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

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-85

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for renewed license filed by Exelon Generation Company, LLC* complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Limerick Generating Station, Unit 2 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-107 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. Constellation Energy Generation, LLC (the licensee) is technically qualified to engage in the activities authorized by this renewed license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

---

* The Commission approved a transaction on November 16, 2021, that resulted in Exelon Generation Company, LLC being renamed Constellation Energy Generation, LLC.

G. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Renewed Facility Operating License No. NPF-85, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

J. 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.

2. Based on the foregoing findings and the Decision of the Atomic Safety and Licensing Board, LBP-85-25, dated July 22, 1985, the Commission's Order dated July 7, 1989, and the Commission's Memorandum and Order dated August 25, 1989, regarding this facility, Renewed Facility Operating License NPF-85 is hereby issued to Constellation Energy Generation, LLC (the licensee), to read as follows:

A. This renewed license applies to the Limerick Generating Station, Unit 2, a boiling water nuclear reactor and associated equipment, owned by Constellation Energy Generation, LLC. The facility is located on the licensee's site in Montgomery and Chester Counties, Pennsylvania on the banks of the Schuylkill River approximately 1.7 miles southeast of the city limits of Pottstown, Pennsylvania and 21 miles northwest of the city limits of Philadelphia, Pennsylvania, and is described in the licensee's Final Safety Analysis Report, as supplemented and amended, and in the licensee's Environmental Report-Operating License Stage, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Constellation Energy Generation, LLC:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in Montgomery and Chester Counties, Pennsylvania, in accordance with the procedures and limitations set forth in this renewed license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and to 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) 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) Pursuant to the Act and 10 CFR Parts 30, 40, 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 calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and 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 subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Constellation Energy Generation, LLC is authorized to operate the facility at reactor core power levels of 3515 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 224, are hereby incorporated into this renewed license. Constellation Energy Generation, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection (Section 9.5, SSER-2, -4)*

Constellation Energy Generation, LLC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report dated August 1983 through Supplement 9, dated August 1989, and Safety Evaluation dated November 20, 1995, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Security and Safeguards

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 provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Limerick Generating Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2." The set contains Safeguards Information protected under 10 CFR 73.21.

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 CSP was approved by License Amendment No. 166 and modified by License Amendment No. 180.

(5) 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.

* The parenthetical notation following the title of license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

(6) Deleted.

(7) Deleted.

(8) Deleted.

(9) Mitigation Strategy License Condition

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

(10) The licensee 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.

(11) Upon implementation of Amendment No. 149 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.2.2.a, in accordance with TS 6.16.c.(i), the assessment of CRE habitability as required by Specification 6.16.c.(ii), and the measurement of CRE pressure as required by Specification 6.16.d, shall be considered met. Following implementation:

(a) The first performance of SR 4.7.2.2.a, in accordance with Specification 6.16.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from September 16, 2004, the date of the most recent successful tracer gas test, as stated in the December 10, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 6.16.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from September 16, 2004, the date of the most recent successful tracer gas test, as stated in the December 10, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 6.16.d, shall be within 24 months, plus the 180 days allowed by SR 4.0.2, as measured from September 16, 2004, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

(12) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), is henceforth part of the UFSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities described in the UFSAR supplement, without prior Commission approval, provided 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.

(13) The licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as revised in accordance with license condition 2.C.(12), describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

(a) Constellation Energy Generation, LLC shall implement those new programs and enhancements to existing programs no later than December 22, 2028.

(b) Constellation Energy Generation, LLC shall complete those activities designated for completion prior to the PEO, as noted in Commitment Nos. 18, 19, 20, 22, 23, 24, 28, 29, 30, 38, 39, 40, 41, 42, 43, and 47, of Appendix A of NUREG-2171, "Safety Evaluation Report Related to the License Renewal of Limerick Generating Station, Units 1 and 2," no later than December 22, 2028, or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.

(c) Constellation Energy Generation, LLC shall notify the NRC in writing within 30 days after having accomplished item (a) above and include the status of those activities that have been or remain to be completed in item (b) above.

(14) The Additional Conditions contained in Appendix C, as revised through Amendment No. 223, are hereby incorporated into this renewed license. Constellation Energy Generation, LLC shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) exemption from the requirement of Appendix J, the testing of containment air locks at times when the containment integrity is not required (Section 6.2.6.1 of the SER and SSER-3), (b) exemption from the requirements of Appendix J, the leak rate testing of the Main Steam Isolation Valves (MSIVs) at the peak calculated containment pressure, Pa, and exemption from the requirements of Appendix J that the measured MSIV leak rates be included in the summation for the local leak rate test (Section 6.2.6.1 of SSER-3), (c) exemption from the requirement of Appendix J, the local leak rate testing of the Traversing Incore Probe Shear Valves (Section 6.2.6.1 of the SER and SSER-3), and (d) an exemption from the schedule requirements of 10 CFR 50.33(k)(l) related to availability of funds for decommissioning the facility (Section 22.1, SSER 8). The special circumstances regarding exemptions (a), (b) and (c) are identified in Sections 6.2.6.1 of the SER and SSER 3. An exemption from the criticality monitoring requirements of 10 CFR 70.24 was previously granted with NRC materials license No. SNM-1977 issued November 22, 1988. The licensee is hereby exempted from the requirements of 10 CFR 70.24 insofar as this requirement applies to the handling and storage of fuel assemblies held under this renewed license.

E. Deleted

F. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

G. This renewed license is effective as of the date of issuance and shall expire at midnight on June 22, 2049.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulation

Enclosures:

1. Appendix A - Technical Specifications
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional Conditions

Date of Issuance: October 20, 2014

NUREG-1376

Technical Specifications
Limerick Generating Station, Unit No. 2

Docket No. 50-353

Appendix "A" to
License No. NPF-85

Issued by the
U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

August 1989

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for OPERABILITY and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by 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.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY of all devices in the channel required for OPERABILITY.

The CHANNEL FUNCTIONAL TEST may be performed by 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.

CORE ALTERATION

1.7 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a) Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special moveable detectors (including undervessel replacement); and
b) Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

1.7a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides the 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 Specifications 6.9.1.9 thru 6.9.12. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the (GEXL) correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same inhalation committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation committed effective dose equivalent (CEDE) conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1989, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE.

DOWNSCALE TRIP SETPOINT (DTSP)

1.9a The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

DRAIN TIME

1.9b The DRAIN TIME is the time it would take for the water inventory in and above the Reactor Pressure Vessel (RPV) to drain to the top of the active fuel (TAF) seated in the RPV assuming:

a) The water inventory above the TAF 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

DRAIN TIME (Continued)

susceptible to a common mode failure, for all penetration flow paths below the TAF except:

1. Penetration flow paths connected to an intact closed system, or isolated by manual or automatic valves that are closed and administratively 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 the TAF 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 TAF 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.

1.10 (Deleted)

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.11 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation set-point at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.12 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

This total system response time consists of two components, the instrumentation response time and the breaker arc suppression time. These times may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

1.13 (Deleted)
1.14 (Deleted)

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

HIGH (POWER) TRIP SETPOINT (HTSP)

1.15a The high power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable above 85% reactor thermal power.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of the leakage detection systems.

INSERVICE TESTING PROGRAM

1.16a The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

INTERMEDIATE (POWER) TRIP SETPOINT (ITSP)

1.16b The intermediate power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 65% and 85% reactor thermal power.

ISOLATION SYSTEM RESPONSE TIME

1.17 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.18 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.19 LINEAR HEAT GENERATION RATE (LHGR) shall be 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 unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.20 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practical up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

LOW (POWER) TRIP SETPOINT (LTSP)

1.20a The low power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 30% and 65% reactor thermal power.

1.21 (Deleted)

MEMBER(S) OF THE PUBLIC

1.22 MEMBER OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

MAPFAC(F)-(MAPLHGR FLOW FACTOR)

1.22a A core flow dependent multiplication factor used to flow bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MAPFAC(P)-(POWER DEPENDENT MAPLHGR MULTIPLIER)

1.22b A core power dependent multiplication factor used to power bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MINIMUM CRITICAL POWER RATIO (MCPR)

1.23 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core (for each class of fuel). Associated with the minimum critical power ratio is a core flow dependent (MCPR(F)) and core power dependent (MCPR(P)) minimum critical power ratio.

OFFSITE DOSE CALCULATION MANUAL

1.24 The OFFSITE DOSE CALCULATION MANUAL (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/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

OPERABLE - OPERABILITY

1.25 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, 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).

OPERATIONAL CONDITION - CONDITION

1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a fault in a reactor coolant system component body, pipe wall or vessel wall. Leakage past seals, packing, and gaskets is not PRESSURE BOUNDARY LEAKAGE.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.28a The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.13.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the solidification or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With solidification or dewatering, the PCP shall identify the process parameters influencing solidification or dewatering based on laboratory scale and full scale testing or experience.

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3515 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

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

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a, except as indicated by the footnote for Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

RECENTLY IRRADIATED FUEL

1.35 RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.2.

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

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

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a, except as indicated by the footnote for Specification 4.6.5.1.2a.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

RESTRICTED AREA

1.37a RESTRICTED AREA means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. RESTRICTED AREA does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a RESTRICTED AREA.

1.38 (Deleted)

SHUTDOWN MARGIN (SDM)

1.39 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.

SITE BOUNDARY

1.40 The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.42 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.43 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.43A The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required position. The response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.44 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.45 UNRESTRICTED AREA means an area, access to which is neither limited nor controlled by the licensee.

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
E	At least once per 18 months (550 days).
R (Refueling Interval)	At least once per 24 months (731 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown# ***	> 200°F
4. COLD SHUTDOWN	Shutdown# ## ***	≤ 200°F ****
5. REFUELING*	Shutdown or Refuel** #	NA

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided that the one-rod-out interlock is OPERABLE.

****See Special Test Exception 3.10.8.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATION CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4, and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable* and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

*The APRM Simulated Thermal Power - Upscale Functional Unit need not be declared inoperable upon entering single reactor recirculation loop operation provided that the flow-biased setpoints are adjusted within 6 hours per Specification 3.4.1.1.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale (Setdown)	≤ 15.0% of RATED THERMAL POWER	≤ 20.0% of RATED THERMAL POWER
b. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ 0.65 W + 61.7% and ≤ 116.6% of RATED THERMAL POWER	≤ 0.65 W + 62.2% and ≤ 117.0% of RATED THERMAL POWER
- Single Recirculation Loop Operation***	≤ 0.65 (W-7.6%) + 61.5% and ≤ 116.6% of RATED THERMAL POWER	≤ 0.65 (W-7.6%) + 62.0% and ≤ 117.0% of RATED THERMAL POWER
c. Neutron Flux - Upscale	118.3% of RATED THERMAL POWER	118.7% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
e. 2-Out-Of-4 Voter	N.A.	N.A.
f. OPRM Upscale	****	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1096 psig	≤ 1103 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation
b. Float Switch	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation

* See Bases Figure B 3/4.3-1.

** Equivalent to 25.58 gallons/scram discharge volume.

*** The 7.6% flow "offset" for Single Loop Operation (SLO) is applied for W ≥ 7.6%. For flows W < 7.6%, the (W-7.6%) term is set equal to zero.

**** See COLR for OPRM period based detection algorithm trip setpoints. OPRM Upscale trip output auto-enable (not bypassed) setpoints shall be APRM Simulated Thermal Power ≥ 29.5% and recirculation drive flow < 60%.

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein, except as provided in Specification 3.0.8. Upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specifications 3.0.5, 3.0.6, and 3.0.9.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in Specifications 3.0.5 and 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

a. At least STARTUP within the next 6 hours,
b. At least HOT SHUTDOWN within the following 6 hours, and
c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITION 4 or 5.

3.0.4 When a Limiting Condition for Operation is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made:

a. When the associated ACTION requirements to be entered permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements or that are part of a shutdown of the unit.

LIMITING CONDITION FOR OPERATION (Continued)

3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONs may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to the second premise of Specification 3.0.1 and is an exception to Specification 3.0.2 (i.e., to not comply with the applicable ACTION(s)) for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0.6 When a supported system Limiting Condition for Operation is not met solely due to a support system Limiting Condition for Operation not being met, the ACTIONs associated with this supported system are not required to be entered. Only the support system Limiting Condition for Operation ACTIONs are required to be entered. This is an exception to the second premise of Specification 3.0.1 and is an exception to Specification 3.0.2 (i.e., to not comply with the applicable ACTION(s)) for the supported system. In this event, an evaluation shall be performed in accordance with Specification 6.17, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate ACTIONs of the Limiting Condition for Operation in which the loss of safety function exists are required to be entered.

When a support system's ACTION directs a supported system to be declared inoperable or directs entry into ACTIONs for a supported system, the applicable ACTIONs shall be entered in accordance with Specification 3.0.1.

3.0.7 Not Used

3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported Limiting Condition(s) for Operation 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 Limiting Condition(s) for Operation shall be declared not met.

LIMITING CONDITION FOR OPERATION (Continued)

3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system Limiting Conditions for Operation 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.

For the purposes of this specification, the High Pressure Coolant Injection system, the Reactor Core Isolation Cooling system, and the Automatic Depressurization System are considered independent subsystems of a single system.

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 Limiting Conditions for Operation shall be declared not met.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. 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 Limiting Condition for Operation. Failure to perform a Surveillance within the specified Surveillance time interval and allowed extension per Specification 4.0.2, shall be failure to meet the Limiting Condition for Operation 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 Surveillance Requirement shall be performed within the specified surveillance time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified Surveillance time interval and allowed extension per Specification 4.0.2, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance time interval, 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 Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of a Limiting Condition for Operation shall only be made when the Limiting Condition for Operation's Surveillance Requirements have been met within their Surveillance time interval, except as provided in Specification 4.0.3. When a Limiting Condition for Operation is not met due to its Surveillance Requirements not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements or that are part of a shutdown of the unit.

4.0.5 Inservice Inspection and Inservice Testing Program

Structures, systems, and components (SSCs) within the INSERVICE TESTING PROGRAM shall be tested in accordance with the requirements of 10 CFR 50.55a(f). SSCs within the Inservice Inspection Program shall be inspected in accordance with the requirements of 10 CFR 50.55a(g). The provisions of SR 4.0.2 and SR 4.0.3 do not apply to the INSERVICE TESTING PROGRAM unless there is a specific SR referencing usage of the program.

SURVEILLANCE REQUIREMENTS (Continued)

SSCs that have been categorized as Risk-Informed Safety Class (RISC) of RISC-3 in accordance with 10 CFR 50.69, and removed from the INSERVICE TESTING PROGRAM or Inservice Inspection Program in accordance with 10 CFR 50.69(b)(1)(v), are subject to the alternative treatment requirements specified in 10 CFR 50.69(d)(2). The SSCs must continue to meet the acceptance criteria specified in the applicable technical specification surveillance requirements; however, the surveillance frequency is determined as part of the alternative treatment.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity difference between the actual core k_{eff} and the predicted core k_{eff} shall not exceed 1% $\Delta k/k$.

APPLICABILITY: OPERATIONAL CONDITION 1 and 2.

ACTION:

With the reactivity difference exceeding 1% $\Delta k/k$:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity difference between the actual core k_{eff} and the predicted core k_{eff} shall be verified to be less than or equal to 1% $\Delta k/k$:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

c. The provisions of Specification 4.0.4 are not applicable.

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods and scram discharge volume vent and drain valves shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3***

ACTION:

a. With one withdrawn control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Within 1 hour:

      a) Verify that the inoperable withdrawn control rod is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.

      b) Disarm the associated directional control valves** either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   2. Restore the inoperable withdrawn control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

   1. If the inoperable control rod(s) is withdrawn, within 1 hour:

      a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

      b) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*.

---

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

***OPERATIONAL CONDITION 3 is only applicable to the scram discharge volume vent and drain valves.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves** either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

2. If the inoperable control rod(s) is inserted, within 1 hour disarm the associated directional control valves** either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d. With one or more scram discharge volume (SDV) vent or drain lines with one valve inoperable, restore the inoperable valve(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours*** and in COLD SHUTDOWN within the following 24 hours.

e. With one or more SDV vent or drain lines with both valves inoperable, isolate the associated line within 8 hours **** or be in at least HOT SHUTDOWN within the next 12 hours*** and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying each valve to be open,* and

b. Cycling each valve through at least one complete cycle of full travel.

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

** May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

***Separate Action entry is allowed for each SDV vent and drain line.

****An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

4.1.3.1.2 When above the preset power level of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. In accordance with the Surveillance Frequency Control Program, and

b. Within 24 hours from discovery that a control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6, and 4.1.3.7.

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE in accordance with the Surveillance Frequency Control Program, by verifying that the drain and vent valves:

   1. Close within 30 seconds after receipt of a signal for control rods to scram, and

   2. Open when the scram signal is reset.

b. Proper level sensor response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation in accordance with the Surveillance Frequency Control Program.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER with reactor coolant pressure greater than or equal to 950 psig, following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.

b. For 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 in accordance with either "1" or "2" as follows:

   1.a Specifically affected individual control rods shall be scram time tested at zero reactor coolant pressure and the scram insertion time from the fully withdrawn position to notch position 05 shall not exceed 2.0 seconds, and

   1.b Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure prior to exceeding 40% of RATED THERMAL POWER.

   2. Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure.

c. For at least 10% of the control rods, with reactor coolant pressure greater than or equal to 950 psig, on a rotating basis, and in accordance with the Surveillance Frequency Control Program.

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

REACTIVITY CONTROL SYSTEMS

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
5	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

a. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one control rod scram accumulator inoperable, within 8 hours:

   a) Restore the inoperable accumulator to OPERABLE status, or
   b) Declare the control rod associated with the inoperable accumulator inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

   a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by verifying that control rod charging water header pressure is ≥1400 psig or by inserting at least one withdrawn control rod at least one notch: If no control rod drive pump is operating and:

      1) If reactor pressure is ≥900 psig, then restart at least one control rod drive pump within 20 minutes or place the reactor mode switch in the shutdown position, or

      2) If reactor pressure is <900 psig, then place the reactor mode switch in the Shutdown position.

   b) Insert the inoperable control rods and disarm the associated control valves either:

      1) Electrically, or
      2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within one hour, either:

   a) Electrically, or
   b) Hydraulically by closing the drive water and exhaust water isolation valves.

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

2. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the indicated pressure is greater than or equal to 955 psig unless the control rod is inserted and disarmed or scrammed.

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 and 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

1. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

a) Observing any indicated response of the nuclear instrumentation, and

b) Demonstrating that the control rod will not go to the overtravel position.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. If recoupling is not accomplished on the first attempt or, if not permitted by the RWM, then until permitted by the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours either:

1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.3.6 Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within 1 hour:

   1. Determine the position of the control rod by using an alternate method, or:

      a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,

      b) Returning the control rod, by single notch movement, to its original position, and

      c) Verifying no control rod drift alarm at least once per 12 hours, or

   2. Move the control rod to a position with an OPERABLE position indicator, or

   3. When THERMAL POWER is:

      a) Within the preset power level of the RWM, declare the control rod inoperable.

      b) Greater than the preset power level of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

---

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

a. In accordance with the Surveillance Frequency Control Program that the position of each control rod is indicated,

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

REACTIVITY CONTROL SYSTEMS

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*, **, when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER.

ACTION:

a. With the RWM inoperable after the first 12 control rods are fully withdrawn, operation may continue provided that control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or technically qualified member of the unit technical staff.

b. With the RWM inoperable before the first 12 control rods are fully withdrawn, one startup per calendar year may be performed provided that control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or technically qualified member of the unit technical staff.

c. Otherwise, with the RWM inoperable, control rod movement shall not be permitted except by full scram.***

*See Special Test Exception 3.10.2.

**Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

***Control rods may be moved, under administrative control, to permit testing associated with demonstrating OPERABILITY of the RWM.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 1 hour after RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

b. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

c. In OPERATIONAL CONDITION 1 within 1 hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

d. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

3.1.4.2 Deleted.

4.1.4.2 Deleted.

REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than the limit specified in the CORE OPERATING LIMITS REPORT (COLR), or THERMAL POWER greater than or equal to 90% of rated with MCPR less than the limit specified in the COLR.

ACTION:

a. With one RBM channel inoperable:

   1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

   2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, a minimum of one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis and calculation* that the available weight of Boron-10 is greater than or equal to 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ \text{wt.}} \times \frac{E}{29\ \text{atom}\ \%} \times \frac{Q}{86\ \text{gpm}} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

4. Verifying that no more than two pumps are aligned for automatic operation.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 37.0 gpm per pump at a pressure of greater than or equal to 1230 ± 25 psig is met.

d. In accordance with the Surveillance Frequency Control Program by:

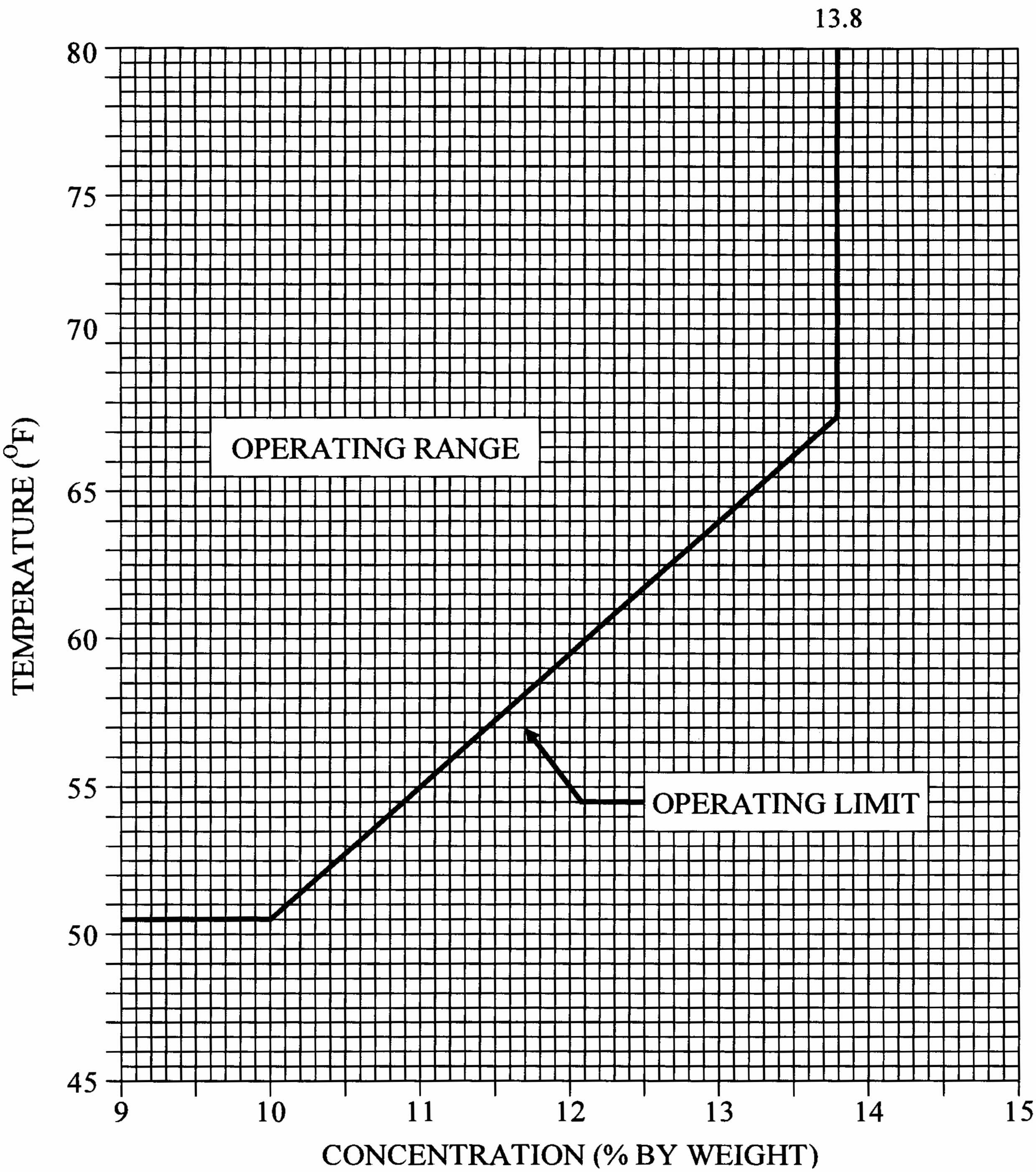
1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 49 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.



SODIUM PENTABORATE SOLUTION
TEMPERATURE/CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-1

(THIS PAGE LEFT BLANK INTENTIONALLY)

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. 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) as shown in the CORE OPERATING LIMITS REPORT (COLR). During operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the appropriate reduction factors for power and flow as defined in the COLR.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limiting value, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limiting value:

a. In accordance with the Surveillance Frequency Control Program,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

Figures on pages
3/4 2-2 thru 3/4 2-6
have been removed from Technical
Specifications, and relocated to the
CORE OPERATING LIMITS REPORT.

Technical Specifications pages
3/4 2-3 thru 3/4 2-6a
have been INTENTIONALLY OMITTED.

Section 3/4.2.2 (DELETED)

INFORMATION CONTAINED ON

THIS PAGE HAS BEEN

DELETED

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the rated MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2 and the main turbine bypass system is OPERABLE per Specification 3.7.8, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.672 + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{1/2} (0.016),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) EOC-RPT inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

c. With the main turbine bypass system inoperable per Specification 3.7.8, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) main turbine bypass valve inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2a and during reactor startups prior to control rod scram time tests in accordance with Specification 4.1.3.2.b.1.b, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit including application of the MCPR(P) and MCPR(F) factors as determined from the CORE OPERATING LIMITS REPORT.

a. In accordance with the Surveillance Frequency Control Program,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

Figures on pages
3/4 2-10 thru 3/4 2-11
have been removed from
Technical Specifications, and
relocated to the CORE OPERATING
LIMITS REPORT.

Technical Specifications pages
3/4 2-10a and 3/4 2-11
have been INTENTIONALLY OMITTED.

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the value in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

a. In accordance with the Surveillance Frequency Control Program,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

Note: Separate condition entry is allowed for each channel.

Note: When Functional Unit 2.b and 2.c channels are inoperable due the calculated power exceeding the APRM output by more than 2% of RATED THERMAL POWER while operating at ≥ 25% of RATED THERMAL POWER, entry into the associated Actions may be delayed up to 2 hours.

a. With the number of OPERABLE channels in either trip system for one or more Functional Units less than the Minimum OPERABLE Channels per Trip System required by Table 3.3.1-1, within one hour or in accordance with the Risk Informed Completion Time Program*** for each affected functional unit either verify that at least one* channel in each trip system is OPERABLE or tripped or that the trip system is tripped, or place either the affected trip system or at least one inoperable channel in the affected trip system in the tripped condition.

b. With the number of OPERABLE channels in either trip system less than the Minimum OPERABLE Channels per Trip System required by Table 3.3.1-1, place either the inoperable channel(s) or the affected trip system** in the tripped condition within 12 hours or in accordance with the Risk Informed Completion Time Program***.

c. With the number of OPERABLE channels in both trip systems for one or more Functional Units less than the Minimum OPERABLE Channels per Trip System required by Table 3.3.1-1, place either the inoperable channel(s) in one trip system or one trip system in the tripped condition within 6 hours** or in accordance with the Risk Informed Completion Time Program***.

d. If within the allowable time allocated by Actions a, b or c, it is not desired to place the inoperable channel or trip system in trip (e.g., full scram would occur), Then no later than expiration of that allowable time initiate the action identified in Table 3.3.1-1 for the applicable Functional Unit.

---

* For Functional Units 2.a, 2.b, 2.c, 2.d, and 2.f, at least two channels shall be OPERABLE or tripped. For Functional Unit 5, both trip systems shall have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or tripped. For Function 9, at least three channels per trip system shall be OPERABLE or tripped.

** For Functional Units 2.a, 2.b, 2.c, 2.d, and 2.f, inoperable channels shall be placed in the tripped condition to comply with Action b. Action c does not apply for these Functional Units.

*** Not applicable when trip capability is not maintained for one or more Functional Units.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.1.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program, except Table 4.3.1.1-1 Functions 2.a, 2.b, 2.c, 2.d, 2.e, and 2.f. Functions 2.a, 2.b, 2.c, 2.d, and 2.f do not require separate LOGIC SYSTEM FUNCTIONAL TESTS. For Function 2.e, tests shall be performed in accordance with the Surveillance Frequency Control Program. LOGIC SYSTEM FUNCTIONAL TEST for Function 2.e includes simulating APRM and OPRM trip conditions at the APRM channel inputs to the voter channel to check all combinations of two tripped inputs to the 2-Out-Of-4 voter logic in the voter channels.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific reactor trip system.

TABLE 1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors[b]:			
a. Neutron Flux - High	2 3(i), 4(i) 5(i)	3 3 3(d)	1 2 3
b. Inoperative	2 3(i), 4(i) 5(i)	3 3 3(d)	1 2 3
2. Average Power Range Monitor[e]:			
a. Neutron Flux - Upscale (Setdown)	2	3(m)	1
b. Simulated Thermal Power - Upscale	1	3(m)	4
c. Neutron Flux - Upscale	1	3(m)	4
d. Inoperative	1, 2	3(m)	1
e. 2-Out-Of-4 Voter	1, 2	2	1
f. OPRM Upscale	1(o)(p)	3(m)	10
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve-Closure	1(g)	1/valve	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. DELETED	DELETED	DELETED	DELETED
7. Drywell Pressure - High	1, 2(h)	2	1
8. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2 5(i)	2 2	1 3
b. Float Switch	1, 2 5(i)	2 2	1 3
9. Turbine Stop Valve - Closure	1(j)	4(k)	6
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1(j)	2(k)	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	2 2 2	1 7 3
12. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

Amendment No. 52
FEB 1 6 1995

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure until the function is automatically bypassed, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 10 - a. If the condition exists due to a common-mode OPRM deficiency*, then initiate alternate method to detect and suppress thermal-hydraulic instability oscillations within 12 hours AND restore required channels to OPERABLE status within 120 days,

OR

b. Reduce THERMAL POWER to < 25% RATED THERMAL POWER within 4 hours.

* Unanticipated characteristic of the instability detection algorithm or equipment that renders all OPRM channels inoperable at once.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) 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 OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall automatically be bypassed when the reactor mode switch is in the Run position.

(c) DELETED

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 6 IRMs.

(e) An APRM channel is inoperable if there are less than 3 LPRM inputs per level or less than 20 LPRM inputs to an APRM channel, or if more than 9 LPRM inputs to the APRM channel have been bypassed since the last APRM calibration (weekly gain calibration).

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 29.5% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) DELETED

(m) Each APRM channel provides inputs to both trip systems.

(n) DELETED

(o) With THERMAL POWER ≥ 25% RATED THERMAL POWER. The OPRM Upscale trip output shall be automatically enabled (not bypassed) when APRM Simulated Thermal Power is ≥ 29.5% and recirculation drive flow is < 60%. The OPRM trip output may be automatically bypassed when APRM Simulated Thermal Power is < 29.5% or recirculation drive flow is ≥ 60%.

(p) A minimum of 23 cells, each with a minimum of 2 OPERABLE LPRMs, must be OPERABLE for an OPRM channel to be OPERABLE.

TA .3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale (Setdown)	N.A.
b. Simulated Thermal Power - Upscale	N.A.
c. Neutron Flux - Upscale	N.A.
d. Inoperative	N.A.
e. 2-Out-Of-4 Voter	≤0.05*
f. OPRM Upscale	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤0.55
4. Reactor Vessel Water Level - Low, Level 3	≤1.05#
5. Main Steam Line Isolation Valve - Closure	≤0.06
6. DELETED	DELETED
7. Drywell Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Stop Valve - Closure	≤0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤0.08**
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

* Neutron detectors, APRM channel and 2-Out-Of-4 Voter channel digital electronics are exempt from response time testing. Response time shall be measured from activation of the 2-Out-Of-4 Voter output relay. For application of Specification 4.3.1.3, the redundant outputs from each 2-Out-Of-4 Voter channel are considered part of the same channel, but the OPRM and APRM outputs are considered to be separate channels, so N = 8. Testing of OPRM and APRM outputs shall alternate.

** Measured from start of turbine control valve fast closure.

\# Sensor is eliminated from response time testing for the RPS circuits. Response time testing and conformance to the administrative limits for the remaining channel including trip unit and relay logic are required.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (n)	CHANNEL FUNCTIONAL TEST (n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	(b)			2
		(j)		3(i), 4(i), 5(i)
b. Inoperative	N.A.	(j)	N.A.	2, 3(i), 4(i), 5(i)
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale (Setdown)	(b)	(l)		2
b. Simulated Thermal Power - Upscale		(e)	(d), (g), (o), (p)	1
c. Neutron Flux - Upscale			(d)	1
d. Inoperative	N.A.		N.A.	1, 2
e. 2-Out-Of-4 Voter			N.A.	1, 2
f. OPRM Upscale		(e)	(c)(g)	1(m)
3. Reactor Vessel Steam Dome Pressure - High				1, 2(h)
4. Reactor Vessel Water Level- Low, Level 3				1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.			1
6. DELETED				
7. Drywell Pressure - High				1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter				1, 2, 5(i)
b. Float Switch	N.A.			1, 2, 5(i)

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (n)	CHANNEL FUNCTIONAL TEST (n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.			1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.			1
11. Reactor Mode Switch Shutdown Position	N.A.		N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.		N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for a least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Calibration includes verification that the OPRM Upscale trip auto-enable (not-bypass) setpoint for APRM Simulated Thermal Power is ≥ 29.5% and for recirculation drive flow is < 60%.
(d) The more frequent calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Verify the calculated power does not exceed the APRM channels by greater than 2% of RATED THERMAL POWER.
(e) CHANNEL FUNCTIONAL TEST shall include the flow input function, excluding the flow transmitter.
(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).
(g) The less frequent calibration includes the flow input function.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) DELETED
(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.
(m) With THERMAL POWER ≥ 25% of RATED THERMAL POWER.
(n) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(o) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(p) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the Trip Setpoint are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the associated Technical Specifications Bases.

3/4.3.2. ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2.-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a) With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b) With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirements for one trip system:

1. If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours or in accordance with the Risk Informed Completion Time Program**#. If this cannot be accomplished, the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken, or the channel shall be placed in the tripped condition.

or

2. If placing the inoperable channel(s) in the tripped condition would not cause an isolation, the inoperable channel(s) and/or that trip system shall be placed in the tripped condition within:

    a) 12 hours or in accordance with the Risk Informed Completion Time Program**# for trip functions common* to RPS Instrumentation,

    b) 24 hours or in accordance with the Risk Informed Completion Time Program**# for trip functions not common* to RPS Instrumentation.

---

* Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1.

** Not applicable when trip capability is not maintained.

\# Not applicable for Function 7, Secondary Containment Isolation.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.2-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.2.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operations of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in accordance with the Surveillance Frequency Control Program, where N is the total number of redundant channels in a specific isolation trip system.

** The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low-Level 2	B	2	1, 2, 3	21
2) Low, Low, Low-Level 1	C	2	1, 2, 3	21
b. DELETED	DELETED	DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	P	2	1	22
d. Main Steam Line Flow - High	E	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	Q	2	1, 2**, 3**	21
f. Outboard MSIV Room Temperature - High	F(f)	2	1, 2, 3	21
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	F(f)	16	1, 2, 3	21
h. Manual Initiation	NA	2	1, 2, 3	24
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level Low - Level 3	A	2	1, 2, 3	23
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	V	2	1, 2, 3	23
c. Manual Initiation	NA	1	1, 2, 3	24

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCS Δ Flow - High	J	1	1, 2, 3	23
b. RWCS Area Temperature - High	J	6	1, 2, 3	23
c. RWCS Area Ventilation Δ Temperature - High	J	6	1, 2, 3	23
d. SLCS Initiation	Y(d)	NA	1, 2, 3	23
e. Reactor Vessel Water Level - Low, Low - Level 2	B	2	1, 2, 3	23
f. Manual Initiation	NA	1	1, 2, 3	24
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High	L	1	1, 2, 3	23
b. HPCI Steam Supply Pressure - Low	LA	2	1, 2, 3	23
c. HPCI Turbine Exhaust Diaphragm Pressure - High	L	2	1, 2, 3	23
d. HPCI Equipment Room Temperature - High	L	1	1, 2, 3	23
e. HPCI Equipment Room Δ Temperature - High	L	1	1, 2, 3	23

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
f. HPCI Pipe Routing Area Temperature - High	L	4	1, 2, 3	23
g. Manual Initiation	NA(e)	1/system	1, 2, 3	24
h. HPCI Steam Line Δ Press Timer	NA	1	1, 2, 3	23
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	K	1	1, 2, 3	23
b. RCIC Steam Supply Pressure - Low	KA	2	1, 2, 3	23
c. RCIC Turbine Exhaust Diaphragm Pressure - High	K	2	1, 2, 3	23
d. RCIC Equipment Room Temperature - High	K	1	1, 2, 3	23
e. RCIC Equipment Room Δ Temperature - High	K	1	1, 2, 3	23
f. RCIC Pipe Routing Area Temperature - High	K	4	1, 2, 3	23
g. Manual Initiation	NA(e)	1/system	1, 2, 3	24
h. RCIC Steam Line Δ Pressure Timer	NA	1	1, 2, 3	23

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL[a]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM [b]	APPLICABLE OPERATIONAL CONDITION	ACTION
6. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low - Level 2	B	2	1, 2, 3	20
2) Low, Low, Low - Level 1	C	2	1, 2, 3	20
b. Drywell Pressure - High	H	2	1, 2, 3	20
c. North Stack Effluent Radiation - High [g]	W	1	1, 2, 3	23
d. Deleted				
e. Reactor Enclosure Ventilation Exhaust Duct-Radiation - High	S	2	1, 2, 3	23
f. Deleted				
g. Deleted				
h. Drywell Pressure - High/ Reactor Pressure - Low	G	2/2	1, 2, 3	26
i. Primary Containment Instrument Gas Line to Drywell Δ Pressure - Low	M	1	1, 2, 3	26
j. Manual Initiation	NA	1	1, 2, 3	24

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL[(a),(c)]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM [(b)]	APPLICABLE OPERATIONAL CONDITION	ACTION
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	B	2	1, 2, 3	25
b. Drywell Pressure - High	H	2	1, 2, 3	25
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	R	2	*#	25
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	R	2	*#	25
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	2	1, 2, 3	25
e. Deleted				
f. Deleted				
g. Reactor Enclosure Manual Initiation	NA	1	1, 2, 3	27
h. Refueling Area Manual Initiation	NA	1	*	25

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated penetration flow path(s) isolated by use of one deactivated automatic valve secured in the isolated position, or one closed manual valve or blind flange*** within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - In OPERATIONAL CONDITION 1 or 2, verify the affected penetration flow path(s) are isolated by use of one deactivated automatic valve secured in the isolated position, or one closed manual valve or blind flange*** within 1 hour and declare the affected system inoperable. In OPERATIONAL CONDITION 3, be in at least COLD SHUTDOWN within 12 hours.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or isolate the affected penetration flow path(s) by use of one deactivated automatic valve secured in the isolated position, or one closed manual valve or blind flange*** within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Isolate the affected penetration flow path(s) by use of one deactivated automatic valve secured in the isolated position, or one closed manual valve or blind flange*** within 1 hour.

ACTION 27 - Restore the manual initiation function to OPERABLE status within 8 hours or establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

TABLE NOTATIONS

* Required when handling RECENTLY IRRADIATED FUEL in the secondary containment.

** May be bypassed under administrative control, with all turbine stop valves closed.

*** Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

(a) DELETED

(b) 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 OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

TABLE 3.3.2-1 (Continued)

TABLE NOTATIONS

(c) Actuates secondary containment isolation valves. Signal B, H, S, and R also start the standby gas treatment system.

(d) RWCU system inlet outboard isolation valve closes on SLCS "B" initiation. RWCU system inlet inboard isolation valve closes on SLCS "A" or SLCS "C" initiation.

(e) Manual initiation isolates the steam supply line outboard isolation valve and only following manual or automatic initiation of the system.

(f) In the event of a loss of ventilation the temperature - high setpoint may be raised by 50°F for a period not to exceed 30 minutes to permit restoration of the ventilation flow without a spurious trip. During the 30 minute period, an operator, or other qualified member of the technical staff, shall observe the temperature indications continuously, so that, in the event of rapid increases in temperature, the main steam lines shall be manually isolated.

(g) Wide range accident monitor per Specification 3.3.7.5.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ 840 psig	≥ 821 psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
3.	REACTOR WATER CLEANUP SYSTEM ISOLATION		
a.	RWCS Δ Flow - High	≤ 54.9 gpm	≤ 65.2 gpm
b.	RWCS Area Temperature - High	≤ 155°F or ≤ 120°F**	≤ 160°F or ≤ 125°F**
c.	RWCS Area Ventilation Δ Temperature - High	≤ 52°F or ≤ 32°F**	≤ 60°F or ≤ 40°F**
d.	SLCS Initiation	N.A.	N.A.
e.	Reactor Vessel Water Level - Low, Low, - Level 2	≥ -38 inches *	≥ -45 inches
f.	Manual Initiation	N.A.	N.A.
4.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a.	HPCI Steam Line Δ Pressure - High	≤ 974" H_2O	≤ 984" H_2O
b.	HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 90 psig
c.	HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d.	HPCI Equipment Room Temperature - High	180°F	≥ 177°F, ≤ 191°F
e.	HPCI Equipment Room Δ Temperature - High	≤ 104°F	≤ 108.5°F
f.	HPCI Pipe Routing Area Temperature - High	180°F	≥ 177°F, ≤ 191°F
g.	Manual Initiation	N.A.	N.A.
h.	HPCI Steam Line Δ Pressure - Timer	$3 \leq \tau \leq 12.5$ seconds	$2.5 \leq \tau \leq 13$ seconds

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
5.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a.	RCIC Steam Line Δ Pressure - High	≤ 373" H_2O	≤ 381" H_2O
b.	RCIC Steam Supply Pressure - Low	≥ 64.5 psig	≥ 56.5 psig
c.	RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10.0 psig	≤ 20.0 psig
d.	RCIC Equipment Room Temperature - High	180°F	≥ 161°F, ≤ 191°F
e.	RCIC Equipment Room Δ Temperature - High	≤ 109°F	≤ 113.5°F
f.	RCIC Pipe Routing Area Temperature - High	180°F	≥ 161°F, ≤ 191°F
g.	Manual Initiation	N.A.	N.A.
h.	RCIC Steam Line Δ Pressure Timer	3 ≤ τ ≤ 12.5 seconds	2.5 ≤ τ ≤ 13 seconds

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1. Low, Low - Level 2	≥ -38 inches*	≥ -45 inches
2. Low, Low, Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. North Stack Effluent Radiation - High	≤ 2.1 μCi/cc	≤ 4.0 μCi/cc
d. Deleted		
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High	≤ 1.35 mR/h	≤ 1.5 mR/h
f. Deleted		
g. Deleted		
h. Drywell Pressure - High/ Reactor Pressure - Low	≤ 1.68 psig/ ≥ 455 psig (decreasing)	≤ 1.88 psig/ ≥ 435 psig (decreasing)
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	≥ 2.0 psi	≥ 1.9 psi
j. Manual Initiation	N.A.	N.A.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
7.	SECONDARY CONTAINMENT ISOLATION		
a.	Reactor Vessel Water Level - Low, Low - Level 2	≥ -38 inches*	≥ -45 inches
b.	Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c.1.	Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	≤ 2.0 mR/h	≤ 2.2 mR/h
2.	Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	≤ 2.0 mR/h	≤ 2.2 mR/h
d.	Reactor Enclosure Ventilation Exhaust Duct Radiation - High	≤ 1.35 mR/h	≤ 1.5 mR/h
e.	Deleted		
f.	Deleted		
g.	Reactor Enclosure Manual Initiation	N.A.	N.A.
h.	Refueling Area Manual Initiation	N.A.	N.A.

* See Bases Figure B 3/4 3-1.

** The low setpoints are for the RWCU Heat Exchanger Rooms; the high setpoints are for the pump rooms.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Low - Level 2	N.A.
2) Low, Low, Low - Level 1	≤1.0###*
b. DELETED	DELETED
c. Main Steam Line Pressure - Low	≤1.0###*
d. Main Steam Line Flow - High	≤1.0###*
e. Condenser Vacuum - Low	N.A.
f. Outboard MSIV Room Temperature - High	N.A.
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
h. Manual Initiation	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low - Level 3	N.A.
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
c. Manual Initiation	N.A.
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCS Δ Flow - High	N.A.##
b. RWCS Area Temperature - High	N.A.
c. RWCS Area Ventilation Δ Temperature - High	N.A.
d. SLCS Initiation	N.A.
e. Reactor Vessel Water Level - Low, Low - Level 2	N.A.
f. Manual Initiation	N.A.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Line Δ Pressure - High	N.A.
b. HPCI Steam Supply Pressure - Low	N.A.
c. HPCI Turbine Exhaust Diaphragm Pressure - High	N.A.
d. HPCI Equipment Room Temperature - High	N.A.
e. HPCI Equipment Room Δ Temperature - High	N.A.
f. HPCI Pipe Routing Area Temperature - High	N.A.
g. Manual Initiation	N.A.
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure - High	N.A.
b. RCIC Steam Supply Pressure - Low	N.A.
c. RCIC Turbine Exhaust Diaphragm Pressure - High	N.A.
d. RCIC Equipment Room Temperature - High	N.A.
e. RCIC Equipment Room Δ Temperature - High	N.A.
f. RCIC Pipe Routing Area Temperature - High	N.A.
g. Manual Initiation	N.A.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Low - Level 2	N.A.
2) Low, Low, Low - Level 1	N.A.
b. Drywell Pressure - High	N.A.
c. North Stack Effluent Radiation - High	N.A.
d. Deleted	
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High	N.A.
f. Deleted	
g. Deleted	
h. Drywell Pressure - High/ Reactor Pressure - Low	N.A.
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.
j. Manual Initiation	N.A.
7. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level Low, Low - Level 2	N.A.
b. Drywell Pressure - High	N.A.
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	N.A.
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	N.A.
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	N.A.
e. Deleted	

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#
f.	Deleted	
g.	Reactor Enclosure Manual Initiation	N.A.
h.	Refueling Area Manual Initiation	N.A.

TABLE NOTATIONS

(a) DELETED

(b) DELETED

* Isolation system instrumentation response time for MSIV only. No diesel generator delays assumed for MSIVs.

** DELETED

\# Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to the isolation time for the valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

\#\# With 45 second time delay.

\#\#\# Sensor is eliminated from response time testing for the MSIV actuation logic circuits. Response time testing and conformance to the administrative limits for the remaining channel including trip unit and relay logic are required.

Amendment No. ~~52~~, ~~74~~, 93, 10
OCT 1 2 2000

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (a)	CHANNEL FUNCTIONAL TEST (a)	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low, Level 2				1, 2, 3
2) Low, Low, Low - Level 1				1, 2, 3
b. DELETED				DELETED
c. Main Steam Line Pressure - Low				1
d. Main Steam Line Flow - High				1, 2, 3
e. Condenser Vacuum - Low				1, 2**, 3**
f. Outboard MSIV Room Temperature - High				1, 2, 3
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High				1, 2, 3
h. Manual Initiation	N.A.		N.A.	1. 2, 3
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level## Low - Level 3				1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High				1, 2, 3
c. Manual Initiation	N.A.		N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (a)	CHANNEL FUNCTIONAL TEST (a)	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCS Δ Flow - High				1, 2, 3
b. RWCS Area Temperature - High				1, 2, 3
c. RWCS Area Ventilation Δ Temperature - High				1, 2, 3
d. SLCS Initiation	N.A.		N.A.	1, 2, 3
e. Reactor Vessel Water Level Low, Low, - Level 2				1, 2, 3
f. Manual Initiation	N.A.		N.A.	1, 2, 3
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High				1, 2, 3
b. HPCI Steam Supply Pressure, Low				1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High				1, 2, 3
d. HPCI Equipment Room Temperature - High				1, 2, 3
e. HPCI Equipment Room Δ Temperature - High				1, 2, 3
f. HPCI Pipe Routing Area Temperature - High				1, 2, 3
g. Manual Initiation	N.A.		N.A.	1, 2, 3
h. HPCI Steam Line Δ Pressure Timer	N.A.			1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (a)	CHANNEL FUNCTIONAL TEST (a)	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High				1, 2, 3
b. RCIC Steam Supply Pressure - Low				1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High				1, 2, 3
d. RCIC Equipment Room Temperature - High				1, 2, 3
e. RCIC Equipment Room Δ Temperature - High				1, 2, 3
f. RCIC Pipe Routing Area Temperature - High				1, 2, 3
g. Manual Initiation	N.A.		N.A.	1, 2, 3
h. RCIC Steam Line Δ Pressure Timer	N.A.			1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (a)	CHANNEL FUNCTIONAL TEST (a)	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low - Level 2				1, 2, 3
2) Low, Low, Low - Level 1				1, 2, 3
b. Drywell Pressure ## - High				1, 2, 3
c. North Stack Effluent Radiation - High				1, 2, 3
d. Deleted				
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High				1, 2, 3
f. Deleted				
g. Deleted				
h. Drywell Pressure - High/ Reactor Pressure - Low				1, 2, 3
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.			1, 2, 3
j. Manual Initiation	N.A.		N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK(a)	CHANNEL FUNCTIONAL TEST(a)	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2				1, 2, 3
b. Drywell Pressure## - High				1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High				*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High				*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High				1, 2, 3
e. Deleted				
f. Deleted				
g. Reactor Enclosure Manual Initiation	N.A.		N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.		N.A.	*

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

*Required when handling RECENTLY IRRADIATED FUEL in the secondary containment.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to Operable status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system subsystem inoperable, restore the inoperable trip system to OPERABLE status within:

   1. 7 days or in accordance with the Risk Informed Completion Time Program, provided that the HPCI and RCIC systems are OPERABLE.

   2. 72 hours or in accordance with the Risk Informed Completion Time Program.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.3.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific ECCS trip system.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM***			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2/pump[b]	1, 2, 3	30
b. Drywell Pressure - High	2/pump[b]	1, 2, 3,	30
c. Reactor Vessel Pressure - Low (Permissive)	6[b]	1, 2, 3	31
d. Manual Initiation	2[e]	1, 2, 3	33
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM***			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	2	1, 2, 3	31
d. Injection Valve Differential Pressure-Low (Permissive)	1/valve	1, 2, 3	31
e. Manual Initiation	1	1, 2, 3	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM##			
a. Reactor Vessel Water Level - Low Low, Level 2	4	1, 2, 3	34
b. Drywell Pressure - High###	4	1, 2, 3	34
c. Condensate Storage Tank Level - Low	2[c]	1, 2, 3	35
d. Suppression Pool Water Level - High	2	1, 2, 3	35
e. Reactor Vessel Water Level - High, Level 8	4[d]	1, 2, 3	31
f. Manual Initiation###	1/system	1, 2, 3	33

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM#***			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	2	1, 2, 3	31
e. RHR LPCI Mode Pump Discharge Pressure High (Permissive)	4	1, 2, 3	31
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
g. Manual Initiation	2	1, 2, 3	33
h. ADS Drywell Pressure Bypass Timer	2	1, 2, 3	31

TRIP FUNCTION	TOTAL NO. OF CHANNELS(f)	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	1/bus	1/bus	1/bus	1, 2, 3, 4**, 5**	36
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	1/source/ bus	1/source/ bus	1/source/ bus	1, 2, 3, 4**, 5**	37

***The Minimum OPERABLE Channels Per Trip Function is per subsystem.

TABLE 3.3.3-1 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
TABLE NOTATIONS

(a) 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 OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also provides input to actuation logic for the associated emergency diesel generators.

(c) One trip system. Provides signal to HPCI pump suction valves only.

(d) On 1 out of 2 taken twice logic, provides a signal to trip the HPCI pump turbine only.

(e) The manual initiation push buttons start the respective core spray pump and diesel generator. The "A" and "B" logic manual push buttons also actuate an initiation permissive in the injection valve opening logic.

(f) A channel as used here is defined as the 127 bus relay for Item 1 and the 127, 127Y, and 127Z feeder relays with their associated time delay relays taken together for Item 2.

* DELETED

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

** Required when ESF equipment is required to be OPERABLE.

\## Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

\### The injection functions of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with reactor steam dome pressure less than 550 psig.

TABLE 3.3.3-1 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
ACTION STATEMENTS

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. 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, or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ECCS inoperable within 24 hours.

ACTION 32 - DELETED

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program*, or declare the associated ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. 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, or declare the HPCI system inoperable.

b. With more than one channel inoperable, declare the HPCI system inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program*, or declare the HPCI system inoperable.

ACTION 36 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator and the associated offsite source breaker that is not supplying the bus inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

---

*Not applicable when trip capability is not maintained.

ACTION 37 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable device in the bypassed condition subject to the following conditions:

Inoperable Device	Condition
127-11X0X	127Y-11X0X and 127Z-11X0X operable
127Y-11X0X	127-11X0X and 127Z-11X0X operable
127Z-11X0X	127-11X0X and 127Y-11X0X operable. 127Z-11Y0Y operable for the other 3 breakers monitoring that source, offsite source grid voltage for that source is maintained at or above 230kV (for the 101 Safeguard Bus Source) or 525kV (for the 201 Safeguard Bus Source), Load Tap Changer for that source is in service and in automatic operation, and the electrical buses and breaker alignments are maintained within bounds of approved plant procedures.

or, place the inoperable channel in the tripped condition within 1 hour and take the Action required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

Operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Reactor Vessel Pressure - Low	$\geq$ 455 psig,(decreasing)	$\geq$ 435 psig, (decreasing)
d. Manual Initiation	N.A.	N.A.
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Reactor Vessel Pressure - Low	$\geq$ 455 psig,(decreasing)	$\geq$ 435 psig, (decreasing)
d. Injection Valve Differential Pressure - Low	$\geq$ 74 psid, (decreasing)	$\geq$ 64 psid and $\leq$ 84 psid
e. Manual Initiation	N.A.	N.A.
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -38 inches*	$\geq$ -45 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Condensate Storage Tank Level - Low	$\geq$ 167.8 inches**	$\geq$ 164.3 inches
d. Suppression Pool Water Level - High	$\leq$ 24 feet 1.5 inches	$\leq$ 24 feet 3 inches
e. Reactor Vessel Water Level - High, Level 8	$\leq$ 54 inches	$\leq$ 60 inches
f. Manual Initiation	N.A.	N.A.
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Core Spray Pump Discharge Pressure - High	$\geq$ 145 psig,(increasing)	$\geq$ 125 psig, (increasing),
e. RHR LPCI Mode Pump Discharge Pressure-High	$\geq$ 125 psig,(increasing)	$\geq$ 115 psig, (increasing)
f. Reactor Vessel Water Level-Low, Level 3	$\geq$ 12.5 inches	$\geq$ 11.0 inches
g. Manual Initiation	N.A.	N.A.
h. ADS Drywell Pressure Bypass Timer	$\leq$ 420 seconds	$\leq$ 450 seconds

*See Bases Figure B 3/4.3-1.

**Corresponds to 2.3 feet indicated.

TABLE 3.3.3-2 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
5. LOSS OF POWER	RELAY		
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	127-11X	NA	NA
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	RELAY		
	127-11X0X and 102-11X0X	a. 4.16 kV Basis 2905 ± 115 volts	2905 ± 145 volts
		b. 120 V Basis 83 ± 3 volts	83 ± 4 volts
		c. ≤ 1 second time delay	≤ 1.5 second time delay
	127Y-11X0X** and 127Y-1-11X0X	a. 4.16 kV Basis 3640 ± 91 volts	3640 ± 182 volts
		b. 120 V Basis 104 ± 3 volts	104 ± 5.2 volts
		c. ≤ 52 second time delay	≤ 60 second time delay
	127Z-11X0X and 162Y-11X0X	a. 4.16 kV Basis 3910 ± 11 volts	3910 ± 19 volts
		b. 120 V Basis 111.7 ± 0.3 volts	111.7 ± 0.5 volts
		c. ≤ 10 second time delay	≤ 11 second time delay
	127Z-11X0X and 162Z-11X0X	a. 4.16 kV Basis 3910 ± 11 volts	3910 ± 19 volts
		b. 120 V Basis 111.7 ± 0.3 volts	111.7 ± 0.5 volts
		c. ≤ 61 second time delay	≤ 64 second time delay

**This is an inverse time delay voltage relay. The voltages shown are the maximum that will not result in a trip. Some voltage conditions will result in decreased trip times.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS		RESPONSE TIME (Seconds)
1.	CORE SPRAY SYSTEM	≤ 27#
2.	LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	≤ 40#
3.	AUTOMATIC DEPRESSURIZATION SYSTEM	N.A.
4.	HIGH PRESSURE COOLANT INJECTION SYSTEM	≤ 60#
5.	LOSS OF POWER	N.A.

ECCS actuation instrumentation is eliminated from response time testing.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK(a)	CHANNEL FUNCTIONAL TEST (a)	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure - Low				1, 2, 3
d. Manual Initiation	N.A.		N.A.	1, 2, 3
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure - Low				1, 2, 3
d. Injection Valve Differential Pressure - Low (Permissive)				1, 2, 3
e. Manual Initiation	N.A.		N.A.	1, 2, 3
3. HIGH PRESSURE COOLANT INJECTION SYSTEM***				
a. Reactor Vessel Water Level - Low Low, Level 2				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. Condensate Storage Tank Level - Low				1, 2, 3
d. Suppression Pool Water Level - High				1, 2, 3
e. Reactor Vessel Water Level - High, Level 8				1, 2, 3
f. Manual Initiation	N.A.		N.A.	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (a)	CHANNEL FUNCTIONAL TEST (a)	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. AUTOMATIC DEPRESSURIZATION SYSTEM#				
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. ADS Timer	N.A.			1, 2, 3
d. Core Spray Pump Discharge Pressure - High				1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High				1, 2, 3
f. Reactor Vessel Water Level - Low, Level 3				1, 2, 3
g. Manual Initiation	N.A.		N.A.	1, 2, 3
h. ADS Drywell Pressure Bypass Timer	N.A.			1, 2, 3
5. LOSS OF POWER				
a. 4.16 kV Emergency Bus Under voltage (Loss of Voltage)##	N.A.		N.A.	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)				1, 2, 3, 4**, 5**

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

* DELETED

** Required OPERABLE when ESF equipment is required to be OPERABLE.

*** Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

\## Loss of Voltage Relay 127-11X is not field setable.

3/4.3.3.A REACTOR PRESSURE VESSEL (RPV) WATER INVENTORY CONTROL (WIC) INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.A The RPV Water Inventory Control (WIC) instrumentation channels shown in Table 3.3.3.A-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.3.A-1

ACTION:

a. With one or more channels inoperable in a trip system, take the ACTION referenced in Table 3.3.3.A-1 for the trip system.

SURVEILLANCE REQUIREMENTS

4.3.3.1.A Each RPV Water Inventory Control (WIC) instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL FUNCTIONAL TEST as shown in Table 4.3.3.A-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.3.A-1.

TABLE 3.3.3.A-1
RPV WATER INVENTORY CONTROL (WIC) INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. DELETED			
2. DELETED			
3. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION			
a. Reactor Vessel Water Level - Low - Level 3	2 in one trip system	(b)	38
4. REACTOR WATER CLEANUP SYSTEM ISOLATION			
a. Reactor Vessel Water Level - Low, Low - Level 2	2 in one trip system	(b)	38

(a) DELETED

(b) When automatic isolation of the associated penetration flow path(s) is credited in calculating DRAIN TIME.

(c) DELETED

TABLE 3.3.3.A-1 (Continued)
RPV WATER INVENTORY CONTROL (WIC) INSTRUMENTATION
ACTION STATEMENTS

ACTION 38 - Immediately initiate action to place the channel in trip, or declare the associated trip system for the penetration flow path(s) incapable of automatic isolation and initiate action to calculate DRAIN TIME.

ACTION 39 - DELETED

ACTION 40 - DELETED

TABLE 3.3.3.A-2
RPV WATER INVENTORY CONTROL (WIC) INSTRUMENTATION SETPOINTS

TRIP FUNCTION	ALLOWABLE VALUE
1. DELETED	
2. DELETED	
3. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level - Low - Level 3	≥ 11.0 inches
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low, Low - Level 2	≥ -45 inches

TABLE 4.3.3.A-1

RPV WATER INVENTORY CONTROL (WIC) INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK(a)	CHANNEL FUNCTIONAL TEST(a)	LOGIC SYSTEM FUNCTIONAL TEST(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. DELETED				
2. DELETED				
3. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low - Level 3			N.A.	(b)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Reactor Vessel Water Level - Low, Low - Level 2			N.A.	(b)

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.
(b) When automatic isolation of the associated penetration flow path(s) is credited in calculating DRAIN TIME.

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program*.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program, or if this action will initiate a pump trip, declare the trip system inoperable.

   2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each of the required ATWS recirculation pump trip system instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

*Not applicable when trip capability is not maintained.

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM *
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure - High	2

* One channel may be placed in an inoperable status for up to 6 hours for required surveillance provided the other channel is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
2. Reactor Vessel Pressure - High	≤ 1149 psig	≤ 1156 psig

* See Bases Figure B3/4.3-1.

INFORMATION ON THIS PAGE HAS BEEN DELETED

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 29.5% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 12 hours or in accordance with the Risk Informed Completion Time Program*.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours or in accordance with the Risk Informed Completion Time Program.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

*Not applicable when trip capability is not maintained.

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each of the required end-of-cycle recirculation pump trip system instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST, including trip system logic testing, and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested in accordance with the Surveillance Frequency Control Program. The measured time shall be added to the most recent breaker arc suppression time and the resulting END-OF-CYCLE-RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be verified to be within its limit.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve-Fast Closure	2**

* A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

** This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER LESS than 29.5% of RATED THERMAL POWER.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	$\leq$ 5% closed	$\leq$ 7% closed
2. Turbine Control Valve-Fast Closure	$\geq$ 500 psig	$\geq$ 465 psig

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milliseconds)
1. Turbine Stop Valve-Closure	$\leq$ 175
2. Turbine Control Valve-Fast Closure	$\leq$ 175

INFORMATION ON THIS PAGE HAS BEEN DELETED

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each of the required RCIC system actuation instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program. CHANNEL CHECK and CHANNEL CALIBRATION are not required for manual initiation.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION*	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	4#	50
b. Reactor Vessel Water Level - High, Level 8	4#	51
c. Condensate Storage Tank Water Level - Low	2**	52
d. Manual Initiation##	1/system***	53

*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 all other channels monitoring that parameter are OPERABLE.

**One trip system with one-out-of-two logic.

***One trip system with one channel.

#One trip system with one-out-of-two twice logic.

##The injection function of Manual Initiation is not required to be OPERABLE with reactor steam dome pressure less than 550 psig.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM
ACTION STATEMENTS

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. 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, or declare the RCIC system inoperable.

b. With more than one channel inoperable, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement, declare the RCIC system inoperable within 24 hours.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 24 hours or in accordance with the Risk Informed Completion Time Program*, or declare the RCIC system inoperable.

ACTION 53 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the RCIC system inoperable.

* Not applicable when trip capability is not maintained.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low, Level 2	≥-38 inches*	≥-45 inches
b. Reactor Vessel Water Level - High, Level 8	≤ 54 inches	≤ 60 inches
c. Condensate Storage Tank Level - Low	≥ 135.8** inches	≥ 132.3 inches
d. Manual Initiation	N.A.	N.A.

*See Bases Figure B 3/4.3-1.

**Corresponds to 2.3 feet indicated.

INFORMATION ON THIS PAGE HAS BEEN DELETED

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint** less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

* 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.

**The APRM Simulated Thermal Power - Upscale Functional Unit need not be declared inoperable upon entering single reactor recirculation loop operation provided that the flow-biased setpoints are adjusted within 6 hours per Specification 3.4.1.1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[a]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Simulated Thermal Power - Upscale	3	1	61
b. Inoperative	3	1, 2	61
c. Neutron Flux - Downscale	3	1	61
d. Simulated Thermal Power - Upscale (Setdown)	3	2	61
e. Recirculation Flow - Upscale	3	1	61
f. LPRM Low Count	3	1, 2	61
3. SOURCE RANGE MONITORS ***			
a. Detector not full in [b]	3	2	61
	2	5	61
b. Upscale[c]	3	2	61
	2	5	61
c. Inoperative[c]	3	2	61
	2	5	61
d. Downscale[d]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5**	61
b. Upscale	6	2, 5**	61
c. Inoperative	6	2, 5**	61
d. Downscale[e]	6	2, 5**	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
6. DELETED	DELETED	DELETED	DELETED
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

ACTION 60 - Declare the affected RBM channel inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 12 hours or place the inoperable channel in the tripped condition.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 12 hours.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

* For OPERATIONAL CONDITION of Specification 3.1.4.3.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

*** These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

(c) This function is automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function is automatically bypassed when the IRM channels are on range 3 or higher.

(e) This function is automatically bypassed when the IRM channels are on range 1.

(f) DELETED

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale[(a)]		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. Downscale (DTSP)	*	*
d. Power Range Setpoint[(b)]		
1) Low Power Setpoint (LPSP)	28.1% RATED THERMAL POWER	28.4% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	63.1% RATED THERMAL POWER	63.4% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	83.1% RATED THERMAL POWER	83.4% RATED THERMAL POWER
2. APRM		
a. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ 0.65 W + 54.3% and ≤ 108.0% of RATED THERMAL POWER	≤ 0.65 W + 54.7% and ≤ 108.4% of RATED THERMAL POWER
- Single Recirculation Loop Operation****	≤ 0.65 (W-7.6%) + 54.1% and ≤ 108.0% of RATED THERMAL POWER	≤ 0.65 (W-7.6%) + 54.5% and ≤ 108.4% of RATED THERMAL POWER
b. Inoperative	N.A.	N.A.
c. Neutron Flux - Downscale POWER	≥ 3.2% of RATED THERMAL POWER	≥ 2.8% of RATED THERMAL
d. Simulated Thermal Power - Upscale (Setdown)	≤ 12.0% of RATED THERMAL POWER	≤ 13.0% of RATED THERMAL POWER
e. Recirculation Flow - Upscale	*	*
f. LPRM Low Count	< 20 per channel < 3 per axial level	< 20 per channel < 3 per axial level
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High a. Float Switch	≤ 257' 7 3/8" elevation***	≤ 257' 9 3/8" elevation
6. DELETED	DELETED	DELETED
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	N.A.

* Refer to the COLR for these setpoints.

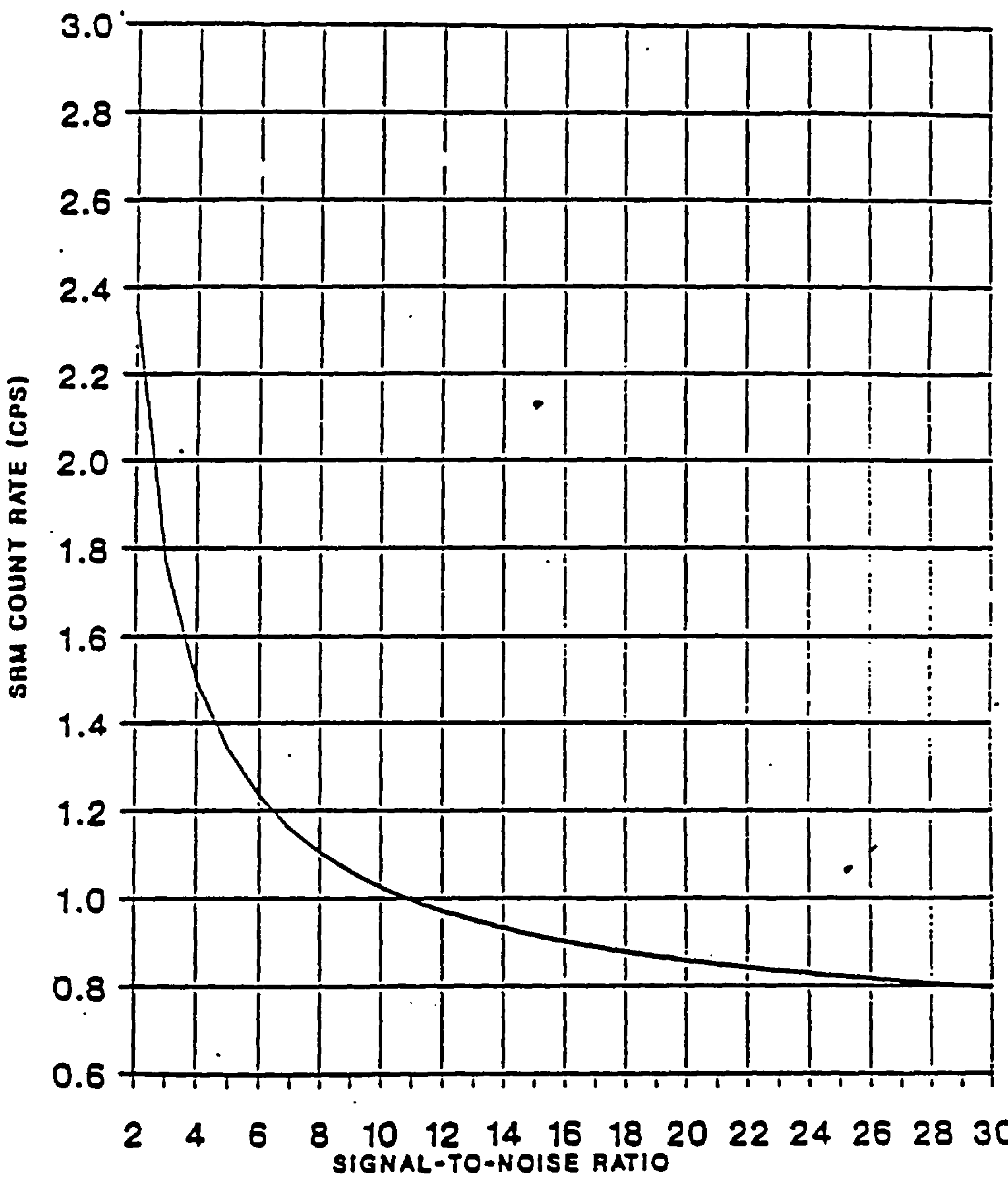
** May be reduced, provided the Source Range Monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1.

*** Equivalent to 13.56 gallons/scram discharge volume.

**** The 7.6% flow "offset" for Single Loop Operation (SLO) is applied for W ≥ 7.6%. For flows W < 7.6%, the (W-7.6%) term is set equal to zero.

(a) There are three upscale trip levels. Each is applicable only over its specified operating core thermal power range. All RBM trips are automatically bypassed below the low power setpoint (LPSP). The upscale LTSP is applied between the low power setpoint (LPSP) and the intermediate power setpoint (IPSP). The upscale ITSP is applied between the intermediate power setpoint and the high power setpoint (HPSP). The HTSP is applied above the high power setpoint.

(b) Power range setpoints control enforcement of appropriate upscale trips over the proper core thermal power ranges. The power signal to the RBM is provided by the APRM.



SRM COUNT RATE VERSUS SIGNAL-TO-NOISE RATIO

Figure 3.3.6-1

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (h)	CHANNEL FUNCTIONAL TEST (h)	CHANNEL CALIBRATION(a)(h)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	(c)		1*
b. Inoperative	N.A.	(c)	N.A.	1*
c. Downscale	N.A.	(c)		1*
2. APRM				
a. Simulated Thermal Power - Upscale	N.A.			1
b. Inoperative	N.A.		N.A.	1, 2
c. Neutron Flux - Downscale	N.A.			1
d. Simulated Thermal Power - Upscale (Setdown)	N.A.			2
e. Recirculation Flow - Upscale	N.A.			1
f. LPRM Low Count	N.A.			1, 2
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	(e)	N.A.	2, 5
b. Upscale	N.A.	(e)		2, 5
c. Inoperative	N.A.	(e)	N.A.	2, 5
d. Downscale	N.A.	(e)		2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.		N.A.	2, 5**
b. Upscale	N.A.			2, 5**
c. Inoperative	N.A.		N.A.	2, 5**
d. Downscale	N.A.			2, 5**
5. SCRAM DISCHARGE VOLUME				
a. Water Level - High	N.A.			1, 2, 5**
6. DELETED				
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	(g)	N.A.	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Deleted.

(c) Includes reactor manual control multiplexing system input.

* For OPERATIONAL CONDITION of Specification 3.1.4.3.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

*** Deleted.

(d) Deleted

(e) The provisions of Specification 4.0.4 are not applicable provided that the surveillance is performed within 12 hours after the IRMs are on Range 2 or below during a shutdown.

(f) Deleted

(g) The provisions of Specification 4.0.4 are not applicable provided that the surveillance is performed within 1 hour after the Reactor Mode Switch has been placed in the shutdown position.

(h) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the conditions shown in Table 4.3.7.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.1-1.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	4	1,2,3, and *	1 x 10^{-5} μCi/cc	70
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	2	(a)	≥ 5 mR/h and ≤20mR/h[(b)]	71
b. Control Room Direct Radiation Monitor	1	At All Times	N.A.[(b)]	73
3. Reactor Enclosure Cooling Water Radiation Monitor	1	At All Times	≤ 3 x Background[(b)]	72

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment with the vessel head removed and fuel in the vessel.

(a) With fuel in the spent fuel storage pool.

(b) Alarm only.

ACTION STATEMENTS

ACTION 70 - With one monitor inoperable, restore the inoperable monitor to the OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the radiation isolation mode of operation.

With two or more of the monitors inoperable, within one hour, initiate and maintain operation of the control room emergency filtration system in the radiation mode of operation.

ACTION 71 - With one of the required monitor inoperable, assure a portable continuous monitor with the same alarm setpoint is OPERABLE in the vicinity of the installed monitor during any fuel movement. If no fuel movement is being made, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 72 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, assure a portable alarming monitor is OPERABLE in the vicinity of the installed monitor or perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK(c)	CHANNEL FUNCTIONAL TEST(c)	CHANNEL CALIBRATION(c)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
1. Main Control Room Normal Fresh Air Supply Radiation Monitor				1, 2, 3, and *
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool				(a)
b. Control Room Direct Radiation Monitor				At All Times
3. Reactor Enclosure Cooling Water Radiation Monitor			(b)	At All Times

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment with the vessel head removed and fuel in the vessel.

(a) With fuel in the spent fuel storage pool.

(b) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(c) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

Section 3.3.7.2 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE TRM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-69 THROUGH 3/4 3-72 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

Section 3.3.7.3 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-74 THROUGH 3/4 3-75 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

effective January 2, 1991

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system functions shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION**:

a. With one or more of the required functions inoperable, restore the inoperable function(s) to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each normally energized required instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK* at the frequency specified in the Surveillance Frequency Control Program.

4.3.7.4.2 Each required control circuit and transfer switch shall be demonstrated OPERABLE by verifying its capability to perform its intended function in accordance with the Surveillance Frequency Control Program.

4.3.7.4.3 Each required instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.

* Control is not required to be transferred to perform the CHANNEL CHECK.

** NOTE: Separate ACTION entry is allowed for each function.

TABLE 3.3.7.4-1

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE TRM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-78 THROUGH 3/4 3-82 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

INFORMATION ON THIS PAGE HAS BEEN DELETED

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.5-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2	80
2. Reactor Vessel Water Level	2	1	1,2	80
3. Suppression Chamber Water Level	2	1	1,2	80
4. Suppression Chamber Water Temperature	8, 6 locations	6, 1/location	1,2	80
5. Deleted				
6. Drywell Pressure	2	1	1,2	80
7. Deleted				
8. Deleted				
9. Deleted				
10. Deleted				
11. Primary Containment Post-LOCA Radiation Monitors	4	2	1,2,3	81
12. North Stack Wide Range Accident Monitor**	3*	3*	1,2,3	81
13. Neutron Flux	2	1	1,2	80

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

TABLE NOTATIONS

*Three noble gas detectors with overlapping ranges (10^{-7} to 10^{-1}, 10^{-4} to 10^{2}, 10^{-1} to 10^{5} μCi/cc).

**High range noble gas monitor.

ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 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 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, initiate the preplanned alternate method of monitoring the appropriate parameters within 72 hours, and

a. Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 - DELETED

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK (a)	CHANNEL CALIBRATION (a)
1. Reactor Vessel Pressure		
2. Reactor Vessel Water Level		
3. Suppression Chamber Water Level		
4. Suppression Chamber Water Temperature		
5. Deleted		
6. Primary Containment Pressure		
7. Deleted		
8. Deleted		
9. Deleted		
10. Deleted		
11. Primary Containment Post LOCA Radiation Monitors		**
12. North Stack Wide Range Accident Monitor***		
13. Neutron Flux		

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

**CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

***High range noble gas monitors.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*#, 3, and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least three source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK in accordance with the Surveillance Frequency Control Program:

   a) in CONDITION 2*, and
   b) in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** in accordance with the Surveillance Frequency Control Program.

b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3.0 cps*** with the detector fully inserted.#

*With IRM's on range 2 or below in CONDITION 2.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.
***May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1.
#During initial startup test program, SRM detectors may be partially withdrawn prior to IRM on-scale indication provided that the SRM channels remain on scale above 100 cps and respond to changes in the neutron flux.

Section 3/4.3.7.7

THE INFORMATION FROM THIS TECHNICAL SPECIFICATION
HAS BEEN RELOCATED TO THE TECHNICAL REQUIREMENTS MANUAL (TRM)

Section 3/4.3.7.8.1 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATION
HAS BEEN RELOCATED TO THE TECHNICAL REQUIREMENTS MANUAL (TRM)

Section 3/4.3.7.8.2 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATION
HAS BEEN RELOCATED TO THE TECHNICAL REQUIREMENTS MANUAL (TRM)

Section 3/4.7.9 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE TECHNICAL REQUIREMENTS MANUAL (TRM) FIRE PROTECTION SECTION. TECHNICAL SPECIFICATIONS PAGES 3/4 3-92 THROUGH 3/4 3-96 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

Section 3.3.7.11 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-99 THROUGH 3/4 3-102 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

effective January 2, 1991

Section 3/4.3.7.12 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM AND THE TRM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-104 THROUGH 3/4 3-108 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

INTENTIONALLY LEFT BLANK

effective January 2, 1991

THE INFORMATION FROM THIS TECHNICAL
SPECIFICATIONS SECTION HAS BEEN
RELOCATED TO THE TRM.
TECHNICAL SPECIFICATIONS PAGE 3/4 3-111
HAS BEEN INTENTIONALLY OMITTED.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in the Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program**, or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each of the required feedwater/main turbine trip system actuation instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.9.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

* 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.
**Not applicable when trip capability is not maintained.

TABLE 3.3.9-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Water Level-High, Level 8	4	1*

* With Thermal Power greater than or equal to 25% of Rated Thermal Power.

TABLE 3.3.9-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level-High, Level 8	$\leq$ 54 inches*	$\leq$ 55.5 inches

*See Bases Figure B 3/4.3-1

INFORMATION ON THIS PAGE HAS BEEN DELETED

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within 4 hours:

      a. Place the recirculation flow control system in the Local Manual mode, and

      b. Reduce THERMAL POWER to $\leq$ 74.9% of RATED THERMAL POWER, and,

      c. Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

      d. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is $\leq$ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is $\leq$ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

---

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. Within 6 hours:

   Reduce the Average Power Range Monitor (APRM) Simulated Thermal Power - Upscale Scram and Rod Block Trip Setpoints and Allowable Values, to those applicable for single recirculation loop operation per Specifications 2.2.1 and 3.3.6, or declare the associated channel(s) inoperable and take the actions required by the referenced specifications.

3. Otherwise be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 DELETED

4.4.1.1.3 DELETED

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, in accordance with the Surveillance Frequency Control Program, verify that:

a. Reactor THERMAL POWER is ≤ 74.9% of RATED THERMAL POWER,

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is ≤ 90% of rated pump speed.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow.

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

CONTENTS OF THIS PAGE HAVE BEEN DELETED

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 All jet pumps shall be demonstrated OPERABLE as follows:

a. During two recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.

    1. The indicated recirculation loop flow differs by more than 10% from the established pump speed-loop flow characteristics.

    2. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

    3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from the established patterns by more than 10%.

SURVEILLANCE REQUIREMENTS (Continued)

b. During single recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that no two of the following conditions occur:

   1. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established pump speed-loop flow characteristics.

   2. The indicated total core flow differs by more than 10% from the established total core flow value derived from single recirculation loop flow measurements.

   3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop patters by more than 10%.

c. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER and upon entering single recirculation loop operation.

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of each other with core flow greater than or equal to 70% of rated core flow.

b. 10% of each other with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Shutdown one of the recirculation loops within the next 8 hours and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F, and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*#

4 safety/relief valves @ 1170 psig ±3%
5 safety/relief valves @ 1180 psig ±3%
5 safety/relief valves @ 1190 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. DELETED

c. DELETED

SURVEILLANCE REQUIREMENTS

4.4.2.1 DELETED

4.4.2.2 Verify the specified safety valve function lift setting of each of the 14 safety/relief valves in accordance with the INSERVICE TESTING PROGRAM requirements of Specification 4.0.5. All safety valves will be recertification tested to meet a ±1% tolerance prior to returning the valves to service.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

\# Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere gaseous radioactivity monitoring system,

b. The drywell sump monitoring system,

c. The drywell unit coolers condensate flow rate monitoring system, and

d. The primary containment pressure and temperature monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.*
* - The primary containment gaseous radioactivity monitor is not required to be operable until Operational Condition 2.

ACTIONS:

A. With the primary containment atmosphere gaseous radioactivity monitoring system inoperable, analyze grab samples of primary containment atmosphere at least once per 12 hours AND restore primary containment atmosphere gaseous radioactivity monitoring system to OPERABLE status within 30 days.

B. With the drywell sump monitoring system inoperable, restore the drywell sump monitoring system to OPERABLE status within 30 days AND increase monitoring frequency of drywell unit cooler condensate flow rate (SR 4.4.3.2.1.c) to once every 8 hours.

C. With the drywell unit coolers condensate flow rate monitoring system inoperable, AND the primary containment atmosphere gaseous radioactivity monitoring system OPERABLE, perform a channel check of the primary containment atmosphere gaseous radioactivity monitoring system (SR 4.4.3.1.a) once per 8 hours.

D. With the primary containment pressure and temperature monitoring system inoperable, restore the primary containment pressure and temperature monitoring system to OPERABLE status within 30 days. Note: All other Tech Spec Limiting Conditions For Operation and Surveillance Requirements associated with the primary containment pressure/temperature monitoring system still apply. Affected Tech Spec Sections include: 3/4.3.7.5, 4.4.3.2.1, 3/4.6.1.6, and 3/4.6.1.7.

E. With the primary containment atmosphere gaseous radioactivity monitoring system inoperable AND the drywell unit coolers condensate flow rate monitoring system inoperable, restore the primary containment atmosphere gaseous radioactivity monitoring system to OPERABLE status within 30 days OR restore the drywell unit coolers condensate flow rate monitoring system to OPERABLE status within 30 days. With the primary containment atmosphere gaseous radioactivity monitoring system inoperable, analyze grab samples of primary containment atmosphere at least once per 12 hours.

ACTIONS (Continued)

F. With the drywell floor drain sump monitoring system inoperable AND the drywell unit coolers condensate flow rate monitoring system inoperable analyze grab samples of the primary containment atmosphere once per 12 hours, AND monitor Reactor Coolant System leakage by administrative means once per 12 hours AND restore either the drywell floor drain sump monitoring system to OPERABLE status within 7 days OR restore the drywell unit coolers condensate flow rate monitoring system to OPERABLE status within 7 days.

G. With any other two or more leak detection systems inoperable other than ACTIONS E and F above OR with required Actions and associated Completion Time of ACTIONS A, B, C, D, E or F not met, be in HOT SHUTDOWN within 12 hours AND in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated operable by:

a. Perform a CHANNEL CHECK of the primary containment atmosphere gaseous radioactivity monitoring system in accordance with the Surveillance Frequency Control Program.

b. Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation in accordance with the Surveillance Frequency Control Program. This does not apply to containment pressure and temperature monitoring system.

c. Perform a CHANNEL CALIBRATION of required leakage detection instrumentation in accordance with the Surveillance Frequency Control Program. This does not apply to containment pressure and temperature monitoring system.

d. Monitor primary containment pressure AND primary containment temperature in accordance with the Surveillance Frequency Control Program.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 30 gpm total leakage.

d. 25 gpm total leakage averaged over any 24-hour period.

e. 1 gpm leakage at a reactor coolant system pressure of 950 ±10 psig from any reactor coolant system pressure isolation valve.**

f. 2 gpm increase in UNIDENTIFIED LEAKAGE over a 24-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, isolate affected component, pipe, or vessel from the reactor coolant system by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours. Otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b, c and/or d above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system leakage greater than the limit in f above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

* Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

** Pressure isolation valve leakage is not included in any other allowable operational leakage specified in Section 3.4.3.2.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric gaseous radioactivity in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),

b. Monitoring the drywell floor drain sump and drywell equipment drain tank flow rate in accordance with the Surveillance Frequency Control Program,

c. Monitoring the drywell unit coolers condensate flow rate in accordance with the Surveillance Frequency Control Program,

d. Monitoring the primary containment pressure in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),

e. Monitoring the reactor vessel head flange leak detection system in accordance with the Surveillance Frequency Control Program, and

f. Monitoring the primary containment temperature in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. In accordance with the Surveillance Frequency Control Program, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints set less than the specified allowable values by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program.

TABLE 3.4.3.2-1 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATION SECTION HAS BEEN RELOCATED TO THE TECHNICL REQUIREMENTS MANUAL (TRM).

3/4.4.4 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE TECHNICAL REQUIREMENTS MANUAL (TRM). TECHNICAL SPECIFICATIONS PAGES 3/4 4-13 AND 3/4 4-14 HAVE BEEN INTENTIONALLY OMITTED.

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcurie per gram DOSE EQUIVALENT I-131.

b. (Deleted)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the specific activity of the primary coolant;

   1. Greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 but less than or equal to 4 microcuries per gram, DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours. The provisions of Specification 3.0.4.c are applicable.

   2. (Deleted)

b. In OPERATIONAL CONDITION 1, 2, 3, or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, perform the sampling and analysis requirements of Item 4.a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

   1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in 1 hour*, or

   2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in 1 hour during steady-state operation at release rates less than 75,000 microcuries per second, or

   3. The off-gas level, at the SJAE, increased by more than 15% in 1 hour during steady-state operation at release rates greater than 75,000 microcuries per second,

   perform the sampling and analysis requirements of Item 4.b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

*Not applicable during the startup test program.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS IS REQUIRED
1. (Deleted)		
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	In accordance with the Surveillance Frequency Control Program	1
3. (Deleted)		
4. Isotopic Analysis for Iodine	a) At least once per 4 hours whenever the specific activity exceeds a limit, as required by ACTION b.	1**, 2**, 3**, 4**
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135, and Kr-88	In accordance with the Surveillance Frequency Control Program	1

**Until the specific activity of the primary coolant system is restored to within its limits.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limits specified in the PTLR, with:

a. A maximum heatup rate within the limits specified in the PTLR,

b. A maximum cooldown rate within the limits specified in the PTLR,

c. A maximum temperature change within the limits specified in the PTLR during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature within the limits specified in the PTLR when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limits specified in the PTLR as applicable, in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limits specified in the PTLR within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and in accordance with the Surveillance Frequency Control Program during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be within the limits specified in the PTLR:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

 1. ≤ 100°F, in accordance with the Surveillance Frequency Control Program.

 2. ≤ 90°F, in accordance with the Surveillance Frequency Control Program.

b. Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during tensioning of the reactor vessel head bolting studs.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1053 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1053 psig, reduce the pressure to less than 1053 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1053 psig in accordance with the Surveillance Frequency Control Program.

*Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one or more MSIVs inoperable:

a. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours or in accordance with the Risk Informed Completion Time Program, either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

3/4.4.8 (DELETED)

PAGE INTENTIONALLY LEFT BLANK

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two (2) independent RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one (1) RHR shutdown cooling subsystem shall be in operation. * ** ***

Each independent RHR shutdown cooling subsystem shall consist of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger, not common to the two (2) independent subsystems.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

a. With less than the above required independent RHR shutdown cooling subsystems OPERABLE, immediately initiate corrective action to return the required independent subsystems to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, verify the availability of at least one alternate method capable of decay heat removal for each inoperable independent RHR shutdown cooling subsystem. Be in at least COLD SHUTDOWN within 24 hours.****

b. With no independent RHR shutdown cooling subsystem in operation, immediately initiate corrective action to return at least one (1) independent subsystem to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 At least one independent RHR shutdown cooling subsystem or alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.4.9.1.2 Verify RHR shutdown cooling subsystem locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.*****

---

*One independent RHR shutdown cooling subsystem may be inoperable for up to 2 hours for surveillance testing provided the other independent subsystem is OPERABLE and in operation.

**The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period provided the other independent subsystem is OPERABLE.

***The independent RHR shutdown cooling subsystem may be removed from operation during hydrostatic testing.

****Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

*****Not required to be performed until 12 hours after reactor steam dome pressure is less than the RHR cut-in permissive setpoint.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two (2) RHR shutdown cooling subsystems shall be OPERABLE, and with no recirculation pump in operation, at least one (1) RHR shutdown cooling subsystem shall be in operation. * ** ***

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION: #

a. With one (1) or two (2) RHR shutdown cooling subsystems inoperable:

   1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.

b. With no RHR shutdown cooling subsystems in operation and no recirculation pump in operation:

   1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulating by an alternate method; and

   2. Once per hour monitor reactor coolant temperature and pressure.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 At least one (1) RHR shutdown cooling subsystem or recirculation pump is operating or an alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.4.9.2.2 Verify RHR shutdown cooling subsystem locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

---

* Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up two (2) hours per eight hour (8) period.

** One (1) RHR shutdown cooling subsystem may be inoperable for up to two (2) hours for the performance of Surveillances.

*** The shutdown cooling subsystem may be removed from operation during hydrostatic testing.

Separate Action entry is allowed for each shutdown cooling subsystem.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE CSS pumps, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of four subsystems with each subsystem comprised of:

   1. One OPERABLE LPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. The high pressure coolant injection (HPCI) system consisting of:

   1. One OPERABLE HPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with at least five OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2* ** #, and 3* ** ##.

---

*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

**The ADS is not required to be OPERABLE when the reactor steam dome pressure is less that or equal to 100 psig.

#See Special Test Exception 3.10.6.

##Two LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when reactor vessel pressure is less than the RHR Shutdown cooling permissive setpoint.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the core spray system:

1. With one CSS subsystem inoperable, provided that at least two LPCI subsystems are OPERABLE, restore the inoperable CSS subsystem to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

1. With one LPCI subsystem inoperable, provided that at least one CSS subsystem is OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHR cross-tie valve (HV-51-282 A or B) open, or power not removed from one closed RHR cross-tie valve operator, close the open valve and/or remove power from the closed valves operator within 72 hours, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With no RHR cross-tie valves (HV-51-282 A, B) closed, or power not removed from both closed RHR cross-tie valve operators, or with one RHR cross-tie valve open and power not removed from the other RHR cross-tie valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. With two LPCI subsystems inoperable, provided that at least one CSS subsystem is OPERABLE, restore at least three LPCI subsystems to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With three LPCI subsystems inoperable, provided that both CSS subsystems are OPERABLE, restore at least two LPCI subsystems to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. With all four LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

*Whenever both shutdown cooling subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. For the HPCI system:

1. With the HPCI system inoperable, provided the CSS, the LPCI system, the ADS and the RCIC system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 200 psig within the following 24 hours.

2. With the HPCI system inoperable, and one CSS subsystem, and/or LPCI subsystem inoperable, and provided at least one CSS subsystem, three LPCI subsystems, and ADS are operable, restore the HPCI to OPERABLE within 8 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT SHUTDOWN in the next 12 hours, and in COLD SHUTDOWN in the next 24 hours.

3. Specification 3.0.4.b is not applicable to HPCI.

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the CSS and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 100 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ 100 psig within the next 24 hours.

e. With a CSS and/or LPCI header ΔP instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or determine the ECCS header ΔP locally at least once per 12 hours; otherwise, declare the associated CSS and/or LPCI, as applicable, inoperable.

f. DELETED

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. In accordance with the Surveillance Frequency Control Program:

1. For the CSS, the LPCI system, and the HPCI system:

a) Verifying locations susceptible to gas accumulation are sufficiently filled with water.

b) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.***

2. For the LPCI system, verifying that both LPCI system subsystem cross-tie valves (HV-51-282 A, B) are closed with power removed from the valve operators.

3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

4. For the CSS and LPCI system, performance of a CHANNEL FUNCTIONAL TEST of the injection header ΔP instrumentation.

b. Verifying that, when tested pursuant to Specification 4.0.5:

1. Each CSS pump in each subsystem develops a flow of at least 2500 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of = 105 psid plus head and line losses.

2. Each LPCI pump in each subsystem develops a flow of at least 8000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of $\geq$ 20 psid plus head and line losses.

3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure which corresponds to a reactor vessel pressure of 1040 psig plus head and line losses when steam is being supplied to the turbine at 1040, +13, -120 psig.**

c. In accordance with the Surveillance Frequency Control Program:

1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. **** Actual injection of coolant into the reactor vessel may be excluded from this test.

---

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 200 psig within the following 72-hours.

*** Not required to be met for system vent flow paths opened under administrative control.

**** Except for valves that are locked, sealed, or otherwise secured in the actuated position.

2. For the HPCI system, verifying that:

   a) The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥ 200 psig plus head and line losses, when steam is being supplied to the turbine at 200 + 15, - 0 psig.**

   b) The suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

3. Performing a CHANNEL CALIBRATION of the CSS, LPCI, and HPCI system discharge line "keep filled" alarm instrumentation.

4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 4.4 psid.

5. Performing a CHANNEL CALIBRATION of the LPCI header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 3.0 psid.

d. For the ADS:

1. In accordance with the Surveillance Frequency Control Program, verify ADS accumulator gas supply header pressure is ≥90 psig.

2. In accordance with the Surveillance Frequency Control Program:

   a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

   b) Verify that when tested pursuant to Specification 4.0.5 that each ADS valve is capable of being opened.

   c) DELETED

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If HPCI OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 200 psig within the following 72 hours.

3/4.5.2 REACTOR PRESSURE VESSEL (RPV) WATER INVENTORY CONTROL (WIC)

LIMITING CONDITION FOR OPERATION

3.5.2 DRAIN TIME of RPV water inventory to the top of active fuel (TAF) shall be ≥ 36 hours

AND

At least one of the following shall be OPERABLE:

a. Core spray system (CSS) subsystem comprised of:

1. Two OPERABLE CSS pumps, and

2. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

a) From the suppression chamber, or

b) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 135,000 available gallons of water, equivalent to a level of 29 feet.

b. Low pressure coolant injection (LPCI) system subsystem comprised of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.**

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

a. With none of the above required subsystems OPERABLE, immediately suspend CORE ALTERATIONS. Restore at least one subsystem to OPERABLE status within 4 hours. Otherwise, initiate action to establish a method of water injection capable of operating without offsite electrical power.

b. DELETED.

**One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

c. With DRAIN TIME less than 36 hours and greater than or equal to 8 hours, within 4 hours:

   1. Verify SECONDARY CONTAINMENT INTEGRITY is capable of being established in less than the DRAIN TIME,

   2. Verify each secondary containment penetration flow path is capable of being isolated in less than the DRAIN TIME, and

   3. Verify one standby gas treatment subsystem is capable of being placed in operation in less than the DRAIN TIME.

d. With DRAIN TIME less than 8 hours, immediately:

   1. Initiate action to establish an additional method of water injection with water sources capable of maintaining RPV water level greater than TAF for greater than or equal to 36 hours,***

   2. Initiate action to establish SECONDARY CONTAINMENT INTEGRITY,

   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 standby gas treatment subsystem is capable of being placed in operation.

e. With required ACTION and associated allowed outage time for ACTIONs c. or d. not met, or DRAIN TIME less than 1 hour, immediately initiate action to restore DRAIN TIME to greater than or equal to 36 hours.

***The required injection/spray subsystem or an additional method of water injection shall be capable of operating without offsite electrical power.

SURVEILLANCE REQUIREMENTS

4.5.2.1 Verify DRAIN TIME is greater than or equal to 36 hours in accordance with the Surveillance Frequency Control Program.*

4.5.2.2 Verify, for a required LPCI subsystem, the suppression chamber water level is greater than or equal to 16 feet 0 inches in accordance with the Surveillance Frequency Control Program.

4.5.2.3 Verify, for a required CSS subsystem, that the suppression chamber water level is greater than or equal to 16 feet 0 inches or the condensate storage tank water level is greater than or equal to 29 feet 0 inches in accordance with the Surveillance Frequency Control Program.

4.5.2.4 Verify, for the required ECCS injection/spray subsystem, locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

4.5.2.5 DELETED

4.5.2.6 Operate the required ECCS injection/spray subsystem for greater than or equal to 10 minutes in accordance with the Surveillance Frequency Control Program.#^

4.5.2.7 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. **

4.5.2.8 Verify the required ECCS injection/spray subsystem can be manually operated in accordance with the Surveillance Frequency Control Program. **

---

* DELETED.

#Operation may be through the test return line.

^Credit may be taken for normal system operation to satisfy this surveillance requirement.

** Except for valves that are locked, sealed, or otherwise secured in the actuated position.

3/4.5.3 SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3 with a contained water volume of at least 122,120 ft^3, equivalent to a level of 22'0".

b. DELETED

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. DELETED

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to, as applicable:

a. 22'0" in accordance with the Surveillance Frequency Control Program.

b. DELETED

4.5.3.2 DELETED

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

b. In accordance with the Surveillance Frequency Control Program by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

c. By verifying the primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

---

* See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A Test) in accordance with the Primary Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted.

c. *Less than or equal to 100 scf per hour through any one main steam isolation valve not to exceed 200 scf per hour for all four main steam lines, when tested at P_t, 22.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 48.4 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate (Type A Test) exceeding the leakage rate specified in the Primary Containment Leakage Rate Testing Program, or

b. The measured combined leakage rate exceeding the leakage rate specified in the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, or

c. The measured leakage rate exceeding 100 scf per hour through any one main steam isolation valve, or exceeding 200 scf per hour for all four main steam lines, or

d. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) (Type A Test) to be in accordance with the Primary Containment Leakage Rate Testing Program, and

*Exemption to Appendix J of 10 CFR Part 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate to be in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, and

c. The leakage rate to ≤100 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage to ≤200 scf per hour, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated to be in accordance with the Primary Containment Leakage Rate Testing Program, or approved exemptions, for the following:

a. Type A Test

b. Type B and C Tests (including air locks)

c. Main Steam Line Isolation Valves

d. Hydrostatically tested Containment Isolation Valves

*Exemption to Appendix "J" to 10 CFR Part 50.

THIS PAGE IS
INTENTIONALLY LEFT BLANK

PRIMARY CONTAINMENT AIR LOCK

LIMITING CONDITION FOR OPERATION

3.6.1.3 The primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Primary Containment Leakage Rate Testing Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary containment air lock shall be demonstrated OPERABLE:

a. By verifying the seal leakage rate is in accordance with the Primary Containment Leakage Rate Testing Program.

b. By conducting an overall air lock leakage test in accordance with the Primary Containment Leakage Rate Testing Program.

c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in the air lock can be opened at a time.***

***Except that the airlock doors need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the airlock doors' interlock is tested within 8 hours after the primary containment has been deinerted and provided the shield door to the airlock is maintained locked closed.

MSIV LEAKAGE ALTERNATE DRAIN PATHWAY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The MSIV Leakage Alternate Drain Pathway shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the MSIV Leakage Alternate Drain Pathway inoperable, restore the pathway to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The MSIV Leakage Alternate Drain Pathway shall be demonstrated OPERABLE:

a. In accordance with Specification 4.0.5, by cycling each motor operated valve, required to be repositioned, through at least one complete cycle of full travel.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined by a visual inspection of those surfaces. This inspection shall be performed in accordance with the Primary Containment Leakage Rate Testing Program.

4.6.1.5.2 DELETED

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -1.0 and +2.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 145°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 145°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program:

	Approximate Elevation	Number of Installed Sensors*
a.	330'	3
b.	320'	3
c.	260'	3
d.	248'	6

* At least one reading from each elevation is required for a volumetric average calculation.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system may be in operation with the supply and exhaust isolation valves in one supply line and one exhaust line open for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With a drywell and/or suppression chamber purge supply and/or exhaust isolation valve open, except as permitted above, close the valve(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8 In accordance with the Surveillance Frequency Control Program, verify each primary containment purge valve [18" or 24"] is closed.*, **

* Only required to be met in OPERATIONAL CONDITIONS 1, 2, and 3.

** Not required to be met when the primary containment purge valves are open for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require these valves to be open.

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume* between 122,120 ft^3 and 134,600 ft^3, equivalent to a level between 22' 0" and 24' 3", and a

2. Maximum average temperature of 95°F except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

c) 120°F with the main steam line isolation valves closed following a scram, one in each of the eigth locations.

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{K}$ design value of 0.0500 ft^2.

c. At least eight suppression pool water temperature instrumentation indicators, one in each of the eight locations.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression chamber average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than:

a) 95°F for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

---

*Includes the volume inside the pedestal.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.

d. With no suppression chamber water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits in accordance with the Surveillance Frequency Control Program.

b. In accordance with the Surveillance Frequency Control Program by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F, and

b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER 12 hours after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

c. By verifying at least 8 suppression pool water temperature indicators in at least 8 locations, OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program with the temperature alarm setpoint for:

   1. High water temperature:

      a) First setpoint ≤ 95°F

      b) Second setpoint ≤ 105°F

      c) Third setpoint ≤ 110°F

      d) Fourth setpoint ≤ 120°F

d. By verifying at least two suppression chamber water level indicators OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program with the water level alarm setpoint for high water level ≤ 24'1-1/2".

e. Drywell-to-suppression chamber bypass leak tests shall be conducted to coincide with the Type A test at an initial differential pressure of 4 psi and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit, at which time the test schedule may be resumed.

f. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area $A/\sqrt{k}$ contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.e is not conducted.

CONTAINMENT SYSTEMS

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger(s).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and the suppression pool spray sparger when tested pursuant to Specification 4.0.5.

c. By verifying RHR suppression pool spray subsystem locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

---

* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours** or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the flow path including the RHR heat exchanger and its associated closed bypass valve, the suppression pool and the full flow test line when tested pursuant to Specification 4.0.5.

c. By verifying RHR suppression pool cooling subsystem locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

**During the extended Allowed Outage Time (AOT) specified by TS LCO 3.7.1.1, Action a.3.a) or a.3.b) to allow for RHRSW subsystem piping repairs, the 72-hour AOT for one inoperable suppression pool cooling loop may also be extended to 7 days or in accordance with the Risk Informed Completion Time Program for the same period.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each primary containment isolation valve and each instrumentation line excess flow check valve shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves inoperable,** maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours or in accordance with the Risk Informed Completion Time Program either:

1. Restore the inoperable valve(s) to OPERABLE status, or

2. Isolate each affected penetration by use of at least one de-activated automatic valve secured in the isolated position,* or

3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within 4 hours either:

1. The inoperable valve is returned to OPERABLE status, or

2. The instrument line is isolated and the associated instrument is declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one or more scram discharge volume vent or drain valves inoperable, perform the applicable actions specified in Specification 3.1.3.1.

---

* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

** Except for the scram discharge volume vent and drain valves.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 A representative sample of instrumentation line excess flow check valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program, such that each valve is tested in accordance with the Surveillance Frequency Control Program, by verifying that the valve checks flow.*

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying the continuity of the explosive charge.

b. In accordance with the Surveillance Frequency Control Program by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested in accordance with the Surveillance Frequency Control Program, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable.

*The reactor vessel head seal leakage detection line (penetration 29A) excess flow check valve is not required to be tested pursuant to this requirement.

TABLE 3.6.3-1 (Deleted)

THE INFORMATION FROM THIS
TECHNICAL SPECIFICATION SECTION
HAS BEEN RELOCATED TO THE
TECHNICAL REQUIREMENTS MANUAL (TRM), PCIV SECTION.

TECHNICAL SPECIFICATION PAGES 3/4 6-19 THROUGH 3/4 6-43a
HAVE BEEN INTENTIONALLY OMITTED.

Amendment No. ~~52~~, ~~53~~, 107
OCT 18 2000

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Three pairs of suppression chamber - drywell vacuum breakers shall be OPERABLE and all suppression chamber - drywell vacuum breakers shall be closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more vacuum breakers in one of the three required pairs of suppression chamber - drywell vacuum breaker pairs inoperable for opening but known to be closed, restore at least one inoperable pair of vacuum breakers to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one position indicator of any suppression chamber - drywell vacuum breaker inoperable:

   1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, or

   2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.7 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed in accordance with the Surveillance Frequency Control Program.

b. Demonstrated OPERABLE:

1. In accordance with the Surveillance Frequency Control Program and within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. In accordance with the Surveillance Frequency Control Program by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.

3. In accordance with the Surveillance Frequency Control Program by:

a) Verifying each valve's opening setpoint, from the closed position, to be 0.5 psid ± 5%, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

c) Verifying that each outboard valve's position indicator is capable of detecting disk displacement ≥0.050", and each inboard valve's position indicator is capable of detecting disk displacement ≥0.120".

3/4.6.5 SECONDARY CONTAINMENT

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY, restore REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying in accordance with the Surveillance Frequency Control Program that the pressure within the reactor enclosure secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.*

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

   1. All reactor enclosure secondary containment equipment hatches and blowout panels are closed and sealed.

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

   3. All reactor enclosure secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.

c. In accordance with the Surveillance Frequency Control Program:

   1. Verifying that one standby gas treatment subsystem will draw down the reactor enclosure secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 916 seconds with the reactor enclosure recirc system in operation, and

   2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the reactor enclosure secondary containment at a flow rate not exceeding 2500 cfm with wind speeds of $\leq$ 7.0 mph as measured on the wind instrument on Tower 1, elevation 30' or, if that instrument is unavailable, Tower 2, elevation 159'.

*Not required to be met for 4 hours if analysis demonstrates one standby gas treatment subsystem is capable of establishing the required secondary containment vacuum.

3/4.6.5 SECONDARY CONTAINMENT

REFUELING AREA SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment.

ACTION:

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying in accordance with the Surveillance Frequency Control Program that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.*

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

   1. All refueling area secondary containment equipment hatches and blowout panels are closed and sealed.

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

   3. All refueling area secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.

c. In accordance with the Surveillance Frequency Control Program:

   Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm.

*Not required to be met for 4 hours if analysis demonstrates one standby gas treatment subsystem is capable of establishing the required secondary containment vacuum.

CONTAINMENT SYSTEMS

REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.1 The reactor enclosure secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one or more of the reactor secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable valves to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2.1 Each reactor enclosure secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

b. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

c. By verifying the isolation time to be within its limit in accordance with the Surveillance Frequency Control Program.

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE TRM.

REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.2 The refueling area secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment.

ACTION:

With one or more of the refueling area secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable valves to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, suspend handling of RECENTLY IRRADIATED FUEL in the refueling area secondary containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2.2 Each refueling area secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

b. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

c. By verifying the isolation time to be within its limit in accordance with the Surveillance Frequency Control Program.

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE TRM.

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS
SECTION HAS BEEN RELOCATED TO THE TRM.

CONTAINMENT SYSTEMS

STANDBY GAS TREATMENT SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

   1. With the Unit 1 diesel generator for one standby gas treatment subsystem inoperable for more than 30 days, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   3. With one standby gas treatment subsystem inoperable and the other standby gas treatment subsystem with an inoperable Unit 1 diesel generator, restore the inoperable subsystem to OPERABLE status or restore the inoperable Unit 1 diesel generator to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   4. With the Unit 1 diesel generators for both standby gas treatment system subsystems inoperable for more than 72 hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. When (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS:

   1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or suspend handling of irradiated fuel in the secondary containment and CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

   2. With both standby gas treatment subsystems inoperable, if in progress, suspend handling of irradiated fuel in the secondary containment and CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 5764 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 0.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

3. Verify that when the fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980.

4. Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gage while operating at a flow rate of 2400 cfm ± 10%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 0.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

d. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the fan starts and isolation valves necessary to draw a suction from the refueling area or the reactor enclosure recirculation discharge open on each of the following test signals, except for valves that are locked, sealed, or otherwise secured in the actuated position:

   a) Manual initiation from the control room, and

   b) Simulated automatic initiation signal.

3. Verifying that the temperature differential across each heater is $\geq$ 15°F when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 5764 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 5764 cfm ± 10%.

g. After any major system alteration:

1. Verify that when the SGTS fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III).

2. Verify that one standby gas treatment subsystem will drawdown reactor enclosure Zone II secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 916 seconds with the reactor enclosure recirculation system in operation and the adjacent reactor enclosure and refueling area zones are in their isolation modes.

REACTOR ENCLOSURE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent reactor enclosure recirculation subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one reactor enclosure recirculation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both reactor enclosure recirculation subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly.

b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 60,000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate of 60,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) an a relative humidity of 70%.

d. In accordance with the Surveillance Frequency Control Program by:

   1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the filter train at a flow rate of 60,000 cfm ± 10%, verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

   2. Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals, except for valves that are locked, sealed, or otherwise secured in the actuated position:

      a. Manual initiation from the control room, and

      b. Simulated automatic initiation signal.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 60,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 60,000 cfm ± 10%.

PRIMARY CONTAINMENT HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 DELETED

CONTAINMENT SYSTEMS

DRYWELL HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.2 Four independent drywell unit cooler hydrogen mixing subsystems (2AV212, 2BV212, 2GV212, 2HV212) shall be OPERABLE with each subsystem consisting of one unit cooler fan.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell unit cooler hydrogen mixing subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each drywell unit cooler hydrogen mixing subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Starting the system from the control room, and

b. Verifying that the system operates for at least 15 minutes.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.3 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours. The provision of Specification 3.0.4.c is applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The drywell and suppression chamber oxygen concentration shall be verified to be within the limit in accordance with the Surveillance Frequency Control Program.

*See Special Test Exception 3.10.5.

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent residual heat removal service water (RHRSW) system subsystems, with each subsystem comprised of:

a. Two OPERABLE RHRSW pumps, and

b. An OPERABLE flow path capable of taking suction from the RHR service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water through one Unit 2 RHR heat exchanger,

shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3, two subsystems.

b. In OPERATIONAL CONDITIONS 4 and 5, the subsystem(s) associated with systems and components required OPERABLE by Specification 3.4.9.2, 3.9.11.1, and 3.9.11.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one RHRSW pump inoperable, restore the inoperable pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHRSW pump in each subsystem inoperable, restore at least one of the inoperable RHRSW pumps to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one RHRSW subsystem otherwise inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE RHRSW pump within 72 hours or in accordance with the Risk Informed Completion Time Program, unless otherwise specified in a) or b) below**, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   a) When the 'A' RHRSW subsystem is inoperable to allow for repairs of the 'A' RHRSW subsystem piping, with Limerick Generating Station Unit 1 shutdown, reactor vessel head removed and reactor cavity flooded, the 72-hour Allowed Outage Time may be extended to 7 days or in accordance with the Risk Informed Completion Time Program once every other calendar year with the following compensatory measures established:

** Only one of these two Actions, either a.3.a) or a.3.b), may be entered on Unit 2 in a calendar year. However, if either Unit 1 TS LCO 3.7.1.1, Action a.3.a) or a.3.b) has previously been entered in the calendar year, then Unit 2 Action a.3.a) or a.3.b) may not be entered during that same calendar year.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

1) The following systems and subsystems will be protected in accordance with applicable station procedures:
   - 'B' RHRSW subsystem
   - 'B' ESW loop
   - 'B' and 'D' RHR subsystems
   - D12, D22, and D24 4kV buses and emergency diesel generators
   - Division 2 and Division 4 Safeguard DC, and

2) The 'A' and 'B' loop of ESW return flow shall be aligned to the operable 'B' RHRSW return header only. The ESW return valves to the 'B' RHRSW return header (i.e., HV-11-015A and HV-11-015B) will be administratively controlled in the open position and de-energized prior to entering the extended AOT. The ESW return valves to the 'A' RHRSW return header (i.e., HV-11-011A and HV-11-011B) will be administratively controlled in the closed position and de-energized as part of the work boundary.

b) When the 'B' RHRSW subsystem is inoperable to allow for repairs of the 'B' RHRSW subsystem piping, with Limerick Generating Station Unit 1 shutdown, reactor vessel head removed and reactor cavity flooded, the 72-hour Allowed Outage Time may be extended to 7 days or in accordance with the Risk Informed Completion Time Program once every other calendar year with the following compensatory measures established:

1) The following systems and subsystems will be protected in accordance with applicable station procedures:
   - 'A' RHRSW subsystem
   - 'A' ESW loop
   - 'A' and 'C' RHR subsystems
   - D11, D21, and D23 4kV buses and emergency diesel generators
   - Division 1 and Division 3 Safeguard DC, and

2) The 'A' and 'B' loop of ESW return flow shall be aligned to the operable 'A' RHRSW return header only. The ESW return valves to the 'A' RHRSW return header (i.e., HV-11-011A and HV-11-011B) will be administratively controlled in the open position and de-energized prior to entering the extended AOT. The ESW return valves to the 'B' RHRSW return header (i.e., HV-11-015A and HV-11-015B) will be administratively controlled in the closed position and de-energized as part of the work boundary.

4. With both RHRSW subsystems otherwise inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

*Whenever both RHRSW subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

5. With two RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. With three RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

7. With four RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 3 or 4 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

c. In OPERATIONAL CONDITION 5 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required residual heat removal service water system subsystem(s) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

*A RHRSW pump/diesel generator pair consists of a RHRSW pump and its associated diesel generator. If either a RHRSW pump or its associated diesel generator becomes inoperable, then the RHRSW pump/diesel generator pair is inoperable.

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least the following independent emergency service water system loops, with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and

b. An OPERABLE flow path capable of taking suction from the emergency service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water to the associated Unit 2 and common safety-related equipment,

shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3, two loops.

b. In OPERATIONAL CONDITIONS 4, 5, and *, one loop.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and *.

ACTION:

a. In OPERATION CONDITION 1, 2, or 3:

1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 45 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one emergency service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one emergency service water system loop otherwise inoperable, declare all equipment aligned to the inoperable loop inoperable**, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours# or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*When handling irradiated fuel in the secondary containment.

**The diesel generators may be aligned to the OPERABLE emergency service water system loop provided confirmatory flow testing has been performed. Those diesel generators not aligned to the OPERABLE emergency service water system loop shall be declared inoperable and the actions of 3.8.1.1 taken.

#During the extended Allowed Outage Time (AOT) specified by TS LCO 3.7.1.1, Action a.3.a) or a.3.b) to allow for RHRSW subsystem piping repairs, the 72-hour AOT for one inoperable emergency service water system loop may also be extended to 7 days or in accordance with the Risk Informed Completion Time Program for the same period.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

4. With three ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With four ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5:

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.

c. In OPERATIONAL CONDITION *

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or verify adequate cooling remains available for the diesel generators required to be OPERABLE or declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENT

4.7.1.2 At least the above required emergency service water system loop(s) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program by verifying that:

1. Each automatic valve actuates to its correct position on its appropriate ESW pump start signal. ***

2. Each pump starts automatically when its associated diesel generator starts.

---

* When handling irradiated fuel in the secondary containment.

** An ESW pump/diesel generator pair consists of an ESW pump and its associated diesel generator. If either an ESW pump or its associated diesel generator becomes inoperable, than the ESW pump/diesel generator pair is inoperable.

*** Except for valves that are locked, sealed, or otherwise secured in the actuated position.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

a. A minimum pond water level at or above elevation 250'-10" Mean Sea Level, and

b. A pond water temperature of less than or equal to 88°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and *.

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. In OPERATIONAL CONDITION *, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

a. By verifying the pond water level to be greater than its limit in accordance with the Surveillance Frequency Control Program.

b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:

1. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than or equal to 80°F; and

2. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than or equal to 85°F; and

3. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than 32°F.

c. By verifying all piping above the frost line is drained:

1. within one (1) hour after being used when ambient air temperature is below 40°F; or

2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

*When handling irradiated fuel in the secondary containment.

3/4.7.2 CONTROL ROOM SYSTEMS - COMMON SYSTEMS

3/4.7.2.1 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2.1 Two control room emergency fresh air supply system subsystems shall be OPERABLE.

NOTE: The main control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: All OPERATIONAL CONDITIONS and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With the Unit 1 diesel generator for one control room emergency fresh air supply subsystem inoperable for more than 30 days, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one control room emergency fresh air supply subsystem inoperable for reasons other than Condition a.5, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one control room emergency fresh air supply subsystem inoperable for reasons other than Condition a.5, and the other control room emergency fresh air supply subsystem with an inoperable Unit 1 diesel generator, restore the inoperable subsystem to OPERABLE status or restore the Unit 1 diesel generator to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With the Unit 1 diesel generators for both control room emergency fresh air supply subsystems inoperable for more than 72 hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With one or more control room emergency fresh air supply subsystems inoperable due to an inoperable CRE boundary,

   a. Initiate action to implement mitigating actions immediately or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours; and

   b. Within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological and chemical hazards will not exceed limits and actions to mitigate exposure to smoke hazards are taken or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours; and

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. Restore CRE boundary to operable status within 90 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. With both control room emergency fresh air supply subsystems inoperable for reasons other than Condition a.5, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or when RECENTLY IRRADIATED FUEL is being handled in the secondary containment:

1. With one control room emergency fresh air supply subsystem inoperable for reasons other than Condition b.3, restore the inoperable subsystem to OPERABLE status within 7 days, or initiate and maintain operation of the OPERABLE subsystem in the radiation isolation mode of operation.

2. With both control room emergency fresh air supply subsystems inoperable for reasons other than Condition b.3, immediately suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment. The provisions of Specification 3.0.3 are not applicable.

3. With one or more control room emergency fresh air subsystems inoperable due to an inoperable CRE boundary, immediately suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.2.1.1 Each control room emergency fresh air supply subsystem shall be demonstrated OPERABLE:

a. DELETED

b. In accordance with the Surveillance Frequency Control Program on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

c. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 3000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate of 3000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

e. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm ± 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Relocated to the TRM

3. Verifying that on each of the below radiation isolation mode actuation test signals, the subsystem automatically switches to the radiation isolation mode of operation:

   a) Outside air intake high radiation, and

   b) Manual initiation from control room.

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 3000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 3000 cfm ± 10%.

4.7.2.1.2 The control room envelope boundary shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Control Room Envelope Habitability Program by performance of control room envelope unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

3/4.7.2.2 CONTROL ROOM AIR CONDITIONING (AC) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2.2 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:

1. With the Unit 1 diesel generator for one control room AC subsystem inoperable for more than 30 days, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one control room AC subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one MCR AC subsystem inoperable and the other control room AC subsystem with an inoperable Unit 1 diesel generator, restore the inoperable subsystem to OPERABLE status or restore the inoperable Unit 1 diesel generator to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With the Unit 1 diesel generators for both control room AC subsystems inoperable for more than 72 hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With two control room AC subsystems inoperable:#

   a. Verify control room air temperature is less than 90°F Wet Bulb Globe Temperature at least once per 4 hours; and

   b. Restore one control room AC subsystem to OPERABLE status within 72 hours.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. When RECENTLY IRRADIATED FUEL is being handled in the secondary containment:

1. With one control room AC subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days; or immediately place the OPERABLE control room AC subsystem in operation; or immediately suspend movement of RECENTLY IRRADIATED FUEL assemblies in the secondary containment*.

2. With two control room AC subsystems inoperable, immediately suspend movement of recently irradiated fuel assemblies in the secondary containment.*

---

* The provisions of Specification 3.0.3 are not applicable.

\# Supplemental cooling provisions, if required, may be implemented under this condition. When Hazard Barriers are unable to perform their support function(s) to allow implementation of DWCW to CECW supplemental cooling, any supported system Limiting Conditions for Operation are not required to be declared not met solely for this reason.

CONTROL ROOM AIR CONDITIONING (AC) SYSTEM

SURVEILLANCE REQUIREMENTS

4.7.2.2 Each control room AC subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying each subsystem has the capability to remove the assumed heat load.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

b. DELETED

c. Specification 3.0.4.b is not applicable to RCIC.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying locations susceptible to gas accumulation are sufficiently filled with water.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.**

3. Verifying that the pump flow controller is in the correct position.

b. In accordance with the Surveillance Frequency Control Program by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1040 + 13, - 120 psig.*

---

* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 150 psig within the following 72 hours.

** Not required to be met for system vent flow paths opened under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program by:

1. Performing a system functional test which includes simulated automatic actuation and restart and verifying that each automatic valve in the flow path actuates to its correct position. ** Actual injection of coolant into the reactor vessel may be excluded.

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, - 0 psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.

4. Performing a CHANNEL CALIBRATION of the RCIC system discharge line "keep filled" level alarm instrumentation.

* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 150 psig within the following 72 hours.

** Except for valves that are locked, sealed, or otherwise secured in the actuated position.

3/4.7.4 DELETED

TABLE 4.7.4.1 (Deleted)

TECHNICAL SPECIFICATION PAGES 3/4 7-11a THROUGH 3/4 7-11b HAVE BEEN INTENTIONALLY OMITTED

SURVEILLANCE REQUIREMENTS (Continued)

THE INFORMATION FROM TECHNICAL SPECIFICATIONS SECTION 4.7.4.d HAS BEEN RELOCATED TO THE TECHNICAL REQUIREMENTS MANUAL (TRM) SNUBBERS SECTION. TECHNICAL SPECIFICATIONS PAGES 3/4 7-13 THROUGH 7-16 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.5.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcurie per test sample.

4.7.5.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below:

a. Sources in use - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source* and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.5.3 DELETED

*Except the Cf-252 startup sources which shall be tested within 6 months prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

Section 3/4.7.6 through 3/4.7.7 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAVE BEEN RELOCATED TO THE TECHNICAL REQUIREMENTS MANUAL (TRM) FIRE PROTECTION SECTION. TECHNICAL SPECIFICATIONS PAGES 3/4 7-19 THROUGH 3/4 7-32 HAVE BEEN INTENTIONALLY OMITTED.

3/4.7.8 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 The main turbine bypass system shall be OPERABLE as determined by the number of operable main turbine bypass valves being greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or take the ACTION required by Specification 3.2.3.c.

SURVEILLANCE REQUIREMENTS

4.7.8 The main turbine bypass system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program:

a. By cycling each turbine bypass valve through at least one complete cycle of full travel,

b. By performing a system functional test which includes simulated automatic actuation, and by verifying that each automatic valve actuates to its correct position, and

c. By determining TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to the value specified in the CORE OPERATING LIMITS REPORT.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Four separate and independent diesel generators, each with:

    1. A separate day tank containing a minimum of 250 gallons of fuel,

    2. A separate fuel storage system containing a minimum of 33,500 gallons of fuel, and

    3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 24 hours and at least once per 7 days thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining operable diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 for one diesel generator at a time, within 24 hours, unless the absence of any potential common-mode failure for the remaining diesel generators is determined. Restore the inoperable diesel generator to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. See also ACTION e.

b. With two diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either of the diesel generators became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 for one diesel generator at a time, within 8 hours, unless the absence of any potential common-mode failure for the remaining diesel generators is determined. Restore at least one of the inoperable diesel generators to OPERABLE status within 72 hours* or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. See also ACTION e.

*During the extended Allowed Outage Time (AOT) specified by TS LCO 3.7.1.1, Action a.3.a) or a.3.b) to allow for RHRSW subsystem piping repairs, the 72-hour AOT for two inoperable diesel generators may also be extended to 7 days or in accordance with the Risk Informed Completion Time Program for the same period.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With three diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and perform Surveillance Requirement 4.8.1.1.2.a.4 for the remaining diesel generator, within 1 hour. Restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. See also ACTION e.

d. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 for one diesel generator at a time, within 8 hours, unless the absence of any potential common-mode failure for the remaining diesel generators is determined. Restore at least two offsite circuits to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program from time of initial loss, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. See also ACTION e.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

e. In addition to the ACTIONS above:

1. For two train systems, with one or more diesel generators of the above required A.C. electrical power sources inoperable, verify within 2 hours and at least once per 12 hours thereafter that at least one of the required two train system subsystem, train, components, and devices is OPERABLE and its associated diesel generator is OPERABLE. Otherwise, restore either the inoperable diesel generator or the inoperable system subsystem to an OPERABLE status within 72 hours* or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI systems, with two or more diesel generators of the above required A.C. electrical power sources inoperable, verify within 2 hours and at least once per 12 hours thereafter that at least two of the required LPCI system subsystems, trains, components and devices are OPERABLE and its associated diesel generator is OPERABLE. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

This ACTION does not apply for those systems covered in Specifications 3.7.1.1 and 3.7.1.2.

*During the extended Allowed Outage Time (AOT) specified by TS LCO 3.7.1.1, Action a.3.a) or a.3.b) to allow for RHRSW subsystem piping repairs, the 72-hour AOT may also be extended to 7 days or in accordance with the Risk Informed Completion Time Program for the same period.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

f. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore at least two offsite circuits to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

g. With two of the above required offsite circuits inoperable, restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program from time of initial loss, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

h. With one offsite circuit and two diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either of the diesel generators became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 for one diesel generator at a time, within 8 hours, unless the absence of any potential common-mode failure for the remaining diesel generators is determined. Restore at least one of the above required inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and at least three of the above required diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. See also ACTION e.

i. Specification 3.0.4.b is not applicable to diesel generators.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank.

4. Verify that the diesel can start* and gradually accelerate to synchronous speed with steady-state generator voltage ≥ 4160 V and ≤ 4400 V and frequency ≥ 59.8 Hz and ≤ 60.8 Hz.

5. Verify diesel is synchronized, gradually loaded* to an indicated 2700-2800 kW** and operates with this load for at least 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 225 psig.

---

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

b. By removing accumulated water:

1) From the day tank in accordance with the Surveillance Frequency Control Program and after each occasion when the diesel is operated for greater than 1 hour, and

2) From the storage tank in accordance with the Surveillance Frequency Control Program.

c. By sampling new fuel oil in accordance with ASTM D4Ø57-81 prior to addition to the storage tanks and:

1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within Ø.3 degrees at 6Ø°F or a specific gravity of within Ø.ØØ16 at 6Ø/6Ø°F, when compared to the supplier's certificate or an absolute specific gravity at 6Ø/6Ø°F of greater than or equal to Ø.83 but less than or equal to Ø.89 or an API gravity at 6Ø°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 4Ø°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

d. In accordance with the Surveillance Frequency Control Program by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 1Ø mg/liter when checked in accordance with ASTM D2276-78, Method A, except that the filters specified in ASTM D2276-78, Sections 5.1.6 and 5.1.7, may have a nominal pore size of up to three (3) microns.

e. In accordance with the Surveillance Frequency Control Program by:

1) Deleted

2) Verifying each diesel generator's capability to reject a load of greater than or equal to that of its single largest post-accident load, and:

a) Following load rejection, the frequency is ≤ 66.5 Hz;

b) Within 1.8 seconds following the load rejection, the voltage is ≥ 3865 V and ≤ 47Ø5 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; and

c) After steady-state conditions are reached, voltage is maintained ≥ 416Ø V and ≤ 44ØØ V and frequency ≥ 59.8 Hz and ≤ 6Ø.8 Hz.

3. Verifying the diesel generator capability to reject a load of 2850 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection.

4. Simulating a loss-of-offsite power by itself, and:

    a) Verifying deenergization of the emergency buses and load shedding from the emergency buses.

    b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency buses within 10 seconds, energizes the auto-connected loads through the individual load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency buses is ≥ 4160 V and ≤ 4400 V and ≥ 59.8 Hz and ≤ 60.8 Hz during the test.

5. Verifying that on an ECCS actuation test signal, without loss-of-offsite power, the diesel generator starts* on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall reach ≥ 4160 V and ≤ 4400 V and ≥ 58.8 Hz and ≤ 61.2 Hz within 10 seconds after auto-start signal; the steady state generator voltage and frequency shall be maintained ≥ 4160 V and ≤ 4400 V and ≥ 59.8 Hz and ≤ 60.8 Hz during this test.

6. Simulating a loss-of-offsite power in conjunction with an ECCS actuation test signal, and:

    a) Verifying deenergization of the emergency buses and load shedding from the emergency buses.

    b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency buses within 10 seconds, energizes the auto-connected shutdown loads through the individual load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency buses shall be maintained at ≥ 4160 V and ≤ 4400 V and ≥ 59.8 Hz and ≤ 60.8 Hz during this test.

7. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential over-current are automatically bypassed upon an ECCS actuation signal.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warm up procedures, and as applicable regarding loading and shutdown recommendations.

8. a) Verifying the diesel generator operates* for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to an indicated 2950-3050 kW** and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 2700-2800 kW**.

   b) Verifying that, within 5 minutes of shutting down the diesel generator after the diesel generator has operated* for at least 2 hours at an indicated 2700-2800 kW**, the diesel generator starts*. The generator voltage and frequency shall reach ≥ 4160 V and ≤ 4400 V and ≥ 58.8 Hz and ≤ 61.2 Hz within 10 seconds after the start signal. After steady-state conditions are reached, voltage is maintained ≥ 4160 V and ≤ 4400 V and frequency is maintained ≥ 59.8 Hz and ≤ 60.8 Hz.

9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3100 kW.

10. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its standby status.

11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within ± 10% of its design interval.

---

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Control Room Switch In Pull-To-Lock (With Local/Remote Switch in Remote)

    b) Local/Remote Switch in Local.

    c) Emergency Stop

f. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting* all four diesel generators simultaneously, during shutdown, and verifying that all four diesel generators accelerate to at least 882 rpm in less than or equal to 10 seconds.

g. In accordance with the Surveillance Frequency Control Program by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

h. In accordance with the Surveillance Frequency Control Program the diesel generator shall be started* and verified to accelerate to synchronous speed in less than or equal to 10 seconds. The generator voltage and frequency shall reach ≥ 4160 V and ≤ 4400 V and ≥ 58.8 Hz and ≤ 61.2 Hz within 10 seconds after the start signal. After steady-state conditions are reached, voltage is maintained ≥ 4160 V and ≤ 4400 V and frequency is maintained ≥ 59.8 Hz and ≤ 60.8 Hz. The diesel generator shall be started for this test by using one of the following signals:

a) Manual***

b) Simulated loss-of-offsite power by itself.

c) Simulated loss-of-offsite power in conjunction with an ECCS actuation test signal.

d) An ECCS actuation test signal by itself.

The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 2700-2800 KW** and operate for at least 60 minutes. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

4.8.1.1.3 Deleted

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

***If diesel generator started manually from the control room, 10 seconds after the automatic prelube period.

INFORMATION ON THIS PAGE HAS BEEN DELETED

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators each with:

   1. A day fuel tank containing a minimum of 250 gallons of fuel.

   2. A fuel storage system containing a minimum of 33,500 gallons of fuel.

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1 and 4.8.1.1.2.

*When handling irradiated fuel in the secondary containment.

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division 1, Consisting of:
   1. 125-Volt Battery 2A1 (2A1D101).
   2. 125-Volt Battery 2A2 (2A2D101).
   3. 125-Volt Battery Charger 2BCA1 (2A1D103).
   4. 125-Volt Battery Charger 2BCA2 (2A2D103).

b. Division 2, Consisting of:
   1. 125-Volt Battery 2B1 (2B1D101).
   2. 125-Volt Battery 2B2 (2B2D101).
   3. 125-Volt Battery Charger 2BCB1 (2B1D103).
   4. 125-Volt Battery Charger 2BCB2 (2B2D103).

c. Division 3, Consisting of:
   1. 125-Volt Battery 2C (2CD101).
   2. 125-Volt Battery Charger 2BCC (2CD103).

d. Division 4, Consisting of:
   1. 125-Volt Battery 2D (2DD101).
   2. 125-Volt Battery Charger 2BCD (2DD103).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or two battery chargers on one division inoperable:

   1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours,

   2. Verify associated Division 1 or 2 float current $\leq$ 2 amps, or Division 3 or 4 float current $\leq$ 1 amp within 18 hours and once per 12 hours thereafter, and

   3. Restore battery charger(s) to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program.

b. With one or more batteries inoperable due to:

   1. One or two batteries on one division with one or more battery cells float voltage < 2.07 volts, perform 4.8.2.1.a.1 and 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore affected cell(s) voltage $\geq$ 2.07 volts within 24 hours.

   2. Division 1 or 2 with float current > 2 amps, or with Division 3 or 4 with float current > 1 amp, perform 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore battery float current to within limits within 18 hours.

ACTION: (Continued)

3. One or two batteries on one division with one or more cells electrolyte level less than minimum established design limits, if electrolyte level was below the top of the plates restore electrolyte level to above top of plates within 8 hours and verify no evidence of leakage(*) within 12 hours. In all cases, restore electrolyte level to greater than or equal to minimum established design limits within 31 days.

4. One or two batteries on one division with pilot cell electrolyte temperature less than minimum established design limits, restore battery pilot cell temperature to greater than or equal to minimum established design limits within 12 hours.

5. Batteries in more than one division affected, restore battery parameters for all batteries in all but one division to within limits within 2 hours.

6. (i) Any battery having both (Action b.1) one or more battery cells float voltage < 2.07 volts and (Action b.2) float current not within limits, and/or

   (ii) Any battery not meeting any Action b.1 through b.5,

   Restore the battery parameters to within limits within 2 hours.

c. With any battery(ies) on one division of the above required D.C. electrical power sources inoperable for reasons other than Action b., restore the inoperable division battery to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(*) Contrary to the provisions of Specification 3.0.2, if electrolyte level was below the top of the plates, the verification that there is no evidence of leakage is required to be completed regardless of when electrolyte level is restored.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required division batteries and chargers shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. Each Division 1 and 2 battery float current is $\leq$ 2 amps, and Division 3 and 4 battery float current is $\leq$ 1 amp when battery terminal voltage is greater than or equal to the minimum established float voltage of 4.8.2.1.a.2, and

2. Total battery terminal voltage for each 125-volt battery is greater than or equal to the minimum established float voltage.

b. In accordance with the Surveillance Frequency Control Program by verifying that:

1. Each battery pilot cell voltage is $\geq$ 2.07 volts,

2. Each battery connected cell electrolyte level is greater than or equal to minimum established design limits, and

3. The electrolyte temperature of each pilot cell is greater than or equal to minimum established design limits.

c. In accordance with the Surveillance Frequency Control Program by verifying that each battery connected cell voltage is $\geq$ 2.07 volts.

d. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The battery chargers will supply the currents listed below at greater than or equal to the minimum established float voltage for at least 4 hours:

Charger	Current (Amperes)
2BCA1	300
2BCA2	300
2BCB1	300
2BCB2	300
2BCC	75
2BCD	75

2. The battery capacity is adequate to supply and maintain in OPERABLE status the required emergency loads for the design duty cycle when subjected to a battery service test.

SURVEILLANCE REQUIREMENTS (Continued)

e. In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test or modified performance discharge test. The modified performance discharge test may be performed in lieu of the battery service test (Specification 4.8.2.1.d.2).

f. Performance discharge tests or modified performance discharge tests of battery capacity shall be given as follows:

   1. In accordance with the Surveillance Frequency Control Program when:
      (a) The battery shows degradation or
      (b) The battery has reached 85% of expected life with battery capacity < 100% of manufacturer's rating, and

   2. In accordance with the Surveillance Frequency Control Program when battery has reached 85% of expected life with battery capacity ≥ 100% of manufacturer's rating.

TABLE 4.8.2.1-1 (DELETED)

THE INFORMATION ON THE PAGE HAS BEEN DELETED

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four divisions of the D.C. electrical power sources system shall be OPERABLE with:

a. Division 1, Consisting of:
   1. 125-Volt Battery 2A1 (2A1D101).
   2. 125-Volt Battery 2A2 (2A2D101).
   3. 125-Volt Battery Charger 2BCA1 (2A1D103).
   4. 125-Volt Battery Charger 2BCA2 (2A2D103).

b. Division 2, Consisting of:
   1. 125-Volt Battery 2B1 (2B1D101).
   2. 125-Volt Battery 2B2 (2B2D101).
   3. 125-Volt Battery Charger 2BCB1 (2B1D103).
   4. 125-Volt Battery Charger 2BCB2 (2B2D103).

c. Division 3, Consisting of:
   1. 125-Volt Battery 2C (2CD101).
   2. 125-Volt Battery Charger 2BCC (2CD103).

d. Division 4, Consisting of:
   1. 125-Volt Battery 2D (2DD101).
   2. 125-Volt Battery Charger 2BCD (2DD103).

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With one or two required battery chargers on one required division inoperable:

   1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours,

   2. Verify associated Division 1 or 2 float current $\leq$ 2 amps, or Division 3 or 4 float current $\leq$ 1 amp within 18 hours and once per 12 hours thereafter, and

   3. Restore battery charger(s) to OPERABLE status within 7 days.

b. With one or more required batteries inoperable due to:

   1. One or two batteries on one division with one or more battery cells float voltage < 2.07 volts, perform 4.8.2.1.a.1 and 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore affected cell(s) voltage $\geq$ 2.07 volts within 24 hours.

*When handling irradiated fuel in the secondary containment.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. Division 1 or 2 with float current > 2 amps, or with Division 3 or 4 with float current > 1 amp, perform 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore battery float current to within limits within 18 hours.

3. One or two batteries on one division with one or more cells electrolyte level less than minimum established design limits, if electrolyte level was below the top of the plates restore electrolyte level to above top of plates within 8 hours and verify no evidence of leakage(*) within 12 hours. In all cases, restore electrolyte level to greater than or equal to minimum established design limits within 31 days.

4. One or two batteries on one division with pilot cell electrolyte temperature less than minimum established design limits, restore battery pilot cell temperature to greater than or equal to minimum established design limits within 12 hours.

5. Batteries in more than one division affected, restore battery parameters for all batteries in one division to within limits within 2 hours.

6. (i) Any battery having both (Action b.1) one or more battery cells float voltage < 2.07 volts and (Action b.2) float current not within limits, and/or

   (ii) Any battery not meeting any Action b.1 through b.5,

   Restore the battery parameters to within limits within 2 hours.

c. 1. With the requirements of Action a. and/or Action b. not met, or

   2. With less than two divisions of the above required D.C. electrical power sources OPERABLE for reasons other than Actions a. and/or b.,

   Suspend CORE ALTERATIONS and handling of irradiated fuel in the secondary containment.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required batteries and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

(*) Contrary to the provisions of Specification 3.0.2, if electrolyte level was below the top of the plates, the verification that there is no evidence of leakage is required to be completed regardless of when electrolyte level is restored.

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized:

a. A.C. power distribution:

1. Unit 2 Division 1, Consisting of:

a)	4160-VAC Bus:	D21 (20A115)
b)	480-VAC Load Center:	D214 (20B201)
c)	480-VAC Motor Control Centers:	D214-R-C (20B213) D214-R-G (20B211) D214-R-G1 (20B215) D214-D-G (20B515)
d)	120-VAC Distribution Panels:	20Y101 20Y206

2. Unit 2 Division 2, Consisting of:

a)	4160-VAC Bus:	D22 (20A116)
b)	480-VAC Load Center:	D224 (20B202)
c)	480-VAC Motor Control Centers:	D224-R-C (20B214) D224-R-G (20B212) D224-R-G1 (20B216) D224-D-G (20B516)
d)	120-VAC Distribution Panels:	20Y102 20Y207

3. Unit 2 Division 3, Consisting of:

a)	4160-VAC Bus:	D23 (20A117)
b)	480-VAC Load Center:	D234 (20B203)
c)	480-VAC Motor Control Centers:	D234-R-H1 (20B221) D234-R-H (20B217) D234-R-E (20B223) D234-D-G (20B517)
d)	120-VAC Distribution Panels:	20Y103 20Y163

4. Unit 2 Division 4, Consisting of:

a)	4160-VAC Bus:	D24 (20A118)
b)	480-VAC Load Center:	D244 (20B204)
c)	480-VAC Motor Control Centers:	D244-R-H1 (20B222) D244-R-H (20B218) D244-R-E (20B224) D244-D-G (20B518)
d)	120-VAC Distribution Panels:	20Y104 20Y164

5. Unit 1 and Common Division 1, Consisting of:

a)	4160-VAC Bus:	D11	(10A115)
b)	480-VAC Load Center:	D114	(10B201)
c)	480-VAC Motor Control Centers:	D114-R-C	(10B213)
		D114-R-C1	(10B219)
		D114-D-G	(10B515)
		D114-S-L	(00B519)
d)	120-VAC Distribution Panels:	10Y101	
		10Y206	
		01Y501	

6. Unit 1 and Common Division 2, Consisting of:

a)	4160-VAC Bus:	D12	(10A116)
b)	480-VAC Load Center:	D124	(10B202)
c)	480-VAC Motor Control Centers:	D124-R-C	(10B214)
		D124-R-C1	(10B220)
		D124-D-G	(10B516)
		D124-S-L	(00B520)
d)	120-VAC Distribution Panels:	10Y102	
		10Y207	
		02Y501	

7. Unit 1 and Common Division 3, Consisting of:

a)	4160-VAC Bus:	D13	(10A117)
b)	480-VAC Load Center:	D134	(10B203)
c)	480-VAC Motor Control Centers:	D134-R-E	(10B223)
		D134-C-B	(00B131)
		D134-D-G	(10B517)
		D234-S-L	(00B521)
d)	120-VAC Distribution:	10Y103	
		10Y163	
		03Y501	

8. Unit 1 and Common Division 4, Consisting of:

a)	4160-VAC Bus:	D14	(10A118)
b)	480-VAC Load Center:	D144	(10B204)
c)	480-VAC Motor Control Centers:	D144-R-E	(10B224)
		D144-C-B	(00B132)
		D144-D-G	(10B518)
		D244-S-L	(00B522)
d)	120-VAC Distribution:	10Y104	
		10Y164	
		04Y501	

LIMITING CONDITION FOR OPERATION (Continued)

b. D.C. Power Distribution Panels

1. Unit 2 Division 1, Consisting of:

a)	250-V DC Fuse Box:	2FA	(2AD105)
b)	250-V DC Motor Control Center:	2DA	(20D201)
c)	125-V DC Distribution Panels:	2PPA1	(2AD102)
		2PPA2	(2AD501)
		2PPA3	(2AD162)

2. Unit 2 Division 2, Consisting of:

a)	250-V DC Fuse Box:	2FB	(2BD105)
b)	250-V DC Motor Control Centers:	2DB-1	(20D202)
		2DB-2	(20D203)
c)	125-V DC Distribution Panels:	2PPB1	(2BD102)
		2PPB2	(2BD501)
		2PPB3	(2BD162)

3. Unit 2 Division 3, Consisting of:

a)	125-V DC Fuse Box:	2FC	(2CD105)
b)	125-V DC Distribution Panels:	2PPC1	(2CD102)
		2PPC2	(2CD501)
		2PPC3	(2CD162)

4. Unit 2 Division 4, Consisting of:

a)	125-V DC Fuse Box:	2FD	(2DD105)
b)	125-V DC Distribution Panels:	2PPD1	(2DD102)
		2PPD2	(2DD501)
		2PPD3	(2DD162)

5. Unit 1 and Common Division 1, Consisting of:

a)	250-V DC Fuse Box:	1FA	(1AD105)
b)	125-V DC Distribution Panels:	1PPA1	(1AD102)
		1PPA2	(1AD501)

6. Unit 1 and Common Division 2, Consisting of:

a)	250-V DC Fuse Box:	1FB	(1BD105)
b)	125-V DC Distribution Panels:	1PPB1	(1BD102)
		1PPB2	(1BD501)

7. Unit 1 and Common Division 3, Consisting of:

a)	125-V DC Fuse Box:	1FC	(1CD105)
b)	125-V DC Distribution Panels:	1PPC1	(1CD102)
		1PPC2	(1CD501)

8. Unit 1 and Common Division 4, Consisting of:

a)	125-V DC Fuse Box:	1FD	(1DD105)
b)	125-V DC Distribution Panels:	1PPD1	(1DD102)
		1PPD2	(1DD501)

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one of the above required Unit 2 A.C. distribution system divisions not energized, reenergize the division within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one of the above required Unit 2 D.C. distribution system divisions not energized, reenergize the division within 8 hours or in accordance with the Risk Informed completion Time Program, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any of the above required Unit 1 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system divisions shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, 2 of the 4 divisions of the power distribution system shall be energized with:

a. A.C. power distribution:

1. Unit 2 Division 1, Consisting of:

a)	4160-VAC Bus:	D21 (20A115)
b)	480-VAC Load Center:	D214 (20B201)
c)	480-VAC Motor Control Centers:	D214-R-C (20B213) D214-R-G (20B211) D214-R-G1 (20B215) D214-D-G (20B515)
d)	120-VAC Distribution Panels:	20Y101 20Y206

2. Unit 2 Division 2, Consisting of:

a)	4160-VAC Bus:	D22 (20A116)
b)	480-VAC Load Center:	D224 (20B202)
c)	480-VAC Motor Control Centers:	D224-R-C (20B214) D224-R-G (20B212) D224-R-G1 (20B216) D224-D-G (20B516)
d)	120-VAC Distribution Panels:	20Y102 20Y207

3. Unit 2 Division 3, Consisting of:

a)	4160-VAC Bus:	D23 (20A117)
b)	480-VAC Load Center:	D234 (20B203)
c)	480-VAC Motor Control Centers:	D234-R-H1 (20B221) D234-R-H (20B217) D234-R-E (20B223) D234-D-G (20B517)
d)	120-VAC Distribution Panels:	20Y103 20Y163

4. Unit 2 Division 4, Consisting of:

a)	4160-VAC Bus:	D24 (20A118)
b)	480-VAC Load Center:	D244 (20B204)
c)	480-VAC Motor Control Centers:	D244-R-H1 (20B222) D244-R-H (20B218) D244-R-E (20B224) D244-D-G (20B518)
d)	120-VAC Distribution Panels:	20Y104 20Y164

5. Unit 1 and Common Division 1, Consisting of:

a)	4160-VAC Bus:	D11	(10A115)
b)	480-VAC Load Center:	D114	(10B201)

LIMITING CONDITION FOR OPERATION (Continued)

c)	480-VAC Motor Control Centers:	D114-R-C	(10B213)
		D114-R-C1	(10B219)
		D114-D-G	(10B515)
		D114-S-L	(00B519)
d)	120-VAC Distribution Panels:	10Y101	
		10Y206	
		01Y501	

6. Unit 1 and Common Division 2, Consisting of:

a)	4160-VAC Bus:	D12	(10A116)
b)	480-VAC Load Center:	D124	(10B202)
c)	480-VAC Motor Control Centers:	D124-R-C	(10B214)
		D124-R-C1	(10B220)
		D124-D-G	(10B516)
		D124-S-L	(00B520)
d)	120-VAC Distribution Panels:	10Y102	
		10Y207	
		02Y501	

7. Unit 1 and Common Division 3, Consisting of:

a)	4160-VAC Bus:	D13	(10A117)
b)	480-VAC Load Center:	D134	(10B203)
c)	480-VAC Motor Control Centers:	D134-R-E	(10B223)
		D134-C-B	(00B131)
		D134-D-G	(10B517)
		D234-S-L	(00B521)
d)	120-VAC Distribution:	10Y103	
		10Y163	
		03Y501	

8. Unit 1 and Common Division 4, Consisting of:

a)	4160-VAC Bus:	D14	(10A118)
b)	480-VAC Load Center:	D144	(10B204)
c)	480-VAC Motor Control Centers:	D144-R-E	(10B224)
		D144-C-B	(00B132)
		D144-D-G	(10B518)
		D244-S-L	(00B522)
d)	120-VAC Distribution:	10Y104	
		10Y164	
		04Y501	

b. D.C. power distribution:

1. Unit 2 Division 1, Consisting of:

a)	250-V DC Fuse Box:	2FA	(2AD105)
b)	250-V DC Motor Control Center:	2DA	(2OD201)

LIMITING CONDITION FOR OPERATION (Continued)

c)	125-V DC Distribution Panels:	2PPA1 2PPA2 2PPA3	(2AD102) (2AD501) (2AD162)

2. Unit 2 Division 2, Consisting of:

a)	250-V DC Fuse Box:	2FB	(2BD105)
b)	250-V DC Motor Control Centers:	2DB-1 2DB-2	(20D202) (20D203)
c)	125-V DC Distribution Panels:	2PPB1 2PPB2 2PPB3	(2BD102) (2BD501) (2BD162)

3. Unit 2 Division 3, Consisting of:

a)	125-V DC Fuse Box:	2FC	(2CD105)
b)	125-V DC Distribution Panels:	2PPC1 2PPC2 2PPC3	(2CD102) (2CD501) (2CD162)

4. Unit 2 Division 4, Consisting of:

a)	125-V DC Fuse Box:	2FD	(2DD105)
b)	125-V DC Distribution Panels:	2PPD1 2PPD2 2PPD3	(2DD102) (2DD501) (2DD162)

5. Unit 1 and Common Division 1, Consisting of:

a)	250-V DC Fuse Box:	1FA	(1AD105)
b)	125-V DC Distribution Panels:	1PPA1 1PPA2	(1AD102) (1AD501)

6. Unit 1 and Common Division 2, Consisting of:

a)	250-V DC Fuse Box:	1FB	(1BD105)
b)	125-V DC Distribution Panels:	1PPB1 1PPB2	(1BD102) (1BD501)

7. Unit 1 and Common Division 3, Consisting of:

a)	125-V DC Fuse Box:	1FC	(1CD105)
b)	125-V DC Distribution Panels:	1PPC1 1PPC2	(1CD102) (1CD501)

8. Unit 1 and Common Division 4, Consisting of:

a)	125-V DC Fuse Box:	1FD	(1DD105)
b)	125-V DC Distribution Panels:	1PPD1 1PPD2	(1DD102) (1DD501)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With less than two divisions of the above required Unit 2 A.C. distribution systems energized, suspend CORE ALTERATIONS and handling of irradiated fuel in the secondary containment.

*When handling irradiated fuel in the secondary containment.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With less than two divisions of the above required Unit 2 D.C. distribution systems energized, suspend CORE ALTERATIONS and handling of irradiated fuel in the secondary containment.

c. With any of the above required Unit 1 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

Section 3/4 8.4.1 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATION SECTION HAS BEEN RELOCATED TO THE TRM. TECHNICAL SPECIFICATIONS PAGES 3/4 8-21 THROUGH 3/4 8-26 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

Section 3/4 8.4.2 (Deleted)

PAGE INTENTIONALLY LEFT BLANK

REACTOR PROTECTION SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two reactor protection system (RPS) electric power monitoring channels for each inservice RPS Inverter or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS Inverter or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS Inverter or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS Inverter or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 24 hours or remove the associated RPS Inverter or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.

b. In accordance with the Surveillance Frequency Control Program by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers and verifying the following Allowable Values.

   1. Overvoltage ≤ 127.6 VAC,

   2. Undervoltage ≥ 110.7 VAC,

   3. Underfrequency ≥ 57.05 Hz.

3.4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. The Refuel position one-rod-out interlock shall be OPERABLE.

b. The following Refuel position interlocks shall be OPERABLE:

1. All rods in.
2. Refuel Platform (over-core) position.
3. Refuel Platform hoists fuel-loaded.
4. Service Platform hoist fuel-loaded (with Service Platform installed).

APPLICABILITY: OPERATIONAL CONDITION 5* **, OPERATIONAL CONDITIONS 3 AND 4 when the reactor mode switch is in the Refuel position.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, verify all control rods are fully inserted and disable withdraw capabilities of all control rods ***, or lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel Platform Refuel position interlocks inoperable, take one of the ACTIONS listed below, or suspend CORE ALTERATIONS.

1. Verify control rods are fully inserted and disable withdraw capabilities of all control rods***, or
2. Verify Refuel Platform is not over-core (limit switches not reached) and disable Refuel Platform travel over-core, or
3. Verify that no Refuel Platform hoist is loaded and disable all Refuel Platform hoists from picking up (grappling) a load.

d. With the Service Platform installed over the vessel and any of the above required Service Platform Refuel position interlocks inoperable, take one of the ACTIONS listed below, or suspend CORE ALTERATIONS.

1. Verify all control rods are fully inserted and disable withdraw capabilities of all control rods***, or
2. Verify Service Platform hoist is not loaded and disable Service Platform hoist from picking up (grappling) a load.

---

* See Special Test Exceptions 3.10.1 and 3.10.3.

** The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

*** Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified in accordance with the Surveillance Frequency Control Program.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

*The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least two source range monitor (SRM) channels* shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. At least one with audible alarm in the control room,

c. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

d. Unless adequate SHUTDOWN MARGIN has been demonstrated, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.**

APPLICABILITY: OPERATIONAL CONDITION 5.***

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. In accordance with the Surveillance Frequency Control Program:

1. Performance of a CHANNEL CHECK,

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

---

*These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core. The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

***See Special Test Exception, Specification 3/4.10.7.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

c. Verifying that the channel count rate is at least 3.0 cps:*

   1. Prior to control rod withdrawal,

   2. Prior to and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS, and

   3. In accordance with the Surveillance Frequency Control Program.

d. Verifying, within 8 hours prior to and in accordance with the Surveillance Frequency Control Program, that the RPS circuitry "shorting links" have been removed during:

   1. The time any control rod is withdrawn**, unless adequate shutdown margin has been demonstrated, or

   2. Shutdown margin demonstrations.

*May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1. These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified in accordance with the Surveillance Frequency Control Program.

*Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**See Special Test Exception 3.10.3.

3/4.9.4 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor shall be subcritical for at least 24 hours.

APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS.*

*Except movement of control rods with their normal drive system.

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling platform main hoist used for handling of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1150 ± 50 pounds.

b. Demonstrating operation of the hoist loaded control rod block interlock on the main hoist when the load exceeds 485 ± 50 pounds.

c. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 + 0, - 115 pounds.

d. Demonstrating operation of the uptravel interlock when uptravel brings the top of the active fuel to not less than 8 feet 0 inches below the normal water level.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.6.2 The refueling platform frame-mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the frame mounted hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the frame mounted hoist when uptravel brings the top of a control rod to not less than 6 feet 6 inches below the normal fuel storage pool water level.

4.9.6.3 The refueling platform monorail mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the monorail hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the monorail hoist when uptravel brings the top of a control rod to not less than 6 feet 6 inches below the normal fuel storage pool water level.

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks which prevent crane travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and in accordance with the Surveillance Frequency Control Program during crane operation.

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 22 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth in accordance with the Surveillance Frequency Control Program.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 One (1) RHR shutdown cooling subsystem shall be OPERABLE and in operation. *

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet above the top of the reactor pressure vessel flange.

ACTION:

a. With the required RHR shutdown cooling subsystem inoperable:

   1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available.

b. With the required action and associated completion time of Action "a" above not met.

   1. Immediately suspend loading of irradiated fuel assemblies into the reactor pressure vessel; and

   2. Immediately initiate action to restore REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY to OPERABLE status; and

   3. Immediately initiate action to restore one (1) Standby Gas Treatment subsystem to OPERABLE status; and

   4. Immediately initiate action to restore isolation capability in each required Refueling Floor secondary containment penetration flow path not isolated.

c. With no RHR shutdown cooling subsystem in operation:

   1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulation by an alternate method; and

   2. Once per hour monitor reactor coolant temperature.

SURVEILLANCE REQUIREMENTS

4.9.11.1.1 At least one (1) RHR shutdown cooling subsystem, or an alternate method, shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.9.11.1.2 Verify required RHR shutdown cooling subsystem locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

---

* The required RHR shutdown cooling subsystem may be removed from operation for up to two (2) hours per eight (8) hour period.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two (2) RHR shutdown cooling subsystems shall be OPERABLE, and one (1) RHR shutdown cooling subsystem shall be in operation. *

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet above the top of the reactor pressure vessel flange.

ACTION:

a. With one (1) or two (2) required RHR shutdown cooling subsystems inoperable:

   1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.

b. With the required action and associated completion time of Action "a" above not met:

   1. Immediately initiate action to restore REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY to OPERABLE status; and

   2. Immediately initiate action to restore one (1) Standby Gas Treatment subsystem to OPERABLE status; and

   3. Immediately initiate action to restore isolation capability in each required Refueling Floor secondary containment penetration flow path not isolated.

c. With no RHR shutdown cooling subsystem in operation:

   1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulation by an alternate method; and

   2. Once per hour monitor reactor coolant temperature.

SURVEILLANCE REQUIREMENTS

4.9.11.2.1 At least one (1) RHR shutdown cooling subsystem, or an alternate method, shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.9.11.2.2 Verify RHR shutdown cooling subsystem locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

---

* The required operating shutdown cooling subsystem may be removed from operation for up to two (2) hours per eight (8) hour period.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3, and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during low power PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod worth minimizer (RWM) per Specification 3.1.4.1 may be suspended for the following tests provided that control rod movement prescribed for this testing is verified by a second licensed operator or other technically qualified member of the unit technical staff present at the reactor console:

a. Shutdown margin demonstration, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER.

ACTION:

With the requirements of the above specifications not satisfied, verify that the RWM is OPERABLE per Specification 3.1.4.1.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RWM are bypassed, verify:

a. That movement of control rods is blocked or limited to the approved control rod withdrawal sequence during scram and friction tests.

b. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3.

c. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3, and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 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.

c. The "continuous rod withdrawal" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 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 procedures, and

c. No other CORE ALTERATIONS are in progress.

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.3 may be suspended until completion of the Startup Test Program or the reactor has operated for 120 Effective Full Power Days.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The Effective Full Power Days of operation shall be verified to be less than 120, by calculation, in accordance with the Surveillance Frequency Control Program during the Startup Test Program.

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during training startups.

SPECIAL TEST EXCEPTIONS

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

LIMITING CONDITION FOR OPERATION

3.10.7 During initial core loading within the Startup Test Program the provisions of Specification 3.9.2 may be suspended provided that at least two source range monitor (SRM) channels with detectors inserted to the normal operating level are OPERABLE with:

a. One of the required SRM channels continuously indicating* in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant,**

c. The RPS "shorting links" shall be removed prior to and during fuel loading,

d. The reactor mode switch is OPERABLE and locked in the REFUEL position.

APPLICABILITY: OPERATIONAL CONDITION 5

ACTION:

With the requirements of the above specifications not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.10.7 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. Within 1 hour prior to and at least once per 12 hours during CORE ALTERATIONS:

   1. Performance of a CHANNEL CHECK***

   2. Confirming that the above required SRM detectors are at the normal operating level and located in the quadrants required by Specification 3.10.7.

---

*Up to 16 fuel bundles may be loaded without a visual indication of count rate.

**The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

***Check may be performed by use of movable neutron source. Movement of the movable neutron source is not a CORE ALTERATION.

SURVEILLANCE REQUIREMENTS (Continued)

4.10.7 (Continued)

3. The RPS "shorting links" are removed.

4. The reactor mode switch is locked in the REFUEL position.

b. Performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start and at least once per 7 days during CORE ALTERATIONS.

c. Verifying for at least one SRM channel that the count rate is at least 0.7 cps*:

1. Immediately following the loading of the first 16 fuel bundles.

2. At least once per 12 hours thereafter during CORE ALTERATIONS.

*Provided signal-to-noise is $\geq$ 2 (for initial startup only). Otherwise, 3 cps.

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

LIMITING CONDITION FOR OPERATION

3.10.8 When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to greater than 200°F, and operation considered not to be in OPERATIONAL CONDITION 3:

- For performance of an inservice leak or hydrostatic test,
- As a consequence of maintaining adequate pressure for an inservice leak or hydrostatic test, or
- As a consequence of maintaining adequate pressure for control rod scram time testing initiated in conjunction with an inservice leak or hydrostatic test,

provided the following OPERATIONAL CONDITION 3 Specifications are met:

a. 3.3.2 ISOLATION ACTUATION INSTRUMENTATION, Functions 7.a, 7.c.1, 7.c.2 and 7.d of Table 3.3.2-1;
b. 3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY;
c. 3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY;
d. 3.6.5.2.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES;
e. 3.6.5.2.2 REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES; and
f. 3.6.5.3 STANDBY GAS TREATMENT SYSTEM.

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature greater than 200°F.

ACTION:

With the requirements of the above Specifications not satisfied:

1. Immediately enter the applicable (OPERATIONAL CONDITION 3) action for the affected Specification; or

2. Immediately suspend activities that could increase the average reactor coolant temperature or pressure and reduce the average reactor coolant temperature to 200°F or less within 24 hours.

SURVEILLANCE REQUIREMENTS

4.10.8 Verify applicable OPERATIONAL CONDITION 3 surveillances for the Specifications listed in 3.10.8 are met.

Section 3/4 11.1.1 through 3/4 11.1.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-2 THROUGH 3/4 11-6 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

effective January 2, 1991

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

Section 3/4 11.2.1 through Section 3/4 11.2.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-9 THROUGH 3/4 11-14 OF THESE SECTIONS HAVE BEEN INTENTIONALLY OMITTED.

effective January 2, 1991

Section 3/4.11.2.5 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE TRM.

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.6 The rate of the sum of the activities of the noble gases Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, and Xe-138 measured at the recombiner after-condenser discharge shall be limited to less than or equal to 330 millicuries/second.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3*.

ACTION:

With the rate of the sum of the activities of the specified noble gases at the recombiner after-condenser discharge exceeding 330 millicuries/second, 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 REQUIREMENTS

4.11.2.6.1 Relocated to the ODCM.

4.11.2.6.2 The rate of the sum of the activities of the specified noble gases from the recombiner after-condenser discharge shall be determined to be within the limits of Specification 3.11.2.6 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner after condenser discharge:

a. In accordance with the Surveillance Frequency Control Program.

b. Within 4 hours following a noted increase of greater than 50%, after factoring out increases due to changes in THERMAL POWER level or air in-leakage, in the nominal steady-state fission gas release from the primary coolant.

c. The provisons of Specification 4.0.4 are not applicable.

*When the main condenser air ejector is in operation.

Section 3/4 11-2.7 (Deleted)

THE INFORMATION FROM THIS
TECHNICAL SPECIFICATIONS
SECTION HAS BEEN RELOCATED
TO THE ODCM.

effective January 2, 1991

Section 3/4 11.3 through 3/4 11.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL
SPECIFICATIONS SECTIONS HAS BEEN RELOCATED
TO THE PCP OR ODCM. TECHNICAL SPECIFICATIONS
PAGES 3/4 11-19 THROUGH 3/4 11-20 OF THESE
SECTIONS HAVE BEEN INTENTIONALLY OMITTED.

effective January 2, 1991

Section 3/4.12 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 12-2 THROUGH 3/4 12-14 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

effective January 2, 1991

SECTION 5.0

DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBER OF THE PUBLIC, shall be as shown in Figures 5.1.3-1a and 5.1.3-1b.

5.1.4 (Deleted)

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined reinforced concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined reinforced concrete vessel in a shape of a truncated cone on top of a water filled suppression chamber and is separated by a diaphragm slab and connected to the suppression chamber through a series of downcomer vents. The drywell has a maximum free air volume of 243,580 cubic feet at a minimum suppression pool level of 22 feet. The suppression chamber has a maximum air region of 159,540 cubic feet and a minimum water region of 122,120 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 55 psig.

b. Maximum internal temperature: drywell 340°F.
suppression pool 220°F.

c. Maximum external to internal differential pressure 5 psid.

d. Maximum floor differential pressure: 30 psid, downward.
20 psid, upward.

effective January 2, 1991

FIGURE 5.1.1-1

EXCLUSION AREA

FIGURE 5.1.2-1

LOW POPULATION ZONE

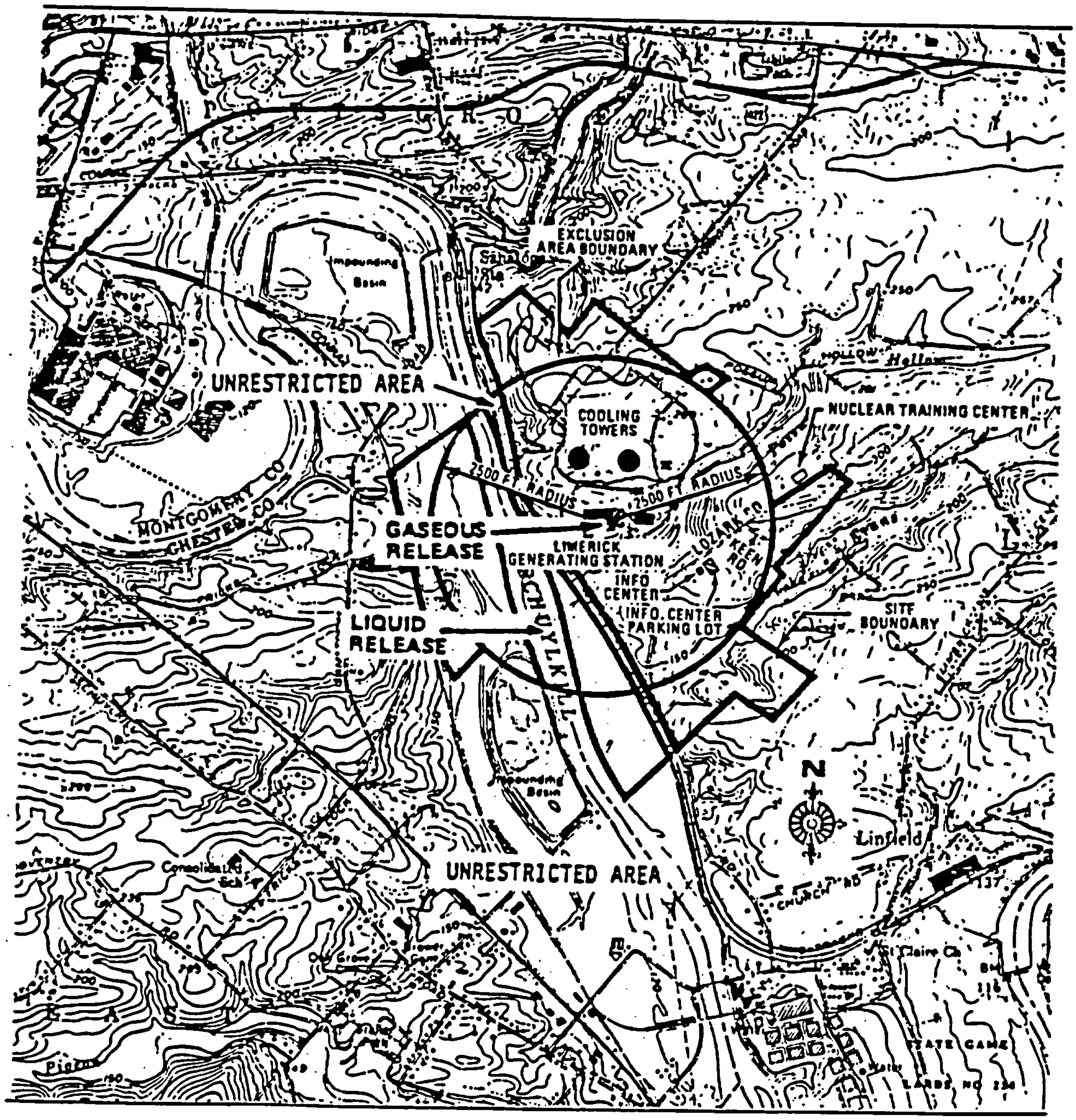


FIGURE 5.1.3-1a

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

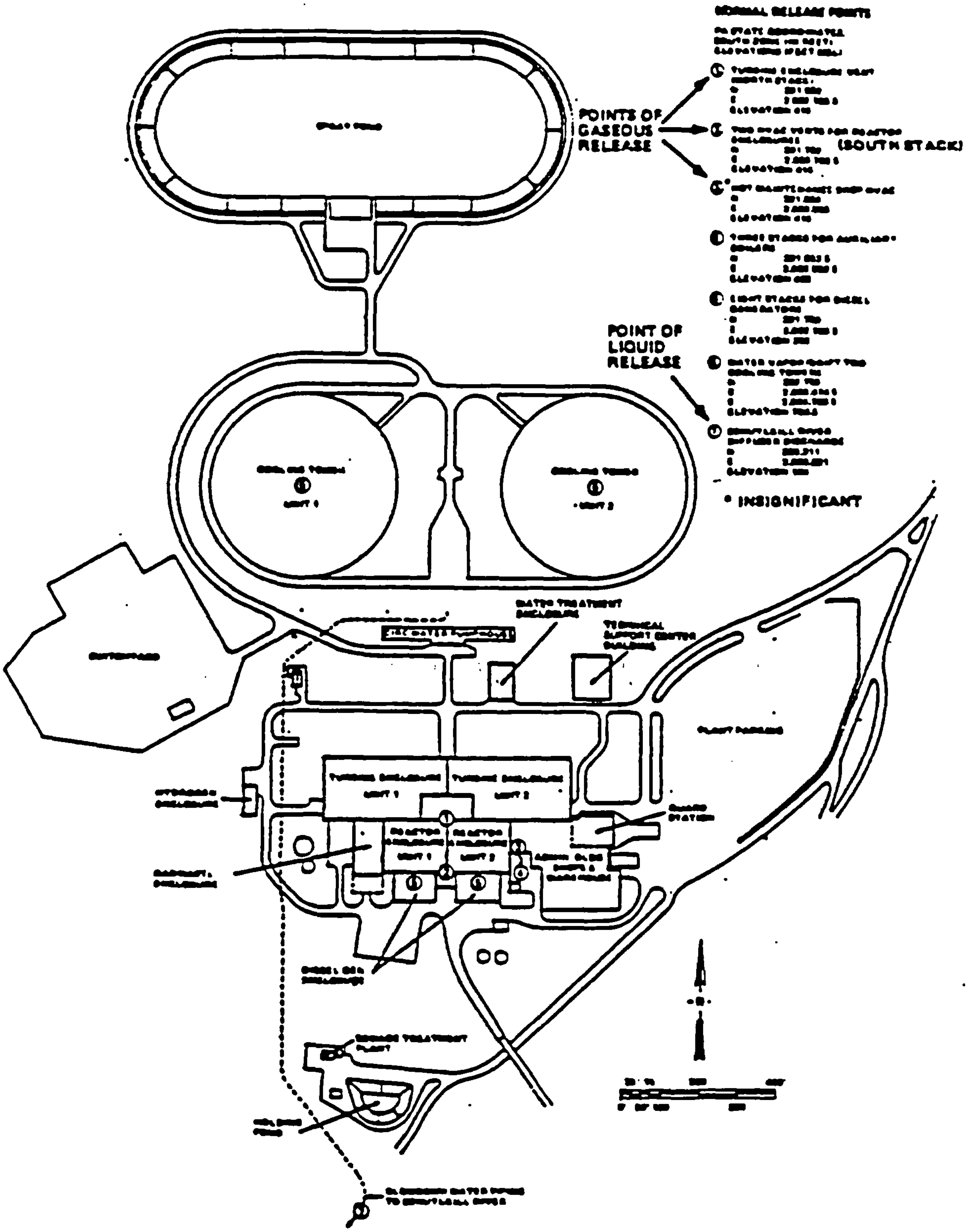


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

THE FIGURE ON THIS PAGE HAS BEEN RELOCATED TO THE ODCM.

effective January 2, 1991

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of three distinct isolatable zones. Zones I and II are the Unit 1 and Unit 2 reactor enclosures respectively. Zone III is the common refueling area. Each zone has an independent normal ventilation system which is capable of providing secondary containment zone isolation as required.

Each reactor enclosure (Zone I or II) completely encloses and provides secondary containment for its corresponding primary containment and reactor auxiliary or service equipment, and has a minimum free volume of 1,800,000 cubic feet.

The common refueling area (Zone III) completely encloses and provides secondary containment for the refueling servicing equipment and spent fuel storage facilities for Units 1 and 2, and has a minimum free volume of 2,200,000 cubic feet.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall consist of not more than 764 fuel assemblies and shall be limited to those fuel assemblies which have been analyzed with NRC approved codes and methods and have been shown to comply with all Safety Design Bases in the Final Safety Analysis Report (FSAR).

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform-shaped control rod assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements, DESIGN PRESSURE AND TEMPERATURE (Continued)

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.

2. 1500 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal steam dome saturation temperature of 552°F.

5.5 FUEL STORAGE

CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal center-to-center distance between fuel assemblies placed in the storage racks of greater than or equal to 6.244 inches.

5.5.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.5.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 346'0".

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4117 fuel assemblies.

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6.1-1.

TABLE 5.6.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	130 hydrostatic pressure and leak tests	Pressurized to $\geq$ 930 and $\leq$ 1250 psig

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager, or during his absence from the control room, a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President, Limerick Generating Station shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE 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 established and defined for the highest management levels through intermediate levels to and including all operating organizational positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Limerick Quality Assurance Program.

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

c. The Vice President, Limerick Generating Station 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 and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2, or 3, at least one licensed Senior Operator shall be in the control room;

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

d. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

f. (Deleted)

---

* The Health Physics Technician position may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

6.2.2 UNIT STAFF (Continued)

g. The individual filling the position of Operations Manager as defined by ANSI/ANS-3.1-1978 or another Manager in Operations shall hold a Senior Reactor Operator License.

INTENTIONALLY LEFT BLANK

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

TWO UNITS WITH A COMMON CONTROL ROOM

WITH UNIT 1 IN CONDITION 4 OR 5 OR DEFUELED

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 OR 5
SM	1*	1*
SRO	2*	2*
RO	2	1
NLO	2	2**
STA	1***	None

WITH UNIT 1 IN CONDITION 1, 2, OR 3

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SM	1*	1*
SRO	2*	2*
RO	2**	1
NLO	2**	1
STA	1*,***	None

TABLE NOTATIONS

* Individual(s) may fill the same position on Unit 1.
** One of the two required individuals may fill the same position on Unit 1.
***The STA position may be filled by an on-shift SM or SRO provided the individual meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.
SM - Shift Manager with a Senior Operator license on Unit 2.
SRO - Individual with a Senior Operator license on Unit 2.
RO - Individual with an Operator license on Unit 2.
NLO - Non-licensed operator properly qualified to support the unit to which assigned.
STA - Shift Technical Advisor

Except for the Shift Manager (SM), the shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 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 of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an upcoming shift crewman being late or absent.

During any absence of the Shift Manager (SM) from the control room while the unit is in OPERATIONAL CONDITION 1, 2, or 3, an individual with a valid Senior Operator license shall be designated to assume the control room command function. Du ring any absence of the Shift Manager (SM) from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

6.2.3 DELETED. The information from this section is located in the UFSAR.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to Shift Supervision in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall meet the qualifications specified by the 1985 NRC 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 DELETED

6.5 DELETED

THE INFORMATION FROM SECTION 6.5 HAS BEEN RELOCATED TO THE QATR

THE INFORMATION FROM SECTION 6.5 HAS BEEN RELOCATED TO THE QATR

THE INFORMATION FROM SECTION 6.5 HAS BEEN RELOCATED TO THE QATR

THE INFORMATION FROM SECTION 6.5 HAS BEEN RELOCATED TO THE QATR

THIS PAGE IS INTENTIONALLY LEFT BLANK.

THE INFORMATION FROM SECTION 6.5 HAS BEEN RELOCATED TO THE QATR

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and submitted to the NRB, Plant Manager and the Vice President, Limerick Generating Station.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President, Limerick Generating Station, Plant Manager, and the NRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the NRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NRB, Plant Manager, and the Vice President, Limerick Generating Station, within the 14 days of the violation.

SAFETY LIMIT VIOLATION (Continued)

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance of Regulatory Guide 4.15, February 1979.

6.8.2 The information from Section 6.8.2 has been relocated to the QATR.

6.8.3 The information from Section 6.8.3 has been relocated to the QATR.

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the core spray, high pressure coolant injection, reactor core isolation cooling, residual heat removal, post-accident sampling system (until such time as a modification eliminates the PASS system as a potential leakage path), safeguard piping fill system, control rod drive scram discharge system, and containment air monitor systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

c. Deleted.

PROCEDURES AND PROGRAMS (Continued)

d. Radioactive Effluent Controls Program

A program shall be provided conforming with 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 (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2,

3) 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,

4) 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 Appendix I to 10 CFR Part 50,

5) Determination of cumulative 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. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) 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 limited to the following:

a. For noble gases: less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to 1500 mrem/yr to any organ.

PROCEDURES AND PROGRAMS (Continued)

8) Limitations on the annual quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) 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 greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on venting and purging of the Mark II containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable, and

11) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

e. Meteorological Monitoring Program

A program shall be provided to provide meteorological information in the environs of the plant. The program shall provide sufficient meteorological data for estimating potential radiation doses to the public.

The program shall (1) be contained in the ODCM, (2) conform to the guidance of Regulatory Guide 1.23, "Safety Guide 23 - Onsite Meteorological Program", and (3) include limitations on the operability of meteorological monitoring instrumentation including surveillance tests in accordance with the methodology in the ODCM.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

effective January 2, 1991

g. Primary Containment Leakage Rate Testing Program

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, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the Limitations and Conditions specified in NEI 94-01, Revision 2-A, dated October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.0 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall airlock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.

2) Seal leakage rate is less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the tests described in the Primary Containment Leakage Rate Testing Program.

h. 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 Bases without prior NRC approval provided the changes do not require either of the following:

A change in the TS incorporated in the license; or

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 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).

PROCEDURES AND PROGRAMS (Continued)

i. Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications," of the following:

a. Actions to restore battery cells with float voltage < 2.13 volts, and

b. Actions to equalize and test battery cells that have been discovered with electrolyte level below the minimum established design limit.

j. Surveillance Frequency Control Program

This program provides controls for 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 those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 0.

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.

k. Snubber Program

This program conforms to the examination, testing, and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

a. This program shall meet 10 CFR 50.55a ISI requirements for supports.

b. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a, subject to its conditions, and subject to Commission approval.

c. The program shall, as allowed by 10 CFR 50.55a, meet Subsection ISTA, "General Requirements," and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," in lieu of Section XI of the ASME B&PV Code ISI requirements for snubbers, or meet authorized alternatives pursuant to 10 CFR 50.55a.

d. The 120-month program updates shall be made in accordance with 10 CFR 50.55a subject to the limitations and conditions listed therein.

l. Explosive Gas Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the off-gas recombiners.

The program shall include:

a. The limit for the concentration of hydrogen downstream of the offgas recombiners and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Explosive Gas Monitoring Program surveillance frequencies.

m. 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 OPERATIONAL CONDITIONs 1 and 2.

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 ACTION allowed outage 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 ACTION allowed outage 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. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the completion times must be PRA methods approved for use with this program in Amendment No. 203, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 Deleted

6.9.1.2 Deleted

6.9.1.3 Deleted

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year unless otherwise noted.

6.9.1.5 Reports required on an annual basis shall include:

a. Deleted

*A single submittal may be made for a multiple unit station.

ANNUAL REPORTS (Continued)

b. (Deleted)

c. (Deleted)

d. (Deleted)

MONTHLY OPERATING REPORTS*

6.9.1.6 Deleted

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality. The report shall include summaries, interpretations, analysis 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 (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

*A single submittal may be made for a multiple unit station.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year 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 Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

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

INTENTIONALLY LEFT BLANK

CORE OPERATING LIMITS REPORT

6.9.1.9 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 CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

c. The MINIMUM CRITICAL POWER RATIO (MCPR) and MCPR(99.9%) for Specification 3.2.3,

d. The MCPR(P) and MCPR(F) adjustment factor for specification 3.2.3,

e. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4,

f. The power biased Rod Block Monitor setpoints of Specification 3.3.6 and the Rod Block Monitor MCPR OPERABILITY limits of Specification 3.1.4.3.

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Oscillation Power Range Monitor (OPRM) period based detection algorithm (PBDA) setpoints for Specification 2.2.1,

i. The minimum required number of operable main turbine bypass valves for Specification 3.7.8 and the TURBINE BYPASS SYSTEM RESPONSE TIME for Specification 4.7.8.c.

6.9.1.10 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:

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision),

b. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, " August 1996.

6.9.1.11 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.12 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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

a. Limiting Condition for Operation Section 3.4.6, "RCS Pressure/Temperature Limits"
b. Surveillance Requirement Section 4.4.6, "RCS Pressure/Temperature Limits"

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 documents:

a. BWROG-TP-11-022-A, Revision 1 (SIR-05-044), “Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013.

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 DELETED

THE INFORMATION FROM SECTION 6.10 HAS BEEN RELOCATED TO THE QATR

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

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:

6.12.1 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.

HIGH RADIATION AREA (Continued)

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.

6.12.2 In addition to the requirements of Specification 6.12.1, 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.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented with the following information:

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

INTENTIONALLY LEFT BLANK

INTENTIONALLY LEFT BLANK

PROCESS CONTROL PROGRAM (Continued)

2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective upon review and acceptance by the PORC and approval of the Plant Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Changes to the ODCM:

a. Shall be documented with the following information:

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

2. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective upon review and acceptance by the PORC and the approval of the Plant Manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to 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 (e.g., month/year) the change was implemented.

6.15 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

6.16 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM

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 Emergency Fresh Air Supply (CREFAS) 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 .

CONTROL ROOM ENVELOPE HABITABILITY PROGRAM (Continued)

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 by one train of the CREFAS, operating at the flow rate required by SR 4.7.2.1 c.1, at a Frequency of 24 months on a STAGGERED TEST BASIS. 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 Specification 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.

6.17 SAFETY FUNCTION DETERMINATION PROGRAM (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into Specification 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system ACTIONs. This program implements the requirements of Specification 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

c. Provisions to ensure that an inoperable supported system's Allowed Outage Time is not inappropriately extended as a result of multiple support system inoperabilities, and

6.17 SAFETY FUNCTION DETERMINATION PROGRAM (SFDP) (Continued)

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable,

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate ACTIONs of the Limiting Condition for Operation in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ACTIONs to enter are those of the support system.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-85

LIMERICK GENERATING STATION

UNIT 2

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. 50-353

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

August 25, 1989

LIMERICK GENERATING STATION

UNIT 2

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

TABLE OF CONTENTS

1.0 OBJECTIVES OF THE ENVIRONMENTAL PROTECTION PLAN

The Environmental Protection Plan (EPP) is to provide for protection of non-radiological environmental values during operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement-Operating License Stage (FES-OL) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility operation and of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's NPDES permit.

2.0 ENVIRONMENTAL PROTECTION ISSUES

In the FES-OL dated April, 1984, the staff considered the environmental impacts associated with the operation of the two unit Limerick Generating Station. Certain environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment.

2.1 Aquatic Issues

(1) During operation, the station blowdown temperature will exceed the maximum permissible temperatures set by the applicable water quality standards. However, the affected area of the Schuylkill River is expected to be smaller than the maximum area permitted by the Delaware River Basin Commission. (FES Section 5.3.2.2)

(2) The water quality of the station discharge, after initial mixing with the Schuylkill River, is predicted to, at times, exceed the applicable quality criteria for some constituents, based on source water maximum constituent concentrations. These exceedances are expected for constituents whose maximum river concentrations also exceed the applicable criteria. (FES Section 5.3.2.3)

(3) Chlorination of station cooling waters for condenser and cooling tower biofouling control may result in some adverse impacts to Schuylkill River biota in the vicinity of the station discharge. (FES Section 5.3.2.3)

(4) Operation of the Point Pleasant Diversion will alter the hydrology, aquatic habitats, and water quality of the headwater section of the East Branch of Perkiomen Creek but the diversion waters are expected to provide beneficial dilution of waste loads entering the stream in its middle and lower reaches. (FES Sections 5.3.2.3 and 5.2.2)

(5) The supplemental cooling water withdrawal from Perkiomen Creek using state-of-the-art technology will result in localized effects from entrainment of fish larvae. (FES Section 5.5.2)

2.2 Terrestrial Issues

No specific terrestrial issues were identified by the NRC staff in the FES-OL.

2.3 Noise Issues

(1) Tones from the Point Pleasant pumphouse transformers are predicted to be audible and may cause annoyance at a nearby residence. Noise monitoring and, if necessary, mitigative measures to make the tones inaudible have been mandated by the ASLB. (FES Sections 5.12.1 and 5.14.4.1)

(2) Noise from transformers and pumps in the Bradshaw Reservoir pumphouse may be audible at nearby residences. The licensee has committed to ambient and operational noise level monitoring and implementation of identified mitigative measures, if necessary, to reduce noise levels below those likely to cause annoyance and complaints. (FES Sections 5.12.2 and 5.14.4.2)

(3) Offsite noise levels in the vicinity of the Limerick site during station operation are not expected to be high enough above ambient levels to annoy nearby residents. But because of uncertainties in the assessment, a confirmatory noise monitoring program and implementation of mitigative measures, if necessary, will be undertaken. (FES Sections 5.12.3 and 5.14.4.3)

NRC requirements with regard to noise issues are specified in Section 4.3 of this EPP.

3.0 CONSISTENCY REQUIREMENTS

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP.* Changes in station design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable non-radiological environmental effects are confined to the on-site areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or

*This provision does not relieve the licensee of the requirements of 10 CFR 50.59

power level; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of the EPP to meet the objectives specified in Section 1.0.

3.2 Reporting Related to the NPDES Permit and State Certification

Changes to, or renewals of, the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments that are either regulated or mandated by Federal, State and local environmental regulations are not subject to the requirements of Section 3.1. However, if any environmental impacts of a change are not evaluated under other Federal, State, or local environmental regulations, then those impacts are subject to the requirements of Section 3.1.

4.0 ENVIRONMENTAL CONDITIONS

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. If an event is reportable under 10 CFR 50.72, then a duplicate immediate report under this subsection is not required. However, a follow-up written report is required in accordance with Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the Commonwealth of Pennsylvania, under the authority of the Clean Water Act, for any requirements for aquatic monitoring.

4.2.2 Terrestrial Monitoring

No terrestrial monitoring is required.

4.2.3 Maintenance of Transmission Line Corridors

The use of herbicides within the Limerick Generating Station transmission line corridors (Limerick to Cromby, Cromby to Plymouth Meeting, Cromby to North Wales, and Limerick to Whitpain) shall conform to the approved use of selected herbicides as registered by the Environmental Protection Agency and approved by Commonwealth authorities and applied as directed on the pesticide label.

4.2.4 Noise Monitoring

All initial Environmental Noise Assessments have been completed.

The information in Subsections 4.2.4.1, 4.2.4.2,

4.2.4.3, and 4.2.4.4 has been deleted.

Pages 4-3, 4-4, and 4-5 have been removed from this Section.

5.0 ADMINISTRATIVE PROCEDURES

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the EPP. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organizational structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records associated with this Environmental Protection Plan shall be made and retained in a manner convenient for review and inspection. These records shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained until the date of termination of the operating license. All other records relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Request for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

5.4 Plant Reporting Requirements

5.4.1 Deleted

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of an Unusual or Important Environmental event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the same time it is submitted to the other agency.

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-85

Constellation Energy Generation, LLC shall comply with the following conditions on the schedule noted below:

Amendment No.	Additional Conditions
193, 217, 223	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; 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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. 193 dated July 31, 2018. In addition, Constellation Energy Generation, LLC (CEG) is approved to implement 10 CFR 50.69 using the alternative seismic approach, as described in the licensee's letters dated December 15, 2021, and February 14, 2022, for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs as specified in Unit 2 License Amendment No. 223 dated May 17, 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).