PMTurkeyCOLPEm Resource

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Sent:	Wednesday, December 07, 2016 3:35 PM
To:	Comar, Manny
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Subject:	[External_Sender] Comment on draft PTN 6&7 licenses
Attachments:	ML052790649_U34_OL.pdf; Location_from_FSAR_SEC02_01_Rev8.pdf

The words in the draft licenses for PTN 6&7 state "…The facility would be located in unincorporated southeast Miami-Dade County, Florida (FL), approximately 30 miles southwest of Miami, FL,…" at the top of page 2.

However, the U3&4 OL and the COLA state "…located on the applicant's Turkey Point site in Miami-Dade County, about 25 miles south of Miami, Florida, and…" and "…property located approximately 25 miles south of Miami in unincorporated Miami-Dade County, Florida…" respectively.

I have attached excerpts from both documents.

Any questions, please call.
Thanks
Steve Franzone
NNP Licensing Manager - COLA
"A doubtful friend is worse than a certain enemy. Let a man be one thing or the other, and we then know how to meet him." ~ Aesop
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Hearing Identifier: TurkeyPoint_COL_Public
Email Number: 1251

Mail Envelope Properties (837bea038cba409f9c06ff97ac347248)

Subject: [External_Sender] Comment on draft PTN 6&7 licenses
Sent Date: 12/7/2016 3:34:42 PM
Received Date: 12/7/2016 3:36:23 PM
From: Franzone, Steve

Created By: Steve.Franzone@fpl.com

Recipients:
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Tracking Status: None

Post Office: fpl.com

Files	**Size**	**Date & Time**
MESSAGE	1437	12/7/2016 3:36:23 PM
ML052790649_U34_OL.pdf	4834384	
Location_from_FSAR_SEC02_01_Rev8.pdf	75869	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

RENEWED FACILITY OPERATING LICENSE NO. DPR-31

The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-31 issued on July 19, 1972, has now found that:

a. The application to renew License No. DPR-31 filed by Florida Power and Light Company, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

b. 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 this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the Turkey Point Unit 3 plant, and that any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations;

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

d. There is reasonable assurance (i) that the facility can be operated at steady state power levels up to 2644 megawatts thermal in accordance with this renewed operating license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. Florida Power and Light Company is technically and financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

f. The applicable provisions of 10 CFR Part 140 have been satisfied;

g. The renewal of this operating license will not be inimical to the common defense and security or to the health and safety of the public; and

h. After weighing the environmental, economic, technical and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Renewed Facility Operating License No. DPR-31 is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

On the basis of the foregoing findings regarding this facility, Facility Operating License No. DPR-31, issued on July 19, 1972, is superseded by Renewed Facility Operating License No. DPR-31, which is hereby issued to Florida Power and Light Company (FPL), to read as follows:

1. This renewed license applies to the Turkey Point Nuclear Generating Unit No. 3 nuclear power reactor, a pressurized, light water moderated and cooled reactor, and associated steam generators and electrical generating equipment (the facility). The facility is located on the applicant's Turkey Point site in Miami-Dade County, about 25 miles south of Miami, Florida, and is described in the Final Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

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

   A. Pursuant to Section 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location on the Turkey Point site, in accordance with the procedures and limitations set forth in this renewed license;

   B. Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

   C. 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;

   D. Pursuant to the Act and 10 CFR Part 30 to receive, possess, and use at any time 100 millicuries each of any byproduct material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

3. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; 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 below:

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 271 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than July 19, 2012.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Fire Protection

FPL shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated June 28, 2012 (and supplements dated September 19, 2012; March 18, April 16, and May 15, 2013; January 7, April 4, June 6, July 18, September 12, November 5, and December 2, 2014; and February 18, 2015), and as approved in the safety evaluation dated May 28, 2015. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

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

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

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

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

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

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

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

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

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated May 28, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. and 3. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Enclosure 1, Attachment S, Table S-2, "Plant Modifications Committed," of FPL letter L-2014-303, dated 11/05/2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the end of the second refueling outage (for each unit) following issuance of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Enclosure 1, Attachment S, Table S-3, "Implementation Items," of FPL letter L-2014-303, dated 11/05/2014, with the exception of items 12, 18 and 19, no later than 12 months after issuance of the license amendment. Items 12, 18 and 19 are associated with modifications in Table S-2 and will be completed in accordance with Transition License Condition 2 above. Item 22 will be completed within 6 months of the NRC approval of the Flowserve RCP Seal Topical Report.

E. The licensee 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 provision of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Florida Power and Light Turkey Point Nuclear Plant Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program - Revision 15" submitted by letter dated August 3, 2012.

The licensee 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 Turkey Point Nuclear Generating Station CSP was approved by License Amendment No. 245 as supplemented by changes approved by Amendment Nos. 256 and 266.

F. 1. The licensee shall restrict the combined number of fuel assemblies loaded in the existing spent fuel pool storage racks and cask pit rack to no more than the capacity of the spent fuel pool storage racks. This condition applies at all times, except during activities associated with a reactor core offload/reload refueling condition. This restriction will ensure the capability to unload and remove the cask pit rack when cask loading operations are necessary.

2. The licensee shall establish two hold points within the rack installation procedure to ensure proper orientation of the cask rack in each unit's spent fuel pool. Verification of proper cask pit rack orientation will be implemented by an authorized Quality Control inspector during installation of the racks to ensure consistency with associated spent fuel pool criticality analysis assumptions.

G. <u>Mitigation Strategy License Condition</u>

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

H. PAD TCD Safety Analyses

1. PAD 4.0 TCD has been specifically approved for use for the Turkey Point licensing basis analyses. Upon NRC's approval of a revised generic version of PAD that accounts for Thermal Conductivity Degradation (TCD), FPL will within six months:

   a. Demonstrate that PAD 4.0 TCD remains conservatively bounding in licensing basis analyses when compared to the new generically approved version of PAD w/TCD, or
   b. Provide a schedule for the re-analysis using the new generically approved version of PAD w/TCD for any of the affected licensing basis analyses.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight July 19, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Signed by
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications for Unit 3
Appendix B – Environmental Protection Plan

Date of Issuance: June 6, 2002

U. S. ATOMIC ENERGY COMMISSION
DOCKET NOS. 50-250
50-251

LICENSE AUTHORITY FILE COPY

TECHNICAL SPECIFICATIONS

Appendix "A"

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT
UNITS NO. 3 & 4

JUL 19 1972

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

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

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 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.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.4.

b. The equipment hatch is closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

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

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or 0-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with NRC approved methodology. Unit operation within these operating limits is addressed in individual specifications. The COLR is submitted to the NRC in accordance with the requirements of 6.9.1.7.

DIGITAL CHANNEL OPERATIONAL TEST

1.11 A DIGITAL CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock, and/or trip functions.

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE - 133

1.13 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

FREQUENCY NOTATION

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

GAS DECAY TANK SYSTEM

1.15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are 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 either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage).

OPERABLE - OPERABILITY

1.17 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 MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 13.5 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.20 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PURGE - PURGING

1.21 PURGE or PURGING shall be any 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.

QUADRANT POWER TILT RATIO

1.22 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.23 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2644 MWt.

SHUTDOWN MARGIN

1.24 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.25 The SITE BOUNDARY shall mean that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOURCE CHECK

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

STAGGERED TEST BASIS

1.27 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, and

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.29 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.30 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.31 An UNRESTRICTED AREA shall mean an area, access to which is neither limited nor controlled by the licensee.

VENTING

1.32 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

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.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.
SFCP	In accordance with the Surveillance Frequency Control Program

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits specified in the COLR, for 3 loop operation; and the following Safety Limits shall not be exceeded:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-1 DNB correlation.

b. The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

Action:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the setpoint consistent with the Trip setpoint value within permissible calibration tolerance.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that the affected channel is OPERABLE; or

   2. 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.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	$\leq$ 108.6% of RTP**	108.0% of RTP**
b. Low Setpoint	$\leq$ 28.0% of RTP**	$\leq$ 25% of RTP**
3. Intermediate Range, Neutron Flux	$\leq$ 31.0% of RTP**	$\leq$ 25% of RTP**
4. Source Range, Neutron Flux	$\leq 1.4 \times 10^5$ cps	$\leq 10^5$ cps
5. Overtemperature ΔT	See Note 2	See Note 1
6. Overpower ΔT	See Note 4	See Note 3
7. Pressurizer Pressure-Low	$\geq$ 1817 psig	$\geq$ 1835 psig
8. Pressurizer Pressure-High	$\leq$ 2403 psig	$\leq$ 2385 psig
9. Pressurizer Water Level-High	$\leq$ 92.2% of instrument span	$\leq$ 92% of instrument span
10. Reactor Coolant Flow-Low	$\geq$ 89.6% of loop design flow*	90% of loop design flow*
11. Steam Generator Water Level Low-Low	$\geq$ 15.5% of narrow range instrument span	16% of narrow range instrument span

* Loop design flow = 86,900 gpm

** RTP = Rated Thermal Power

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
12. Steam/Feedwater Flow Mismatch Coincident with	Feed Flow ≤ 20.7% below rated Steam Flow	Feed Flow 20% below rated Steam Flow
Steam Generator Water Level-Low	≥ 15.5% of narrow range instrument span	16% of narrow range instrument span
13. Undervoltage – 4.16 kV Busses A and B	≥ 69% bus voltage	≥ 70% bus voltage
14. Underfrequency – Trip of Reactor Coolant Pump Breaker(s) Open	≥ 55.9 Hz	≥ 56.1 Hz
15. Turbine Trip		
a. Emergency Trip Header Pressure	≥ 901 psig	1000 psig
b. Turbine Stop Valve Closure	Fully Closed***	Fully Closed***
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	≥ 6.0×10^{-11} amps	Nominal 1×10^{-10} amps

*** Limit switch is set when Turbine Stop Valves are fully closed.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	≤ 13.0% RTP**	Nominal 10% of RTP**
2) Turbine Inlet Pressure	≤ 13.0% Turbine Power	Nominal 10% Turbine Power
c. Power Range Neutron Flux, P-8	≤ 48.0% RTP**	Nominal 45% of RTP**
d. Power Range Neutron Flux, P-10	≥ 7.0% RTP**	Nominal 10% of RTP**
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

** RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT (Those values denoted with [*] are specified in the COLR.)

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \frac{1}{(1+\tau_6 S)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Instrumentation

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = Lead/Lag compensator on measured ΔT; τ_1 = [*]s, τ_2 = [*]s

$\frac{1}{1+\tau_3 S}$ = Lag compensator on measured ΔT; τ_3 = [*]s

ΔT_0 = Indicated ΔT at RATED THERMAL POWER

K_1 = [*];

K_2 = [*]/°F;

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4 , τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = [*]s, τ_5 = [*]s;

T = Average temperature, °F;

$\frac{1}{1+\tau_6 S}$ = Lag compensator on measured T_{avg}; τ_6 = [*]s

T′ $\leq$ [*]°F (Indicated Loop T_{avg} at RATED THERMAL POWER);

K_3 = [*]/psi;

P = Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P' $\geq$ [*] psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

And $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between – [*]% and + [*]%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds – [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER.

NOTE 2: The Overtemperature ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.2% ΔT span for the Pressurizer Pressure channel, and 0.4% ΔT span for the $f(\Delta I)$ channel. No separate Allowable Value is provided for T_{avg} because this function is part of the ΔT value.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT (Those values denoted with [*] are specified in the COLR.)

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 S}{1+\tau_7 S} \left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left[T \frac{1}{1+\tau_6 S} - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	As defined in Note 1,
$\frac{1}{1+\tau_3 S}$	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	=	[*],
K_5	$\geq$	[*]/°F for increasing average temperature and [*]/°F for decreasing average temperature,
$\frac{\tau_7 s}{1+\tau_7 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_7 \geq$ [*] s,
$\frac{1}{1+\tau_6 S}$	=	As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	[*]/°F for T > T”
	=	[*] for T ≤ T”,
T	=	As defined in Note 1,
T”	≤	[*]°F (Indicated Loop T_{avg} at RATED THERMAL POWER)
S	=	As defined in Note 1, and
f_2 (ΔI)	=	[*]

NOTE 4: The Overpower ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel. No separate Allowable Value is provided for T_{avg} because this function is part of the ΔT value.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specification 3.0.6.

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 Specification 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 1 hour action shall be initiated to place the unit, as applicable, in:

a. At least HOT STANDBY 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 MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

LIMITING CONDITIONS FOR OPERATION (Continued)

3.0.5 Limiting Conditions for Operation including the associated ACTION requirements shall apply to each unit individually unless otherwise indicated as follows:

a. Whenever the Limiting Conditions for Operation refers to systems or components which are shared by both units, the ACTION requirements will apply to both units simultaneously.

b. Whenever the Limiting Conditions for Operation applies to only one unit, this will be identified in the APPLICABILITY section of the specification; and

c. Whenever certain portions of a specification contain operating parameters, Setpoints, etc., which are different for each unit, this will be identified in parentheses, footnotes or body of the requirement.

3.0.6 Equipment removed from service or declared inoperable to comply with ACTION requirements may be returned to service under administrative controls solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.1 and 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. If an ACTION item requires periodic performance on a "once per . . ." basis, the above frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition of Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. 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 of Operation must immediately be declared not met, and the applicable ACTION(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition of Operation must immediately be declared not met, and the applicable ACTION(s) must be entered.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with a Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a.

   Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50, Section 50.55a.

APPLICABILITY

SURVEILLANCE REQUIREMENTS (CONTINUED)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda and the ASME OM Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and the ASME OM Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

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

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

f. Each reactor coolant pump flywheel shall be inspected at least once every 20 years, by either conducting an in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or conduct a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

4.0.6 Surveillance Requirements shall apply to each unit individually unless otherwise indicated as stated in Specification 3.0.5 for individual specifications or whenever certain portions of a specification contain surveillance parameters different for each unit, which will be identified in parentheses, footnotes or body of the requirement.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN not within limits, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the limits specified in the COLR:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6 in accordance with the Surveillance Frequency Control Program;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. In accordance with the Surveillance Frequency Control Program, when in MODE 3 or 4 by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 When in Mode 1 or 2, the overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1% $\Delta k/k$ in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limit specified in the COLR.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN not within limits, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be within the limit specified in the COLR:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

    1) Reactor Coolant System boron concentration,

    2) Control rod position,

    3) Reactor Coolant System average temperature,

    4) Fuel burnup based on gross thermal energy generation,

    5) Xenon concentration, and

    6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less positive than or equal to $+5.0 \times 10^{-5}$ $\Delta k/k/°F$ for all the rods withdrawn, beginning of cycle life (BOL), for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 $\Delta k/k/°F$ at 100 % RATED THERMAL POWER.

APPLICABILITY: Beginning of cycle life (BOL) - MODES 1 and 2* only**.
End of life (EOL) - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive or equal to the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

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* With K_{eff} greater than or equal to 1.

** See Special Test Exceptions Specification 3.10.3.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm*. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

* Measurement of the MTC in accordance with Surveillance Requirement 4.1.1.3.b may be suspended provided that the benchmark criteria in WCAP-13749-P-A and the Revised Prediction specified in the COLR are satisfied.

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 541°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 547°F with the $T_{avg} - T_{ref}$ Deviation Alarm not reset.

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* With K_{eff} greater than or equal to 1.

** See Special Test Exceptions Specification 3.10.3.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid storage tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.4a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.4b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the temperature of the rooms containing flow path components is greater than or equal to 62°F when a flow path from the boric acid tanks is used, and

b. 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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

a. The source path from a boric acid storage tank via a boric acid transfer pump to the charging pump suction*, and

b. At least one of the two source paths from the refueling water storage tank to the charging pump suction; and,

c. The flow path from the charging pump discharge to the Reactor Coolant System via the regenerative heat exchanger.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With no boration source path from a boric acid storage tank OPERABLE,

1. Demonstrate the OPERABILITY of the second source path from the refueling water storage tank to the charging pump suction by verifying the flow path valve alignment; and

2. Restore the boration source path from a boric acid storage tank to OPERABLE status within 70 hours or be in at least HOT STANDBY and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F within the next 8 hours; restore the boration source path from a boric acid storage tank to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With only one boration source path OPERABLE or the regenerative heat exchanger flow path to the RCS inoperable, restore the required flow paths to OPERABLE status within 70 hours or be in at least HOT STANDBY and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F within the next 8 hours; restore at least two boration source paths to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

c. With the boration source path from a boric acid storage tank and the charging pump discharge path via the regenerative heat exchanger inoperable, within one hour initiate boration to a boron concentration equivalent to the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F and go to COLD SHUTDOWN as soon as possible within the limitations of the boration and pressurizer level control functions of the CVCS.

* The flow required in Specification 3.1.2.2.a above shall be isolated from the other unit from the boric acid transfer pump discharge to the charging pump suction.

SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the temperature of the rooms containing flow path components is greater than or equal to 62°F when a flow path from the boric acid tanks is used;

b. 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;

c. In accordance with the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2a. and c. delivers at least 16 gpm to the RCS.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 70 hours or be in at least HOT STANDBY and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F within 8 hours; restore at least two charging pumps to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUI REMENTS

4.1.2.3.1 The required charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.4 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum indicated borated water volume of 2,900 gallons per unit,

2) A boron concentration between 3.0 wt% (5245 ppm) and 4.0 wt.% (6993 ppm), and

3) A minimum boric acid tanks room temperature of 62°F.

b. The refueling water storage tank (RWST) with:

1) A minimum indicated borated water volume of 20,000 gallons,

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 39°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEI LLANCE REQUIREMENTS

4.1.2.4 The above required borated water source shall be demonstrated OPERABLE:

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

1) Verifying the boron concentration of the water,

2) Verifying the indicated borated water volume, and

3) Verifying that the temperature of the boric acid tanks room is greater than or equal to 62°F, when it is the source of borated water.

SURVEILLANCE REQUIREMENTS (Continued)

b. By verifying the RWST temperature is above its limit whenever the outside air temperature is less than 39° at the following frequencies:

1) Within one hour when the outside temperature is below 39° for 23 consecutive hours, and

2) At least once per 24 hours when the outside temperature is below 39°.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.5 The following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum indicated borated water volume in accordance with Figure 3.1-2,

2) A boron concentration in accordance with Figure 3.1-2. and

3) A minimum boric acid tanks room temperature of 62°F.

b. The refueling water storage tank (RWST) with:

1) A minimum indicated borated water volume of 320,000 gallons,

2) A boron concentration between 2400 ppm and 2600 ppm.

3) A minimum solution temperature of 39°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

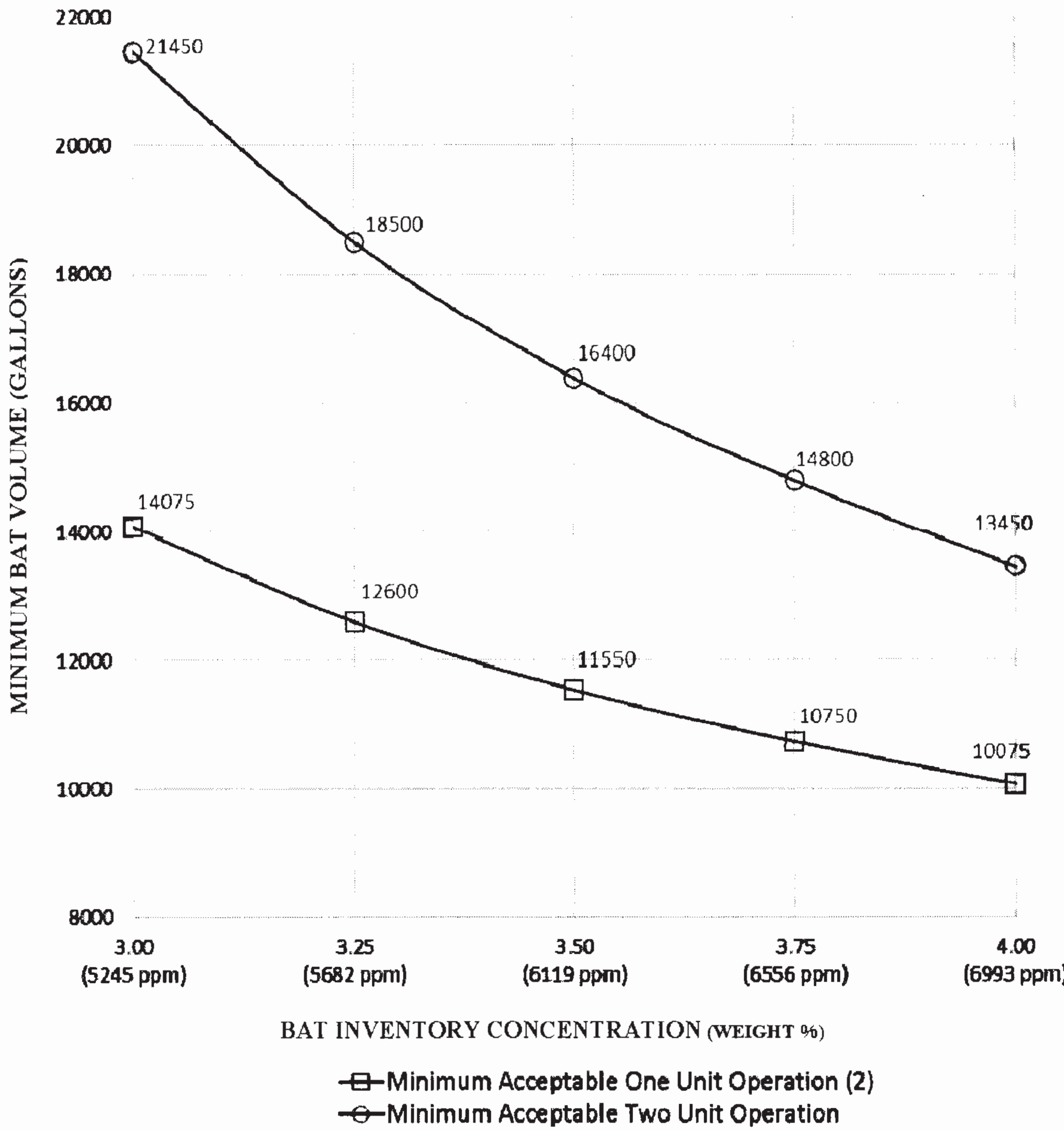
a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 70 hours or be in at least HOT STANDBY within the next 8 hours* and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the boric acid tank inventory concentration greater than 4.0 wt%, verify that the boric acid solution temperature for boration sources and flow paths is greater than the solubility limit for the concentration.

* If this action applies to both units simultaneously, be in at least HOT STANDBY within the next sixteen hours.

Figure 3.1-2
BORIC ACID TANK MINIMUM VOLUME (1)
Modes 1, 2, 3 and 4



Notes:

(1) Combined volume of all available boric acid tanks assuming RWST boron concentration between 2400 ppm and 2600 ppm.

(2) Includes 2900 gallons for the shutdown unit.

SURVEILLANCE REQUIREMENTS

4.1.2.5 Each borated water source shall be demonstrated OPERABLE:

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

1) Verifying the boron concentration in the water,

2) Verifying the indicated borated water volume of the water source, and

3) Verifying that the temperature of the boric acid tanks room is greater than or equal to 62°F, when it is the source of borated water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for 23 consecutive hours, and

2) At least once per 24 hours while the outside temperature exceeds its limits.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within the Allowed Rod Misalignment between the Analog Rod Position Indication and the group step counter demand position within one hour after rod motion. The Allowed Rod Misalignment shall be defined as:

a. for THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 18 steps, and

b. for THERMAL POWER greater than 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 12 steps.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps and THERMAL POWER greater than 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER and confirm that all indicated rod positions are within the Allowed Rod Misalignment, or

   3. Be in HOT STANDBY within the following 6 hours.

c. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 18 steps and THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Be in HOT STANDBY within the following 6 hours.

---

* See Special Test Exceptions 3.10.2 and 3.10.3.

d. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the Allowed Rod Misalignment of Specification 3.1.3.1, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the Allowed Rod Misalignment of Specification 3.1.3.1, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.d.3.c and 3.1.3.1.d.3.d below are demonstrated, and

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and F_Q (Z) and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the Allowed Rod Misalignment of the group step counter demand position in accordance with the Surveillance Frequency Control Program (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

LIMITING CONDITION FOR OPERATION (Continued)

3.1.3.2 The Analog Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the respective actual and demanded shutdown and control rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

   All Shutdown Banks: within the Allowed Rod Misalignment of Specification 3.1.3.1 of the group demand counters for withdrawal ranges of 0-30 steps and 200-All Rods Out as defined in the Core Operating Limits Report.

   Control Bank A and B: within the Allowed Rod Misalignment of Specification 3.1.3.1 of the group demand counters for withdrawal ranges of 0-30 steps and 200-All Rods Out as defined in the Core Operating Limits Report.

   Control Banks C and D: within the Allowed Rod Misalignment of Specification 3.1.3.1 of the group demand counters for withdrawal range of 0-All Rods Out as defined in the Core Operating Limits Report.

b. Group demand counters; ± 2 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one analog rod position indicator per bank inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2** a). Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours and once every 31 Effective Full Power Days thereafter, and within 1 hour if rod control system parameters indicate unintended movement, or if the rod with an inoperable position indicator is moved greater than 12 steps, and

   b). Review the parameters of the rod control system for indications of unintended rod movement for the rod with an inoperable indicator within 8 hours and once per 8 hours thereafter, and

   c). Determine the position of the non-indicating rod indirectly by the movable incore detectors prior to increasing THERMAL POWER above 50% RATED THERMAL POWER and within 8 hours of reaching 100% RATED THERMAL POWER, or

   3. Reduce THERMAL POWER to less than 75% of RATED THERMAL POWER within 8 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued):

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all analog rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within the Allowed Rod Misalignment of Specification 3.1.3.1 at least once per 8 hours, or

2. Reduce THERMAL POWER to less than 75% of RATED THERMAL POWER within 8 hours.

**Rod position monitoring by Actions a.2.a), a.2.b), and a.2.c) may only be applied to one inoperable rod position indicator per unit and shall only be allowed until an entry into MODE 3.

SURVEILLANCE REQUIREMENTS

4.1.3.2.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Analog Rod Position Indication System agree within the Allowed Rod Misalignment of Specification 3.1.3.1 (allowing for one hour thermal soak after rod motion) in accordance with the Surveillance Frequency Control Program except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the Demand Position Indication System and the Analog Rod Position Indication System at least once per 4 hours.

4.1.3.2.2 Each of the above required analog rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST performed in accordance with the Table 4.1-1.

TABLE 4.1-1

ROD POSITION INDICATOR SURVEILLANCE REQUIREMENTS

Functional Unit	Check	Calibration	Operational Test
Individual Rod Position	SFCP	SFCP	SFCP
Demand Position	SFCP	N/A	SFCP

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group step counter demand position indicator shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown and control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **

ACTION:

With less than the above required group step counter demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each of the above required group step counter demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

4.1.3.3.2 OPERABILITY of the group step counter demand position indicator shall be verified in accordance with Table 4.1-1.

---

* With the Reactor Trip System breakers in the closed position.

** See Special Test Exceptions Specification 3.10.5.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. In accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2* **

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. In accordance with the Surveillance Frequency Control Program thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.0

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion specified in the Rod Bank Insertion Limits curve, defined in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2* **

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2 either:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position specified in the Rod Bank Insertion Limits curve, defined in the CORE OPERATING LIMITS REPORT, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits in accordance with the Surveillance Frequency Control Program, except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

---

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

** With K_{eff} greater than or equal to 1.0

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed Relaxed Axial Offset Control (RAOC) operational space as defined in the CORE OPERATING LIMITS REPORT (COLR), or

b. within a +/- 2% or +/- 3% target band about the target flux difference during Base Load operation.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION***:

a. For RAOC operation with the indicated AFD outside of the limits specified in the COLR, either

   1. Restore the indicated AFD to within the RAOC limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above P_T** with the indicated AFD outside of the applicable target band about the target flux difference, either

   1. Restore the indicated AFD to within the Peaking Factor Limit Report target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than P_T and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless indicated AFD is within the limits specified in the COLR.

---

* See Special Test Exceptions Specification 3.10.2.

** P_T = Reactor Power at which predicted F_Q would exceed its limit (consistent with Specification 4.2.2.1).

*** The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) In accordance with the Surveillance Frequency Control Program when the alarm used to monitor the AFD is OPERABLE, and

   2) At least once per hour for the first 6 hours after restoring the alarm used to monitor the AFD to OPERABLE status.*

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the alarm used to monitor the AFD is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The target flux difference of each OPERABLE excore channel shall be determined by measurement in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.

4.2.1.3 In accordance with the Surveillance Frequency Control Program, the target flux difference shall be updated by either determining the target flux difference pursuant to Specification 4.2.1.2 above or by linear interpolation between the most recently measured value and the predicted value at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

* Performance of a functional test to demonstrate OPERABILITY of the alarm used to monitor the AFD may be substituted for this requirement.

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q{}^{(Z)}$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q^L(Z)$ shall be limited by the following relationships:

$$F_Q^M(Z) \le \frac{[F_Q]^{Lx}}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q^M(Z) \le \frac{[F_Q]^{Lx}}{0.5} [K(Z)] \text{ for } P \le 0.5$$

where: $[F_Q]^L$ = F_Q limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

$$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}},$$

$[F_Q]^M$ = The Measured Value, and

K(Z) for a given core height, is specified in the K(Z) curve, defined in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1

ACTION:

With the measured value of $F_Q^M(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^M(Z)$ exceeds $F_Q^L(Z)$ within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours: POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta-T Trip Setpoints (value of K_4) have been reduced at least 1% for each 1%$F_Q^M(Z)$ exceeds the $F_Q^L(Z)$; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced power limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q^M(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 If $[F_Q]^P$ as predicted by approved physics calculations is greater than $[F_Q]^L$ and P is greater than P_T* as defined in 4.2.2.2, $F_Q(Z)$ shall be evaluated by MIDS (Specification 4.2.2.2), BASE LOAD (Specification 4.2.2.3) or RADIAL BURNDOWN (Specification 4.2.2.4) to determine if F_Q is within its limit $[F_Q]^P$ = Predicted F_Q).

If $[F_Q]^P$, is less than $[F_Q]^L$ or P is less than P_T, $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit as follows:

a. Using the movable incore detectors to obtain power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verifying that the requirements of Specification 3.2.2 are satisfied.

c. $F_Q^M(Z) \leq F_Q^L(Z)$

Where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowance for manufacturing tolerances and measurement uncertainty and $F_Q^L(Z)$ is the F_Q limit defined in 3.2.2.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

1. Prior to exceeding 75% of RATED THERMAL POWER,** after refueling,

2. In accordance with the Surveillance Frequency Control Program.

e. With the relationship specified in Specification 4.2.2.1.c above not being satisfied:

1) Calculate the percent $F_Q^M(Z)$ exceeds its limit by the following expression:

$$\left[\left[\frac{F_Q^M(Z)}{[F_Q]^L \ X \ K(Z)/P}\right] - 1\right] X100 \text{ for } P \geq 0.5$$

$$\left[\left[\frac{F_Q^M(Z)}{[F_Q]^L \ X \ K(Z)/0.5}\right] - 1\right] X100 \text{ for } P < 0.5$$

* P_T = Reactor power level at which predicted F_Q would exceed its limit.

** During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map obtained.

2) The following action shall be taken:

a) Comply with the requirements of Specification 3.2.2 for F_Q^M (Z) exceeding its limit by the percent calculated above.

4.2.2.2 MIDS

Operation is permitted at power above P_T where P_T equals the ratio of $[F_Q]^L$ divided by $[F_Q]^P$ if the following Augmented Surveillance (Movable Incore Detection System, MIDS) requirements are satisfied:

a. The axial power distribution shall be measured by MIDS when required such that the limit of $[F_Q]^L/P$ times K(Z) is not exceeded. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation (Z).

1) If $F_j(Z)$ exceeds $[F_j(Z)]_s$* as defined in the bases by $\leq$ 4%, immediately reduce thermal power one percent for every percent by which $[F_j(Z)]_s$ is exceeded.

2) If $F_j(Z)$ exceeds $[F_j(Z)]_s$ by > 4% immediately reduce thermal power below P_T. Corrective action to reduce $F_j(Z)$ below the limit will permit return to thermal power not to exceed current P_L** as defined in the bases.

b. $F_j(Z)$ shall be determined to be within limits by using MIDS to monitor the thimbles required per Specification 4.2.2.2.c at the following frequencies.

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

2. Immediately following and as a minimum at 2, 4 and 8 hours following the events listed below and in accordance with the Surveillance Frequency Control Program thereafter.

1) Raising the thermal power above P_T, or

2) Movement of control-bank D more than an accumulated total of 15 steps in any one direction.

c. MIDS shall be operable when the thermal power exceeds P_T with:

1) At least two thimbles available for which $\overline{R_j}$ and σ_j as defined in the bases have been determined.

---

* $[F_j(Z)]_s$ is the alarm setpoint for MIDS.

** P_L is reactor thermal power expressed as a fraction of the Rated Thermal Power that is used to calculate $[F_j(Z)]_s$.

SURVEILLANCE REQUIREMENTS (Continued)

2. At least two movable detectors available for mapping $F_j(Z)$.

3. The continued accuracy and representativeness of the selected thimbles shall be verified by using the most recent flux map to update the $\bar{R}$ for each selected thimble. The flux map must be updated in accordance with the Surveillance Frequency Control Program.

where:

$\bar{R}$ = Total peaking factor from a full flux map ratioed to the axial peaking factor in a selected thimble.

j - The thimble location selected for monitoring.

4.2.2.3 Base Load

Base Load operation is permitted at powers above P_T if the following requirements are satisfied:

a. Either of the following preconditions for Base Load operation must be satisfied.

1. For entering Base Load operation with power less than P_T,

a) Maintain THERMAL POWER between $P_T/1.05$ and P_T for at least 24 hours,

b) Maintain the AFD (Delta-I) to within a ± 2% or ± 3% target band for at least 23 hours per 24-hour period.

c) After 24 hours have elapsed, take a full core flux map to determine $F_Q^M(Z)$ unless a valid full core flux map was taken within the time period specified in 4.2.2.1d.

d) Calculate P_{BL} per 4.2.2.3b.

2. For entering Base Load operation with power greater than P_T,

a) Maintain THERMAL POWER between P_T and the power limit determined in 4.2.2.2 for at least 24 hours, and maintain Augmented Surveillance requirements of 4.2.2.2 during this period.

b) Maintain the AFD (Delta-I) to within a ± 2% or ± 3% target band for at least 23 hours per 24-hour period,

SURVEILLANCE REQUIREMENTS (Continued)

c) After 24 hours have elapsed, take a full core flux map to determine $F^M_Q(Z)$ unless a valid full core flux map was taken within the time period specified in 4.2.2.1d.

d) Calculate P_{BL} per 4.2.2.3b.

b. Base Load operation is permitted provided:

1. THERMAL POWER is maintained between P_T and P_{BL} or between P_T and 100% (whichever is most limiting).

2. AFD (Delta-I) is maintained within a ± 2% or ± 3% target band.

3. Full core flux maps are taken at least once per 31 effective Full Power Days.

P_{BL} and P_T are defined as:

$$P_{BL} = \frac{[F_Q]^L \text{ X } K(Z)}{F^M_Q(Z) \text{ X } W(Z)\text{ BL X } 1.09}$$

$$P_T = [F_Q]^L/[F_Q]^P$$

where: $F^M_Q(Z)$ is the measured $F_Q(Z)$ with no allowance for manufacturing tolerances or measurement uncertainty. For the purpose of this Specification $[F^M_Q(Z)]$ shall be obtained between elevations bounded by 10% and 90% of the active core height. $[F_Q]^L$ is the F_Q limit. K(Z) is given in the CORE OPERATING LIMITS REPORT. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation.

The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6. The 9% uncertainty factor accounts for manufacturing tolerance, measurement error, rod bow and any burnup and power dependent peaking factor increases.

c. During Base Load operation, if the THERMAL Power is decreased below P_T, then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

d. If any of the conditions of 4.2.2.3b are not maintained, reduce THERMAL POWER to less than or equal to P_T, or, within 15 minutes initiate the Augmented Surveillance (MIDS) requirements of 4.2.2.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.4 RADIAL BURNDOWN

Operation is permitted at powers above P_T if the following Radial Burndown conditions are satisfied:

a. Radial Burndown operation is restricted to use at powers between P_T and P_{RB} or P_T and 1.00 (whichever is most limiting). The maximum relative power permitted under Radial Burndown operation, P_{RB}, is equal to minimum value of the ratio of $[F^L_Q Z)]/[F_Q(Z)]_{RB}$ Meas.

where: $[F_Q(Z)]_{RB\ Meas.} = [F_{xy}(Z)]_{Map}$ Meas. x $F_Z(Z)$ x 1.09 and

$[F^L_Q(Z)]$ is equal to $[F^L_Q]$ x K(Z).

b. A full core flux map to determine $[F_{xy}(Z)]_{Map}$ Meas. shall be taken within the time period specified in Section 4.2.2.1d.2. For the purpose of the specification, $[F_{xy}(Z)]_{Map}$ Meas. shall be obtained between the elevations bounded by 10% and 90% of the active core height.

c. The function $F_z(Z)$, provided in the Peaking Factor Limit Report (6.9.1.6), is determined analytically and accounts for the most perturbed axial power shapes which can occur under axial power distribution control. The uncertainty factor of 9% accounts for manufacturing tolerances, measurement error, rod bow, and any burnup dependent peaking factor increases.

d. Radial Burndown operation may be utilized at powers between P_T and P_{RB}, or P_T and 1.00 (whichever is most limiting) provided that the AFD (Delta-I) is within ± 5% of the target axial offset.

e. If the requirements of Section 4.2.2.4d are not maintained, then the power shall be reduced to less than or equal to P_T, or within 15 minutes Augmented Surveillance of hot channel factors shall be initiated if the power is above P_T.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specifications 4.2.2.1, 4.2.2.2, 4.2.2.3 or 4.2.2.4 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^{N}_{\Delta H}$ shall be limited by the following relationship:

$$F^{N}_{\Delta H} \leq F^{RTP}_{\Delta H}[1.0 + PF_{\Delta H}(1-P)],$$

Where: $F^{RTP}_{\Delta H}$ = $F_{\Delta}H$ limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

$PF_{\Delta}H$ = Power Factor Multiplier for $F_{\Delta}H$ as specified in the CORE OPERATING LIMITS REPORT

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^{N}_{\Delta H}$ exceeding its limit:

a. Within 2 hours either:

1. Restore $F^{N}_{\Delta H}$ to within the above limit, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^{N}_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and /or b., above; subsequent POWER OPERATION may proceed provided that $F^{N}_{\Delta H}$ is demonstrated, through incore flux mapping, to be within the limit of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% or RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 When a measurement of $F^N_{\Delta H}$ is taken, the measured $F^N_{\Delta H}$ shall be increased by 4% to account for measurement error.

4.2.3.3 This corrected $F^N_{\Delta H}$ shall be determined to be within its limit through incore flux mapping:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. In accordance with the Surveillance Frequency Control Program.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of -limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio in accordance with the Surveillance Frequency Control Program when the Power Range Upper Detector High Flux Deviation and Power Range Lower Detector High Flux Deviation Alarms are OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when either alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained either from two sets of four symmetric thimble locations or full-core flux map, or by incore thermocouple map is consistent with the indicated QUADRANT POWER TILT RATIO in accordance with the Surveillance Frequency Control Program.

4.2.4.3 If the QUADRANT POWER TILT RATIO is not within its limit within 24 hours and the POWER DISTRIBUTION LIMITS of 3.2.2 and 3.2.3 are within their limits, a Special Report in accordance with 6.9.2 shall be submitted within 30 days including an evaluation of the cause of the discrepancy.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} is less than or equal to the limit specified in the COLR

b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and

c. Reactor Coolant System Flow $\geq$ 270,000 gpm

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less then 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program

4.2.5.2 RCS flow rate shall be monitored for degradation in accordance with the Surveillance Frequency Control Program.

4.2.5.3 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

4.2.5.4 After each fuel loading, and in accordance with the Surveillance Frequency Control Program, the RCS flow rate shall be determined by precision heat balance after exceeding 90% RATED THERMAL POWER. The measurement instrumentation shall be calibrated within 90 days prior to the performance of the calorimetric flow measurement. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirement specified in Table 4.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen.	2/stm. gen.	1, 2	6
12. Steam Generator Water Level--Low Coincident With Steam/ Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen.	1 stm. gen. level coin-cident with 1 stm./feed-water flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6
13. Undervoltage--4.16 KV Busses A and B (Above P-7)	2/bus	1/bus on both busses	2/bus	1	12
14. Underfrequency-Trip of Reactor Coolant Pump Breaker(s) Open (Above P-7)	2/bus	1 to trip RCPs***	2/bus	1	11
15. Turbine Trip (Above P-7)					
a. Emergency Trip Header Pressure	3	2	2	1	12
b. Turbine Stop Valve Closure	2	2	2	1	12

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Safety Injection Input from ESF	2	1	2	1, 2	8
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2#	7
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	7
or					
Turbine Inlet Pressure	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
18. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	1/breaker	1	1/breaker	1	11
b. Above P-7 and below P-8	1/breaker	2	1/breaker	1	11
19. Reactor Trip Breakers	2	1	2	1, 2	8, 10
	2	1	2	3*, 4*, 5*	9
20. Automatic Trip and Interlock logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

* When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

** When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.

*** Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1 (UF-4A1) or UF-3B1(UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).

Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 10- With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ACTUATION LOGIC TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 13 - With the number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	SFCP(11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	SFCP	SFCP(2, 4), SFCP(3, 4), SFCP(4, 6),[(a), (b)] SFCP(4)[(a), (b)]	SFCP[(a), (b)]	N.A.	N.A.	1, 2
b. Low Setpoint	SFCP	SFCP(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Intermediate Range, Neutron Flux	SFCP	SFCP(4)	S/U(1)	N.A.	N.A.	1***, 2
4. Source Range, Neutron Flux	SFCP	SFCP(4)	S/U(1), SFCP(9)	N.A.	N.A.	2**,3, 4, 5
5. Overtemperature ΔT	SFCP	SFCP[(a), (b)]	SFCP[(a), (b)]	N.A.	N.A.	1, 2
6. Overpower ΔT	SFCP	SFCP[(a), (b)]	SFCP[(a), (b)]	N.A.	N.A.	1, 2
7. Pressurizer Pressure--Low	SFCP	SFCP	SFCP	N.A.	N.A.	1
8. Pressurizer Pressure--High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
9. Pressurizer Water Level--High	SFCP	SFCP	SFCP	N.A.	N.A.	1
10. Reactor Coolant Flow--Low	SFCP	SFCP[(a), (b)]	SFCP[(a), (b)]	N.A.	N.A.	1
11. Steam Generator Water Level--Low-Low	SFCP	SFCP[(a), (b)]	SFCP[(a), (b)]	N.A.	N.A.	1, 2

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	SFCP	SFCP[(a), (b)]	SFCP[(a), (b)]	N.A.	N.A.	1, 2
13. Undervoltage – 4.16 kV Busses A and B	N.A.	SFCP	N.A.	N.A.	N.A.	1
14. Underfrequency – Trip of Reactor Coolant Pump Breakers(s) Open	N.A.	SFCP	N.A.	N.A.	N.A.	1
15. Turbine Trip						
a. Emergency Trip Header Pressure	N.A.	SFCP[(a), (b)]	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	SFCP	N.A.	S/U(1, 10)	N.A.	1
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	SFCP(4)	SFCP	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7 (includes P-10 input and Turbine Inlet Pressure)	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
17. Reactor Trip System Interlocks (Continued)						
d. Power Range Neutron Flux, P-10	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1, 2
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	SFCP	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	SFCP(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	SFCP(7,14)	1, 2, 3*, 4*, 5*
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	SFCP(13), SFCP(15)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(a) 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.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in UFSAR Section 7.2.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power level indication above 15% of RATED THERMAL POWER (RTP). Adjust excore channel gains consistent with calorimetric power level if the absolute difference is greater than 2%. Below 70% RTP, downward adjustments of NIS excore channel gains to match a lower calorimetric power level are not required. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(5) This table Notation number is not used.

(6) Incore-Excore Calibration, above 75% of RATED THERMAL POWER (RTP). If the quarterly surveillance requirement coincides with sustained operation between 30% and 75% of RTP, calibration shall be performed at this lower power level. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(8) DELETED

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissive P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of 1/2 decade above the existing count rate.

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the undervoltage and shunt trip attachment of the Reactor Trip Breakers.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(12) NOT USED.

(13) Remote manual undervoltage trip when breaker placed in service.

(14) Interlock Logic Test shall consist of verifying that the interlock is in its required state by observing the permissive annunciator window.

(15) Automatic undervoltage trip.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value within permissible calibration tolerance.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-3, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-3 and determine within 12 hours that the affected channel is OPERABLE; or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection					
a. Manual Initiation	2	1	2	1 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1 2, 3, 4	14
c. Containment Pressure - High	3	2	2	1 2, 3	15
d. Pressurizer Pressure - Low	3	2	2	1 2, 3#	15
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1 2, 3#	15

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15
Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements. (Manual S.I. initiation will not initiate Phase A Isolation.)				
b. Phase "B" Isolation					
1) Manual Initiation	2	2 (Both buttons must be pushed simultaneously to actuate)	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15
c. Containment Ventilation Isolation					
1) Containment Isolation Manual Phase A or Manual Phase B	See Items 3.a.1 and 3.b.1 above for all Manual Containment Ventilation functions and requirements.				

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	16
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions requirements.				
4) Containment Radioactivity-High	2##	1	1	1, 2, 3, 4	16
4. Steam Line Isolation					
a. Manual Initiation (individual)	1/operating steam line	1/operating steam line	1/operating steam line	1, 2, 3	21
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/Loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	25
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Steam Generator Water Level -- High-High####	3/steam generator	2/steam generator in any operating steam generator	2/steam generator in any operating steam generator	1, 2, 3	15
6. Auxiliary Feedwater###					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level--Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	15
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23
e. Trip of All Main Feed-water Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Loss of Power (Continued)					
c. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C 4D Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure	3	2	2	1, 2, 3	19
b. T_{avg} - Low	3	2	2	1, 2, 3	19
9. Control Room Ventilation Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4,6**	16
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Containment Radio-activity--High	2	1	1	1, 2, 3, 4,6**	16
d. Containment Isolation Manual Phase A or Manual Phase B	2	1	2	1, 2, 3, 4	17
e. Control Room Air Intake Radiation Level	2	1	2	All	24

TABLE 3.3-2 (Continued)

TABLE NOTATION

Trip function may be blocked in this MODE below the Pressurizer Pressure Interlock Setpoint of 2000 psig.

Channels are for particulate radioactivity and for gaseous radioactivity.

Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

Steam Generator overfill protection is not part of the Engineered Safety Features Actuation System (ESFAS), and is added to the Technical Specifications only in accordance with NRC Generic Letter 89-19.

* Trip function may be blocked in this MODE below the Tavg--Low Interlock Setpoint.

** Only during CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 16 - With less than the Minimum Channels OPERABLE requirement, comply with the ACTION statement requirements of Specification 3.3.3.1 Item 1a of Table 3.3-4.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. Both channels of any one load center may be taken out of service for up to 8 hours in order to perform surveillance testing per Specification 4.3.2.1.

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, comply with Specification 3.0.3.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the control room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 25 - With number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

TABLE 3.3-3
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A	N.A
c. Containment Pressure--High	≤4.5 psig	≤4.0 psig
d. Pressurizer Pressure--Low	≥1712 psig	≥1730 psig
e. High Differential Pressure Between the Steam Line Header and any Steam Line.	≤114 psig	≤100 psi
f. Steam Line Flow--High	≤A function defined as follows: A ΔP corresponding to 41.2% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 114.4% steam flow at full load	A function defined as follows: A ΔP corresponding to 40% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 114% steam flow at full load

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
Coincident with: Steam Generator Pressure--Low(4)	≥607 psig	614 psig
or T_{avg}--Low	≥542.5°F	≥543°F
2. Containment Spray		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Containment Pressure--High-High Coincident with:	≤22.6 psig	≤20.0 psig
Containment Pressure--High	≤4.5 psig	≤4.0 psig
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1 above for all Safety Injection Allowable Values.	See Item 1 above for all Safety Injection Trip Setpoints.
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.

TABLE 3. ntinued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
3. Containment Isolation (Continued)		
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High Coincident with:	$\leq$22.6 psig	$\leq$20.0 psig
Containment Pressure--High	$\leq$4.5 psig	$\leq$4.0 psig
c. Containment Ventilation Isolation		
1) Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
4) Containment Radioactivity--High (1)	Particulate (R-11) $\leq 6.8 \times 10^5$ CPM Gaseous (R-12) See Note 2	Particulate (R-11) $\leq 6.1 \times 10^5$ CPM Gaseous (R-12) See Note 2
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (Continued)		
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High Coincident with:	≤22.6 psig	≤20. psig
Containment Pressure--High	≤ 4.5 psig	≤4.0 psig
d. Steam Line Flow--High	≤A function defined as follows: A ΔP corresponding to 41.2% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 114.4% steam flow at full load.	A function defined as follows: A ΔP corresponding to 40% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 114% steam flow at full load.
Coincident with: Steam Line Pressure--Low(4)	≥607 psig	614 psig
or T_{avg} -- Low	≥542.5°F	≥543°F
5. Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. for all Safety Injection Allowable Valves.	See Item 1. above for all Safety Injection Trip Setpoints.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
5. Feedwater Isolation (Continued)		
c. Steam Generator Water Level High-High	≤80.5% of narrow range instrument span	80% of narrow range instrument span
6. Auxiliary Feedwater (3)		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level–Low-Low	≥15.5% of narrow range instrument span.	16% of narrow range instrument span
c. Safety Injection	See Item 1. for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
d. Bus Stripping	See Item 7. below for all Bus Stripping Allowable Values.	See Item 7. below for all Bus Stripping Trip Setpoints.
e. Trip of All Main Feedwater Pump Breakers	N.A.	N.A.
7. Loss of Power		
a. 4.16 kV Busses A and B (Loss of Voltage)	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE#	TRIP SETPOINT
7. Loss of Power (Continued)		
b. 480V Load Centers Undervoltage		
Load Center		
3A	[ ]	430V ±3V (10 sec ±1 sec delay)
3B	[ ]	438V ±3V (10 sec ±1 sec delay)
3C	[ ]	434V ±3V (10 sec ±1 sec delay)
3D	[ ]	434V ±3V (10 sec ±1 sec delay)
4A	[ ]	435V ±3V (10 sec ±1 sec delay)
4B	[ ]	434V ±3V (10 sec ±1 sec delay)
4C	[ ]	434V ±3V (10 sec ±1 sec delay)
4D	[ ]	430V ±3V (10 sec ±1 sec delay)
Coincident with: Safety Injection and	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
Diesel Generator Breaker Open	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE#	TRIP SETPOINT
7. Loss of Power (Continued)		
c. 480V Load Centers Degraded Voltage		
Load Center		
3A	[ ]	424V ±3V (60 sec ±30 sec delay)
3B	[ ]	427V ±3V (60 sec ±30 sec delay)
3C	[ ]	437V ±3V (60 sec ±30 sec delay)
3D	[ ]	435V ±3V (60 sec ±30 sec delay)
4A	[ ]	430V ±3V (60 sec ±30 sec delay)
4B	[ ]	436V ±3V (60 sec ±30 sec delay)
4C	[ ]	434V ±3V (60 sec ±30 sec delay)
4D	[ ]	434V ±3V (60 sec ±30 sec delay)
Coincident with: Diesel Generator Breaker Open	N.A.	N.A.

TABLE 3.3-3 (continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
8. Engineering Safety Features Actuation System Interlocks		
a. Pressurizer Pressure	$\leq$2018 psig	Nominal 2000 psig
b. Tavg--Low	$\geq$542.5°F	Nominal 543°F
9. Control Room Ventilation Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
c. Containment Radioactivity--High (1)	Particulate (R-11) $\leq 6.8 \times 10^5$ CPM Gaseous (R-12) See Note 2	Particulate (R-11) $\leq 6.1 \times 10^5$ CPM Gaseous (R-12) See Note 2
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.
e. Air Intake Radiation Level	$\leq$2.83 mR/hr	$\leq$2 mR/hr

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

(1) Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

(2) Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(\quad F \quad)}$ CPM,

Containment Gaseous Monitor Allowable Value = $\frac{(3.5 \times 10^4)}{(\quad F \quad)}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

(3) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

(4) Time constants utilized in lead-lag controller for Steam Generator Pressure-Low and Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

If no Allowable Value is specified, as indicated by [ ], the trip setpoint shall also be the allowable value.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection						
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	1, 2, 3(3)
c. Containment Pressure--High	N.A.	SFCP	N.A.	N.A.	SFCP(1)	1, 2, 3
d. Pressurizer Pressure--Low	SFCP	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3(3)
e. High Differential Pressure Between the Steam Line Header and any Steam Line	SFCP	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3(3)
f. Steam Line Flow--High Coincident with:	SFCP	SFCP(a)(b)	SFCP(5)(a)(b)	N.A	N.A	1, 2, 3(3)
Steam Generator Pressure--Low	SFCP	SFCP(a)(b)	SFCP(5)(a)(b)	N.A.	N.A.	1, 2, 3(3)
or						
Tavg--Low	SFCP	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3(3)

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Containment Spray						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	1, 2, 3, 4
b. Containment Pressure--High-High	N.A.	SFCP	N.A.	SFCP	SFCP(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	SFCP	N.A.	SFCP	SFCP(1)	1, 2, 3
3. Containment Isolation						
a. Phase "A" Isolation						
1) Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
b. Phase "B" Isolation						
1) Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)						
3) Containment Pressure--High-High Coincident	N.A.	SFCP	N.A.	SFCP	SFCP(1)	1, 2, 3
with: Containment Pressure--High	N.A.	SFCP	N.A.	SFCP	SFCP(1)	1, 2, 3
c. Containment Ventilation Isolation						
1) Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
4) Containment Radioactivity--High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2, 3, 4
4. Steam Line Isolation						
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	1, 2, 3(3)

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steamline Isolation (Continued)						
c. Containment Pressure--High-High	N.A.	SFCP	N.A.	SFCP	SFCP(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	SFCP	N.A	SFCP	SFCP(1)	1, 2, 3
d. Steam Line Flow--High	SFCP(3)	SFCP[(a)(b)]	SFCP(5)[(a)(b)]	N.A.	N.A.	1, 2, 3
Coincident with: Steam Generator Pressure--Low	SFCP(3)	SFCP[(a)(b)]	SFCP(5)[(a)(b)]	N.A.	N.A.	1, 2, 3
or Tavg--Low	SFCP(3)	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP	1, 2, 3
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Steam Generator Water Level--High-High	SFCP	SFCP[(a)(b)]	SFCP[(a)(b)]	N.A.	N.A.	1, 2, 3
6. Auxiliary Feedwater (2)						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP	1, 2, 3
b. Steam Generator Water Level--Low-Low	SFCP	SFCP[(a)(b)]	SFCP[(a)(b)]	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)						
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
d. Bus Stripping	N.A.	SFCP	N.A.	SFCP	N.A.	1, 2, 3
e. Trip of All Main Feedwater Pump Breakers.	N.A.	N.A.	N.A.	SFCP	N.A.	1. 2
7. Loss of Power						
a. 4.16 kV busses A and B (Loss of Voltage)	N.A.	SFCP	N.A.	SFCP	N.A.	1, 2, 3, 4
b. 480V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	SFCP	SFCP	N.A.	SFCP(1)	N.A.	1, 2, 3, 4
Coincident with: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. 480V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Degraded Voltage	SFCP	SFCP	N.A.	SFCP(1)	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Engineering Safety Features Actuation System Interlocks						
a. Pressurizer Pressure	N.A.	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3(3)
b. Tavg--Low	N.A.	SFCP	SFCP(5)	N.A.	N.A.	1, 2, 3(3)
9. Control Room Ventilation Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Containment Radioactivity--High	SFCP	SFCP	SFCP	N.A.	N.A.	(4)
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2, 3, 4
e. Control Room Air Intake Radiation Level	SFCP	SFCP	SFCP	N.A.	N.A.	All

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

In accordance with the Surveillance Frequency Control Program each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay.

(a) 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.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in UFSAR Section 7.2

(1) Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

(3) The provisions of Specification 4.0.4 are not applicable for entering Mode 3, provided that the applicable surveillances are completed within 96 hours from entering Mode 3.

(4) Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment.

(5) Test of alarm function not required when alarm locked in.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-4 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-4, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-4.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous (See Note 1.))	1	1*	All*	Particulate $\leq 6.1 \times 10^5$ CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 for MODES 5 and 6
b. RCS Leakage Detection Particulate Radio-activity or Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26
2. Spent Fuel Storage Pool Areas					
a. Unit 3 Radioactivity – High Gaseous	1	1	**	$<5.5 \times 10^{-2} \frac{\mu Ci}{cc}$	28
b. Unit 4 Radioactivity – High Gaseous**#**	1	1	**	$<2.8 \times 10^{-2} \frac{\mu Ci}{cc}$ (SPING) or $<1.0 \times 10^6$ CPM (PRMS)	28

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

* During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.

** With irradiated fuel in the spent fuel pits.

Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(\quad F \quad)}$ CPM,

$$\text{Where } F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment sump level monitoring system is OPERABLE,

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,

3) A Reactor Coolant System water inventory balance is performed at least once per 8*** hours except when operating in shutdown cooling mode, and

4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.****

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

*** Not required to be performed until 12 hours after establishment of steady state operation.

**** Instrument Air Bleed Valves may be opened intermittently under administrative controls.

TABLE 3.3-4 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - In MODES 5 or 6 (except during CORE ALTERATION or movement of irradiated fuel within the containment): With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement perform the following:

1) Obtain and analyze appropriate grab samples at least once per 24 hours, and

2) Monitor containment atmosphere with area radiation monitors.

Otherwise, isolate all penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

During CORE ALTERATION or movement of irradiated fuel within the containment: With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements, comply with ACTION statement requirements of Specification 3.9.9 and 3.9.13.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, immediately suspend operations in the Spent Fuel Pool area involving spent fuel manipulations.

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Radioactivity--High	SFCP	SFCP	SFCP	All
b. RCS Leakage Detection				
1) Particulate Radio-activity	SFCP	SFCP	SFCP	1, 2, 3, 4
2) Gaseous Radioactivity	SFCP	SFCP	SFCP	1, 2, 3, 4
2. Spent Fuel Pool Areas				
a. Unit 3 Radioactivity--High Gaseous	SFCP	SFCP	SFCP	*
b. Unit 4 (Plant Vent) Radioactivity--High Gaseous# (SPING and PRMS)	SFCP	SFCP	SFCP	*

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.

\# Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 16 detector thimbles when used for recalibration and check of the Excore Neutron Flux Detection System and monitoring the QUANDRANT POWER TILT RATIO*, and at least 38 detector thimbles when used for monitoring $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABLITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

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* Exception to the 16 detector thimble requirement of monitoring the QUADRANT POWER TILT RATIO is acceptable when performing Specification 4.2.4.2 using two sets of four symmetric thimbles.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
1. Containment Pressure (Wide Range)	2	1	1, 2, 3	31, 32
2. Containment Pressure (Narrow Range)	2	1	1, 2, 3	36
3. Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
4. Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
5. Reactor Coolant Pressure – Wide Range	2	1	1, 2, 3	31, 32
6. Pressurizer Water Level	2	1	1, 2, 3	31, 32
7. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator	1, 2, 3	31, 32
8. Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1, 2, 3	31, 32
9. PORV Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	33
10. PORV Block Valve Position Indicator	1/valve	1/valve	1, 2, 3	33
11. Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	32
12. Containment Water Level (Narrow Range)	2	1	1, 2, 3	36
13. Containment Water Level (Wide Range)	2	1	1, 2, 3	31, 32

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 33 Close the associated block valve and open its circuit breaker.

ACTION 34 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

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

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

ACTION 35 DELETED

ACTION 36 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 30 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 38 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 7 days. If repairs are not feasible without shutting down:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

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

3. Restore at least one channel to OPERABLE status at the next scheduled refueling.

ACTION 39 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, verify position by an alternate means (e.g. administrative controls, ERDADS, alternate position indication, or visual observation) within 2 hours, and restore the inoperable channel(s) within 7 days, or comply with the provisions of Specification 3.6.4 for an inoperable containment isolation valve.

TABLE 4.3-4

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure (Wide Range)	SFCP	SFCP
2.	Containment Pressure (Narrow Range)	SFCP	SFCP
3.	Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	SFCP	SFCP
4.	Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	SFCP	SFCP
5.	Reactor Coolant Pressure - Wide Range	SFCP	SFCP
6.	Pressurizer Water Level	SFCP	SFCP
7.	Auxiliary Feedwater Flow Rate	SFCP	SFCP
8.	Reactor Coolant System Subcooling Margin Monitor	SFCP	SFCP
9.	PORV Position Indicator (Primary Detector)	SFCP	SFCP
10.	PORV Block Valve Position Indicator	SFCP	SFCP
11.	Safety Valve Position Indicator (Primary Detector)	SFCP	SFCP
12.	Containment Water Level (Narrow Range)	SFCP	SFCP
13.	Containment Water Level (Wide Range)	SFCP	SFCP
14.	In Core Thermocouples (Core Exit Thermocouples)	SFCP	SFCP
15.	Containment - High Range Area Radiation Monitor	SFCP	SFCP*
16.	Reactor Vessel Level Monitoring System	SFCP	SFCP
17.	Neutron Flux, Backup NIS (Wide Range)	SFCP	SFCP
18.	DELETED		
19.	High Range - Noble Gas Effluent Monitors		
	a. Plant Vent Exhaust	SFCP	SFCP
	b. Unit 3 - Spent Fuel Pit Exhaust	SFCP	SFCP
	c. Condenser Air Ejectors	SFCP	SFCP
20.	RWST Water Level	SFCP	SFCP
21.	Steam Generator Water Level (Narrow Range)	SFCP	SFCP
22.	Containment Isolation Valve Position Indication	SFCP	SFCP

*Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

INSTRUMENTATION

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The explosive gas monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.7.8 are not exceeded.

APPLICABILITY: As shown in Table 3.3-8

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-8. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful prepare and submit a special report to the Commission within 30 days to explain why this inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-6.

TABLE 3.3-8
EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTIONS
1. WASTE GAS DISPOSAL SYSTEM (Explosive Gas Monitoring System)			
a. Hydrogen and Oxygen Monitors	1	*	49

TABLE NOTATION

* During GAS DECAY TANK SYSTEM operation.

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the GAS DECAY TANK SYSTEM may continue provided that grab samples are collected and analyzed for hydrogen and oxygen concentration at least a) once per 8 hours during degassing operations, and b) once per day during other operations.

TABLE 4.3-6

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GAS DECAY TANK SYSTEM (Explosive Gas Monitoring System)					
a. Hydrogen and Oxygen Monitors	SFCP	N.A.	SFCP(1,2)	SFCP	*

TABLE NOTATION

* During GAS DECAY TANK SYSTEM operation.

(1) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal.

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

(2) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 All of the reactor coolant loops listed below shall be OPERABLE with all reactor coolant loops in operation when the Reactor Trip breakers are closed and two reactor coolant loops listed below shall be OPERABLE with at least one reactor coolant loop in operation when the Reactor Trip breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump, and

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than three reactor coolant loop in operation and the Reactor Trip breakers in the closed position, within 1 hour open the Reactor Trip breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE by verifying correct breaker alignments and indicated power availability in accordance with the Surveillance Frequency Control Program.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% in accordance with the Surveillance Frequency Control Program.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR Loop A, and

e. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 275°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE by verifying correct breaker alignments and indicated power availability in accordance with the Surveillance Frequency Control Program.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% in accordance with the Surveillance Frequency Control Program.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.4.1.3.4 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.*

* Not required to be performed until 12 hours after entering MODE 4.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than 10%.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits in accordance with the Surveillance Frequency Control Program.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.4.1.4.1.3 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

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* The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE.

*** A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 275°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR loop shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.4.1.4.2.2 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

* One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE.

** The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE* with a lift setting of 2465 psig + 2%, -3%.** ***

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

* While in MODE 5, an equivalent size vent pathway may be used provided that the vent pathway is not isolated or sealed.

** The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

*** All valves tested must have "as left" lift setpoints that are within ± 1% of the lift setting value.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2465 psig + 2%, -3%.* **

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** All valves tested must have "as left" lift setpoints that are within ± 1% of the lift setting value.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of indicated level, and at least two groups of pressurizer heaters each having a capacity of at least 125 kW and capable of being supplied by emergency power.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 125 kw in accordance with the Surveillance Frequency Control Program.

** 14 days if the inoperability is associated with an inoperable diesel generator.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable because of excessive leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive leakage, within 1 hour either restore at least one PORV to OPERABLE status or close each PORV's associated block valve and remove power from the block valve; with both block valves closed with power removed, restore at least one PORV to OPERABLE status within 30 days and restore power to its associated block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

d. With one or both block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, place its associated PORV in manual control within the next hour and be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours. Restore at least one block valve to OPERABLE status within 30 days if both block valves are inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 In accordance with the Surveillance Frequency Control Program each block valve shall be demonstrated OPERABLE by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of Specification 3.4.4 or is closed to provide an isolation function.

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the SG Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION*:

a. With one or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program;

   1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

   2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. With the requirements and associated allowable outage time of Action a above not met or SG tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

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*Separate Action entry is allowed for each SG tube.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and

b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment Sump Level Monitoring System is OPERABLE;

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;

3) A Reactor Coolant System water inventory balance is performed at least once per 8* hours except when operating in shutdown cooling mode; and

4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.**

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With no Containment Sump Level Monitoring System operable, restore at least one Containment Sump Level Monitoring System to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

* Not required to be performed until 12 hours after establishment of steady state operation.

** Instrument Air Bleed valves may be opened intermittently under administrative controls.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATING

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4-1 up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than allowed by 3.4.6.2.e above operation may continue provided:

   1. Within 4 hours verify that at least two valves in each high pressure line having a non-functional valve are in, and remain in that mode corresponding to the isolated condition, i.e., manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized. Follow applicable ACTION statement for the affected system, and

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* Test pressure less than 2235 psig are allowed. Minimum differential test pressure shall not be less than 150 psid. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

2. The leakage* from the remaining isolating valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1, as listed in Table 3.4-1, shall be determined and recorded daily. The positions of the other valves located in the high pressure line having the leaking valve shall be recorded daily unless they are manual valves located inside containment.

Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than 5 gpm, reduce leakage to below 5 gpm within 1 hour, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous or particulate radioactivity monitor in accordance with the Surveillance Frequency Control Program.

b. Monitoring the containment sump level in accordance with the Surveillance Frequency Control Program.

c.** Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program***; and

d. Monitoring the Reactor Head Flange Leakoff System in accordance with the Surveillance Frequency Control Program; and

e. Verifying primary-to-secondary leakage is $\leq$ 150 gallons per day through any one SG in accordance with the Surveillance Frequency Control Program***.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage* to be within its limit:

a. In accordance with the Surveillance Frequency Control Program.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months, and

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

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

** Not applicable to primary-to-secondary leakage.

*** Not required to be performed until 12 hours after establishment of steady state operation.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION (Continued)

d. Following valve actuation due to automatic or manual action or flow through the valve:

   1. Within 24 hours by verifying valve closure, and

   2. Prior to entering Mode 2 by verifying leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER		FUNCTION
Unit 3	Unit 4	High-Head Safety Injection Check Valves
3-874A	4-874A	Loop A, hot leg
3-875A	4-875A	cold leg
3-873A	4-873A	cold leg
3-874B	4-874B	Loop B, hot leg
3-875B	4-875B	cold leg
3-873B	4-873B	cold leg
3-875C	4-875C	Loop C,cold leg
3-873C	4-873C	cold leg
		Residual Heat Removal Line Check Valves
3-876A	4-876A 4-876E	Loop A, cold leg
3-876B 3-876D	4-876B 4-876D	Loop B, cold leg
3-876C 3-876E	4-876C	Loop C, cold leg
	MOV4-750 MOV4-751	Loop A, hot leg to RHR
MOV3-750 MOV3-751		Loop C, hot leg to RHR

ACCEPTABLE LEAKAGE LIMITS

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable provided that the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

REACTOR COOLANT SYSTEM

3/4.4.7 DELETED

TABLE 3.4-2

DELETED

TABLE 4.4-3

DELETED

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 0.25 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries per gram once per 4 hours.

b. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 60 microcuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 0.25 microcuries per gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131 for greater than or equal to 48 hours during one continuous time interval, or greater than 60 microcuries per gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

d. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 447.7 microcuries per gram limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133 for greater than or equal to 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performing the sampling and analysis described in Table 4.4-4.

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. NOT USED		
2. Tritium Activity Determination	SFCP.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131	a) SFCP. b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period.	1, 2, 3, 4
4. Radiochemical Isotopic Determination Including Gaseous Activity	SFCP	1, 2, 3, 4
5. Isotopic Analysis for DOSE EQUIVALENT XE-133	SFCP	1, 2, 3, 4
6. NOT USED		

3/4.4.9 PRESSURE TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 5°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit 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 operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: Intermediate/Lower Shell Circumferential Weld Seams Ht. # 71249 and Upper Shell Forging

LIMITING ART VALUES AT 48 EFPY: 1/4 T, 231°F (Circ Flaw ART), 141°F (Axial Flaw ART)
3/4 T, 192°F (Circ Flaw ART), 124°F (Axial Flaw ART)

FPL/FLA 48 EFPY HEATUP CURVES

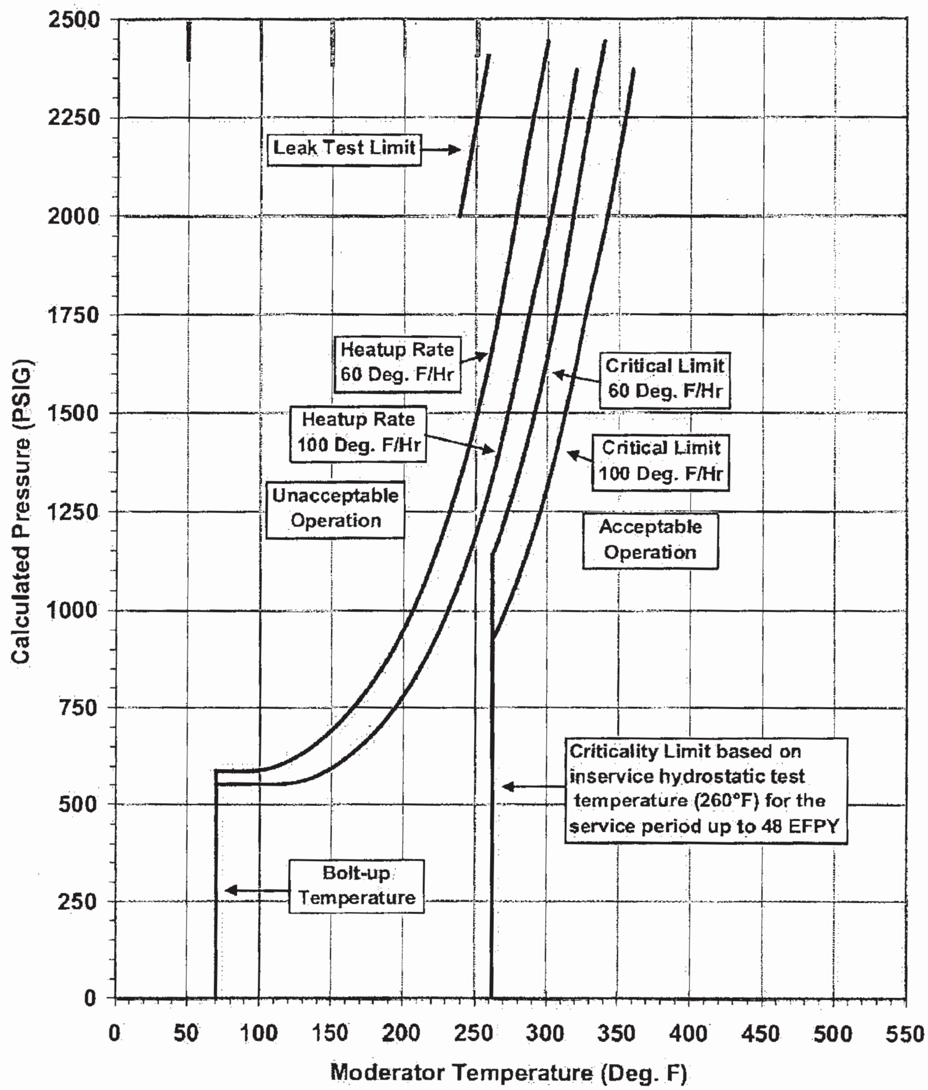

FIGURE 3.4-2 Turkey Point Units 3 & 4 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 48 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: Intermediate/Lower Shell Circumferential Weld Seams Ht. # 71249 and Upper Shell Forging

LIMITING ART VALUES AT 48 EFPY: 1/4 T, 231°F (Circ Flaw ART), 141°F (Axial Flaw ART)
3/4 T, 192°F (Circ Flaw ART), 124°F (Axial Flaw ART)

FPL 48 EFPY COOLDOWN CURVES

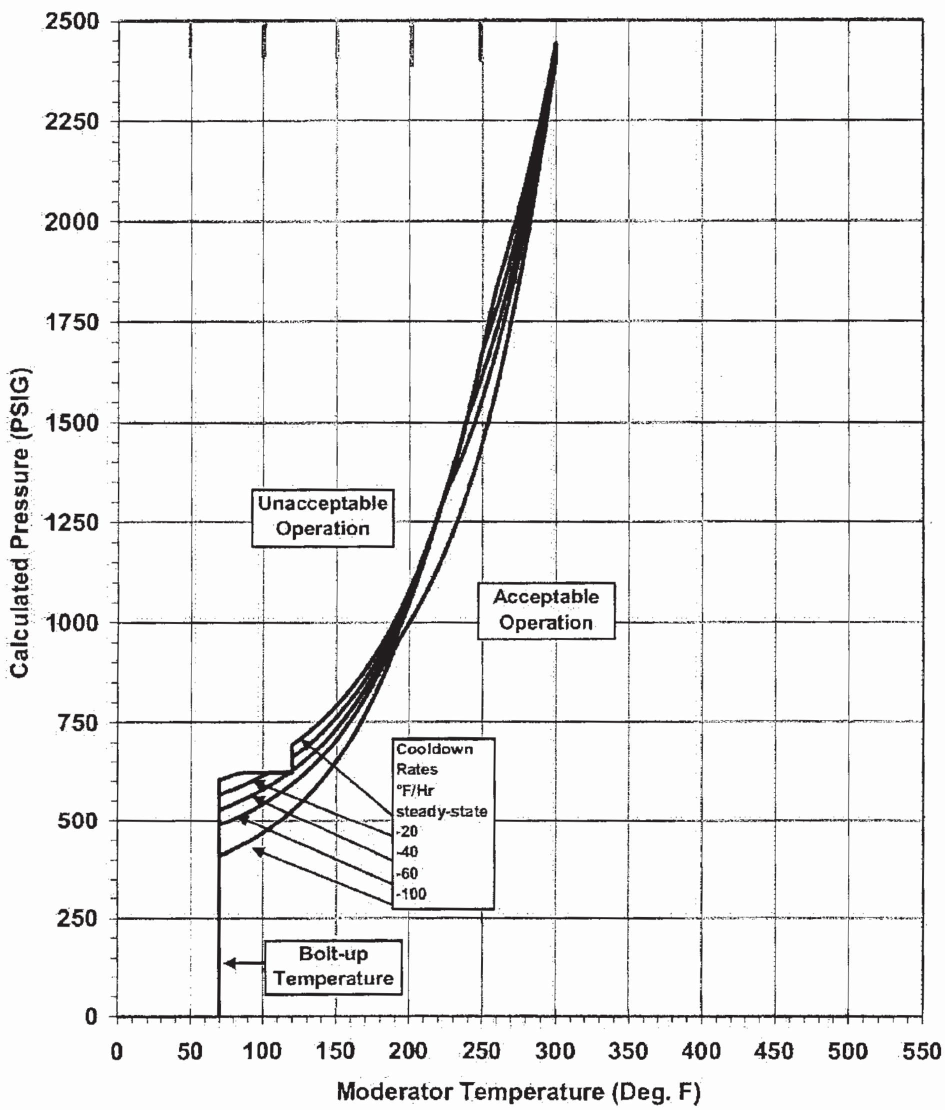

FIGURE 3.4-3 Turkey Point Units 3 & 4 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for 48 EFPY (Without Margins for Instrumentation Errors)

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature 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 pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of ≤ 448 psig, or

b. The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), 5, and 6 with the reactor vessel head on.

ACTION:

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours.

b. With one PORV inoperable in MODE 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

c. With one PORV inoperable in Modes 5 or 6 with the reactor vessel head on, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.20 square inch vent within a total of 32 hours, or (3) complete depressurization and venting of the RCS through at least one open PORV and associated block valve within a total of 32 hours.

d. With both PORVs inoperable, either restore one PORV to OPERABLE status or complete depressurization and venting of the RCS through at least a 2.20 square inch vent within 24 hours.

e. In the event either the PORVs or a 2.20 square inch vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. A Special Report is not required when such a transient is the result of water injection into the RCS for test purposes with an open vent path.

f. The provisions of Specification 3.0.4 are not applicable.

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST* on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and in accordance with the Surveillance Frequency Control Program thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel in accordance with the Surveillance Frequency Control Program; and

c. Verifying the PORV block valve is open in accordance with the Surveillance Frequency Control Program when the PORV is being used for overpressure protection.

d. While the PORVs are required to be OPERABLE, the backup nitrogen supply shall be verified OPERABLE in accordance with the Surveillance Frequency Control Program.*

4.4.9.3.2 The 2.20 square inch vent shall be verified to be open in accordance with the Surveillance Frequency Control Program** when the vent(s) is being used for overpressure protection.

4.4.9.3.3 Verify the high pressure injection flow path to the RCS is isolated in accordance with the Surveillance Frequency Control Program by closed valves with power removed or by locked closed manual valves.

* Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ 275°F.

** Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open in accordance with the Surveillance Frequency Control Program.

3/4.4.10 DELETED

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of at least two vent valves in series and powered from emergency busses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each Reactor Coolant System vent path shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of boron concentration not being within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the boron concentration not being within the limits, restore boron concentration back to the required limits within 72 hours, or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

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

1) Verifying the borated water volume in each accumulator is between 6520 and 6820 gallons, and

2) Verifying that the nitrogen cover pressure in each accumulator is between 600 and 675 psig, and

3) Verifying that each accumulator isolation valve is open by control room indication (power may be restored to the valve operator to perform this surveillance if redundant indicator is inoperable).

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the solution in the water-filled accumulator is between 2300 and 2600 ppm;

c. In accordance with the Surveillance Frequency Control Program, when the RCS pressure is above 1000 psig, by verifying that the power to the isolation valve operator is disconnected by a locked open breaker.

d. In accordance with the Surveillance Frequency Control Program*, each accumulator check valve shall be checked for operability.

* During Unit 3 Cycle 26 only, in lieu of the Technical Specification specified 18 month refueling frequency and 4.5 month grace period allowance, the maximum allowed surveillance test interval will be extended to no more than 24.5 months.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four Safety Injection (SI) pumps, each capable of being powered from its associated OPERABLE diesel generator#, with discharge flow paths aligned to the RCS cold legs,*

b. Two RHR heat exchangers,

c. Two RHR pumps with discharge flow paths aligned to the RCS cold legs,

d. A flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

e. Two flow paths capable of taking suction from the containment sump.

APPLICABILITY: MODES 1, 2, and 3**.

ACTION:

a. With one RHR heat exchanger or suction flow path from the containment sump inoperable, restore the inoperable RHR heat exchanger or suction flow path from the containment sump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water in the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

c. With one of the four required Safety Injection pumps or its associated discharge flow path inoperable and the opposite unit in MODE 1, 2, or 3, restore the pump or flow path to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.***

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*Only three Safety Injection (SI) pumps (two associated with the unit and one from the opposite unit), each capable of being powered from its associated OPERABLE diesel generator#, with discharge flow paths aligned to the RCS cold leg are required if the opposite unit is in MODE 4, 5, or 6.

**The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to Tavg exceeding 380°F. Safety Injection flow paths may be isolated when Tavg is less than 380°F.

***The provisions of Specifications 3.0.4 and 4.0.4 are not applicable.

#Inoperability of the required diesel generators does not constitute inoperability of the associated Safety Injection pumps.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

d. With two of the four required Safety Injection pumps or their associated discharge flow paths inoperable and the opposite unit in MODE 1, 2, or 3, restore one of the two inoperable pumps or flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With one of the three required Safety Injection pumps or its associated discharge flow path inoperable and the opposite unit in MODE 4, 5, or 6, restore the pump or flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. With a required Safety Injection pump OPERABLE but not capable of being powered from its associated diesel generator, restore the capability within 14 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

g. With an ECCS subsystem inoperable due to an RHR pump or its associated discharge flow path being inoperable, restore the inoperable RHR pump or its associated discharge flow path to OPERABLE status within 7 days or be in as least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

h. With the suction flow path from the refueling water storage tank inoperable, restore the suction flow path to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS component and flow path shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying by control room indication that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
864A and B	Supply from RWST to ECCS	Open
862A and B	RWST Supply to RHR pumps	Open
863A and B	RHR Recirculation	Closed
866A and B	H.H.S.I. to Hot Legs	Closed
HCV-758*	RHR HX Outlet	Open

To permit positive valve position indication for surveillance or maintenance purposes in the event that continuous valve position indication is unavailable in the control room, power may be restored to these valves for a period not to exceed 1 hour.

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

1) Verifying ECCS locations susceptible to gas accumulation are sufficiently filled with water, and

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.**

c. By verifying that each SI and RHR pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5:

1) SI pump $\geq$ 1083 psid at a metered flowrate $\geq$ 300 gpm (normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST), or

$\geq$ 1113 psid at a metered flowrate $\geq$ 280 gpm (Unit 3 SI pumps aligned to Unit 4 RWST).

2) RHR pump Develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1.

*Air Supply to HCV-758 shall be verified shut off and sealed closed in accordance with the Surveillance Frequency Control Program.

**Not required to be met for system vent flow paths opened under administrative control.

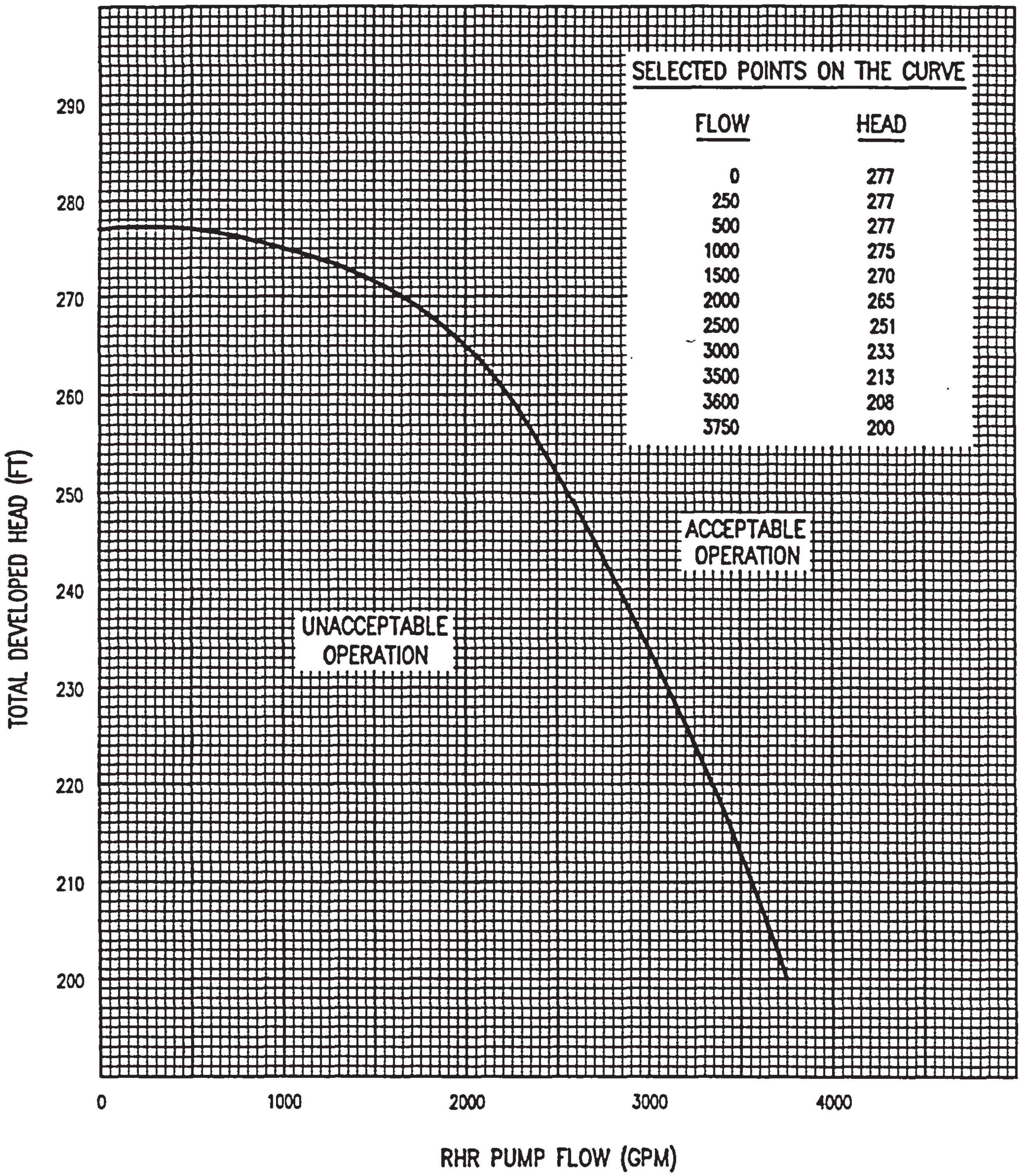

Figure 3.5-1
RHR Pump Curve

d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. The visual inspection shall be performed:

   1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2) At least once daily of the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

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

   1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 525 psig the interlocks cause the valves to automatically close and prevent the valves from being opened, and

   2) Verifying correct interlock action to ensure that the RWST is isolated from the RHR System during RHR System operation and to ensure that the RHR System cannot be pressurized from the Reactor Coolant System unless the above RWST Isolation Valves are closed.

   3) A visual inspection of the containment sump and verifying that the suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

f. In accordance with the Surveillance Frequency Control Program, during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation test signal, and

   2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

      a) Safety Injection pump, and

      b) RHR pump.

SURVEILLANCE REQUIREMENTS

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS components are required to be OPERABLE, and

2) In accordance with the Surveillance Frequency Control Program.

RHR System
Valve Number

HCV-*-758
MOV-*-872

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, the following ECCS components and flow path shall be OPERABLE:

a. One OPERABLE RHR heat exchanger,

b. One OPERABLE RHR pump, and

c. An OPERABLE flow path capable of (1) taking suction from the refueling water storage tank upon being manually realigned and (2) transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no OPERABLE ECCS flow path from the refueling water storage tank, restore at least one ECCS flow path to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With either the residual heat removal heat exchanger or RHR pump inoperable, restore the components to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

SURVEILLANCE REQUIREMENTS

4.5.3 The above ECCS components shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 For single Unit operation, one refueling water storage tank (RWST) shall be OPERABLE or for dual Unit operation two RWSTs shall be OPERABLE with:

a. A minimum indicated borated water volume of 320,000 gallons per RWST,

b. A boron concentration between 2400 ppm and 2600 ppm,

c. A minimum solution temperature of 39°F, and

d. A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With less than the required number of RWST(s) OPERABLE, restore the tank(s) to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The required RWST(s) shall be demonstrated OPERABLE:

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

1) Verifying the indicated borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for consecutive 23 hours, and

2) At least once per 24 hours while the outside temperature exceeds its limit.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that all 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 their closed positions;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

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* Exception may be taken under Administrative Controls for opening of valves and airlocks necessary to perform surveillance, testing requirements and/or corrective maintenance. In addition, Specification 3.6.4 shall be complied with.

** 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 that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the measured overall integrated containment leakage rate exceeding 1.0 L_a within one hour, initiate action to be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. Restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the required test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each 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, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one 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 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 STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment air lock inoperable, except as the 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 STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Following each closing, at the frequency specified in the Containment Leakage Rate Testing Program, by verifying that the seals have not been damaged and have seated properly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

b. By conducting overall air lock leakage tests in accordance with the Containment Leakage Rate Testing Program.

c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and +1 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 125°F and shall not exceed 120°F by more than 336 equivalent hours* during a calendar year.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 125°F or greater than 120°F for more than 336 equivalent hours* during a calendar year, reduce the average air temperature to within the applicable limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined in accordance with the Surveillance Frequency Control Program:

Approximate Location

a. 0° Azimuth 58 feet elevation

b. 120° Azimuth 58 feet elevation

c. 240° Azimuth 58 feet elevation

* Equivalent hours are determined from actual hours using the time-temperature relationships that support the environmental qualification requirements of 10 CFR 50.49.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY MODES 1, 2, 3, and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between 90% and 95% of the predicted force, or with one tendon below 90% of the predicted force, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the average of all measured tendon forces for each type of tendon (dome, vertical, and hoop), including those measured in ACTION a., less than the predicted force, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any abnormal degradation of the structural integrity other than ACTION a. and ACTION b., at a level below the acceptance criteria of Specifications 4.6.1.6.1, 4.6.1.6.2 and 4.6.1.6.3, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUREMENTS

4.6.1 .6.1 Containment Tendons. The containment tendons and the containment exterior surfaces shall be examined in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants," and the modifications presented in 10 CFR 50.55a(b)(2)(viii), "Examination of concrete containments," as modified by approved exemptions. The containment structural integrity shall be demonstrated during the inspection periods specified in IWL-2410 and IWL-2420. The tendons' structural integrity shall be demonstrated by:

SURVEILLANCE REQUIREMENTS (Continued)

a. Determining that tendons, selected in accordance with IWL-2521, have the average of all measured tendon forces for each type of tendon (dome, vertical and hoop) equal to or greater than the minimum required prestress specified at the anchorage for that type of tendon.

b. Assuring that the measured force in each individual tendon is not less than 95% of the predicted force unless the following conditions are satisfied:

1) The measured force in no tendon is below 90% of the predicted force and the measured force in no more than one tendon is between 90% and 95% of the predicted force;
2) The measured force in two tendons located adjacent to the tendon in 1) are not less than 95% of the predicted forces; and
3) The measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

The predicted force for each tendon shall be calculated individually for each inspection prior to the beginning of each inspection, and should consider such factors as:

- Prestressing history;
- Friction losses; and
- Time-dependent losses (creep, shrinkage, relaxation), considering time elapsed from prestressing.

When evaluation of consecutive surveillances of prestressing forces for the same tendon or tendons in a group indicates a trend of prestress loss such that the tendon force(s) would be less than the minimum design prestress requirements before the next inspection interval, an evaluation shall be performed and reported in the Engineering Evaluation Report as prescribed in IWL-3300.

c. Performing tendon detensioning, examinations, and testing on a sample tendon of each type (dome, vertical, and hoop). A single wire or strand shall be removed from each detensioned tendon. Each removed wire or strand shall be examined over its entire length for corrosion and mechanical damage. Tension tests shall be performed on each removed wire or strand: one at each end, one at mid-length, and one in the location of the most corroded area, if any. The following information shall be obtained from each test:

1) Yield strength;
2) Ultimate tensile strength;
3) Elongation.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

The condition of wire or strand is acceptable if:

1) Samples are free of physical damage;
2) Sample ultimate tensile strength and elongation are not less than minimum specified values.

d. Performing tendon retensioning of those tendons that have been detensioned to at least the force predicted for the tendon at the time of the test. However, the retensioning force shall not exceed 70% of the specified minimum ultimate tensile strength of the tendon based on the number of effective wires or strands in the tendon at the time of retensioning. During retensioning of these tendons, if the elongation corresponding to a specific load (adjusted for effective wires or strands) differs by more than 10% from that recorded during the last measurement, an evaluation must be performed to determine whether the difference is related to wire failures or slip of wires in anchorage. A difference of more than 10% must be identified in the ISI Summary Report required by IWA-6000.

e. Performing examination of corrosion protection medium and free water in accordance with IWL-2525, with acceptance standards prescribed in IWL-3221.4. The following conditions, if they occur, shall be reported in the ISI Summary Report required by IWA-6000:

1) The sheathing filler grease contains chemically combined water exceeding 10% by weight or the presence of free water;
2) The absolute difference between the amount removed and the amount replaced exceeds 10% of the tendon net duct volume.
3) Grease leakage is detected during general visual examination of the containment surface.

4.6.1.6.2 End Anchorages and Containment Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the containment concrete surfaces shall be demonstrated by performing examination of tendon anchorage areas and containment concrete surfaces in accordance with IWL-2000, with acceptance standards prescribed in IWL-3000. Acceptability of inaccessible areas shall be evaluated when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the following shall be provided in the ISI Summary Report required by IWA-6000:

1) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
2) An evaluation of each area, and the result of the evaluation; and
3) A description of necessary corrective actions.

4.6.1.6.3. Containment Surfaces Inspection for Containment Leakage Rate Testing Program. In accordance with the Containment Leakage Rate Testing Program, a visual inspection of the accessible interior and exterior surfaces of the containment, including the liner plate, shall be performed. The purpose of this inspection shall be to identify any evidence of structural deterioration which may affect containment structural integrity or leaktightness. The visual inspection shall be general in nature; its intent shall be to detect gross areas of widespread cracking, spalling, gouging, rust, weld degradation, or grease leakage. The visual examination may include the utilization of binoculars or other optical devices. Corrective actions taken, and recording of structural deterioration and corrective actions, shall be in accordance with the Containment Leakage Rate Testing Program. Records of previous inspections shall be reviewed to verify no apparent changes in appearance. The first inspection performed will form the baseline for future surveillances.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. The containment purge supply and exhaust isolation valves shall be sealed closed to the maximum extent practicable but may be open for purge system operation for pressure control, for environmental conditions control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valve to be open.

b. The purge supply and exhaust isolation valves shall not be opened wider than 33 or 30 degrees, respectively (90 degrees is fully open).

APPLICABILITY: MODES 1, 2, 3, AND 4.

ACTION:

a. With a containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in 3.6.1.7.a above, close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status or isolate the penetrations such that the measured leakage rate does not exceed the limits of Specification 4.6.1.7.2 within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7.a in accordance with the Surveillance Frequency Control Program.

4.6.1.7.2 In accordance with the Surveillance Frequency Control Program, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

4.6.1.7.3 In accordance with the Surveillance Frequency Control Program, the mechanical stop on each containment purge supply and exhaust isolation valve shall be verified to be in place and that the valves will open no more than 33 or 30 degrees, respectively.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and manually transferring suction to the containment sump via the RHR System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one Containment Spray System inoperable restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Containment Spray Systems inoperable restore at least one Spray System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray 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* and that power is available to flow path components that require power for operation;

b. By verifying that on recirculation flow, each pump develops the indicated differential pressure, when tested pursuant to Specification 4.0.5:

   Containment Spray Pump $\geq$241.6 psid while aligned in recirculation mode.

c. In accordance with the Surveillance Frequency Control Program by verifying containment spray locations susceptible to gas accumulation are sufficiently filled with water.

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*Not required to be met for system vent flow paths opened under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

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

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal, and

   2) Verifying that each spray pump starts automatically on a containment spray actuation test signal. The manual isolation valves in the spray lines at the containment shall be locked closed for the performance of these tests.

e. In accordance with the Surveillance Frequency Control Program by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

EMERGENCY CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three emergency containment cooling units shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

a. With one of the above required emergency containment cooling units inoperable restore the inoperable cooling unit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above required emergency containment cooling units inoperable, restore at least two cooling units to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each emergency containment cooling unit shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by starting each cooler unit from the control room and verifying that each unit motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes.

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

   1) Verifying that two emergency containment cooling units start automatically on a safety injection (SI) test signal, and

   2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

3/4.6.2.3 RECIRCULATION pH CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The Recirculation pH Control System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Recirculation pH Control System inoperable, restore the buffering agent to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 72 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The Recirculation pH Control System shall be demonstrated OPERABLE:

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

   1. Verifying that the buffering agent baskets are in place and intact;

   2. Collectively contain $\geq$ 7500 pounds (154 cubic feet) of sodium tetraborate decahydrate, or equivalent.

3/4.6.3 DELETED

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE with isolation times less than or equal to required isolation times.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

*With one or more isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic containment isolation valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 The isolation valves 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 performance of a cycling test, and verification of isolation time.

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*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase “A” Isolation test signal, each Phase “A” isolation valve actuates to its isolation position;

b. Verifying that on a Phase “B” Isolation test signal, each Phase “B” isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each purge, exhaust and instrument air bleed valve actuates to its isolation position.

4.6.4.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

3/4.6.5 DELETED

3/4.6.6 DELETED

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With (3) reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, and

a. in MODES 1 and 2, with a positive Moderator Temperature Coefficient, operation may continue provided that, within 4 hours, either the inoperable valve(s) are restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced to the maximum allowable percent of RATED THERMAL POWER listed in Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours, or

b. in MODES 1 and 2, with a negative or zero Moderator Temperature Coefficient; or in Mode 3, with a positive, negative or zero Moderator Temperature Coefficient, operation may continue provided that, within 4 hours, either the inoperable valve(s) are restored to OPERABLE status or reactor power is reduced to less than or equal to the maximum allowable percent of RATED THERMAL POWER listed in Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL WITH
INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER LEVEL (PERCENT OF RATED THERMAL POWER)
1	44
2	27
3	10

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER				LIFT SETTING (±3%)* **	ORIFICE SIZE SQUARE INCHES
	Loop A	Loop B	Loop C		
1.	RV1400	RV1405	RV1410	1085 psig	16
2.	RV1401	RV1406	RV1411	1100 psig	16
3.	RV1402	RV1407	RV1412	1105 psig	16
4.	RV1403	RV1408	RV1413	1105 psig	16

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**All valves tested must have "as left" lift setpoints that are within ±1% of the lift setting value listed in Table 3.7-2.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours.

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify the OPERABILITY of both standby feed-water pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

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

   1) Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS (Continued)

equal to 373 gpm to the entrance of the steam generators. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3;

2) Verifying by control panel indication and visual observation of equipment that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance test above;

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that power is available to those components which require power for flow path operability.

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

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each Auxiliary Feedwater Actuation test signal.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 1 by verifying normal flow to each steam generator.

TABLE 3.7-3

AUXILIARY FEEDWATER SYSTEM OPERABILITY

UNIT	TRAIN	STEAM SUPPLY FLOWPATH(3)	PUMP	DISCHARGE WATER FLOWPATH(3)
3	1	SG 3C via MOV-3-1405 or SG 3B via MOV-3-1404(1)	A or C(2)	SG 3A via CV-3-2816 SG 3B via CV-3-2817 SG 3C via CV-3-2818
3	2	SG 3A via MOV-3-1403 or SG 3B via MOV-3-1404(1)	B or C(2)	SG 3A via CV-3-2831 SG 3B via CV-3-2832 SG 3C via CV-3-2833
4	1	SG 4C via MOV-4-1405 or SG 4B via MOV-4-1404(1)	A or C(2)	SG 4A via CV-4-2816 SG 4B via CV-4-2817 SG 4C via CV-4-2818
4	2	SG 4A via MOV-4-1403 or SG 4B via MOV-4-1404(1)	B or C(2)	SG 4A via CV-4-2831 SG 4B via CV-4-2832 SG 4C via CV-4-2833

NOTES:

(1) Steam admission valves MOV-3-1404 and MOV-4-1404 can be aligned to either train (but not both) to restore OPERABILITY in the event MOV-3-1403 or MOV-3-1405, or MOV-4-1403 or MOV-4-1405 are inoperable.

(2) During single and two unit operation, one pump shall be OPERABLE in each train and the third auxiliary feedwater pump shall be OPERABLE and capable of being powered from, and supplying water to either train, except as noted in ACTION 3 of Technical Specification 3.7.1.2. The third auxiliary feedwater pump (normally the "C" pump) can be aligned to either train to restore OPERABILITY in the event one of the required pumps is inoperable.

(3) If any local manual realignment of valves is required when operating the auxiliary feedwater pumps, a dedicated individual, who is in communication with the control room, shall be stationed at the auxiliary feedwater pump area. Upon instructions from the control room, this operator would realign the valves in the AFW system train to its normal operational alignment.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CST) system shall be OPERABLE with:

Opposite Unit in MODES 4, 5 or 6

A minimum indicated water volume of 210,000 gallons in either or both condensate storage tanks.

Opposite Unit in MODES 1, 2 or 3

A minimum indicated water volume of 420,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Opposite Unit in MODES 4, 5 or 6

With the CST system inoperable, within 4 hours restore the CST system to OPERABLE status or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Opposite Unit in MODES 1, 2 or 3

1) With the CST system inoperable due to indicating less than 420,000 gallons, but greater than or equal to 210,000 gallons indicated, within 4 hours restore the inoperable CST system to OPERABLE status or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With the CST system inoperable with less than 210,000 gallons indicated, within 1 hour restore the CST system to OPERABLE status or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.1.3 The condensate storage tank (CST) system shall be demonstrated OPERABLE by verifying the indicated water volume is within its limit when the tank is the supply source for the auxiliary feedwater pumps in accordance with the Surveillance Frequency Control Program.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.10 microCurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.10 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performing the sampling and analysis described in Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	SFCP
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

PLANT SYSTEMS

STANDBY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two Standby Steam Generator Feedwater Pumps shall be OPERABLE* and at least 145,000 gallons of water (indicated volume), shall be in the Demineralized Water Storage Tank.**

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one Standby Steam Generator Feedwater Pump inoperable, restore the inoperable pump to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6d.

b. With both Standby Steam Generator Feedwater Pumps inoperable, restore at least one pump to OPERABLE status within 24 hours, or:

    1. Notify the NRC within the following 4 hours, and provide cause for the inoperability and plans to restore pump(s) to OPERABLE status and,

    2. Submit a SPECIAL REPORT per 3.7.1.6d.

c. With less than 145,000 gallons of water indicated in the Demineralized Water Storage Tank restore the available volume to at least 145,000 gallons indicated within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the inoperability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 The Demineralized Water Storage tank water volume shall be determined to be within limits in accordance with the Surveillance Frequency Control Program.

4.7.1.6.2 In accordance with the Surveillance Frequency Control Program verify the standby feedwater pumps are OPERABLE by testing in recirculation.

4.7.1.6.3 In accordance with the Surveillance Frequency Control Program, verify operability of the respective standby steam generator feedwater pump by starting each pump and providing feedwater to the steam generators.

*These pumps do not require plant safety related emergency power sources for operability and the flowpath is normally isolated.

**The Demineralized Water Storage Tank is non-safety grade.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.1.6.4 The diesel engine for the diesel-driven Standby Steam Generator Feedwater Pump shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program, by testing with the associated standby steam generator feedwater pump in recirculation.

b. In accordance with the Surveillance Frequency Control Program, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

3/4.7.1.7 FEEDWATER ISOLATION

LIMITING CONDITION FOR OPERATION

3.7.1.7 Six Feedwater Control Valves (FCVs) both main and bypass and six Feedwater Isolation Valves (FIVs) both main and bypass shall be OPERABLE.*

APPLICABILITY: MODES 1, 2 and 3**

ACTION:

a. With one or more FCVs inoperable, restore operability, or close or isolate the inoperable FCVs within 72 hours and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more FIVs inoperable, restore operability, or close or isolate the inoperable FIV(s) within 72 hours and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more bypass valves in different steam generator flow paths inoperable, restore operability, or close or isolate the inoperable bypass valve(s) within 72 hours and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With two valves in the same steam generator flow paths inoperable, restore operability, or isolate the affected flowpath within 8 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours..

SURVEILLANCE REQUIREMENTS

4.7.1.7 Each FCV, FIV and bypass valve shall be demonstrated OPERABLE:

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

1) Verifying that each FCV, FIV and bypass valve actuates to the isolation position on an actual or simulated actuation signal.

b. In accordance with the Inservice Testing Program by:

1) Verifying that each FCV, FIV and bypass valve isolation time is within limits.

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*Separate Condition entry is allowed for each valve.

**The provisions of specification 3.0.4 and 4.0.4 are not applicable.

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

b. 1) In accordance with the Surveillance Frequency Control Program verify that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position.

2) In accordance with the Surveillance Frequency Control Program verify by a performance test the heat exchanger surveillance curves.*

c. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Component Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for CCW operability are OPERABLE.

*Technical specification 4.7.2.b.2 is not applicable for entry into MODE 4 or MODE 3, provided that:

1) Surveillance 4.7.2.b.2 is performed no later than 72 hours after reaching a Reactor Coolant System Tavg of 547°F, and
2) MODE 2 shall not be entered prior to satisfactory performance of this surveillance.

PLANT SYSTEMS

3/4.7.3 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Intake Cooling Water System (ICW) shall be OPERABLE with:

a. Three ICW pumps, and
b. Two ICW headers.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only two ICW pumps with independent power supplies OPERABLE, restore the inoperable ICW pump to OPERABLE status within 14 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one ICW pump OPERABLE or with two ICW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With only one ICW header OPERABLE, restore two headers to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The Intake Cooling Water System (ICW) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Intake Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for system operability are OPERABLE.

PLANT SYSTEMS

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 The ultimate heat sink shall be OPERABLE with an average supply water temperature less than or equal to 104°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION shall be applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.4 The ultimate heat sink shall be determined OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying the average supply water temperature* is less than or equal to 104°F.

b. At least once per hour by verifying the average supply water temperature* is less than or equal to 104°F, when water temperature exceeds 100°F.

*Portable monitors may be used to measure the temperature.

PLANT SYSTEMS

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5 The Control Room Emergency Ventilation System shall be OPERABLE* with:

a. Three air handling units,

b. Two of three condensing units,

c. Two control room recirculation fans,

d. Two recirculation dampers,

e. One filter train,

f. Two isolation dampers in the normal outside air intake duct,

g. Two isolation dampers in the emergency outside air intake duct,

h. Two isolation dampers in the kitchen area exhaust duct, and

i. Two isolation dampers in the toilet area exhaust duct.

j. Control Room Envelope.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

a.1 With one air handling unit inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable air handling unit to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.2 With two condensing units inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore at least one of the inoperable condensing units to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.3 With one recirculation fan inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable fan to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

*The Control Room Envelope (CRE) boundary may be opened intermittently under administrative control.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

a.4 With one recirculation damper inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the recirculation dampers in the open position and place the system in recirculation mode** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.5 With the filter train inoperable, e.g., an inoperable filter, and/or two inoperable recirculation fans, and/or two inoperable recirculation dampers, immediately suspend all movement of irradiated fuel, and, immediately, initiate action to implement mitigating actions, [e.g., use of the compensatory filtration unit is required to be immediately initiated], and, within 24 hours, verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits and, within 7 days, restore the filter train to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.6 With an inoperable damper in the normal outside air intake, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the normal outside air intake isolation dampers in the closed position and place the system in recirculation mode** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.7 With an inoperable damper in the emergency outside air intake, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the emergency outside air intake isolation dampers in the open position** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.8 With an isolation damper inoperable in the kitchen area exhaust duct, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or isolate the flow path** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.9 With an isolation damper inoperable in the toilet area exhaust duct, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or isolate the flow path** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

**If action is taken such that indefinite operation is permitted and the system is placed in recirculation mode, then movement of irradiated fuel may resume.

LIMITING CONDITION FOR OPERATION (continued)

b. With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary, immediately suspend all movement of irradiated fuel in the spent fuel pool, and immediately initiate action to implement mitigating actions, and within 24 hours, verify mitigating actions ensure CRE occupant radiological and chemical hazards will not exceed limits, and CRE occupants are protected from smoke hazards, and restore CRE boundary to OPERABLE status within 90 days, or:

1) With the requirements not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2) If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

c. With the Control Room Emergency Ventilation System inoperable++, immediately suspend all operations involving CORE ALTERATIONS, movement of irradiated fuel in the spent fuel pool, or positive reactivity changes. This ACTION shall apply to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The Control Room Emergency Ventilation System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 120°F;

b. 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 system operates for at least 15 minutes***;

c. In accordance with the Surveillance Frequency Control Program or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system, or (4) after complete or partial replacement of a filter bank by:

++ If action per ACTIONS a.4, a.6, a.7, a.8, or a.9 is taken that permits indefinite operation and the system is placed in recirculation mode, then CORE ALTERATIONS, movement of irradiated fuel in the spent fuel pool, and positive reactivity changes may resume.

***As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and g.

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99.95% DOP and 99% halogenated hydrocarbon removal at a system flow rate of 1000 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, and analyzed per ASTM D3803 - 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 2.5% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement***, and

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration***.

d.1 In accordance with the Surveillance Frequency Control Program by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%***;

d.2 In accordance with the Surveillance Frequency Control Program, test the supply fans (trains A and B) and measure CRE pressure relative to external areas adjacent to the CRE boundary.***

e. In accordance with the Surveillance Frequency Control Program by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation,

f. In accordance with the Surveillance Frequency Control Program by verifying operability of the kitchen and toilet area exhaust dampers, and

g. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.***

***As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and g.

3/4.7.6 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.6 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.6.f on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-2 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment 151 and 146.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual

TABLE 4.7-2

SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extended Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less that the number in Column A, the next inspection interval may be twice the previous interval but not greater that 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of

unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.6e. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Functional Tests

For each unit during refueling shutdown, a representative sample of snubbers shall be tested using the following sample plan:

1) At least 10% of the total number of safety related snubbers for the respective unit identified by site records shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.6e, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested;

2) The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following categories;

   A. Snubbers within 5 feet of heavy equipment (ex. valves, pumps, turbines, motors, etc.)

   B. Snubbers within 10 feet of the discharge from a safety relief valve.

3) Snubbers identified by site records as “Especially Difficult to Remove” or in “High Radiation Zones During Shutdown” shall also be included in the representative sample.*

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*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber OPERABILITY for all design conditions at either the completion of their fabrication or at a subsequent date.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved with the specified range of velocity or acceleration in both tension and compression;

2) Snubber release rate, where required, is within the specified range in tension and compression,

3) The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

f. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated under the provisions of 10 CFR Part 21.

Should the results of the evaluation indicate that the failure was caused by either manufacturer or design deficiency, further action shall be taken, if needed, based on manufacturer or engineering recommendations.

For the snubber(s) found inoperable, an evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this evaluation shall be to determine if the components to which the inoperable snubber(s) are attached were adversely affected by the inoperability of the snubber(s) in order to ensure that the component remains capable of meeting the designed service.

SURVEILLANCE REQUIREMENTS (Continued)

g. Snubber Service Life Monitoring Program

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained.

Concurrent with the first inservice visual inspection and during refueling shutdown thereafter, the installation and maintenance records for each safety related snubber as identified by site records shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

PLANT SYSTEMS

3/4.7.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.7 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 limits, immediately 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

4.7.7.1 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.

4.7.7.2 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 materials:

   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; and

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.

4.7.7.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

4.7.7.4 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

PLANT SYSTEMS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.7.8 The concentration of oxygen in the GAS DECAY TANK SYSTEM (as measured in the inservice gas decay tank) shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the inservice gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the inservice gas decay tank greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the gas decay tanks and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8 The concentrations of hydrogen and oxygen in the inservice gas decay tanks shall be determined to be within the above limits by continuously* monitoring the waste gases in the inservice gas decay tank with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-8 of Specification 3.3.3.6.

*When continuous monitoring capability is inoperable, Table 3.3-8 allows the use of grab samples.

PLANT SYSTEMS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.7.9 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases (DOSE EQUIVALENT Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank and the Reactor Coolant System total activity exceeds the limit of Specification 3.4.8.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 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 startup transformers and their associated circuits, and

b. Three separate and independent diesel generators* including,

1) For Unit 3, two (3A and 3B); for Unit 4, one (3A or 3B) each with:

a) A separate skid-mounted fuel tank and a separate day fuel tank with an OPERABLE solenoid valve to permit gravity flow from the day tank to the skid mounted tank, and with the two tanks together containing a minimum of 2000 gallons of fuel oil.

b) A common Fuel Storage System containing a minimum volume of 38,000 gallons of fuel.**

c) A separate fuel transfer pump.**

d) Lubricating oil storage containing a minimum volume of 120 gallons of lubricating oil,

e) Capability to transfer lubricating oil from storage to the diesel generator unit, and

f) Energized MCC bus (MCC 3A vital section for EDG 3A, MCC 3K for EDG 3B).

2) For Unit 3, one (4A or 4B); for Unit 4, two (4A and 4B) each with:

a) A separate day fuel tank containing a minimum volume of 230 gallons of fuel,

b) A separate Fuel Storage System containing a minimum volume of 34,700 gallons of fuel,

c) A separate fuel transfer pump, and

d) Energized MCC bus (MCC 4J for EDG 4A, MCC 4K for EDG 4B).

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*Whenever one or more of the four EDG's is out-of-service, ensure compliance with the EDG requirements specified in Specifications 3.5.2 and 3.8.2.1.

** A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i.1 for the Unit 3 storage tank while Unit 3 is in Modes 5, 6, or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days, Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of two startup transformers or an associated circuit inoperable, demonstrate the OPERABILITY of the other startup transformer and its associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit is in MODE 1, reduce THERMAL POWER to ≤ 30% RATED THERMAL POWER within 24 hours, or restore the inoperable startup transformer and associated circuits to OPERABLE status within the next 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If THERMAL POWER is reduced to ≤ 30% RATED THERMAL POWER within 24 hours or if the inoperable startup transformer is associated with the opposite unit restore the startup transformer and its associated circuits to OPERABLE status within 30 days of the loss of OPERABILITY, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit was in MODE 2, 3, or 4 restore the startup transformer and its associated circuits to OPERABLE status within 24 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

b. With one of the required diesel generators inoperable, demonstrate the OPERABILITY of the above required startup transformers and their associated circuits 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 preventative maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours, unless the absence of any potential common mode failure for the remaining diesel generators is determined. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the inoperable diesel generator to OPERABLE status within 14 days** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one startup transformer and one of the required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a on the remaining

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** 72 hours if inoperability is associated with Action Statement 3.8.1.1.c.

ACTION (Continued)

startup transformer and associated circuits within one hour and at least once per 8 hours thereafter; and 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 required diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable diesel generator does not exist on the remaining required diesel generators, unless the diesel generators are already operating; restore one of the inoperable sources to OPERABLE status in accordance with Action Statements a and b, as appropriate. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Notify the NRC within 4 hours of declaring both a start-up transformer and diesel generator inoperable. Restore the other A.C. power source (startup transformer or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source.

d. With one diesel generator inoperable, in addition to ACTION b. or c. above, verify that:

1. All required systems, subsystems, trains, components, and devices (except safety injection pumps) that depend on the remaining required OPERABLE diesel generators as a source of emergency power are also OPERABLE.

   If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. At least two Safety Injection pumps are OPERABLE and capable of being powered from their associated OPERABLE diesel generators.

   If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With two of the above required startup transformers or their associated circuits inoperable notify the NRC within 4 hours; restore at least one of the inoperable startup transformers to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours* and in COLD

*If the opposite unit is shutdown first, this time can be extended to 42 hours.

ACTION (Continued)

SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously. With only one startup transformer and associated circuits restored, perform Surveillance Requirement 4.8.1.1.1a on the OPERABLE Startup transformer at least once per 8 hours, and restore the other startup transformer and its associated circuits to OPERABLE status or shutdown in accordance with the provisions of Action Statement 3.8.1.1a with time requirements of that Action Statement based on the time of initial loss of a startup transformer. This ACTION applies to both units simultaneously.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two startup transformers and their associated circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all required diesel generators to OPERABLE status within 14 days from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

g. Following the addition of the new fuel oil* to the Diesel Fuel Oil Storage Tanks, with one or more diesel generators with new fuel oil properties outside the required Diesel Fuel Oil Testing Program limits, restore the stored fuel oil properties to within the required limits within 30 days.

h. With one or more diesel generators with stored fuel oil total particulates outside the required Diesel Fuel Oil Testing Program limits, restore the fuel oil total particulates to within the required limits within 7 days.

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* The properties of API Gravity, specific gravity or an absolute specific gravity; kinematic viscosity; clear and bright appearance; and flash point shall be confirmed to be within the Diesel Fuel Oil Testing Program limits, prior to the addition of the new fuel oil to the Diesel Fuel Oil Storage Tanks.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required startup transformers and their associated circuits shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program while shutting down, by transferring manually unit power supply from the auxiliary transformer to the startup transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE*:

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

1) Verifying the fuel volume in the day and skid-mounted fuel tanks (Unit 4-day tank only),

2) Verifying the fuel volume in the fuel storage tank,

3) Verifying the lubricating oil inventory in storage,

4) Verifying the diesel starts and accelerates to reach a generator voltage and frequency of 3950-4350 volts and 60 $\pm$ 0.6 Hz. In accordance with the Surveillance Frequency Control Program, these conditions shall be reached within 15 seconds after the start signal from normal conditions. For all other starts, warmup procedures, such as idling and gradual acceleration as recommended by the manufacturer may be used. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded** to 2300 - 2500 kW (Unit 3), 2650-2850 kW (Unit 4)***, operates at this loaded condition for at least 60 minutes and for Unit 3 until automatic transfer of fuel from the day tank to the skid mounted tank is demonstrated, and the cooling system is demonstrated OPERABLE.

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

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* All diesel generator starts for the purpose of these surveillances may be proceeded by a prelube period as recommended by the manufacturer.

** May include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

***Momentary transients outside these load bands do not invalidate this test.

SURVEILLANCE REQUIREMENTS (Continued)

b. Demonstrating that a fuel transfer pump starts automatically and transfers fuel from the storage system to the day tank, in accordance with the Surveillance Frequency Control Program;

c. In accordance with the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day and skid-mounted fuel tanks (Unit 4-day tank only);

d. In accordance with the Surveillance Frequency Control Program by checking for and removing accumulated water from the fuel oil storage tanks;

e. By verifying fuel oil properties of new fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.

f. By verifying fuel oil properties of stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.

g. In accordance with the Surveillance Frequency Control Program, during shutdown (applicable to only the two diesel generators associated with the unit):

1) Deleted

2)* Verifying the generator capability to reject a load of greater than or equal to 380 kw while maintaining voltage at 3950-4350 volts and frequency at 60 ± 0.6 Hz;

3)* Verifying the generator capability to reject a load of greater than or equal to 2500 kW (Unit 3), 2874 kW (Unit 4) without tripping. The generator voltage shall return to less than or equal to 4784 volts within 2 seconds following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b. Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently

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* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

connected loads within 15 seconds, energizes the auto–connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the auto-connected shutdown loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at 3950-4350 volts and 60 ± 0.6 Hz during this test.

5) Verifying that on an ESF 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 be 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently connected loads within 15 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at 3950-4350 volts and 60 ± 0.6 Hz during this test; and

   c) Verifying that diesel generator trips that are made operable during the test mode of diesel operation are inoperable.

7)* # 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 2550-2750 kW (Unit 3), 2950-3150 kW (Unit 4)** and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2300-2500 kW (Unit 3), 2650-2850 kW (Unit 4)**. The generator voltage and frequency shall be 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds after the start signal; the steady-state generator voltage and frequency

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* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

** Momentary transients outside these load bands do not invalidate this test.

This test may be performed during POWER OPERATION

shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, verify the diesel starts and accelerates to reach a generator voltage and frequency of 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds after the start signal.**

8) Verifying that the auto-connected loads to each diesel generator do not exceed 2500 kW (Unit 3), 2874 kW (Unit 4);

9) 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.

10) Verifying that the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from the fuel storage tank (Unit 3), fuel storage tanks (Unit 4) to the day tanks of each diesel associated with the unit via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

13) Verifying that the diesel generator lockout relay prevents the diesel generator from starting;

** If verification of the diesel's ability to restart and accelerate to a generator voltage and frequency of 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds following the 24 hour operation test of Specification 4.8.1.1.2.g.7) is not satisfactorily completed, it is not necessary to repeat the 24-hour test. Instead, the diesel generator may be operated between 2300-2500 kW Unit 3, 2650-2850 kW (Unit 4) for 2 hours or until operating temperature has stabilized (whichever is greater). Following the 2 hours/operating temperature stabilization run, the EDG is to be secured and restarted within 5 minutes to confirm its ability to achieve the required voltage and frequency within 15 seconds.

SURVEILLANCE REQUIREMENTS (Continued)

h. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting all required diesel generators simultaneously and verifying that all required diesel generators provide 60 ± 0.6 Hz frequency and 3950-4350 volts in less than or equal to 15 seconds: and

i. In accordance with the Surveillance Frequency Control Program, by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.*

j. At least once per 10 years, for Unit 4 only, by 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 Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda.

4.8.1.1.3 Reports - (Not Used)

* A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i for the Unit 3 storage tank while Unit 3 is in Modes 5, 6, or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days, Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

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 startup transformer and associated circuits, or an alternate circuit, between the offsite transmission network and the 4160 volt bus, A or B, and

b. One diesel generator with:

1) For Unit 3 (3A or 3B)

A skid-mounted fuel tank and a day fuel tank, with an OPERABLE solenoid valve to permit gravity flow from the day tank to the skid mounted tank, with the two tanks together containing a minimum of 2000 gallons of fuel oil

For Unit 4 (4A or 4B)

A day fuel tank containing a minimum volume of 230 gallons of fuel

2) A fuel storage system containing a minimum volume of fuel of 38,000 gallons (Unit 3), 34,700 gallons (Unit 4)**

3) An associated fuel transfer pump**

4) For Unit 3 only, lubricating oil storage containing a minimum volume of 120 gallons of lubricating oil

5) For Unit 3 only capability to transfer lubricating oil from storage to the diesel generator unit and

6) Energized MCC bus (as identified by Specification 3.8.1.1.b.).

APPLICABILITY: MODES 5* and 6*.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 2.2 square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible and increase RCS inventory as soon as possible.

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* CAUTION - If the opposite unit is in MODES 1, 2, 3, or 4 see Specification 3.8.1.1

**A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i.1 for the Unit 3 storage tank while Unit 3 is in Modes 5, 6, or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days, Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1.a and 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5).

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. electrical sources shall be OPERABLE:*#

a. 125-volt D.C. Battery Bank 3A or spare battery bank D-52 and associated full capacity charger(s)

   1) 3A1 powered by motor control center (MCC) 3C with EDG 3A OPERABLE, or

   2) 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or

   3) 3A1 powered by MCC 3C with EDG 3A OPERABLE and 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

b. 125-volt D.C. Battery Bank 3B or spare battery bank D-52 and associated full capacity charger(s)

   1) 3B1 powered by MCC 3B with EDG 3B OPERABLE, or

   2) 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or

   3) 3B1 powered by MCC 3B with EDG 3B OPERABLE and 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

c. 125-volt D.C. Battery Bank 4A or spare battery bank D-52 and associated full capacity charger(s)

   1) 4A1 powered by MCC 4C with EDG 4A OPERABLE, or

   2) 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or

   3) 4A1 powered by MCC 4C with EDG 4A OPERABLE and 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE,

d. 125-volt D.C. Battery Bank 4B or spare battery bank D-52 and associated full capacity charger(s)

   1) 4B1 powered by MCC 4B with EDG 4B OPERABLE, or

   2) 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or

   3) 4B1 powered by MCC 4B with EDG 4B OPERABLE and 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one or more of the required battery chargers OPERABLE but not capable of being powered from its associated OPERABLE diesel generator(s), restore the capability within 72 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

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*All battery chargers required to satisfy the LCO shall be powered from separate MCCs.

#Inoperability of the required EDG's specified in the LCO requirements below does not constitute inoperability of the associated battery chargers or battery banks.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With one of the required battery banks inoperable, or with none of the full-capacity chargers associated with a battery bank OPERABLE, restore all battery banks to OPERABLE status and at least one charger associated with each battery bank to OPERABLE status within two hours* or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and its associated full capacity charger(s) shall be demonstrated OPERABLE:

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

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 129 volts on float charge and the battery charger(s) output voltage is ≥ 129 volts, and

3) If two battery chargers are connected to the battery bank, verify each battery charger is supplying a minimum of 10 amperes, or demonstrate that the battery charger supplying less than 10 amperes will accept and supply the D.C. bus load independent of its associated battery charger.

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 105 volts (108.6 volts for spare battery D-52), or battery overcharge with battery terminal voltage above 143 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) The average electrolyte temperature of every sixth cell is above 60°F, and

3) There is no visible corrosion at either terminals or connectors, or verify battery connection resistance is:

Battery	Connection	Limit (Micro-Ohms)
3B, 4A	inter-cell / termination	≤ 29
	inter-cell (brace locations)	≤ 30
	transition cables	≤ 125
	or	
	total battery connections	≤ 1958
Battery	Connection	Limit (Micro-Ohms)
3A, 4B, D-52	inter-cell / termination	≤ 35
	inter-cell (brace locations)	≤ 40
	transition cables	≤ 125
	or	
	total battery connections	≤ 2463

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

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

*Can be extended to 24 hours if the oppsite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

SURVEILLANCE REQUIREMENTS (Continued)

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) Each 400 amp battery charger (associated with Battery Banks 3A and 4B) will supply at least 400 amperes at ≥ 129 volts for at least 8 hours, and each 300 amp battery charger (associated with Battery Banks 3B and 4A) will supply at least 300 amperes at ≥ 129 volts for at least 8 hours, and

4) Battery Connection resistance is:

Battery 3B, 4A	Connection inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	Limit (Micro-Ohms) ≤ 29 ≤ 30 ≤ 125 ≤ 1958
Battery 3A, 4B, D-52	Connection inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	Limit (Micro-Ohms) ≤ 35 ≤ 40 ≤ 125 ≤ 2463

d. In accordance with the Surveillance Frequency Control Program, during shutdown**, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 12 months, during shutdown**, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% [75% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% [7% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of rated capacity from its average on previous performance tests, or is below 90% [93% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of the manufacturer's rating.

f. In accordance with the Surveillance Frequency Control Program, during shutdown**, by verifying that the battery capacity is at least 80% [87% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.d.

**Except that the spare battery bank D-52, and any other battery out of service when spare battery bank D-52 is in service may be tested with simulated loads during operation.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(6)	≥ 2.07 volts
Specific Gravity(4)	≥ 1.200(5)	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(5)

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, three 125 volt battery banks, each with at least one associated full capacity charger capable of being powered by an OPERABLE diesel generator, shall be OPERABLE.

APPLICABILITY: MODES 5* and 6*.

ACTION:

With one or more of the required 125 volt battery banks or required associated full-capacity chargers inoperable or not capable of being powered from an OPERABLE diesel generator, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery banks and associated full-capacity chargers to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2.2 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125 volt battery banks and associated full-capacity chargers shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

*CAUTION - If the opposite unit is in MODES 1, 2, 3 or 4, see the corresponding Limiting Condition for Operation 3.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses* shall be energized in the specified manner with the tie breakers open between redundant busses within the unit** and between the busses of Units 3 and 4.

a. One train of A.C. Busses consisting of:

1) 4160-Volt Bus A,
2) 480-Volt Load Center Busses A, C and H***, and
3) 480-Volt Motor Control Center Busses A (Unit 4 only),C and D***,

b. One train of A.C. Busses consisting of:

1) 4160-Volt Bus B
2) 480-Volt Load Center Busses B, D and H***, and
3) 480-Volt Motor Control Center Busses B and D***

c. One opposite unit train of AC busses consisting of either:

1) 4160-Volt Bus A, 480-Volt Load Center Busses A, C and H***, and 480-Volt Motor Control Center Busses A (Unit 4 only), C and D***, or

2) 4160-Volt Bus B, 480-Volt Load Center Busses B, D and H***, and 480-Volt Motor Control Center Busses B and D***.

d. 120 Volt AC Vital Panel 3P06 and 3P21 energized from its associated inverter connected to D.C. Bus 3B.****

e. 120 Volt AC Vital Panel 4P06 and 4P21 energized from its associated inverter connected to D.C. Bus 3B.****

f. 120 Volt AC Vital Panel 3P07 and 3P22 energized from its associated inverter connected to D.C. Bus 3A.****

g. 120 Volt AC Vital Panel 4P07 and 4P22 energized from its associated inverter connected to D.C. Bus 3A.****

h. 120 Volt AC Vital Panel 3P08 and 3P23 energized from its associated inverter connected to D.C. Bus 4B.****

i. 120 Volt AC Vital Panel 4P08 and 4P23 energized from its associated inverter connected to D.C. Bus 4B.****

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*For Motor Control Center busses, vital sections only.

**With the opposite unit in MODE 5 or 6, its 480-Volt Load Center can be cross-tied under conditions specified in Specification 3.8.3.2.a.

***Electrical bus can be energized from either train of its unit and swing function to opposite train must be OPERABLE for the Unit(s) in MODES 1,2,3, and 4.

****A back-up inverter may be used to replace the normal inverter provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

j. 120 Volt AC Vital Panel 3P09 and 3P24 energized from its associated inverter connected to D.C. Bus 4A.****

k. 120 Volt AC Vital Panel 4P09 and 4P24 energized from its associated inverter connected to D.C. Bus 4A.****

l. 125 Volt D.C. Bus 3D01 energized from an associated battery charger and from Battery Bank 3A or spare battery bank D-52,

m. 125 Volt D.C. Bus 3D23 energized from an associated battery charger and from Battery Bank 3B or spare battery bank D-52,

n. 125 Volt D.C. Bus 4D01 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52, and

o. 125 Volt D.C. Bus 4D23 energized from an associated battery charger and from Battery Bank 4A or spare battery bank D-52

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required trains (3.8.3.1a., b., and c) of A.C. emergency busses not fully energized (except for the required LC's and MCC's associated with the opposite unit), reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any of the required LC's and/or MCC's associated with the opposite unit inoperable, restore the inoperable LC or MCC to OPERABLE status in accordance with Table 3.8-1 or Table 3.8-2 as applicable or place the unit in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) Reenergize the A.C. vital panel within 2 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from an inverter connected to its associated D.C. bus

****A back-up inverter may be used to replace the normal inverter, provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

within 24 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

d. With one D.C. bus not energized from its associated battery bank or associated charger, reenergize the D.C. bus from its associated battery bank within 2 hours* or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized and aligned in the required manner by verifying correct breaker alignment and indicated voltage on the buses in accordance with the Surveillance Frequency Control Program.

* Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

TABLE 3.8-1

APPLICABLE TO UNIT 3 BASED ON UNIT 4 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 4 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 3 - MODES 1, 2, 3 or 4		
	With AC Trains 3A, 3B, 4A, & 4B OPERABLE	With AC Trains 3A, 3B, & 4A OPERABLE	With AC Trains 3A, 3B, & 4B OPERABLE
LC 4A	N/A	72	N/A
MCC 4A	N/A	N/A	N/A
LC 4C and/or MCC 4C	2*	2*	N/A
LC 4H and/or MCC 4D	2**	2**	2**
LC 4B and/or MCC 4B	2*	N/A	2*
LC 4D	N/A	N/A	72

*If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).
**If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours.

TABLE 3.8-2

APPLICABLE TO UNIT 4 BASED ON UNIT 3 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 3 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 4 - MODES 1, 2, 3 or 4		
	With AC Trains 4A, 4B, 3A, & 3B OPERABLE	With AC Trains 4A, 4B, & 3A OPERABLE	With AC Trains 4A, 4B, & 3B OPERABLE
LC 3A	N/A	72	N/A
LC 3C and/or MCC 3C	2*	2*	N/A
LC 3H and/or MCC 3D	2**	2**	2**
LC 3B and/or MCC 3B	2*	N/A	2*
LC 3D	N/A	N/A	72

*If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).
**If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours.

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. emergency busses associated with the unit (3.8.3.1a. or b.) consisting of one 4160-volt and three 480-volt A.C. emergency busses load centers* and three (four for Unit 4 Train A) vital sections of motor control center busses,

b. Two 120-volt A.C. vital busses for the unit energized from their associated inverters** connected to their respective D.C. busses, and

c. Three 125-volt D.C. busses energized from their associated battery banks.

APPLICABILITY MODES 5*** and 6***.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours, depressurize and vent the RCS through at least a 2.2 square inch vent.

SURVEI LLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner by verifying correct breaker alignment and indicated voltage on the busses in accordance with the Surveillance Frequency Control Program.

* With the opposite unit in MODE 1, 2, 3, or 4, the 480-volt load centers can only be cross-tied upon issuance of an engineering evaluation to prevent exceeding required electrical components maximum design ratings and to ensure availability of the minimum required equipment.

** A backup inverter may be used to replace the normal inverter provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

*** CAUTION - If the opposite unit is in MODES 1, 2, 3, or 4, see the corresponding Limiting Condition for Operation 3.8.3.1.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2300 ppm.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2300 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

4.9.1.3 Valves isolating unborated water sources** shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power in accordance with the Surveillance Frequency Control Program.

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* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** The primary water supply to the boric acid blender may be opened under administrative controls for makeup.

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, one primary Source Range Neutron Flux Monitor with continuous visual indication in the control room and audible indication in the containment and control room, and one of the remaining three Source Range Neutron Flux Monitors (one primary or one of the two backup monitors) with continuous visual indication in the control room shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each required Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts.

b. A minimum of one door in each airlock is closed, or, both doors of the containment personnel airlock may be open if:
   1) at least one personnel airlock door is capable of being closed.
   2) The plant is in MODE 6 with at least 23 feet of water above the reactor vessel flange, and
   3) a designated individual is available outside the personnel airlock to close the door.

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:*

   1) Closed by an isolation valve, blind flange, or manual valve, or

   2) Be capable of being closed by an OPERABLE automatic containment ventilation isolation valve.

APPLICABILITY: During movement of recently irradiated fuel within the containment.

ACTION:
With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of recently irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment ventilation isolation valve within 100 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment ventilation isolation valves per the applicable portions of Specification 4.6.4.2.

*Exception may be taken under Administrative Controls for opening of certain valves and airlocks necessary to perform surveillance or testing requirements.

3/4.9.5 DELETED

3/4.9.6 DELETED

3/4.9.7 DELETED

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm in accordance with the Surveillance Frequency Control Program.

4.9.8.1.2 The RHR flow indicator shall be subjected to a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

4.9.8.1.3 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

*The required RHR loop may be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation*.

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm in accordance with the Surveillance Frequency Control Program.

4.9.8.2.2 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

* One required RHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other RHR loop is OPERABLE and in operation.

REFUELING OPERATIONS

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the containment ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REOUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS by verifying that Containment Ventilation Isolation occurs on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

REFUELING OPERATIONS

3/4.9.10 REFUELING CAVITY WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.10 Refueling cavity water level shall be maintained $\geq$ 23 feet above the top of the reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTION:

With the refueling cavity water level not within limit, suspend movement of irradiated fuel assemblies within containment immediately.

SURVEILLANCE REQUIREMENTS

4.9.10 Verify refueling cavity water level is $\geq$ 23 feet above the top of the reactor vessel flange within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program thereafter during movement of irradiated fuel assemblies within containment.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10" the spent fuel storage pool.*

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

*The requirements of this specification may be suspended for more than 4 hours to perform maintenance provided a 10 CFR 50.59 evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

3/4.9.12 DELETED

3/4.9.13 RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.9.13 The Containment Radiation monitors which initiate containment and control room ventilation isolation shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a) With one or both radiation monitors inoperable, operation may continue provided the containment ventilation isolation valves are maintained closed.

b) With one or both radiation monitors inoperable, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

SURVEILLANCE REQUIREMENTS

4.9.13 Each Containment Radiation monitor shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-3.

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. The minimum boron concentration in the Spent Fuel Pit shall be 2300 ppm.

b. The combination of initial enrichment, burnup, and cooling time of each fuel assembly stored in the Spent Fuel Pit shall be in accordance with Specification 5.5.1.

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With boron concentration in the Spent Fuel Pit less than 2300 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 2300 ppm or greater.

b. With condition b not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEI LLANCE REQUIREMENTS

4.9.14.1 The boron concentration of the Spent Fuel Pit shall be verified to be 2300 ppm or greater in accordance with the Surveillance Frequency Control Program.

4.9.14.2 A representative sample of inservice Metamic inserts shall be visually inspected in accordance with the Metamic Surveillance Program described in UFSAR Section 16.2. The surveillance program ensures that the performance requirements of Metamic are met over the surveillance interval.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm). boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Specifications 4.2.2.1 and 4.2.2.5, and

b. Specification 4.2.3.3.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

3/4.10.4 (This specification number is not used)

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and in accordance with the Surveillance Frequency Control Program thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Analog Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

5.0 DESIGN FEATURES

5.1 SITE

5.1.1 The site is approximately 25 miles south of Miami, 8 miles east of Florida City, and 9 miles southeast of Homestead, Florida

5.2 DELETED

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4, ZIRLO®, or Optimized ZIRLO™ except that replacement of fuel rods by filler rods consisting of stainless steel, or by vacant rod positions, may be made in fuel assemblies if justified by cycle-specific reload analysis using NRC-approved methodology. The reactor core contains approximately 71 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. Each fuel rod shall have a nominal active fuel length of 144 inches. Should more than 30 individual rods in the core, or 10 fuel rods in any fuel assembly, be replaced per refueling, a Special Report discussing the rod replacements shall be submitted to the Commission within 30 days after cycle startup.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be silver, indium, and cadmium. All control rods shall be clad with stainless steel tubing.

5.4 DELETED

5.5 FUEL STORAGE

5.5.1 CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} less than 1.0 when flooded with unborated water, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

b. A k_{eff} less than or equal to 0.95 when flooded with water borated to 500 ppm, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

c. A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for the two region spent fuel pool storage racks. A nominal 10.1 inch center-to-center distance in the east-west direction and a nominal 10.7 inch center-to-center distance in the north-south direction for the cask area storage rack.

d. A maximum enrichment loading for fuel assemblies of 5.0 weight percent of U-235.

e. No restriction on storage of fresh or irradiated fuel assemblies in the cask area storage rack.

f. Fresh or irradiated fuel assemblies not stored in the cask area storage rack shall be stored in accordance with Specification 5.5.1.3.

g. The Metamic neutron absorber inserts shall have a minimum certified ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

5.5.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than a nominal 4.5 weight percent of U-235 if the assembly contains no burnable absorber rods and no more than 5.0 weight percent of U-235 if the assembly contains at least 16 IFBA rods.

5.5.1.3 Credit for burnup and cooling time is taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. Fresh or irradiated fuel assemblies in the Region I or Region II racks shall be stored in compliance with the following:

a. Any 2x2 array of Region I storage cells containing fuel shall comply with the storage patterns in Figure 5.5-1 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array.

b. Any 2x2 array of Region II storage cells containing fuel shall:

i. Comply with the storage patterns in Figure 5.5-2 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array,

ii. Have the same directional orientation for Metamic inserts in a contiguous group of 2x2 arrays where Metamic inserts are required, and

iii. Comply with the requirements of 5.5.1.3.c for cells adjacent to Region I racks.

c. Any 2x2 array of Region II storage cells that interface with Region I storage cells shall comply with the rules of Figure 5.5-3.

d. Any fuel assembly may be replaced with a fuel rod storage basket or non-fuel hardware.

e. Storage of Metamic inserts or RCCAs is acceptable in locations designated as empty (water-filled) cells.

DRAINAGE

5.5.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1535 fuel assemblies.

Table 5.5-1

Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-6 for use of Table 5.5-1

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	5.66439153	-14.7363682	-7.74060457	-7.63345029	24.4656526	8.5452608	26.2860949
A2	-7.22610116	11.0284547	5.13978237	10.7798957	-20.3141124	-4.47257395	-18.0738662
A3	2.98646188	-1.80672781	-0.360186309	-2.81231555	6.53101471	2.09078914	5.8330891
A4	-0.287945644	0.119516492	0.0021681285	0.29284474	-0.581826027	-0.188280562	-0.517434342
A5	-0.558098618	0.0620559676	-0.0304713673	0.0795058096	-0.16567492	0.157548739	-0.0614152031
A6	0.476169245	0.0236575787	0.098844889	-0.0676341983	0.243843226	-0.0593584027	0.134626308
A7	-0.117591963	-0.0088144551	-0.0277584786	0.0335130877	-0.0712130368	0.0154678626	-0.0383060399
A8	0.0095165354	0.0008957348	0.0024057185	-0.0040803875	0.0063998706	-0.0014068318	0.0033419846
A9	-47.1782783	-20.2890089	-21.424984	14.6716317	-41.1150	-0.881964768	-12.1780
A10	33.4270029	14.7485847	16.255208	-10.0312224	43.9149156	9.69128392	23.6179517
A11	-6.11257501	-1.22889103	-1.77941882	5.62580894	-9.6599923	-0.18740168	-4.10815592
A12	0.490064351	0.0807808548	0.127321203	-0.539361868	0.836931842	0.0123398618	0.363908736

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ - (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal central zone U-235 enrichment. Axial blanket material is not considered when determining enrichment. Any enrichment between 2.0 and 5.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 0 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. DELETED

5. DELETED

6. This Table applies for any blanketed fuel assembly.

Table 5.5-2

Non-Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-4 for use of Table 5.5-2

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	2.04088171	-27.6637884	-11.2686777	20.7284208	29.8862876	-83.5409405	35.5058622
A2	-4.83684164	26.1997193	2.0659501	11.9673275	-37.0771132	94.7973724	-30.1986997
A3	2.59801889	-7.2982252	2.66204924	-14.4072388	16.3986049	-31.9583373	11.0102438
A4	-0.300597247	0.723731768	-0.513334362	2.83623963	-2.1571669	3.55898487	-1.27269125
A5	-0.610041808	0.401332891	-0.0987986108	-1.49118695	1.02330848	0.299948492	1.34723758
A6	0.640497159	-0.418616707	-0.0724198633	1.75361041	-1.21889631	-0.312341996	-1.19871392
A7	-0.219000712	0.144304039	0.106248806	-0.659046438	0.467440882	0.107463895	0.352920811
A8	0.0252870451	-0.0154239536	-0.0197359109	0.080884618	-0.0560129443	-0.0108814287	-0.0325155213
A9	-4.48207836	-5.54507376	-1.34620551	-245.825283	12.1549	39.4975573	-5.2576
A10	-2.12118634	-5.76555416	-10.1728821	243.59979	-22.7755385	-50.5818253	10.1733379
A11	2.91619317	6.29118025	8.71968815	-75.7805818	14.3755458	23.3093829	0.369083041
A12	-0.196645176	-0.732079719	-1.14461356	8.10936356	-1.80803352	-2.69466612	0.0443577624

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ -\ (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal U-235 enrichment. Any enrichment between 1.8 and 4.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 15 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. This Table applies only for pre-EPU non-blanketed fuel assemblies. If a non-blanketed assembly is depleted at EPU conditions, none of the burnup accrued at EPU conditions can be credited (i.e., only burnup accrued at pre-EPU conditions may be used as burnup credit).

Table 5.5-3

Fuel Categories Ranked by Reactivity

See notes 1-5 for use of Table 5.5-3

Region I	I-1	High Reactivity
	I-2	
	I-3	
	I-4	Low Reactivity
Region II	II-1	High Reactivity
	II-2	
	II-3	
	II-4	
	II-5	Low Reactivity

Notes:

1. Fuel Category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category I-2 is less reactive than Category I-1, etc. The more reactive fuel categories require compensatory measures to be placed in Regions I and II of the SFP, e.g., use of water filled cells, Metamic inserts, or full length RCCAs.
2. Any higher numbered fuel category can be used in place of a lower numbered fuel category from the same Region.
3. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.
4. Category I-2 is fresh unburned fuel that obeys the IFBA requirements of Table 5.5-4.
5. All Categories except I-1 and I-2 are determined from Tables 5.5-1 and 5.5-2.

Table 5.5-4

IFBA Requirements for Fuel Category I-2

Nominal Enrichment (wt% U-235)	Minimum Required Number of IFBA Pins
Enr. ≤ 4.3	0
4.3 < Enr. ≤ 4.4	32
4.4 < Enr. ≤ 4.7	64
4.7 < Enr. ≤ 5.0	80

FIGURE 5.5-1

ALLOWABLE REGION I STORAGE ARRAYS
See notes 1-8 for use of Figure 5.5-1

DEFINITION	**ILLUSTRATION**

Array I-A
Checkerboard pattern of Category I-1 assemblies and empty (water-filled) cells.

I-1	X
X	I-1

Array I-B
Category I-4 assembly in every cell.

I-4	I-4
I-4	I-4

Array I-C
Combination of Category I-2 and I-4 assemblies. Each Category I-2 assembly shall contain a full length RCCA.

I-2	I-4
I-4	I-4

I-2	I-2
I-4	I-4

I-2	I-2
I-2	I-4

I-2	I-2
I-2	I-2

Array I-D
Category I-3 assembly in every cell. One of every four assemblies contains a full length RCCA.

I-3	I-3
I-3	I-3

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.
3. Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 5.5-4.
4. Categories I-3 and I-4 are determined from Tables 5.5-1 and 5.5-2.
5. Shaded cells indicate that the fuel assembly contains a full length RCCA.
6. **X** indicates an empty (water-filled) cell.
7. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-2

ALLOWABLE REGION II STORAGE ARRAYS

See notes 1-6 for use of Figure 5.5-2

DEFINITION	**ILLUSTRATION**

Array II-A

Category II-1 assembly in three of every four cells; one of every four cells is empty (water-filled); the cell diagonal from the empty cell contains a Metamic insert or full length RCCA.

II-1	II-1
X	II-1

X	II-1
II-1	II-1

Array II-B

Checkerboard pattern of Category II-3 and II-5 assemblies with two of every four cells containing a Metamic insert or full length RCCA.

II-3	II-5
II-5	II-3

II-3	II-5
II-5	II-3

Array II-C

Category II-4 assembly in every cell with two of every four cells containing a Metamic insert or full length RCCA.

II-4	II-4
II-4	II-4

II-4	II-4
II-4	II-4

Array II-D

Category II-2 assembly in every cell with three of every four cells containing a Metamic insert or full length RCCA.

II-2	II-2
II-2	II-2

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
6. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-3

INTERFACE RESTRICTIONS BETWEEN REGION I AND REGION II ARRAYS

See notes 1-8 for use of Figure 5.5-3

DEFINITION	**ILLUSTRATION**

Array II-A, as defined in Figure 5.5-2, when placed on the interface with Region I shall have the empty cell in the row adjacent to the Region I rack.

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-1	X	II-1	X
II-1	II-1	II-1	II-1

Array II-A

Arrays II-B, II-C and II-D, as defined in Figure 5.5-2, when placed on the interface with Region I shall have an insert in every cell in the row adjacent to the Region I rack.

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Array II-B

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-C

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2

Array II-D

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.

2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.

3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.

4. **X** indicates an empty (water-filled) cell.

5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only. Region I Array I-B is depicted as the example; however, any Region I array is allowed provided that

    a. For Array I-D, the RCCA shall be in the row adjacent to the Region II rack, and

    b. Array I-A shall not interface with Array II-D.

6. If no fuel is stored adjacent to Region II in Region I, then the interface restrictions are not applicable.

7. Figure 5.5-3 is applicable only to the Region I - Region II interface. There are no restrictions for the interfaces with the cask area rack.

8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant General Manager shall be responsible for overall unit operation of both units and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Nuclear Plant Supervisor (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 Site Vice President shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to, and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety, and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

c. The Plant General Manager shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the Health Physics Supervisor shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

PLANT STAFF

6.2.2 The plant organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. DELETED

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

e. 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; and

f. DELETED

h. The Operations Supervisor shall hold a Senior Reactor Operator License.

i. The Operations Manager shall either:

1. hold or have held a Senior Reactor Operator License on the Turkey Point Plant; or,

2. have held a Senior Reactor Operator License on a similar plant (i.e., another pressurized water reactor); or

3. have completed the Turkey Point Plant Senior Management Operations Training Course. (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level.)

* The Health Physics Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
NPS	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

NPS - Nuclear Plant Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 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 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Nuclear Plant Supervisor from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear Plant Supervisor from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position may be filled by the Nuclear Plant Supervisor or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION

6.2.3.1 An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit and the opposite unit. This individual shall meet the qualifications specified by the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for

6.3.1.1 The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.1.2 The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i.

6.3.1.3 The licensed operators, who shall comply only with the requirements of 10 CFR 55.

6.3.1.4 The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant

c. Training: Complete the Multi-Discipline Supervisor training program

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

6.3.3 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1.3, perform the functions described in 10 CFR 50.54(m)

6.4 DELETED

6.5 DELETED

6.6 DELETED

6.7 DELETED

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 required by the Quality Assurance Topical Report.

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Process Control Program implementation;

d. Offsite Dose Calculation Manual implementation;

e. Quality Control Program for effluent monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974;

f. DELETED

g. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975; and

h. Diesel Fuel Oil Testing Program implementation.

6.8.2 DELETED

6.8.3 DELETED

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 Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at every 18 months.

The provisions of Specification 4.0.2 are applicable.

b. DELETED

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

PROCEDURES AND PROGRAMS (Continued)

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. DELETED

e. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API Gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for Grade No. 2-D fuel oil per ASTM D975, and

3. a clear and bright appearance with proper color;

b. Other properties for Grade No. 2-D fuel oil per ASTM D975 are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/liter when tested every 31 days in accordance with either ASTM D-2276 or ASTM D-5452.

PROCEDURES AND PROGRAMS (Continued)

f. Radioactive Effluent Controls Program

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

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 ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

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 10 CFR 50, Appendix I;

5. Determination of cumulative dose 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 appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

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 in accordance with the following:

   a. For noble gases: a dose rate less or equal to 500 mrems/yr to the whole body and a dose rate less than or equal to 3000 mrems/yr to the skin, and

   b. For iodine 131, iodine 133 tritium and all radionuclides in particulate form with half live greater than 8 days: a dose rate less than or equal to 1500 mrems/yr to any organ.

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

PROCEDURES AND PROGRAMS (Continued)

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 10 CFR 50, Appendix I;

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

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. DELETED

h. 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, and as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following deviations or exemptions:

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) Type A testing frequency in accordance with NEI 94-01, Revision 0, Section 9.2.3, except:

   a) For Unit 3, the first Type A test performed after the November 1992 Type A test shall be performed no later than November 2007.

   b) For Unit 4, the first Type A test performed after October 1991 shall be performed no later than October 2006.

3) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is defined here as the containment design pressure of 55 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding 1.0 L_a, the As-left leakage rate acceptance criterion is $\leq$0.75 L_a, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than 0.60 L_a, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

- The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than 0.6 L_a, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criteria is $\leq$ 0.05 L_a, when pressurized to P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

i. 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:

1. Change in the TS incorporated in the license or

2. A change to the updated FSAR 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 FSAR.

d. Proposed changes that meet the criteria of Specification 6.8.4 i.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).

j. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met..

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located greater than 18.11 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.11 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The portion of the tube below 18.11 inches from the top of the tubesheet is excluded from inspection. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period beings at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;

   b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

   c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

3. If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-secondary leakage.

k. 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 Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

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

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

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

d. Measurement, at designated locations, of the CRE pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the CREVS, operating at the flow rate required by Surveillance Requirement 4.7.5.d, at a Frequency of 18 months. Additionally, the supply fans (trains A and B) will be tested on a staggered test basis (defined in Technical Specification definition 1.29 every 36 months). The results shall be trended and the CRE boundary assessed every 18 months.

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.

PROCEDURES AND PROGRAMS (Continued)

l. 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 Operations 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 1.

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

6.8.5 DELETED

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 U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1 .1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 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.

Reports required on an annual basis shall include:

The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded 0.25 microcuries per gram DOSE EQUIVALENT I-131.

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* 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.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the Offsite Dose Calculation Manual (ODCM), and in (2) 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT**

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

6.9.1.5 DELETED

*A single submittal may be made for a multiple unit station.

**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; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The W(Z) function(s) for Base-Load Operation corresponding to a ± 2% band about the target flux difference and/or a ± 3% band about the target flux difference, the Load-Follow function $F_z(Z)$ and the augmented surveillance turnon power fraction P_T shall be provided to the U.S. Nuclear Regulatory Commission, whenever P_T is <1.0. In the event, the option of Baseload Operation (as defined in Section 4.2.2.3) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.

The analytical methods used to generate the Peaking Factor limits shall be those previously reviewed and approved by the NRC. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1. Reactor Core Safety Limits for Specification 2.1.1.
2. Overtemperature ΔT, Note 1 of Table 2.2-1 for Specification 2.2.1, determination of values K_1, K_2, K_3, T´, P´, τ_1, τ_2, τ_3, τ_4, τ_5, τ_6, and the breakpoint and slope values for the f_1 (ΔI).
3. Overpower ΔT, Note 3 of Table 2.2-1 for Specification 2.2.1, determination of values for K_4, K_5, K_6, T”, τ_7 and f_2 (ΔI).
4. Shutdown Margin - T_{avg} >200°F for Specification 3/4.1.1.1.
5. Shutdown Margin - T_{avg} $\leq$200°F for Specification 3/4.1.1.2.
6. Moderator Temperature Coefficient for Specification 3/4.1.1.3.
7. Axial Flux Difference for Specification 3.2.1.
8. Control Rod Insertion Limits for Specification 3.1.3.6.
9. Heat Flux Hot Channel Factor - $F_Q(Z)$ for Specification 3/4.2.2.
10. All Rods Out position for Specification 3.1.3.2.
11. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.
12. DNB Parameters for Specification 3.2.5, determination of values for Reactor Coolant System T_{avg} and Pressurizer Pressure.

The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION,” June 1983.

2. WCAP-8385, “POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT,” September 1974.

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine F_Q (Z), $F_\Delta H$ and the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

3. WCAP-10054-P-A, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

4. WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", January 2005.

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,' " June 28, 1996.**

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."**

7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

8. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006.

The analytical methods used to determine Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, "Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions, " September 1986

2. WCAP-9272-P-A, " Westinghouse Reload Safety Evaluation Methodology," July 1985

The analytical methods used to determine Safety Limits, Shutdown Margin - $T_{avg} > 200°F$, Shutdown Margin - $T_{avg} \leq 200°F$, Moderator Temperature Coefficient, DNB Parameters, Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The analytical methods used to support the suspension of the measurement of the Moderator Temperature Coefficient in accordance with Surveillance Requirement 4.1.1.3.b shall be those previously reviewed and approved by the NRC in:

1. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997.

2. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

3. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

4. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007.

**As evaluated in NRC Safety Evaluation dated December 20, 1997.

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants."

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

The AFD, $F_Q(Z)$, $F_{\Delta}H$, K(Z), Safety Limits, Overtemperature ΔT, Overpower ΔT, Shutdown Margin - $T_{avg} > 200°F$, Shutdown Margin -$T_{avg} \leq 200°F$, Moderator Temperature Coefficient, DNB Parameters, and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector, unless otherwise approved by the Commission.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 18.11 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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 as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

6.10 DELETED

6.11 DELETED

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the “control device” or “alarm signal” required by paragraph 20.1601(a), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mrem/hr but equal to or less than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physics Shift Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) and less than 500 rads/hr at 1 meter from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/hr and less than 500 rads/hr that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 DELETED

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall contain the following:

a. The methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and

b. The radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Annual Radioactive Effluent Release Reports required by Specification 6.9.1.3 and Specification 6.9.1.4.

6.14.2 Licensee initiated changes to the ODCM:

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

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

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

b. Shall become effective after approval of the Plant General Manager; and

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

APPENDIX B

TO RENEWED FACILITY OPERATING LICENSES NOS. DPR-31 and DPR-41

TURKEY POINT NUCLEAR GENERATING UNITS NOS. 3 AND 4

ENVIRONMENTAL PROTECTION PLAN (EPP)

(NON-RADIOLOGICAL)

1.0 Objectives of the Environmental Protection Plan

The objective of the Environmental Protection Plan (EPP) is to provide for protection of the environment at the Turkey Point Plant and immediate adjacent areas.

The principal objectives of the EPP are to:

1. Aid in determining that the plant is operated in an environmentally acceptable manner, as established by NRC environmental impact assessments.

2. Provide for review of NRC requirements to maintain consistency with other Federal and State requirements for environmental protection.

3. Keep NRC informed of any significant environmental impacts due to facility operation and of actions taken in response to any impacts.

Environmental concerns which relate to any water quality and biological monitoring matters will be regulated by way of EPA through the licensee's National Pollutant Discharge Elimination System (NPDES) permit.

2.0 Environmental Protection Issues

With assumption of aquatic monitoring programs by U.S. Environmental Protection Agency (EPA) through the NPDES program, NRC will rely on EPA for resolution of issues involving the monitoring of water quality and biological monitoring programs.

3.0 Consistency Requirements

3.1 Facility Design and Operation

The licensee may make changes in facility design or operation or perform tests or experiments affecting the environment provided such changes, tests or experiments do not involve an unreviewed environmental question. Changes in plant design or operation or performance of tests or experiments which do not significantly affect the environment are not subject to this requirement.

Before engaging in construction or operational activities which may significantly affect the environment, the licensee shall perform an environmental evaluation of such activity.* When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activities and obtain prior approval from the NRC.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns a matter which may result in 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 a written evaluation which provides bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question.

*Activities are excluded from this requirement if all measurable nonradiological effects are confined to the on-site areas previously disturbed during site preparation, plant construction and previous plant operation.

Activities governed by Section 3.3 of this EPP are not subject to the requirements of Section 3.1.

3.2 Reporting Related to the NPDES Permit and State 401 Certification

1. Violations of the NPDES Permit or the State 401 Certification Conditions shall be reported to the NRC by submittal of copies of the reports required by the NPDES Permit or State 401 Certification.

2. Changes and additions to the NPDES Permit or the State 401 Certification shall be reported to the NRC within 30 days following the date the change 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. The NRC shall be notified of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in facility design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, or local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Administrative Procedures

4.1 Review and Audit

The licensee shall provide for review and audit of compliance with the Environmental Protection Plan. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

4.2 Records Retention

Records and logs relative to the environmental aspects of facility operation which have significant environmental impact shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to plant structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the facility. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

4.3 Changes in Environmental Protection Plan

Request for change in the Environmental Protection Plan 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 Environmental Protection Plan.

5.0 Facility Reporting Requirements

5.1 A written report shall be submitted to the NRC within 30 days of occurrence of any event having significant environmental impact. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and facility 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.

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 within 10 working days of the time it is submitted to the other agency.

5.2 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 facility operation shall be recorded and promptly reported to the NRC within 5 working days followed by a written report within 30 days. No routine monitoring programs are required to implement this condition.

2.1 GEOGRAPHY AND DEMOGRAPHY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD DEP 1.1-1 Subsection 2.1.1 of the DCD is renumbered as Subsection 2.1.4 and moved to the end of Section 2.1. This is being done to accommodate the incorporation of RG 1.206 numbering conventions for Section 2.1.

2.1.1 SITE LOCATION AND DESCRIPTION

PTN COL 2.1-1

2.1.1.1 Site Location

The Units 6 & 7 plant area is part of the larger Turkey Point plant property located approximately 25 miles south of Miami in unincorporated Miami-Dade County, Florida. The approximate 9400-acre Turkey Point plant property includes two gas/oil-fired steam electric generating units, Units 1 & 2, one natural gas combined cycle plant, Unit 5, and four nuclear powered steam electric generating units, Units 3 & 4 and Units 6 & 7 (Reference 201). Figure 2.1-201 shows the Turkey Point site and the surrounding area within 50 miles. Figure 2.1-202 illustrates the general location of the Turkey Point plant property and localities surrounding the site within 10 miles.

The prominent natural features of the region surrounding the Turkey Point plant property, as shown in Figures 2.1-201 and 2.1-202 include Biscayne Bay and the Everglades National Park. Biscayne Bay is surrounded by the barrier islands, which eventually become part of the Florida Keys, and is connected to the Atlantic Ocean by many natural and man-made channels. Tributaries to Biscayne Bay in the region surrounding the Turkey Point plant property include several man-made canals (Reference 206). The Turkey Point plant property is also located near the eastern edge of the Everglades, a vast area of marshland that ranges from Lake Okeechobee to the southern tip of the Florida peninsula (Reference 207). The Turkey Point plant property lies on the Floridian plateau, a partly submerged peninsula of the continental shelf. The peninsula is underlain by approximately 4000 to 15,000 feet of sedimentary rocks consisting of limestone and associated formations (Reference 201).