water inventory.

Objective:

To verify the operability of the core spray system and RPV water inventory.

Specification:

a. Verify drain time ≥36 hours in accordance with the Surveillance Frequency Control Program.

b. Verify, for a required core spray subsystem, the downcomers in the suppression chamber have greater than or equal to three and one half foot of submergence or the condensate storage tank inventory is not less than 300,000 gallons, in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
d. If drain time <36 hours and ≥8 hours, within 4 hours perform the following actions:	c. Verify each valve credited for automatically isolating a penetration flow path actuates to the isolation position on an actual or simulated isolation signal, in accordance with the Surveillance Frequency Control Program.
(1) Verify secondary containment boundary is capable of being established in less than the drain time,	d. Verify the required core spray subsystem can be manually operated, in accordance with the Surveillance Frequency Control Program. Vessel spray may be excluded.
and	
(2) Verify each secondary containment penetration flow path is capable of being isolated in less than the drain time,	
and	
(3) Verify one Reactor Building Emergency Ventilation System (RBEVS) circuit is capable of being placed in operation in less than the drain time.	
e. If drain time <8 hours, immediately perform the following actions:	
(1) Initiate action to establish an additional method of water injection with water sources capable of maintaining RPV water level above -10 inches indicator scale for ≥ 36 hours without offsite electrical power,	
and	
(2) Initiate action to establish secondary containment boundary,	
and	
(3) Initiate action to isolate each secondary containment penetration flow path or verify it can be automatically or manually isolated from the control room,	
and	
(4) Initiate action to verify one RBEVS circuit is capable of being placed in operation.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
f. Specifications d or e not met, or drain time is <1 hour, immediately initiate action to restore drain time to ≥36 hours.	

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 REACTOR VESSEL HEATUP AND COOLDOWN RATES Applicability: Applies to the reactor vessel heating or cooling rate. Objective: To assure that thermal stress resulting from reactor heatup and cooldown are within allowable code limits. Specification: During the startup and shutdown operations of the reactor, the reactor vessel heatup and cooldown rates shall be maintained within the limits specified in the Pressure and Temperature Limits Report (PTLR).	

BASES FOR 3.2.1 REACTOR VESSEL HEATUP AND COOLDOWN

Design calculations reported in Volume I, Section V-A, 4.0 (page V-6)* have demonstrated that the heatup and cooldown rate of 100°F/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 215°F 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 1050°F/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 250°F in 7.5 minutes (Volume IV, Section I-B)*. Reactor vessel heatup of 300°F/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 be maintained within the limits specified in the Pressure and Temperature Limits Report (PTLR).

b. During reactor vessel heatup and cooldown when the reactor is critical, the reactor vessel temperature and pressure shall be maintained within the limits specified in the PTLR.

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 specified in the PTLR are met.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. During leakage and hydrostatic testing, the reactor vessel temperature and pressure shall be maintained within the limits specified in the PTLR. 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 the values specified in the PTLR.	

Pages 85 Through 94 Deleted

Pages 94a Through 94d Deleted

BASES FOR 3.2.2 AND 4.2.2 MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION

Figures 3.2.2.a, 3.2.2.b, 3.2.2.c, and 3.2.2.d are plots of pressure versus temperature for heatup and cooldown rates of up to 100°F/hr. maximum (Specification 3.2.1). Figures 3.2.2.e, 3.2.2.f, and 3.2.2.g are plots of pressure versus temperature for leakage and hydrostatic testing. When the minimum temperature for leakage and hydrostatic testing is reached, a thermal soak shall be performed to ensure that the thermal gradient across the vessel wall is negligible. 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 fast neutron fluence at twenty-eight effective full power years of operation were incorporated into the figures. These shifts were calculated using the procedure presented in Regulatory Guide 1.99, Revision 2. 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 account for instrument uncertainties and to reflect the calculated elevation head difference between the pressure sensing instrument locations and the pressure sensitive area of the core beltline region. The temperature readings on the figures have been adjusted to account for instrument uncertainties.

The reactor vessel 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 vessel 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 vessel 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 an NDT temperature of 40°F and they are not subject to any appreciable neutron radiation exposure. Therefore, the minimum vessel flange and head flange temperature for bolting is established at 40°F + 60°F or 100°F.

Figures 3.2.2.a, 3.2.2.b, 3.2.2.c, 3.2.2.d, 3.2.2.e, 3.2.2.f and 3.2.2.g have incorporated a temperature shift due to the calculated fast neutron fluence. The neutron flux at the vessel wall is calculated from core physics data and has been determined using flux monitors installed inside the vessel. The curves, except for 3.2.2.f and 3.2.2.g, are applicable for up to twenty-eight effective full power years of operation. Curves 3.2.2.f and 3.2.2.g are applicable for up to twenty and twenty-four effective full power years, respectively.

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

- effective as of the date of its issuance to be implemented before core operation exceeds 18 effective full-power years

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 for > 24 hours with the coolant temperature ≥200 degrees F and reactor thermal power ≤ 10%, or a shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours.

Conductivity	1 μmho/cm*
Chloride ion	100 ppb
Sulfate ion	100 ppb

* During Noble Metal Chemical Addition (NMCA), the limit is 20 μmho/cm. Post NMCA, the conductivity limit is 2 μmho/cm for up to a 5 month period at power operation.

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, chloride and sulfate ion content in accordance with the Surveillance Frequency Control Program. 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.

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

b. The reactor coolant water shall not exceed the following limits for > 24 hours with reactor thermal power > 10%, or a shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to < 200 degrees F within 10 hours.

Conductivity	1 μmho/cm*
Chloride ion	20 ppb
Sulfate ion	20 ppb

c. In no case shall the reactor coolant exceed the following limits at the specified conditions or, a shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours.

1. With reactor coolant temperature ≥ 200 degrees F, the conductivity has a maximum limit of 5 μmho/cm**, or

2. With reactor coolant temperature ≥ 200 degrees F and reactor thermal power ≤ 10%, the maximum limit of chloride or sulfate ion concentration is 200 ppb, or

3. With reactor thermal power > 10%, the maximum limit of chloride or sulfate ion concentration is 100 ppb.

* Post NMCA, the conductivity limit is 2 μmho/cm for up to a 5 month period at power operation.
** During NMCA, the limit is 20 μmho/cm.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
d. If the continuous conductivity monitor is inoperable for more than seven days, a shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 24 hours.	

In its May 8, 1997 letter, the NRC required that the licensee submit an application for amendment to address the differences between the current TS conductivity limits for reactor coolant chemistry and the analysis assumptions for the core shroud crack growth evaluations. The purpose of this specification is to limit intergranular stress corrosion cracking (IGSCC) crack growth rates through the control of reactor coolant chemistry. The LCO values ensure that transient conditions are acted on to restore reactor coolant chemistry values to normal in a reasonable time frame. Under transient conditions, potential crack growth rates could exceed analytical assumptions, however, the duration will be limited so that any effect on potential crack growth is minimized and the design basis assumptions are maintained. The plant is normally operated such that the average coolant chemistry for the operating cycle is maintained at the conservative values of <0.19 µmho/cm for conductivity and <5 ppb for chloride ions and <5 ppb for sulfate ions. This will ensure that the crack growth rate is bounded by the core shroud analysis assumptions. Since these are average values, there are no specific LCO actions to be taken if these values are exceeded at a specific point in time. The EPRI "BWR Water Chemistry Guidelines-1996 Revision" (EPRI TR-103515-R1, BWRVIP-29) action level 1 guidelines suggest that if conductivity is above 0.3 µS/cm, or chloride or sulfate ions exceed 5 ppb, that corrective action be initiated as soon as possible and to restore levels below level 1 within 96 hours. If the parameters are not reduced to below these levels within 96 hours, complete a review and implement a program and schedule for implementing corrective measures.

Specifications 3.2.3a, b, and c are consistent with the licensee's commitment to Table 4.4 of the BWR water chemistry guidelines. The 24 hour action time period for exceeding the coolant chemistry limits described in 3.2.3a and b ensures that prompt action is taken to restore coolant chemistry to normal operating levels. The requirement to commence a shutdown within 1 hour, and to be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours minimizes the potential for IGSCC crack growth. Reactor water samples are analyzed daily to ensure that reactor water quality remains within the BWR water chemistry guidelines. These samples are analyzed and compared to action level 1 values.

The conductivity of the reactor coolant is continuously monitored. The continuous conductivity monitor is visually checked shiftly in accordance with procedures. The monitor alarms at the local panel. The recorder, which is located in the Control Room, alarms in the Control Room. The samples of the coolant which are analyzed for conductivity daily will serve as a comparison with the continuous conductivity monitor. The primary sample point for the reactor water conductivity samples is the non-regenerative heat exchanger in the reactor water cleanup system. An alternate sample point is the #11 recirculation loop. The reactor coolant samples will also be used to determine the chloride and sulfate concentrations. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride and sulfate ion content. However, if the conductivity becomes abnormal (>0.19 µmho/cm), other than short term spikes, chloride and sulfate measurements will be made within 8 hours to assure that the normal limits (<5 ppb of chloride or sulfate ions) are maintained. A short term spike is defined as a rise in conductivity (>0.19 µmho/cm) such as that which could arise from injection of additional feedwater flow for a duration of approximately 30 minutes in time. These actions will minimize the potential for IGSCC crack growth.

NMP1 will use Noble Metal Chemical Addition (NMCA) as a method to enhance the effectiveness of Hydrogen Water Chemistry (HWC) in mitigating IGSCC. NMCA will result in temporary increases in reactor coolant conductivity values during and following application. During application, the conductivity limit specified in 3.2.3a and 3.2.3c.1 is increased to 20 µmho/cm. The application period includes post-NMCA injection cleanup activities conducted prior to returning the plant to power operation. An increase in conductivity is expected principally due to residual ionic species from the NMCA. However, these species have minor effects on IGSCC and are, therefore, acceptable. During NMCA, samples will be obtained from the temporary skid which is placed in service during the NMCA injection process.

Following NMCA application, industry experience indicates that there may be an elevated conductivity approaching the 1 μmho/cm conductivity limit delineated in TS 3.2.3a and 3.2.3b. To provide operating margin, a conductivity limit of 2 μmho/cm is allowed for up to 5 months of power operation. The increase in the conductivity is attributed to an increase in soluble iron and pH in the reactor water, which results from the application of the noble metals and its affect on deposits on the fuel. Soluble iron nor increased pH contribute to IGSCC crack growth. The existing 1 μmho/cm limit is based on EPRI guidelines action level 2 for power operation, which assumes normal conductivity below .3 μmho/cm. Increasing the limit to 2 μmho/cm during the period when soluble iron levels are high provides an equivalent operating margin consistent with the chloride and sulfate limits. Accordingly, a temporary (<5 month period at power operation) elevated conductivity is acceptable and not considered "abnormal" as discussed in TS 4.2.3 and Bases (i.e., > 0.19 μmho/cm). Therefore, following NMCA, increased sampling (i.e., every 8 hours versus daily) is not warranted.

LIMITING CONDITION FOR OPERATION

3.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

Applicability:

Applies to the limits on reactor coolant specific activity.

Objective:

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

Specification:

a. During the power operating and hot shutdown conditions, the specific activity of the reactor coolant shall be limited to Dose Equivalent I-131 specific activity $\leq 0.2\ \mu Ci/gm$.

b. If reactor coolant specific activity is $> 0.2\ \mu Ci/gm$ and $\leq 4.0\ \mu Ci/gm$ Dose Equivalent I-131, determine the Dose Equivalent I-131 once per 4 hours and restore Dose Equivalent I-131 to within the limit of Specification 3.2.4.a within 48 hours.

SURVEILLANCE REQUIREMENT

4.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

Applicability:

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

Objective:

To assure that limits on coolant specific activity are not exceeded.

Specification:

a. When the unit is in the power operating condition, verify that reactor coolant Dose Equivalent I-131 specific activity is $\leq 0.2\ \mu Ci/gm$ in accordance with the Surveillance Frequency Control Program.

b. Verify that reactor coolant Dose Equivalent I-131 specific activity is $\leq 0.2\ \mu Ci/gm$ within 24 hours prior to raising the reactor coolant temperature > 215°F, with the reactor not critical, and with primary containment integrity not established.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If the required actions and completion times of Specification 3.2.4.b cannot be met, or if reactor coolant specific activity is > 4.0 μCi/gm Dose Equivalent I-131, place the reactor in the hot shutdown condition within 12 hours and in the cold shutdown condition within the following 24 hours. d. The steady state specific activity of the reactor coolant shall be limited to Dose Equivalent I-131 specific activity ≤ 0.2 μCi/gm when the reactor coolant temperature is > 215°F, the reactor is not critical, and primary containment has not been established.	

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 1m/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 0.1 μCi/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.

A more restrictive reactor coolant total iodine limit has been imposed for Control Room habitability purposes only. A limit of 9.47 μCi/g is imposed based on the most limiting small break LOCA outside containment. Provided reactor coolant iodine is maintained at or below this value, the Control Room Air Treatment System would not be required to maintain the radiological effects of the line break below GDC19 dose limits.

In the event of a large primary system break under reactor vessel hydrostatic or leakage test conditions with the reactor coolant temperature >215°F, the reactor not critical, and primary containment integrity not established, calculations show the resultant radiological dose at the exclusion area boundary to be conservatively bounded by the dose calculated for a main steam line break outside primary containment. This dose was calculated on the basis of the radioiodine concentration limit of 1.5 μCi of total iodine per gram of water. The reactor coolant sample required by Specification 4.2.4.c will be used to assure that the limit of Specification 3.2.4.c is not exceeded. The sample shall be taken during steady state conditions to ensure the results are representative of the steady state radioactive concentration for reactor vessel hydrostatic or leakage test conditions.

LIMITING CONDITION FOR OPERATION

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.

SURVEILLANCE REQUIREMENT

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 in accordance with the Surveillance Frequency Control Program.

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 the 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 in accordance with the Surveillance Frequency Control Program. (2) An instrument functional test in accordance with the Surveillance Frequency Control Program.

BASES FOR 3.2.5 AND 4.2.5 REACTOR COOLANT SYSTEM LEAKAGE RATE

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.

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.

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 tanks. 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

Another method of determining reactor coolant leakage rate is by monitoring for excess leakage in the drywell floor and equipment drain tanks. This system monitors the change in tank volume over accurate time periods for the full range of tank instrumentation. If the leakage is high enough, an alarm is actuated indicating a leak rate above the predetermined limit (Section V.B)*.

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 excess leakage detection system or the integrated flow can be utilized to satisfy Specification 3.2.5.b.

*FSAR

(Deleted)

(Deleted)

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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, and to minimize potential leakage paths from the primary containment in the event of a loss-of-coolant accident.

Specification:

a. Whenever fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, all reactor coolant system isolation valves on lines connected to the reactor coolant system shall be operable except as specified in Specification 3.2.7.b below.

b. In the event any isolation valve becomes inoperable whenever fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, the system shall be considered operable provided that within 4 hours at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition, except as noted in Specification 3.1.1.e.

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, and to limit potential leakage paths from the primary containment in the event of a loss-of-coolant accident.

Specification:

The reactor coolant system isolation valves surveillance shall be performed as indicated below.

a. In accordance with the Surveillance Frequency Control Program the operable automatically initiated power-operated isolation valves shall be tested for automatic initiation and closure times.

b. Additional surveillances shall be performed as required by the INSERVICE TESTING PROGRAM.

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. In accordance with the Surveillance Frequency Control Program the feedwater and main-steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening. d. In accordance with the INSERVICE TESTING PROGRAM the feedwater and main steam line power-operated isolation valves shall be fully closed and reopened.

PAGES 110 THROUGH 114 ARE NOT USED

The list of reactor coolant isolation valves is contained in the procedure governing controlled lists and have been removed from the Technical Specifications per Generic Letter 91-08. Revisions will be processed in accordance with Quality Assurance Program requirements.

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. Valve closure times are selected to minimize coolant losses in the event of the specific line rupturing and are procedurally controlled. Using the longest closure time on the main-steam-line valves following a main-steam-line break (Section XV C.1.0)(1), 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-D.1.0(1).

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)(2) that a line will not isolate. Additional surveillances are in accordance with the Inservice Testing Program described in Specification 6.5.4.

(1) UFSAR
(2) FSAR

LIMITING CONDITION FOR OPERATION

3.2.7.1 <u>REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE (PIV) LEAKAGE</u>

<u>Applicability</u>:

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

<u>Objective</u>:

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

<u>Specification</u>:

-----------------------------NOTES-----------------------------

1. Separate specification entry is allowed for each flow path.
2. Enter applicable specifications for systems made inoperable by PIVs.

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a. The integrity of each pressure isolation valve shall be demonstrated. Valve leakage shall be within limit during the power operating and hot shutdown reactor operating conditions.

SURVEILLANCE REQUIREMENT

4.2.7.1 <u>REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE (PIV) LEAKAGE</u>

<u>Applicability</u>:

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

<u>Objective</u>:

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

<u>Specification</u>:

------------------------------NOTE------------------------------

Not required to be performed in the hot shutdown reactor operating condition.

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a. The equivalent leakage of each reactor coolant system PIV shall be verified to be $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a reactor coolant system pressure $\geq$ 1010 psig and $\leq$ 1050 psig, in accordance with the INSERVICE TESTING PROGRAM.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. If one or more flow paths with leakage from one or more PIVs is not within limit: 1. Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve within 4 hours, and 2. Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve within 72 hours. Each valve used to satisfy Specifications b.1 and b.2 above must have been verified to meet Specification 4.2.7.1.a and be in the reactor coolant system pressure boundary or the high pressure portion of the system. c. If Specification 3.2.7.1.b 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.	

TABLE 3.2.7.1

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LIMITING CONDITION FOR OPERATION

3.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the operational status of the safety valves.

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.

Specification:

a. During power operating conditions and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than saturation temperature all nine of the safety valves shall be operable.

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.

SURVEILLANCE REQUIREMENT

4.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the periodic testing requirements for the safety valves.

Objective:

To assure the capability of the safety valves to limit reactor overpressure to below the safety limit.

Specification:

In accordance with the Surveillance Frequency Control Program, the number of safety valves as determined by the IST Program Plan shall be removed, tested for set point and partial lift, and then returned to operation or replaced.

BASES FOR 3.2.8 AND 4.2.8 PRESSURE RELIEF SYSTEM-SAFETY VALVES

The required number of operable safety valves is based on a condition of main steam isolation valve closure while operating at 1850 Mwt, followed by a reactor scram on high neutron flux. Operation of all 9 safety valves will limit reactor pressure below the safety limit of 1375 psig.

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

LIMITING CONDITION FOR OPERATION

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.

SURVEILLANCE REQUIREMENT

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	b. In accordance with the Surveillance Frequency Control Program, verify each valve actuator strokes when manually actuated. c. In accordance with the Surveillance Frequency Control Program, 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, the Allowable Value for each valve can be as much as ± 24 psig. The as found value for at least 2 relief valves must be greater than the as found high reactor pressure scram value.

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 215°F and fuel is in the reactor vessel except while performing (1) low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mwt, (2) reactor vessel hydrostatic or leakage tests with the mode switch in refuel or shutdown, (3) scram time testing with the mode switch in refuel when performed in conjunction with reactor vessel hydrostatic or leakage tests, or (4) excess flow check valve testing with the mode switch in refuel or shutdown when performed in conjunction with reactor vessel hydrostatic or leakage tests.

LIMITING CONDITION FOR OPERATION

3.3.1 OXYGEN CONCENTRATION

Applicability:

Applies to the limit on oxygen concentration within the primary containment system in the power operating condition.

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 while in the power operating condition, except as specified in "b" below.

SURVEILLANCE REQUIREMENT

4.3.1 OXYGEN CONCENTRATION

Applicability:

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

Objective:

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

Specification:

In accordance with the Surveillance Frequency Control Program, oxygen concentration shall be determined.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. If the containment oxygen concentration is greater than or equal to the four percent by volume limit while in the power operating condition then restore oxygen concentration to within limit within 72 hours. c. If Specifications "a" or "b" above are not met, exit the power operating condition within 12 hours.	

BASES FOR 3.3.1 AND 4.3.1 OXYGEN CONCENTRATION

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-II.5.2)*. The only way that significant quantities of hydrogen could be generated by metal-water reaction would be if the core spray system failed to sufficiently cool the core. As discussed in Section VII-A.2.0*, each core spray system will deliver, as a minimum, core spray sparger flow as shown on Figure VII-2*. In addition to hydrogen generated by metal-water reaction, significant quantities can be generated by radiolysis. (Technical Supplement 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 344°F.

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.

*FSAR

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 and one half feet and a maximum submergence of four and one quarter feet whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required.

b. During normal power operation, suppression chamber water temperature shall be less than or equal to 85°F.

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. In accordance with the Surveillance Frequency Control Program, 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 in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

c. If Specifications a and b above are not met within 24 hours, the reactor shall be shut down using normal shutdown procedures.

d. During testing of relief valves which add heat to the torus pool, bulk pool temperature shall not exceed 10°F 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. The reactor shall be scrammed from any operating condition when the suppression pool bulk temperature reaches 110°F. 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 120°F.

SURVEILLANCE REQUIREMENT

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.

d. Whenever operation of a relief valve is indicated and the bulk suppression pool temperature reaches 160°F 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.

e. Whenever there is indication of relief valve operation with the local temperature of the suppression pool reaching 200°F or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

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

The combination of three and one-half foot downcomer submergence, 85°F suppression chamber water temperature at lake water temperature defined by specification 3.3.7/4.3.7 will maintain post-accident system temperature and pressure within FSAR design limits (FSAR Section VI, XV, XVI).

The three and one-half foot minimum and the four and one-quarter foot maximum submergence are a result of Suppression Chamber Heat-up Analysis and the Mark I Containment Program respectively. The minimum submergence provides sufficient water to meet the Suppression Chamber Heat-up Analysis post LOCA and the maximum submergence limits the torus levels to be consistent with the Mark I Plant Unique Analysis.

The 215°F 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 312°F 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 160°F 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 200°F. 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 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 to be the points of highest stress.

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

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 50.67 in the event of a loss-of-coolant accident accompanied by significant fuel cladding failure and hydrogen generation from a metal-water reaction.

To assure that periodic surveillances of reactor containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

Specification:

Whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required, the primary containment leakage rate shall be limited to:

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. The primary containment leakage rates shall be demonstrated at test schedules and in conformance with the criteria specified in the 10 CFR 50 Appendix J Testing Program Plan as described in Specification 6.5.7.

b. The provisions of Specification 4.0.2 are not applicable, and the surveillance interval extensions are in accordance with the 10 CFR 50 Appendix J Testing Program Plan.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
a. An overall integrated leakage rate of less than 1.5% by weight of the containment air per day (La), at 35 psig (Pac). b. A combined leakage rate on a minimum pathway basis of less than 0.6 La for all penetrations and all Primary Containment Isolation Valves subject to Types B and C tests when pressurized to 35 psig (Pac).	

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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 suppression chamber are 62 psig and 35 psig, respectively.(2) 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 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.

The maximum allowable leakage rate (L_a) is 1.5%/day at a pressure of 35 psig (P_a). 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.

Closure of the containment isolation valves for the purpose of the test is accomplished by the means provided for normal operation of the valves. The reactor is vented to the containment atmosphere during ILRT testing.

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 Option B of 10 CFR 50 Appendix J.

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.

Leakage from airlocks is measured under accident pressures in accordance with Option B of 10 CFR 50 Appendix J.

The Type A test follows the guidelines stated in ANSI/ANS-56.8[8] and/or the Bechtel Topical Report.[4] This program provides adequate assurance that the test results realistically estimates the degree of containment leakage following a loss-of-coolant accident. The containment leakage rate is calculated using the Absolute Methodology.[8]

The specific treatment of selective valve arrangements including the acceptability of the interpretations of 10 CFR 50 Appendix J requirements are given in References 5, 6, and 7. They serve as the bases for alternative test configurations (e.g., reverse accident, multi-valve, water leakage flow tests) as well as relaxations from previous leakage limits or constraints.

References:

(1) FSAR, Volume II, Appendix E

(2) UFSAR, Section VI B.2.1

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

(4) BN-TOP-1 "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants," Revision 1, Bechtel Corporation, November 1, 1972

(5) NRC Safety Evaluation Report dated May 6, 1988, "Regarding Proposed Technical Specifications and Exemption Requests Related to Appendix J."

(6) Niagara Mohawk Letter dated July 28, 1988, "Clarifications, Justifications & Conformance with 10 CFR 50 Appendix J SER."

(7) NRC Letter dated November 9, 1988, "Review of the July 28, 1988 Letter on Appendix J Containment Leakage Rate Testing at Nine Mile Point Unit 1."

(8) ANSI/ANS - 56.8 - 1994, "Containment System Leakage Testing Requirements."

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 215°F and primary containment integrity is required, 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 isolation valve becomes inoperable the system shall be considered operable provided that within 4 hours or in accordance with the Risk Informed Completion Time Program 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 below.

a. In accordance with the Surveillance Frequency Control Program, the operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure times.

b. In accordance with the Surveillance Frequency Control Program, all normally open power operated isolation valves shall be fully closed and reopened.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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 215°F within ten hours.	c. In accordance with the Surveillance Frequency Control Program, a representative sample of instrument-line flow check valves will be tested for operability.

PAGES 145 THROUGH 149 ARE NOT USED

The list of primary containment isolation valves is contained in the procedure governing controlled lists have been removed from the Technical Specifications per Generic Letter 91-08. Revisions will be processed in accordance with Quality Assurance Program requirements.

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 valves shall be capable of automatic initiation. 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-D.[1] For allowable leakage rate specification, see Section 3.3.3/4.3.3.

For the design basis loss-of-coolant accident fuel rod perforation would not occur until the fuel temperature reached 1700°F which occurs in approximately 100 seconds.[2] The required closing times for all primary containment isolation valves are established to prevent fission product release through lines connecting to the primary containment.

For reactor coolant system temperatures less than 215°F, the containment could not become pressurized due to a loss-of-coolant accident. The 215°F 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).[3] More frequent testing for valve operability results in a more reliable system.

In addition to routine surveillance as outlined in Section VI-D.1.0[1] 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.

(1) UFSAR
(2) Nine Mile Point Nuclear Generation Station Unit 1 Safer/Corecool/GESTR-LOCA Loss of Coolant Accident Analysis, NEDC-31446P, Supplement 3, September, 1990.
(3) FSAR

LIMITING CONDITION FOR OPERATION

3.3.5 ACCESS CONTROL

Applicability:

Applies to the access control to the primary containment.

Objective:

To assure the integrity of the primary containment system.

Specification:

Whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required, the following shall be in effect.

a. Only one door in each of the two double-door drywell access locks will be opened at one time.

b. The equipment hatch and drywell head and other flanged openings will be secured.

c. If following a routine surveillance check "a" or "b" is not met, initiate normal orderly shutdown within one hour and have reactor in the cold shutdown condition within ten hours.

SURVEILLANCE REQUIREMENT

4.3.5 ACCESS CONTROL

Applicability:

Applies to the surveillance on primary containment access control.

Objective:

To assure the operability of the primary containment access control interlocks.

Specification:

A mechanical interlock will be maintained to prevent simultaneous opening of two doors.

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	SURVEILLANCE REQUIREMENT

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 not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

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

In accordance with the Surveillance Frequency Control Program 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 in accordance with the Surveillance Frequency Control Program and following any maintenance on the valves and associated equipment.

LIMITING CONDITION FOR OPERATION

(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.

SURVEILLANCE REQUIREMENT

b. <u>In accordance with the Surveillance Frequency Control Program</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) In accordance with the Surveillance Frequency Control Program, 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.

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.	c. Pressure Suppression Chamber - Reactor Building Vacuum Breakers
d. One drywell-suppression chamber vacuum breaker may be secured in the closed position.	(1) The pressure suppression chamber-reactor building vacuum breaker systems and associated instrumentation, including set point, shall be checked for proper operation in accordance with the Surveillance Frequency Control Program.
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.	(2) In accordance with the Surveillance Frequency Control Program, 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.
f. Pressure Suppression Chamber - Reactor Building Vacuum Breakers	
(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 the pressure suppression chamber-reactor building air-operated vacuum breakers shall be ≤ 0.5 psid.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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 or in accordance with the Risk Informed Completion Time Program unless such vacuum breaker system is sooner made operable, provided that the procedure does not violate containment integrity.	

BASES FOR 3.3.6 AND 4.3.6 VACUUM RELIEF

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 ± 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 215°F, 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 ± 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.

*FSAR

BASES FOR 3.3.6 AND 4.3.6 VACUUM RELIEF

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

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 and primary containment integrity is required; 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., 3.3.7.c, 3.3.7.d, or 3.3.7.e.

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 or in accordance with the Risk Informed Completion Time Program and that the additional surveillance required is performed.

SURVEILLANCE REQUIREMENT

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) In accordance with the Surveillance Frequency Control Program, automatic startup of the containment spray pump shall be demonstrated.

(2) In accordance with the Surveillance Frequency Control Program, pump operability shall be checked.

b. Nozzles

Following maintenance that could result in nozzle blockage, a test shall be performed on the spray 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 or in accordance with the Risk Informed Completion Time Program and that the additional surveillance required is performed.	c. Raw Water Cooling Pumps In accordance with the Surveillance Frequency Control Program, 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 or in accordance with the Risk Informed Completion Time Program.	d. Surveillance with Inoperable Components When a component or system becomes inoperable its redundant component or system shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.
e. If both containment spray system loops become inoperable, or if one containment spray system loop and one train in the redundant loop become inoperable, the reactor may remain in operation for a period not to exceed 8 hours.	
f. If Specifications “a,” “b,” “c,” “d,” or “e” are not met, the reactor shall be in the hot shutdown condition within 12 hours and the reactor shall be in the cold shutdown condition within 36 hours and no work 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").	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
g. The containment spray system shall be considered operable by verifying that lake water temperature does not exceed 83°F. h. If specification "g" cannot be met commence shutdown within one hour and be in hot shutdown within 8 hours and cold shutdown within 24 hours.	g. Lake Water Temperature Record in accordance with the Surveillance Frequency Control Program and at least once per 8 hours when latest recorded water temperature is greater than or equal to 75°F and at least once per 4 hours when the latest recorded water temperature is greater than or equal to 79°F.

BASES FOR 3.3.7 AND 4.3.7 CONTAINMENT SPRAY SYSTEM

For reactor coolant temperatures less than 215°F not enough steam is generated during a loss-of-coolant accident to pressurize the containment. For reactor 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 cooling flow.[1] The specified flow of 3600 gpm at 87.7 psid primary, 89 psid secondary (approximately 95 percent to the drywell and the balance to the suppression chamber) is sufficient to remove post accident core energy released (FSAR Section VII). Requiring both pumps systems operable (400 percent redundancy) will assure the availability of the containment spray system.[1]

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

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 side of the containment spray heat exchangers is 10 psig greater than that on the torus water side (not less than 141 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.

Automatic initiation of the containment spray system assures that the containment will not be overpressurized. This automatic feature would only be required if all core spray systems malfunctioned and significant metal-water reaction occurred. For the normal operation condition of 85°F suppression chamber water, containment spray actuation would not be necessary for about 15 minutes.

[1]With two of the containment spray intertie valves open, operation of two containment spray pumps is required to assure the proper flow distribution to the containment spray headers to reduce containment pressure during the first fifteen minutes of the LOCA. Requiring two containment spray pumps to operate reduces the 400 percent redundance of the containment spray system, but there are still six combinations (two out of four pumps) that will assure two pump operation.

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.

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*FSAR

3.4.0 <u>REACTOR BUILDING</u>

APPLICABILITY

Applies to the operating status of the reactor building (secondary containment).

OBJECTIVE

To assure the integrity of the reactor building.

SPECIFICATION

Reactor building integrity must be in effect for the following conditions:

a. Power operating condition,

b. When the reactor water temperature is above 215°F, or

c. Whenever recently irradiated fuel or an irradiated fuel cask is being handled in the Reactor Building.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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:

At all times when secondary containment integrity is required, the reactor building leakage rate as determined by Specification 4.4.1 shall not exceed 1600 cfm. If this cannot be met after a routine surveillance check, then the actions listed below shall be taken:

a. Suspend any of the following activities:

1. Handling of recently irradiated fuel in the reactor building.

2. Irradiated fuel cask 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:

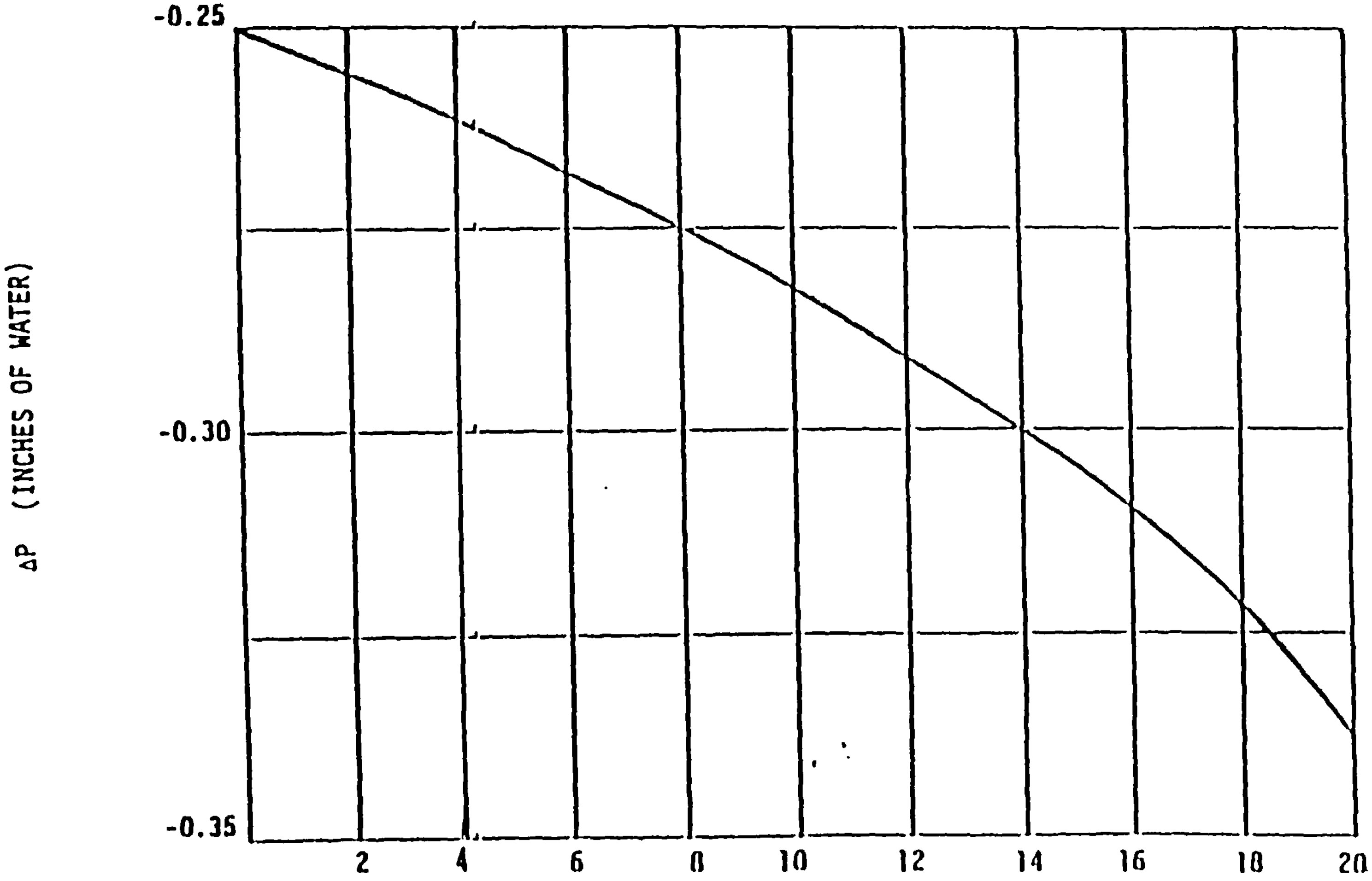
To assure the capability of the secondary containment to maintain leakage within allowable limits.

Specification:

In accordance with the Surveillance Frequency Control Program - 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



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 1600 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.

*FSAR

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 at all times when secondary containment integrity is required.

b. If Specification 3.4.2.a is not met, then the actions listed below shall be taken:

1. The reactor shall be in the cold shutdown condition within ten hours.

2. Suspend any of the following activities:

   a. Handling of recently irradiated fuel in the reactor building,

   b. Irradiated fuel cask handling operations in the reactor building.

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 building isolation valves.

Specification:

In accordance with the Surveillance Frequency Control Program, 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	SURVEILLANCE REQUIREMENT

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. At all times when secondary containment integrity is required, the following conditions will be met:

   1. Only one door in each of the double-doored access ways shall be opened at one time, except when the access opening is being used for entry and exit.

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

   3. 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.

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 and to assure the integrity of the secondary containment system by verifying that reactor building access doors are closed, as required by Specifications 3.4.3.a.1 and 3.4.3.a.2.

Specification:

a. The core and containment spray pump compartments shall be checked in accordance with the Surveillance Frequency Control Program and after each entry.

LIMITING CONDITION FOR OPERATION

b. If these conditions cannot be met, then the actions listed below shall be taken:

1. If in the power operating condition, restore reactor building integrity within 4 hours or be in at least the hot shutdown condition within the next 12 hours and in the cold shutdown condition within the following 24 hours.

   OR

   If the reactor coolant system temperature is above 215°F, restore reactor building integrity within 4 hours or be in cold shutdown within the following 24 hours.

2. Suspend any of the following activities:

   a. Handling of recently irradiated fuel in the reactor building,

   b. Irradiated fuel cask handling operations in the reactor building.

SURVEILLANCE REQUIREMENT

b. Verify in accordance with the Surveillance Frequency Control Program that:

1. At least one door in each access to the secondary containment is closed, except when the access opening is being used for entry and exit.

2. At least one door or closeup of the railroad bay is closed.

BASES FOR 3.4.3 AND 4.4.3 ACCESS CONTROL

The reactor building serves as a secondary containment when the reactor coolant system temperature is above 215°F and during normal Station operations and as a primary containment during refueling and other periods when the reactor coolant system temperature is above 215°F and the pressure suppression system is open or not required. 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 Section XV.C.3 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. Surveillance of the reactor building access doors provides additional assurance 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

LIMITING CONDITION FOR OPERATION

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 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.

SURVEILLANCE REQUIREMENT

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. In accordance with the Surveillance Frequency Control Program, 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) Deleted

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. The results of laboratory carbon sample analysis shall show ≥95% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 at 30°C and 95% R.H.

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

e. During reactor operation, including when the reactor coolant system temperature is above 215°F, 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 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.

During handling of recently irradiated fuel in the reactor building or handling of an irradiated fuel cask in the reactor building from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, recently irradiated fuel handling in the reactor building or irradiated fuel cask handling in the reactor building is permissible during the succeeding seven days unless such circuit is sooner made operable, provided that

b. The tests and sample analysis of Specification 3.4.4b, c and d shall be performed in accordance with the Surveillance Frequency Control Program, 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 at least 15 minutes in accordance with the Surveillance Frequency Control Program.

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
during such seven days all active components of the other emergency ventilation circuit shall be operable. Recently irradiated fuel handling in the reactor building or irradiated fuel cask handling in the reactor building may continue beyond seven days provided the operable emergency ventilation circuit is in operation.	g. In accordance with the Surveillance Frequency Control Program, automatic initiation of each branch of the emergency ventilation system shall be demonstrated.
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.	h. In accordance with the Surveillance Frequency Control Program, 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 verified to be operable within two hours and in accordance with the Surveillance Frequency Control Program thereafter.

BASES FOR 3.4.4 AND 4.4.4 EMERGENCY VENTILATION SYSTEM

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 the 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 removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 95 percent, which is derived from applying a safety factor of 2 to the charcoal filter efficiency of 90 percent assumed in analyses of design basis accidents. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 and General Design Criterion 19 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 should allow for charcoal sampling to be conducted using an ASTM D3803-1989 approved method. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent meeting the physical property specifications of 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 verified to be operable daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation may continue during this period.

LIMITING CONDITION FOR OPERATION

3.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the operating status of the control room air treatment system and Control Room Envelope (CRE) boundary.

---------------------------NOTE---------------------------
The CRE boundary may be opened intermittently under administrative control.

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 below, the control room air treatment system shall be operable for the following conditions:

   1. Power operating condition, and whenever the reactor coolant system temperature is greater than 212°F.

   2. Whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building.

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.

SURVEILLANCE REQUIREMENT

4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the testing of the control room air treatment system and CRE boundary.

Objective:

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

Specification:

a. In accordance with the Surveillance Frequency Control Program, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 1.5 inches of water at system design flow rate (± 10%).

b. The tests and sample analysis of Specification 3.4.5b, c and d shall be performed in accordance with the Surveillance Frequency Control Program, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release any ventilation zone communicating with the system.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. The results of laboratory carbon sample analysis shall show ≥95% radioactive methyl iodine removal when tested in accordance with ASTM D3803-1989 at 30°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, except for an inoperable CRE boundary during the power operating condition, restore the system to operable within the succeeding seven days.

f. If the control room air treatment system is made or found to be inoperable due to an inoperable CRE boundary during the power operating condition: immediately initiate action to implement mitigating actions; within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits; and within 90 days, restore the CRE boundary to operable status.

g. If Specifications 3.4.5.e or 3.4.5.f cannot be met during the power operating condition, or when reactor coolant system temperature is greater than 212°F, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours.

h. If Specification 3.4.5.e cannot be met whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, immediately suspend handling of recently irradiated fuel or the irradiated fuel cask in the reactor building.

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 15 minutes in accordance with the Surveillance Frequency Control Program.

f. In accordance with the Surveillance Frequency Control Program, automatic initiation of the control room air treatment system shall be demonstrated.

g. In accordance with the frequency and specifications of the Control Room Envelope Habitability Program, perform required CRE unfiltered air inleakage testing.

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 maintain the control room pressure to the design positive pressure (one-sixteenth inch water) so that all leakage should be out leakage. The control room air treatment system starts automatically upon receipt of a LOCA (high drywell pressure or low-low reactor water level) or Main Steam Line Break (MSLB) (high steam flow main-steam line or high temperature main-steam line tunnel) signal. The system can also be manually initiated.

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 95 percent, which is derived from applying a safety factor of 2 to the charcoal filter efficiency of 90 percent assumed in analyses of design basis accidents. 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 inoperable, 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 1.5 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.

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 should allow for charcoal sampling to be conducted using an ASTM D3803-1989 approved method. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent meeting the physical property specifications of 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 CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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.

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.

In accordance with the Surveillance Frequency Control Program - 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	SURVEILLANCE REQUIREMENT
3.5.2 REFUELING PLATFORM INTERLOCK	4.5.2 REFUELING PLATFORM INTERLOCK
Applicability:	Applicability:
Applies to the refueling platform on interlocks.	Applies to the periodic testing requirements for the refueling platform interlocks.
Objective:	Objective:
To assure that a loaded refueling platform hoist is never over the core when one or more control rods are withdrawn.	To assure the operability of the refueling platform interlock.
Specification:	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.	The refueling platform interlocks shall be tested prior to any fuel handling with the head off the reactor vessel, and in accordance with the Surveillance Frequency Control Program 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

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:

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.

SURVEILLANCE REQUIREMENT

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
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 necessary.	

BASES FOR 3.5.3 EXTENDED CORE AND CONTROL ROD DRIVE MAINTENANCE

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 and after withdrawing the corresponding control rod, the refueling interlock will be bypassed on that rod. After 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 mechanical vacuum pump isolation, 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	SURVEILLANCE REQUIREMENT
3.6.1 MECHANICAL VACUUM PUMP ISOLATION	4.6.1 MECHANICAL VACUUM PUMP ISOLATION
a. (Deleted)	a. (Deleted)
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 main steam line.	b. In accordance with the Surveillance Frequency Control Program (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 3.6.2m.

   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 the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Tables 4.6.2a to 4.6.2m.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(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.
(3) Emergency Cooling Initiation or Isolation - The emergency cooling system shall be considered inoperable and Specification 3.1.3 shall be applied.	
(4) Core Spray Initiation - The core spray system shall be considered inoperable and Specification 3.1.4 shall be applied.	
(5) Containment Spray Initiation - The containment spray system shall be considered inoperable and Specification 3.3.7 shall be applied.	
(6) Auto Depressurization Initiation - The auto depressurization system shall be considered inoperable and Specification 3.1.5 shall be applied.	
(7) Control Rod Withdrawal Block - No control rods shall be withdrawn.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(8) Mechanical Vacuum Pump Isolation - The mechanical vacuum pump 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.	
(13) Reactor Pressure Vessel Water Inventory Control - Specification 3.1.9 shall be applied for the applicable drain time.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. During operation with the Core Maximum Fraction of Limiting Power Density (CMFLPD) greater than the Fraction of Rated Thermal Power (FRTP), either: (1) The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.2a; or (2) Action shall be taken within two (2) hours to adjust the APRM gain in accordance with Specification 2.1.2a; or (3) The power distribution shall be changed such that the CMFLPD no longer exceeds FRTP.	c. During reactor power operation at ≥25 percent rated thermal power, the Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked in accordance with the Surveillance Frequency Control Program and the flow-referenced APRM scram and rod block signals shall be adjusted, if necessary, as specified by Specification 2.1.2a.

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	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(o)	≤ 1080 psig		(p)	x	x
(3)	High Drywell Pressure	2	2(o)	≤ 3.5 psig		x	(a)	(a)
(4)	Low Reactor Water Level	2	2(o)	≥ 53 inches (Indicator Scale)		x	x	x
(5)	High Water Level Scram Discharge Volume	2	2(o)	≤ 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	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)(o)	≤ 10 percent valve closure from full open		(c)	(c)	x
(7)	Deleted							
(8)	Shutdown Position of Reactor Mode Switch	2	1	---		(k)	x	x
(9)	Neutron Flux							
	(a) IRM							
	(i) Upscale	2	3(d)(o)	≤96 percent of full scale		x	x	

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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	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(ii) Inoperative	2	3(d)(o)	---		x	x	
(b) APRM							
(i) Upscale	2	3(e)(o)	Specification 2.1.2a			x	x
(ii) Inoperative	2	3(e)(o)	---			x	x
(10) Turbine Stop Valve Closure	2	4(o)	≤ 10% valve closure				(i)
(11) Generator Load Rejection	2	2(o)	(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	Note 1	None
(2)	High Reactor Pressure	None	Note 1[(l)]	Note 1[(l)]
(3)	High Drywell Pressure	None	Note 1[(l)]	Note 1[(l)]
(4)	Low Reactor Water Level	Note 1	Note 1[(l)]	Note 1[(l)]
(5)	High Water Level Scram Discharge Volume	None	Note 1	Note 1
(6)	Main-Steam-Line Isolation Valve Position	None	Note 1	Note 1
(7)	Deleted			

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	Note 1[f]	Note 1[g]	Note 1[n]
(ii) Inoperative	Note 1[f]	Note 1[g]	Note 1[n]
(b) APRM			
(i) Upscale	None	Note 1	Note 1[m] Note 1[n]
(ii) Inoperative	None	Note 1	None
(10) Turbine Stop Valve Closure	None	Note 1	Note 1
(11) Generator Load Rejection	None	Note 1	Note1

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, or for the purpose of performing reactor coolant system pressure testing and/or control rod scram time testing with the reactor mode switch in the refuel position.

(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 Traversing 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) Verify SRM/IRM channels overlap during startup after the mode switch has been placed in startup. Verify IRM/APRM channels overlap at least 1/2 decade during entry into startup from run (normal shutdown) if not performed within the previous 7 days.

(g) Within 24 hours before startup, if not performed within the previous 7 days. Not required to be performed during shutdown until 12 hours after entering startup from run.

(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) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(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. Verify the calculated power does not exceed the APRM channels by greater than 2% of rated thermal power. Any APRM channel gain adjustment made in compliance with Specification 2.1.2a shall not be included in determining the difference.

(n) Neutron detectors are excluded.

(o) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same trip system is monitoring that parameter.

With one channel required by Table 3.6.2a inoperable in one or more Parameters, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours or in accordance with the Risk Informed Completion Time Program (see Note 2).

With two or more channels required by Table 3.6.2a inoperable in one or more Parameters:

1. Within one hour, verify sufficient channels remain Operable or tripped* to maintain trip capability for the Parameter, and

2. Within 6 hours or in accordance with the Risk Informed Completion Time Program (see Note 2), place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and

3. Within 12 hours or in accordance with the Risk Informed Completion Time Program (see Note 2), restore the inoperable channels in the other trip system to an Operable status or tripped*.

Otherwise, take the ACTION required by Specification 3.6.2a for that Parameter.

* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to Operable status within the required time, the ACTION required by Specification 3.6.2a for the parameter shall be taken.

** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

(p) May be bypassed during reactor coolant system pressure testing and/or control rod scram time testing.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2a
Note 2: Risk Informed Completion Time can only be applied when trip capability is maintained.

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	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
PRIMARY COOLANT ISOLATION							
(Main Steam, Cleanup, and Shutdown Cooling)							
(1) Low-Low Reactor Water Level							
(a) Main Steam and Cleanup	2	2(f)	≥ 5 inches (Indicator Scale)	(k)		x	x
(b) Shutdown Cooling	2	2(f)	≥ 5 inches (Indicator Scale)	(k)		x	x
(2) Manual	2	1	---	(k)		x	x
MAIN-STEAM-LINE ISOLATION							
(3) High Steam Flow Main-Steam Line	2	2(f)	≤ 105 psid			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	Set Point	Reactor Mode Switch Position In Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(4)	Deleted							
(5)	Low Reactor Pressure	2	2(f)	≥ 850 psig			(h)	x
(6)	Low-Low-Low Condenser Vacuum	2	2(f)	≥ 7 in. mercury vacuum			(a)	x
(7)	High Temperature Main Steam Line Tunnel	2	2(f)	≤ 200°F			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	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(g)	≤ 190°F	(i)		x	x
SHUTDOWN COOLING SYSTEM ISOLATION							
(9) High Area Temperature	1	1	≤ 170°F	(i)		x	x
CONTAINMENT ISOLATION							
(10) Low-Low Reactor Water	2	2(f)	≥ 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	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(11) High Drywell Pressure	2	2(f)	≤ 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 Cooling)			
(1) Low-Low Reactor Water Level	Note 1	Note 1[(d)]	Note 1[(d)]
(2) Manual	---	Note 1	---
MAIN-STEAM-LINE ISOLATION			
(3) High Steam Flow Main-Steam Line	Note 1	Note 1[(d)]	Note 1[(d)]
(4) Deleted			
(5) Low Reactor Pressure	Note 1	Note 1[(d)]	Note 1[(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	Note 1	Note 1
(7) High Temperature Main-Steam-Line Tunnel	None	Note 1	Note 1
CLEANUP SYSTEM ISOLATION			
(8) High Area Temperature	Note 1	Note 1	Note 1
SHUTDOWN COOLING SYSTEM ISOLATION			
(9) High Area Temperature	Note 1	Note 1	Note 1

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	Note 1	Note 1[d]	Note 1[d]
(11)	High Drywell Pressure	Note 1	Note 1[d]	Note 1[d]
(12)	Manual	---	Note 1	---

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) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(e) Deleted.

(f) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable Channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within

    a. 12 hours for Parameters common to SCRAM Instrumentation or in accordance with the Risk Informed Completion Time Program (see Note 2), and
    b. 24 hours for Parameters not common to SCRAM Instrumentation or in accordance with the Risk Informed Completion Time Program (see Note 2).

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable Channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour.

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within

        (1) 12 hours for Parameters common to SCRAM Instrumentation, and
        (2) 24 hours for Parameters not common to SCRAM Instrumentation.

    or

    b. take the ACTION required by Specification 3.6.2a for that Parameter.

(g) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

(h) Only applicable during startup mode while operating in IRM range 10.

(i) May be bypassed in the cold shutdown condition.

(j) Deleted.

(k) The Primary Coolant Isolation Parameters for Cleanup and Shutdown Cooling in Table 3.6.2b are only applicable in the Shutdown Condition - Hot. See Table 3.6.2m for Parameter applicability in the Shutdown Condition - Cold.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2b.

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(e)	≤ 1080 psig	(b)		x	x
(2) Low-Low Reactor Water Level	2	2(e)	≥ 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)(f)	≤ 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	Note 1[(c)]	Note 1[(c)]
(2) Low-Low Reactor Water Level	Note 1	Note 1[(c)]	Note 1[(c)]
EMERGENCY COOLING ISOLATION (for each of two systems)			
(3) High Steam Flow Emergency Cooling System	None	Note 1[(c)]	Note 1[(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) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

(e) With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program (see Note 2), or take the action required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

(f) With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program (see Note 2).

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 24 hours.

   or

   b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2c.
Note 2: Risk Informed Completion Time can only be applied when the trip capability is maintained.

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 (f)	Set Point	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)(g)	(h)	(a)	(a)
(2)	Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)	(b)(g)	(h)	x	x
OPEN CORE SPRAY DISCHARGE VALVES								
(3)	Reactor Pressure and either (1) or (2) above.	2	2	≥ 365 psig	(g)	(h)	x	x

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	Note 1	Note 1[(c)]	Note 1[(c)]
(2)	Low-Low Reactor Water Level	Note 1	Note 1[(c)]	Note 1[(c)]
OPEN CORE SPRAY DISCHARGE VALVES				
(3)	Reactor Pressure and either (1) or (2) above	None	Note 1[(c)]	Note 1[(c)]

(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) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(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) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program (see Note 2), or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

(g) The Parameters for Start Core Spray Pumps and Open Core Spray Discharge Valves in Table 3.6.2d are only applicable in the Shutdown Condition - Hot. See Table 3.6.2m for Parameter applicability in the Shutdown Condition - Cold.

(h) The Parameters are required when Reactor Coolant Temperature is greater than 212°F.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2d.
Note 2: Risk Informed Completion Time can only be applied when trip capability is maintained.

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	Note 1	Note 1(b)	Note 1(b)
	b.	Low-Low Reactor Water Level	Note 1	Note 1(b)	Note 1(b)

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) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(c) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip system in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program (see Note 2), or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2e.
Note 2: Risk Informed Completion Time can only be applied when trip capability is maintained.

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: Shutdown	Refuel	Startup	Run
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	Note 1(c)	Note 1(c)
and			
b. High Drywell Pressure	Note 1	Note 1(c)	Note 1(c)

(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) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(d) A channel may be placed in an inoperable status for up to 6 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.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program (see Note 2), or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2f.
Note 2: Risk Informed Completion Time can only be applied when trip capability is maintained.

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 (i)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(1) Deleted							
(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 (i)	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
b. Upscale (Biased by Recirculation Flow)	2(h)	3(c)	Specification 2.1.2a(h)			x	x
c. Downscale	2(h)	3(c)	≥ [5.28/125] divisions of full scale				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
(4)	Deleted							
(5)	Refuel Platform and Hoists	2(f)	1	---		x		
(6)	Mode Switch in Shutdown	1	1	---	x			
(7)	Mode Switch in Refuel (Blocks withdrawal of more than 1 rod)	1	1	---		x		
(8)	Deleted							

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TABLE 4.6.2g

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

	Parameter	**Sensor Check**	**Instrument Channel Test**	**Instrument Channel Calibration**
(1)	Deleted			
(2)	IRM			
	a. Detector not in Startup Position	N/A	Note 1[(g)]	N/A
	b. Inoperative	N/A	Note 1[(g)]	N/A
	c. Downscale	N/A	Note 1(g)	Note 1[(j)]
	d. Upscale	N/A	Note 1[(g)]	Note 1[(j)]
(3)	APRM			
	a. Inoperative	None	Note 1	None
	b. Upscale (Biased by Recirculation Flow)	None	Note 1	Note 1[(j)]
	c. Downscale	None	Note 1	Note 1[(j)]
(4)	Deleted			

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	---	Note 1	---
(7)	Mode Switch in Refuel (Blocks withdrawal of more than 1 rod)	---	Note 1	---
(8)	Deleted			

NOTES FOR TABLES 3.6.2G AND 4.6.2G

(a) Deleted

(b) No more than one of the four IRM inputs to each instrument channel shall be bypassed. These signals may be bypassed when the APRMs 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) Deleted

(e) Deleted

(f) One sensor provides input to each of two instrument channels. Each instrument channel is in a separate trip system.

(g) Within 24 hours before startup, if not performed within the previous 7 days. Not required to be performed during shutdown until 12 hours after entering startup from run.

(h) The actuation of either or both trip systems will result in a rod block.

(i) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Parameter.

(j) Neutron detectors are excluded.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2g.

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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			
				Shutdown	Refuel	Startup	Run
MECHANICAL VACUUM PUMP							
High Radiation Main Steam Line	2	2	≤ 5 times normal 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	Note 1	Note 1	Note 1

NOTES FOR TABLES 3.6.2h and 4.6.2h

(a) Deleted.

(b) A channel may be placed in an inoperable status for up to 6 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.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 12 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour.

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 12 hours.

   or

   b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2h.

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 (cont'd)

DIESEL GENERATOR INITIATION

Limiting Condition for Operation

Parameter	Set-Point: Relay Dropout	Set-Point: Operating Time
Loss of Power		
a. 4.16kV PB 102/103 Emergency Bus Undervolt (Loss of Voltage)	≥3200 volts	0 volts ≤3.2 seconds[(a)]
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	≥3705 volts	>3.4 seconds[(b)] ≤24 seconds[(c)]

(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.

(b) The operating time indicated in the table is the minimum time required to clear voltage transients due to load sequencing to avoid spurious separation from offsite power.

(c) The operating time indicated in the table is the maximum time allowable to preclude load damage or trip device actuation at voltages below the degraded voltage setpoint of 3705 volts.

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	Note 1	Note 1
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	NA	Note 1	Note 1

(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.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2i.

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(d)	≤ 5mr/hr	x	(a)	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	Note 1	Note 1	Note 1
(2)	High Radiation Refueling Platform	Note 1	(c)	Note 1

(a) This function shall be operable whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building.

(b) Deleted.

(c) Immediately prior to when function is required and in accordance with the Surveillance Frequency Control Program thereafter until function is no longer required.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1) Place the inoperable channel(s) in the tripped condition within 24 hours.

or

2) Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2j.

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	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(1)	Low Reactor Water Level	2	2(c)	≥ 53 inches (Indicator Scale)	(a)		(a)	x
(2)	Automatic Turbine Trip	1	1	---			(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	Note 1[(a)]	Note 1[(a)(b)]	Note 1[(a)(b)]
(2)	Automatic Turbine Trip	None	Note 1[(a)]	None

(a) Not required in the Shutdown Condition – Cold. 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) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(c) A channel may be placed in an inoperable status for up to 6 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.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program (see Note 2), or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2k.
Note 2: Risk Informed Completion Time can only be applied when trip capability is maintained.

TABLE 3.6.2I

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)	Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)	(c)		x	x
(2)	High Steam Flow Main-Steam Line	2	2	≤ 105 psid			x	x
(3)	High Temperature Main-Steam Line Tunnel	2	2	≤ 200°F			x	x
(4)	High Drywell Pressure	2	2	≤ 3.5 psig	(c)		(a)	(a)

TABLE 4.6.2I

CONTROL ROOM AIR TREATMENT SYSTEM INITIATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low-Low Reactor Water Level	Note 1	Note 1[(b)]	Note 1[(b)]
(2)	High Steam Flow Main-Steam Line	Note 1	Note 1[(b)]	Note 1[(b)]
(3)	High Temperature Main-Steam Line Tunnel	---	Note 1	Note 1
(4)	High Drywell Pressure	Note 1	Note 1[(b)]	Note 1[(b)]

NOTES FOR TABLES 3.6.2I AND 4.6.2I

(a) May be bypassed when necessary for containment inerting.

(b) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(c) May be bypassed in the cold shutdown condition.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2I.

TABLE 3.6.2m

RPV WATER INVENTORY CONTROL INSTRUMENTATION

Limiting Condition for Operation

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TABLE 3.6.2m

RPV WATER INVENTORY CONTROL INSTRUMENTATION

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
PRIMARY COOLANT ISOLATION							
(1) Low-Low Reactor Water Level							
(a) Cleanup	2	1(c)	≥ 5 inches (Indicator Scale)	(a) (b)	(a) (b)		
(b) Shutdown Cooling	2	1(c)	≥ 5 inches (Indicator Scale)	(a) (b)	(a) (b)		

TABLE 4.6.2m

RPV WATER INVENTORY CONTROL INSTRUMENTATION

Surveillance Requirement

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TABLE 4.6.2m

RPV WATER INVENTORY CONTROL INSTRUMENTATION

Surveillance Requirement

Parameter	**Sensor Check**	**Instrument Channel Test**	**Instrument Channel Calibration**
PRIMARY COOLANT ISOLATION (Cleanup and Shutdown Cooling)			
(2) Low-Low Reactor Water Level	Note 1	Note 1	---

NOTES FOR TABLES 3.6.2m AND 4.6.2m

(a) The Parameters in this table are only applicable in the Shutdown Condition – Cold and Refuel. See Table 3.6.2b or Table 3.6.2d for Parameter applicability in the Shutdown Condition – Hot.

(b) Applicable when automatic isolation of the associated penetration flow path(s) is credited in calculating drain time.

(c) With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement, immediately either

1. Place the inoperable channel(s) in the trip condition,

or

2. a. Declare associated penetration flow path(s) incapable of automatic isolation,

   and

   b. Initiate action to calculate drain time,

   and

   c. Take the Action required by Specification 3.6.2a for that parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2m.

BASES FOR 3.6.2 AND 4.6.2 PROTECTIVE INSTRUMENTATION

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 initiated rod block.

BASES FOR 3.6.2 AND 4.6.2 PROTECTIVE INSTRUMENTATION

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 ≤ 11.5 psi differential. This represents a flow of approximately 9.8×10^5 lb/hr at rated conditions.

The automatic initiation signals for the emergency cooling systems have to be sustained for more than 12 seconds to cause opening of the return valves. If the signals last for less than 12 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-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 to provide 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

The APRM scram and rod block setpoints have been derived based on GE setpoint methodology as outlined in NEDC-31336, "GE Instrumentation Setpoint Methodology." In this methodology, the setpoints are defined as three values, Nominal Trip Setpoints, Allowable Values, and Analytical Limits. The analytical limits are listed in Specification 2.1.2a. The allowable values are listed below:

The minimum of:

<u>For W ≥ 0%</u>:

$S \leq (0.55W + 64.46\%)$ T with a maximum value of 119.5%
$S_{RB} \leq (0.55W + 59.46\%)$ T with a maximum value of 114.5%

AND:

<u>For 14.42% ≤ W ≤ 45%</u>:

$S \leq (1.287W + 16.6\%)$
$S_{RB} \leq (1.287W + 9.312\%)$

WHERE:

S or S_{RB} = The respective scram or rod block allowable value
W = Loop Recirculation Flow as a percentage of the loop recirculation flow which produces a rated core flow of 67.5 MLB/HR
T = FRTP/CMFLPD (T is applied only if less than or equal to 1.0)
FRTP = Fraction of Rated Thermal Power where Rated Thermal Power equals 1850 MW
CMFLPD = Core Maximum Fraction of Limiting Power Density

IRM, ± 2.5% of rated neutron flux

The APRM downscale rod block setpoint has been derived based on GE setpoint methodology as outlined in NEDC-31336, "GE Instrumentation Setpoint Methodology." In this methodology, the setpoint is defined as three values, Nominal Trip Setpoint, Allowable Value, and Analytical Limit. Table 3.6.2g shows the nominal trip setpoints. The corresponding allowable value is as follows:

APRM Downscale Rod Block, allowable value is ≥[4.24/125] divisions of full scale

Recirculation Flow Upscale, ±1.6% of rated recirculation flow (analytical limit is 107.1% of rated flow)
Recirculation Flow Comparator, ±2.09% of rated recirculation flow (analytical limit is 10% flow differential)

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, ±10°F

High Area Temperature-Clean-up and Shutdown, ±6°F

High Radiation-Main Steam Line, +100% and -50% of set point value

High Radiation-Reactor Building Vent, +100% and -50% of set point

High Radiation-Refueling Platform, +100% and -50% of set point

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," and MDE-77-0485, "Technical Specification Improvement Analysis for Nine Mile Point Nuclear Station, Unit 1."

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A Suppl2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," and with NEDC-31677P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation." 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 and isolation valve position scram.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2 and RE-003, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Nine Mile Point Nuclear Station, Unit 1."

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

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.

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. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A Suppl 1, "Technical Specification Improvement Analyses for BWR Control Rod Block Instrumentation," and with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

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).

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 or in accordance with the Risk Informed Completion Time Program.

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. In accordance with the Surveillance Frequency Control Program - test for automatic startup and pickup of load required for a loss-of-coolant accident.

b. In accordance with the Surveillance Frequency Control Program - 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	SURVEILLANCE REQUIREMENT
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 14 days or in accordance with the Risk Informed Completion Time Program. 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 or in accordance with the Risk Informed Completion Time Program.	c. In accordance with the Surveillance Frequency Control Program – determine the cell voltage and specific gravity of the pilot cells of each battery.
d. If a reserve power transformer becomes inoperable, it shall be returned to service within seven days or in accordance with the Risk Informed Completion Time Program.	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. 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.	e. Surveillance for operation with an inoperable diesel-generator – If a diesel-generator becomes inoperable from any cause other than an inoperable support system or preplanned maintenance or testing, within 8 hours, either determine that the cause of the diesel-generator being inoperable does not impact the operability of the operable diesel-generator or demonstrate operability by testing the operable diesel-generator. Operability by testing will be demonstrated by achieving steady state voltage and frequency.

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 or in accordance with the Risk Informed Completion Time Program, 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 or in accordance with the Risk Informed Completion Time Program.	

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 a LOCA. Engineering calculations show that a LOCA concurrent with a loss of offsite power and the single failure of one of the diesel-generators results in a loading for the remaining diesel-generator that is below the unit's 2000 hour/year rating. 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. The major equipment comprising the maximum diesel-generator loading is given in Figure IX-6*.

As mentioned above, a single diesel-generator is capable of providing the required power to equipment following a LOCA. 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 Section XV.B.3.23* 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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.4 Deleted	4.6.4 Deleted

Pages 261 through 264 Deleted

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 inspecton frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval is based on the number of unacceptable snubbers found during the previous inspection in proportion to the population of the various snubber types and categories. The inspection schedule is based on the guidance provided in Generic Letter 90-09. 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.

LIMITING CONDITION FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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 in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3. A complete inventory of radioactive by-product materials, exceeding the limits set forth in 10 CFR 30.71, in sealed sources in possession shall be maintained current at all times.	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.

BASES FOR 3.6.5 AND 4.6.5 RADIOACTIVE MATERIAL SOURCES

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.

LIMITING CONDITION FOR OPERATION

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.

SURVEILLANCE REQUIREMENT

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 the frequency specified in the Surveillance Frequency Control Program unless otherwise noted 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)	Deleted			
2)	Deleted			
3)	Reactor Vessel Water Level	2	1*	2
4)	Drywell Pressure Monitor	2	1	4
5)	Suppression Chamber Water Level	2	1*	4
6)	Deleted			
7)	Containment High Range Radiation Monitor	2	1	3
8)	Suppression Chamber Water Temperature	2	1	2

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance provided at least one Operable channel is monitoring that Parameter.

TABLE 3.6.11-2

ACCIDENT MONITORING INSTRUMENTATION ACTION STATEMENTS

ACTION - 1 Deleted

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.

TABLE 3.6.11-2 (cont'd)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

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 REQUIREMENTS

	Parameter	Instrument Channel Test	Instrument Channel Calibration
(1)	Deleted		
(2)	Deleted		
(3)	Reactor vessel water level	Note 1	Note 1
(4)	Drywell Pressure Monitor	Note 1	Note 1
(5)	Suppression Chamber Water Level Monitor	Note 1	Note 1
(6)	Deleted		
(7)	Containment High Range Radiation Monitor	Note 1	Note 1
(8)	Suppression Chamber Water Temperature	Note 1	Note 1

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program.

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.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

LIMITING CONDITION FOR OPERATION

3.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

Applicability:

Applies to the operability of instrumentation that provides protection of the reactor protection system and reactor trip system.

Objective:

To assure the operability of the instrumentation monitoring the 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 power supply.

SURVEILLANCE REQUIREMENT

4.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

Applicability:

Applies to the surveillance of instrumentation that provides protection of the reactor protection system and reactor trip system.

Objective:

To verify the operability of protection instrumentation monitoring the power to the reactor protection and reactor trip buses.

Specification:

a. In accordance with the Surveillance Frequency Control Program –
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 power conditions by applying from a test source, an overvoltage signal, an undervoltage signal and an under-frequency signal to verify that the tripping logic up to but not including the output contactors functions properly.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. With one protective relaying system inoperable, restore the inoperable system to an operable status within 72 hours or remove the power supply from service. c. With both protective relaying systems inoperable, restore at least one to an operable status within 30 minutes or remove the power supply from service.	b. <u>In accordance with the Surveillance Frequency Control Program</u> 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 power 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 ≤132 volts, ≤4 seconds ii. Undervoltage ≥108 volts, ≤4 seconds iii. Underfrequency ≥ 57 hertz, ≤2 seconds

BASES FOR 3.6.12 AND 4.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

To eliminate the potential for undetectable single component failure which could adversely affect the operability of the reactor protection system and reactor trip system, protective relaying schemes are installed on Motor Generator Sets 131 and 141, Static Uninterruptible Power Supply Systems 162 and 172, and maintenance bus 130A. This provides for overvoltage, undervoltage and underfrequency protection.

LIMITING CONDITION FOR OPERATION

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, the remote shutdown panels' Functions in Table 3.6.13-1 shall be operable.

SURVEILLANCE REQUIREMENT

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 at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.13-1.

b. In accordance with the Surveillance Frequency Control Program

   1. Each remote shutdown panel shall be demonstrated to initiate the emergency condensers independent of the main/auxiliary control room.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. With the valve control Function inoperable, restore the required Function to operable status within 30 days.	2. Each remote shutdown panel shall be demonstrated to open both the motor-operated steam valves.
c. With one or more required monitoring instrument Functions inoperable, restore the required Function to operable status within 30 days or establish an alternate method of monitoring the parameter within 30 days and restore the required Function to operable status within 90 days.	
d. If the required action and associated completion time is not met, be in hot shutdown within the next 12 hours.	

TABLE 3.6.13-1

REMOTE SHUTDOWN PANELS FUNCTIONS

Limiting Condition for Operation

FUNCTION	MINIMUM NUMBER OF OPERABLE CHANNELS PER FUNCTION
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
Emergency Condenser Condensate Return Valve and Motor-Operated Steam Supply Valves Control on the Same Panel	1

TABLE 4.6.13-1

REMOTE SHUTDOWN PANEL MONITORING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Calibration
Reactor Pressure	Note 1	Note 1[(a)]
Reactor Water Level	Note 1	Note 1[(a)]
Reactor Water Temperature	Note 1	Note 1
Torus Water Temperature	Note 1	Note 1
Drywell Pressure	Note 1	Note 1[(a)]
Emergency Condenser Water Level	Note 1	Note 1
Drywell Temperature	Note 1	Note 1
"All Rods In" Light	Note 1	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.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.13-1.

BASES FOR 3.6.13 AND 4.6.13 REMOTE SHUTDOWN PANELS

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 channel of each Function provides the necessary capabilities consistent with 10CFR50 Appendix R. Therefore, only one channel of either remote shutdown panel monitoring instrument or control 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.

Pages 282 Through 294 Deleted

LIMITING CONDITION FOR OPERATION

3.6.15 MAIN CONDENSER OFFGAS

Applicability:

Applies to the radioactive effluents from the main condenser.

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:

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 μCi/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.

SURVEILLANCE REQUIREMENT

4.6.15 MAIN CONDENSER OFFGAS

Applicability:

Applies to the periodic test and recording requirements of main condenser offgas.

Objective:

To ascertain that radioactive effluents from the main condenser are within allowable values of 10CFR20, Appendix B and 10CFR50, Appendix I.

Specification:

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:

In accordance with the Surveillance Frequency Control Program.

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.

BASES FOR 3.6.15 AND 4.6.15 MAIN CONDENSER OFFGAS

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.

Pages 297 Through 338 Deleted

LIMITING CONDITION 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 REQUIREMENT

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 in accordance with the Surveillance Frequency Control Program 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 CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) The continuous rod withdrawal control shall not be used during out-of-sequence movement of the control rods.	(3) No core alterations are in progress.
(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.	

BASES FOR 3.7.1 AND 4.7.1 SHUTDOWN MARGIN DEMONSTRATION

The shutdown margin demonstration may 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.

This special test exception provides the appropriate additional controls to allow the shutdown margin demonstration to be performed in the cold shutdown condition with the vessel head in place. Compliance with this special test exception is optional and applies only if 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

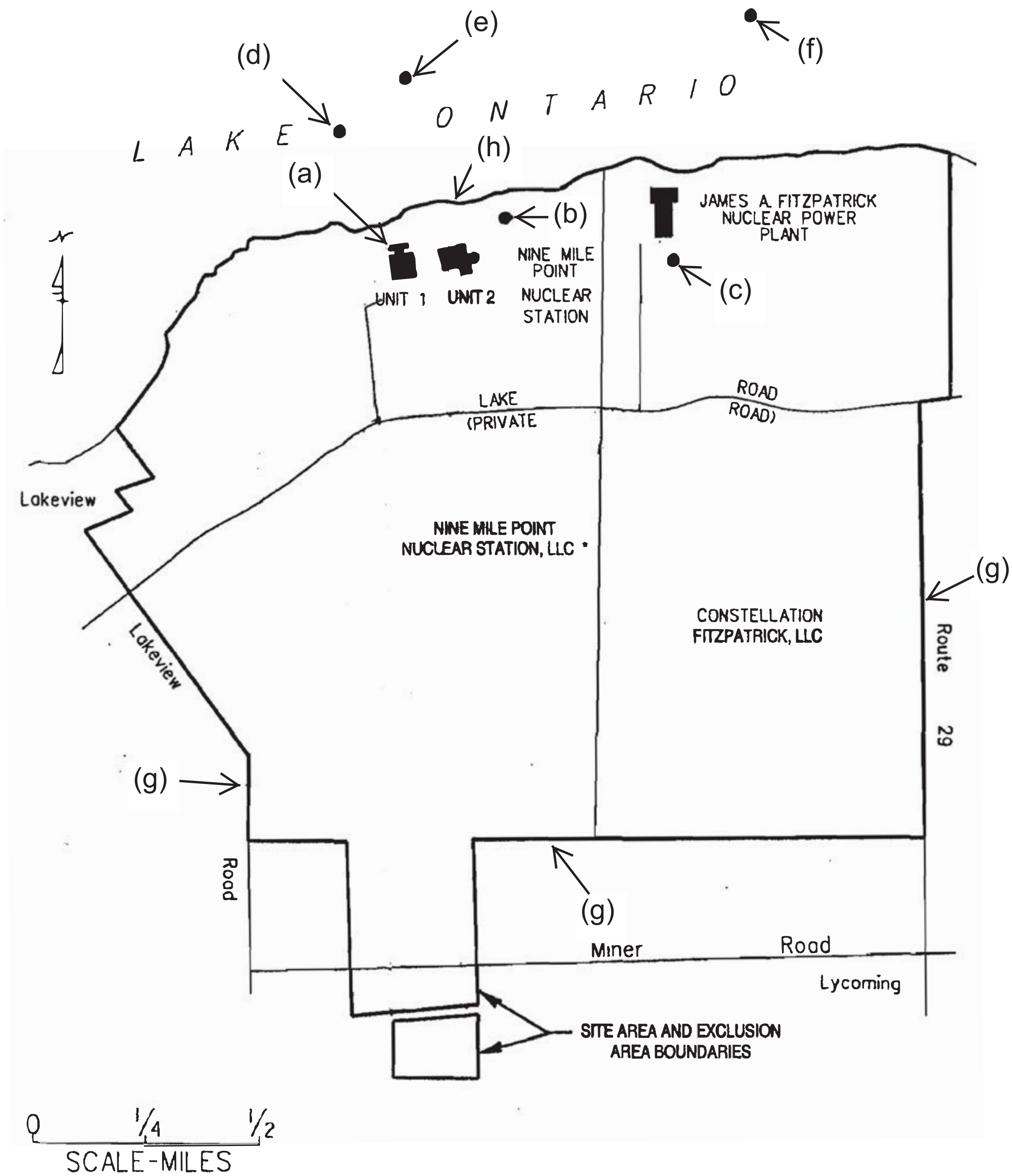
The Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant site comprising approximately 1500 acres, is located on the shores of Lake Ontario, about seven miles northeast of Oswego, New York. An exclusion distance of nearly 4000 feet is provided between the Station and the nearest 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 (Deleted)



* Niagara Mohawk Power Corporation retains ownership in certain transmission line and switchyard facilities within the exclusion area boundary. Access and usage are controlled by Nine Mile Point Nuclear Station, LLC by Agreement.

FIGURE 5.1-1
SITE BOUNDARIES
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) 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 (Deleted)

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 north and south half of the spent fuel pool is analyzed to store 1840 and 2246 spent fuel assemblies, respectively, in storage racks containing the neutron absorber material Boral. The spent fuel pool is analyzed to store a total of 306 spent fuel assemblies in storage racks containing Boraflex. Both types of storage racks will maintain a k_{eff} of less than 0.95 under normal, abnormal and accident conditions. The spent fuel stored in both types of storage racks must have a peak lattice enrichment of 4.6% or less and the k-inf in the standard cold core geometry must be less than or equal to 1.31.

5.6 (Deleted)

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.

The plant manager or a designee shall approve, prior to implementation, each proposed test and experiment not addressed in the UFSAR or Technical Specifications, and each modification to systems or equipment that affect nuclear safety.

6.1.2 The Station Shift Supervisor – Nuclear (SSS) shall be responsible for the control room command function. During any absence of the SSS from the control room while the unit is in the power operating or hot shutdown conditions, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SSS from the control room while the unit is in the cold shutdown or refueling conditions, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 Organization

6.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. The organization chart and the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR. The functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions shall be documented in procedures.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out radiation protection, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Unit Staff

The unit staff organization shall include the following:

a. At least two non-licensed operators shall be assigned when the unit is in the power operating condition; and at least one non-licensed operator shall be assigned when the unit is in the hot shutdown, cold shutdown, or refueling conditions. In addition, if the process computer is out of service for greater than 8 hours, at least three non-licensed operators shall be assigned when the unit is in the power operating, hot shutdown, cold shutdown, or refueling conditions.

b. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specification 6.2.2.a for a period of time not to exceed 2 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.

c. An individual qualified to implement radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence of on-duty personnel, provided immediate action is taken to fill the required position.

d. Deleted

e. As a minimum, either the Manager Operations or the General Supervisor Operations shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the shift supervision in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

6.3 Unit Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the Constellation Energy Generation, LLC Quality Assurance Topical Report.

6.4 Procedures

6.4.1 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 cover the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 3, 1972;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for radioactive effluent and radiological environmental monitoring; and

d. Deleted

e. All programs specified in Specification 6.5.

6.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

6.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 6.6.2 and Specification 6.6.3.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

(a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

(b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager or a designee; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

6.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, Containment Spray, Emergency Cooling, Shutdown Cooling, Reactor Cleanup, Vacuum Relief, Reactor Water Sampling, Containment Atmosphere Dilution (CAD) H_2O_2 Monitor, Drywell Containment Atmosphere Monitoring (CAM), Post Accident Sampling, Radioactive Gaseous Effluent Monitoring (RAGEMS) (the program requirements shall apply to the Post Accident Sampling System and RAGEMS until such time as administrative controls provide for continuous isolation of the associated penetration(s) or a modification eliminates the potential leakage path(s)), Offgas Effluent Stack Monitoring (OGESMS), and Post Accident Vent to Reactor Building Emergency Ventilation. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. System leak test requirements for each system at 24 month intervals.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the 24 month frequency for performing system leak test activities.

6.5.3 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate ≤500 mrems/yr to the whole body and a dose rate ≤3000 mrems/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days: a dose rate ≤1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary; conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives >8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and

k. Limitations on venting and purging of the primary containment through the Emergency Ventilation System to maintain releases as low as reasonably achievable.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequencies.

6.5.4 DELETED

6.5.5 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in all outside temporary liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is ≤10 Ci, excluding tritium and dissolved or entrained noble gases.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

6.5.6 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to the Bases without prior NRC approval provided the changes do not involve either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of 6.5.6.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.5.7 <u>10 CFR 50 Appendix J Testing Program Plan</u>

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

   1. Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel Topical Report BN-TOP-1, and

   2. The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.

   3. Exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Section 9.2.3: The first Type A test performed after the June 8, 1999 Type A test shall be performed no later than June 8, 2014.

b. The peak calculated containment internal pressure (Pac) for the design basis loss of coolant accident is 35 psig.

c. The maximum allowable primary containment leakage rate (La) at Pac shall be 1.5% of primary containment air weight per day.

d. Leakage Rate Surveillance Test acceptance criteria are:

   1. The as-found Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than 1.0 L_a.

   2. The as-left Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than or equal to 0.75 L_a, prior to entering a mode of operation where containment integrity is required.

   3. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

   4. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a minimum pathway basis, at all times when containment integrity is required.

e. The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the 10 CFR 50 Appendix J Testing Program Plan.

The provisions of Specification 4.0.3 are applicable to the 10 CFR 50 Appendix J Testing Program Plan.

6.5.8 <u>Control Room Envelope Habitability Program</u>

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Air Treatment (CRAT) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation of the CRAT System, operating at a flow rate of 2025-2475 cfm, at a Frequency of 24 months. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of TS 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

6.5.9 Surveillance Frequency Control Program

This program provides controls for the Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of the Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Controlled Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequency," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

6.5.10 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines."

The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in the Power Operating Condition;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

    1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

    2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

    3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

   1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

   2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. A RICT calculation must include the follow hazard groups: internal flood and internal events using a PRA model, internal fires using a PRA model, seismic hazards using penalty factors, and configuration-specific straight wind and tornado wind pressure / tornado missile hazards using penalty factors. Changes to these means of assessing the hazard groups require prior NRC approval.

f. The PRA models used to calculate a RICT shall be maintained and upgraded in accordance with the processes endorsed in the regulatory positions of Regulatory Guide 1.200, Revision 3, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities.”

g. A report shall be submitted in accordance with Specification 6.6.8 before a newly developed method is used to calculate a RICT.

6.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

6.6.1 Deleted

6.6.2 Annual Radiological Environmental Operating Report*

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses 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 ODCM, 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 in a supplementary report as soon as possible.

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* A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

6.6.3 Radioactive Effluent Release Report*

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.6.4 Deleted

6.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specifications 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 Specifications 3.1.7.c and 3.1.7.e.

4. The LINEAR HEAT GENERATION RATE for Specification 3.1.7.b.

5. The Power/Flow relationship for Specifications 3.1.7.d and 3.1.7.e.

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* A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

b. 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," U.S. Supplement, (NRC approved version specified in the COLR).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as shutdown margin (SDM), transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.6.6 Special Reports

Special reports shall be submitted 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. – f. (Deleted)

g. Sealed Source Leakage In Excess Of Limits, Specification 3.6.5.2 (Three months).

h. Accident Monitoring Instrumentation Report, Specification 3.6.11.a (Table 3.6.11-2, Action 3 or 4) (Within 14 days following the event).

6.6.7 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and inservice leakage and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Condition for Operation Section 3.2.1, "Reactor Vessel Heatup and Cooldown Rates."

   2. Limiting Condition for Operation Section 3.2.2, "Minimum Reactor Vessel Temperature for Pressurization."

   3. Surveillance Requirement Section 4.2.2, "Minimum Reactor Vessel Temperature for Pressurization."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

   1. SIR-05-044-A, “Pressure-Temperature Limits Report Methodology for Boiling Water Reactors,” Revision 0, April 2007.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

6.6.8 <u>Risk Informed Completion Time (RICT) Program Upgrade Report</u>

A report describing newly developed methods and their implementation must be submitted following a probabilistic risk assessment (PRA) upgrade associated with newly developed methods and prior to the first use of those methods to calculate a RICT. The report shall include:

a. The PRA models upgraded to include newly developed methods;

b. A description of the acceptability of the newly developed methods consistent with Section 5.2 of PWROG-19027-NP, Revision 2, “Newly Developed Method Requirements and Peer Review;”

c. Any open findings from the peer-review of the implementation of the newly developed methods and how those findings were dispositioned; and

d. All changes to key assumptions related to newly developed methods or their implementations.

6.7 High Radiation Area

Pursuant to 10 CFR Part 20, paragraph 20.1601(c), in lieu of the requirements of paragraph 20.1601(a) and 20.1601(b) of 10 CFR Part 20:

6.7.1 Access to each high radiation area, as defined in 10 CFR 20, in which an individual could receive a deep dose equivalent > 0.1 rem in one hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation) shall be controlled as described below to prevent unauthorized entry.

a. Each area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Entrance shall be controlled by requiring issuance of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rate in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may, for the performance of their assigned duties In high radiation areas, be exempt from the preceding requirements for issuance of an RWP or equivalent provided they are otherwise following plant radiation protection procedures for entry into, exit from, and work in such high radiation areas.

d. Each individual or group of individuals permitted to enter such areas shall possess, or be accompanied by, one or more of the following:

1. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset setpoint is reached. Entry into high radiation areas with this monitoring device may be made after the dose rate in the area has been determined and personnel have been made knowledgeable of it.

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area.

4. An individual qualified in radiation protection procedures equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by radiation protection supervision.

6.7.2 In addition to the requirements of Specification 6.7.1, high radiation areas in which an individual could receive a deep dose equivalent > 1.0 rem in one hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) shall be provided with a locked or continuously guarded door, or gate, or equivalent to prevent unauthorized entry.

a. The keys to such locked doors or gates, or equivalent, shall be administratively controlled in accordance with a program approved by the radiation protection manager.

b. Doors and gates, or equivalent, shall remain locked except during periods of access by personnel under an approved RWP, or equivalent, to ensure individuals are informed of the dose rate in the immediate work areas prior to entry.

c. Individual high radiation areas in which an individual could receive a deep dose equivalent > 1.0 rem in one hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), accessible to personnel, that are located within larger areas where no enclosure exists to enable locking, or that are not continuously guarded, and where no lockable enclosure can be reasonably constructed around the individual area require both of the following access controls:

   1. Each area shall be barricaded and conspicuously posted.

   2. A flashing light shall be activated as a warning device.

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APPENDIX "B"

TO

NINE MILE POINT, UNIT 1

HAS BEEN DELETED PER

AMENDMENT NO. 66
(NOVEMBER 2, 1984)