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

July 16, 2015

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 – ISSUANCE OF AMENDMENTS REGARDING RISK-INFORMED JUSTIFICATIONS FOR THE RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE-CONTROLLED PROGRAM (TAC NOS. MF3931 AND MF3932)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 263 to Renewed Facility Operating License (RFOL) No. DPR-31 and Amendment No. 258 to RFOL No. DPR-41 for the Turkey Point Nuclear Generating Unit Nos. 3 and 4, respectively. The amendments change the Technical Specifications (TSs) in response to Florida Power & Light Company's (the licensee's) application dated April 9, 2014, as supplemented by letters dated August 29, 2014, and February 20, April 3, and July 7, 2015.

The amendments revise the TSs by relocating specfic surveillance frequencies to a licensee-controlled program. The TSs are revised to require that changes to such surveillance frequencies will be made in accordance with Nuclear Energy Institute 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies." The changes are consistent with NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specifications change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Specifications Task Force (RITSTF) Initiative 5b," Revision 3. The *Federal Regsiter* (FR) notice published on July 6, 2009 (74 FR 31996), announced the availability of the TSTF.

The NRC staff's safety evaluation of the amendments is enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Audrey L. Klett, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 263 to DPR-31
2. Amendment No. 258 to DPR-41
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 263
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated April 9, 2014, as supplemented by letters dated August 29, 2014, and February 20, April 3, and July 7, 2015, 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;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-31 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 263 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.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Operating License DPR-31
and the Technical Specifications

Date of Issuance: July 16, 2015

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
Renewed License No. DPR-41

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated April 9, 2014, as supplemented by letters dated August 29, 2014, and February 20, April 3, and July 7, 2015, 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;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-41 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 258 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.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Operating License DPR-41
and the Technical Specifications

Date of Issuance: July 16, 2015

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 263 RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 258 RENEWED FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace page 3 of Renewed Facility Operating License DPR-31 with the attached page 3. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Replace page 3 of Renewed Facility Operating License DPR-41 with the attached page 3. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert	Remove	Insert	Remove	Insert	Remove	Insert
vii	vii	3/4 3-33	3/4 3-33	3/4 5-8	3/4 5-8	3/4 8-14	3/4 8-14
x	x	3/4 3-34	3/4 3-34	3/4 5-10	3/4 5-10	3/4 8-15	3/4 8-15
xvi	xvi	3/4 3-35	3/4 3-35	3/4 6-1	3/4 6-1	3/4 8-20	3/4 8-20
xvii	-----	3/4 3-36	3/4 3-36	3/4 6-4	3/4 6-4	3/4 8-23	3/4 8-23
1-7	1-7	3/4 3-37	3/4 3-37	3/4 6-5	3/4 6-5	3/4 9-1	3/4 9-1
3/4 1-1	3/4 1-1	3/4 3-38	3/4 3-38	3/4 6-6	3/4 6-6	3/4 9-2	3/4 9-2
3/4 1-2	3/4 1-2	3/4 3-42	3/4 3-42	3/4 6-11	3/4 6-11	3/4 9-4	3/4 9-4
3/4 1-3	3/4 1-3	3/4 3-43	3/4 3-43	3/4 6-12	3/4 6-12	3/4 9-5	3/4 9-5
3/4 1-7	3/4 1-7	3/4 3-44	3/4 3-44	3/4 6-13	3/4 6-13	3/4 9-7	3/4 9-7
3/4 1-9	3/4 1-9	3/4 3-50	3/4 3-50	3/4 6-14	3/4 6-14	3/4 9-8	3/4 9-8
3/4 1-11	3/4 1-11	3/4 3-53	3/4 3-53	3/4 6-15	3/4 6-15	3/4 9-9	3/4 9-9
3/4 1-15	3/4 1-15	3/4 4-1	3/4 4-1	3/4 6-17	3/4 6-17	3/4 9-10	3/4 9-10
3/4 1-17	3/4 1-17	3/4 4-2	3/4 4-2	3/4 7-3	3/4 7-3	3/4 9-11	3/4 9-11
3/4 1-21	3/4 1-21	3/4 4-4	3/4 4-4	3/4 7-4	3/4 7-4	3/4 9-13	3/4 9-13
3/4 1-22	3/4 1-22	3/4 4-5	3/4 4-5	3/4 7-7	3/4 7-7	3/4 10-1	3/4 10-1
3/4 1-23	3/4 1-23	3/4 4-6	3/4 4-6	3/4 7-8	3/4 7-8	3/4 10-2	3/4 10-2
3/4 1-24	3/4 1-24	3/4 4-9	3/4 4-9	3/4 7-9	3/4 7-9	3/4 10-3	3/4 10-3
3/4 1-25	3/4 1-25	3/4 4-11	3/4 4-11	3/4 7-11	3/4 7-11	3/4 10-4	3/4 10-4
3/4 1-26	3/4 1-26	3/4 4-13	3/4 4-13	3/4 7-12	3/4 7-12	6-13	6-13
3/4 2-2	3/4 2-2	3/4 4-15	3/4 4-15	3/4 7-13	3/4 7-13	6-14	6-14
3/4 2-4	3/4 2-4	3/4 4-20	3/4 4-20	3/4 7-14	3/4 7-14	6-15	6-15
3/4 2-5	3/4 2-5	3/4 4-21	3/4 4-21	3/4 7-15	3/4 7-15	6-16	6-16
3/4 2-6	3/4 2-6	3/4 4-22	3/4 4-22	3/4 7-16	3/4 7-16	6-17	6-17
3/4 2-10	3/4 2-10	3/4 4-23	3/4 4-23	3/4 7-17	3/4 7-17	6-18	6-18
3/4 2-13	3/4 2-13	3/4 4-26	3/4 4-26	3/4 7-20	3/4 7-20	6-19	6-19
3/4 2-14	3/4 2-14	3/4 4-28	3/4 4-28	3/4 7-21	3/4 7-21	6-20	6-20
3/4 3-8	3/4 3-8	3/4 4-29	3/4 4-29	3/4 7-28	3/4 7-28	6-21	6-21
3/4 3-9	3/4 3-9	3/4 5-1	3/4 5-1	3/4 8-5	3/4 8-5	-----	6-22
3/4 3-10	3/4 3-10	3/4 5-2	3/4 5-2	3/4 8-6	3/4 8-6		
3/4 3-11	3/4 3-11	3/4 5-5	3/4 5-5	3/4 8-7	3/4 8-7		
3/4 3-32	3/4 3-32	3/4 5-7	3/4 5-7	3/4 8-10	3/4 8-10		

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. 263 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.

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. 258 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 April 10, 2013.

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.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

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

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

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*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 ± 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

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.

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

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

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.

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.

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

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.

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:

$$\left[\left[\frac{F_Q^M(Z)}{[F_Q]^L \ \mathrm{X} \ K(Z)/P}\right] - 1\right] \mathrm{X}100 \text{ for } P \geq 0.5$$

$$\left[\left[\frac{F_Q^M(Z)}{[F_Q]^L \ \mathrm{X} \ K(Z)/0.5}\right] - 1\right] \mathrm{X}100 \text{ for } P < 0.5$$

* 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.

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* $[F_j(Z)]_s$ 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 $\overline{R}$ for each selected thimble. The flux map must be updated in accordance with the Surveillance Frequency Control Program.

where:

$\overline{R}$ = 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

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.

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 ≥ 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.

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 4.3-2

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	SFCP	SFCP[(a)(b)]	SFCP(5)[(a)(b)]	N.A	N.A	1, 2, 3(3)
Coincident with: 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.

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.

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.

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.

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.

REACTOR COOLANT SYSTEM

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.

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

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* 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.

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.

RELIEF VALVES

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

3/4.4.6 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.

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.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	SFCP
Chloride**	SFCP
Fluoride**	SFCP

* Not required with average reactor coolant temperature less than or equal to 250°F.

** Not required when reactor is defueled and RCS forced circulation is unavailable.

REACTOR COOLANT SYSTEM

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		

REACTOR COOLANT SYSTEM

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.

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

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

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.

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.

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 temporary operation of these valves for surveillance or maintenance purposes, power may be restored to these valves for a period not to exceed 24 hours.

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

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping, 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.

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, 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

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.

* 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.

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.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

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 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.

SURVEILLANCE REOUIREMENTS (Continued)

c. 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.

d. 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.

CONTAINMENT SYSTEMS

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.

CONTAINMENT SYSTEMS

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

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

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.

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.

PLANT SYSTEMS

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

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.

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*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.

PLANT SYSTEMS

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.

PLANT SYSTEMS

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.

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.

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.

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.

3/4.7.7 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

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.

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.

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.

D.C. SOURCES

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 $\geq$ 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 3B, 4A	Connection inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	Limit (Micro-Ohms) $\leq$ 29 $\leq$ 30 $\leq$ 125 $\leq$ 1958
Battery 3A, 4B, D-52	Connection inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	Limit (Micro-Ohms) $\leq$ 35 $\leq$ 40 $\leq$ 125 $\leq$ 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	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

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.

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).

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.

* 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.

3/4.9.4 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.

REFUELING OPERATIONS

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS.

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.

*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.

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 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.

* 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.

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.

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

3/4.9.14 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.

SPECIAL TEST EXCEPTIONS

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.

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_{Δ}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 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.

**As evaluated in NRC Safety Evaluation dated December 20, 1997.

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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR

AMENDMENT NO. 263 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31 AND

AMENDMENT NO. 258 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated April 9, 2014, as supplemented by letters dated August 29, 2014, and February 20, April 3, and July 7, 2015,[1] Florida Power & Light Company (the licensee) requested changes to the Technical Specifications (TSs) for the Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Turkey Point 3 and 4), which are contained in Appendix A of Renewed Facility Operating Licenses DPR-31 and DPR-41. The licensee requested to revise the Turkey Point 3 and 4 TSs by relocating specfic surveillance requirement (SR) frequencies to a licensee-controlled program. The licensee requested to revise the TSs to require that changes to such surveillance frequencies will be made in accordance with Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specification Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies."[2] The requested change is the adoption of the U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specifications (STSs) change, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF [Risk Informed TSTF] Initiative 5b."[3] The *Federal Regsiter* (FR) notice published on July 6, 2009 (74 FR 31996), announced the availability of the TSTF.

By letter dated August 7, 2014, and electronic mail (e-mail) dated January 22, 2015,[4] the NRC sent requests for additional information (RAIs) to the licensee. By letters dated August 29, 2014, and February 20, 2015, the licensee responded to these requests. These letters provided clarifying information that did not expand the scope of the application and did not change the NRC staff's original proposed no significant hazards consideration (NSHC) determination, as published in the FR on July 22, 2014 (79 FR 42551). By letter dated

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[1] Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14105A042, ML14252A228, ML15069A153, ML15113A311, and ML15194A120, respectively.
[2] ADAMS Accession No. ML071360456.
[3] ADAMS Accession No. ML090850642.
[4] ADAMS Accession Nos. ML14212A713 and ML15023A080, respectively.

April 3, 2015, the licensee supplemented its amendment request with a proposed change that expanded the scope of the request. Therefore, the NRC published a second proposed NSHC determination in the FR on May 12, 2015 (80 FR 27199), which superseded the notice dated July 22, 2014 (79 FR 44551). The licensee's supplement dated July 7, 2015, did not expand the scope of the submittal and did not change the NRC staff's proposed NSHC determination, as published in the notice dated May 12, 2015 (80 FR 27199).

2.0 REGULATORY EVALUATION

2.1 Description of the Proposed Changes

The licensee proposed to modify the Turkey Point 3 and 4 TSs by relocating specific surveillance frequencies to a licensee-controlled program (i.e., the Surveillance Frequency Control Program (SFCP)) in accordance with NEI 04-10, Revision 1. The proposed change is consistent with the adoption of NRC-approved TSTF-425, Revision 3. When implemented, TSTF-425 relocates most periodic frequencies of TS surveillances to the SFCP and provides requirements for the new program in the Administrative Controls sections of the TSs. All surveillance frequencies can be relocated except the following:

- Frequencies that reference other approved programs for the specific interval, such as the In-Service Testing Program or the Primary Containment Leakage Rate Testing Program;
- Frequencies that are purely event-driven (e.g., "each time the control rod is withdrawn to the 'full out' position");
- Frequencies that are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching ≥ 95% RTP"); and
- Frequencies that are related to specific conditions (e.g., battery degradation, age and capacity) or conditions for the performance of an SR (e.g., "drywell to suppression chamber differential pressure decrease").

The licensee proposed to add the SFCP to TSs Section 6.0, "Administrative Controls," Subsection 6.8, "Procedures and Programs." The SFCP describes the requirements for controlling changes to the relocated surveillance frequencies. The TS Bases for each affected surveillance would be revised to state that the frequency is controlled under the SFCP. Some SR TS Bases do not contain a discussion of the frequency. The proposed TS Bases changes revise only those TS Bases that currently discuss surveillance frequencies. The existing TS Bases information describing the basis for the surveillance frequencies will be relocated to the SFCP. The proposed changes to the Administrative Controls sections of the TSs to incorporate the SFCP include a specific reference to NEI 04-10, Revision 1 as the basis for making any changes to the surveillance frequencies once they are relocated out of the TSs.

In a letter dated September 19, 2007,[5] the NRC staff approved Topical Report NEI 04-10, Revision 1 as acceptable for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04-10, Revision 1, and the safety evaluation (SE) providing the basis for NRC acceptance of NEI 04-10, Revision 1.

The licensee proposed other changes and deviations from TSTF-425, which are discussed in Section 3.3 of this SE.

2.2 Applicable Commission Policy Statements

In the "Final Policy Statement: Technical Specifications for Nuclear Power Plants," dated July 22, 1993 (58 FR 39132), the NRC addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment or PRA) in STSs. In this 1993 publication, the NRC states:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed.
>
> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "* * * probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made * * * about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety."
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

Approximately 2 years later, the NRC provided additional detail concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities," dated August 16, 1995 (60 FR 42622). In this publication, the NRC states:

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner

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[5] ADAMS Accession No. ML072570267.

that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach....

PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner....

Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data....

Therefore, the Commission adopts the following policy statement regarding the expanded NRC use of PRA:

(1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

(2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

(3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of

uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

2.3 Applicable Regulations

In 10 CFR, Section 50.36, the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) SRs; (4) design features; and (5) administrative controls. These categories will remain in the Turkey Point 3 and 4 TSs.

Paragraph 50.36(c)(3) of 10 CFR states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The FR notice published on July 6, 2009 (74 FR 31996), which announced the availability of TSTF-425, Revision 3, states that the addition of the SFCP to the TSs provides the necessary administrative controls to require that surveillance frequencies relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The FR notice also states that changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, Revision 1, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and are required to be documented.

Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," (i.e., the Maintenance Rule) and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. Such failures can result in the licensee increasing the frequency of a surveillance test. In addition, by having the TSs require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04-10, Revision 1, the licensee will be required to monitor the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs.

2.4 Applicable NRC Regulatory Guides and Review Plans

Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,"[6] describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

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[6] ADAMS Accession No. ML100910006.

RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,"[7] describes an acceptable risk-informed approach specifically for assessing proposed TS changes.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,"[8] describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light water-reactors.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance."[9] Guidance on evaluating PRA technical adequacy is provided in the SRP, Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed License Amendment Requests after Initial Fuel Load."[10] More specific guidance related to risk-informed TS changes is provided in SRP, Section 16.1, Revision 1, "Risk-Informed Decisionmaking: Technical Specifications,"[11] which includes changes to surveillance test intervals (i.e., surveillance frequencies) as part of risk-informed decision making. Section 19.2 of the SRP references the same criteria as RG 1.177, Revision 1, and RG 1.174, Revision 2, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes result in an increase in core damage frequency (CDF) or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- The impact of the proposed change should be monitored using performance measurement strategies.

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[7] ADAMS Accession No. ML100910008.
[8] ADAMS Accession No. ML090410014.
[9] ADAMS Accession No. ML071700658.
[10] ADAMS Accession No. ML12193A107.
[11] ADAMS Accession No. ML070380228.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425, Revision 3 provides for administrative relocation of applicable surveillance frequencies, and it provides for the addition of the SFCP to the Administrative Controls sections of TSs. The changes to the Administrative Controls section of the TSs will also require the application of NEI 04-10, Revision 1 for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes described in TSTF-425, Revision 3 included documentation regarding the PRA technical adequacy consistent with RG 1.200, Revision 2. NEI 04-10, Revision 1, states that PRA methods are used with plant performance data and other considerations to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is consistent with guidance provided in RG 1.174, Revision 2, and RG 1.177, Revision 1, in support of changes to surveillance test intervals.

3.1 Review Methodology

RG 1.177, Revision 1, identifies five key safety principles to be applied to risk-informed changes to TSs. Each of these principles is addressed by NEI 04-10, Revision 1.

3.1.1 The Proposed Change Meets Current Regulations

Paragraph 50.36(c)(3) of 10 CFR requires that TSs include surveillances, which are "requirements relating to test, calibration, or inspection to assure that necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The licensee is required by its TSs to perform surveillance tests, calibration, or inspection on specific safety-related equipment (e.g., reactivity control, power distribution, electrical, and instrumentation) to verify system operability. Surveillance frequencies are based primarily on deterministic methods, such as engineering judgment, operating experience, and manufacturer's recommendations. The licensee's use of NRC-approved methodologies identified in NEI 04-10, Revision 1, provides a way to establish risk-informed surveillance frequencies that complements the deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

The SRs themselves are remaining in the TSs, as required by 10 CFR 50.36(c)(3). This change is analogous with other NRC-approved TS changes in which the SRs are retained in TSs, but the related surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the In-Service Testing Program and the Primary Containment Leakage Rate Testing Program. Thus, this proposed change complies with 10 CFR 50.36(c)(3) by retaining the requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulatory requirements in 10 CFR 50.65 and 10 CFR Part 50, Appendix B, and the monitoring required by NEI 04-10, Revision 1 ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified and appropriate corrective actions taken. The licensee's SFCP ensures that SRs specified in the TSs are performed at intervals sufficient to assure the above regulatory requirements are met. In light of the above, the staff concludes that the proposed

change meets the first key safety principle of RG 1.177, Revision 1, by complying with current regulations.

3.1.2 The Proposed Change Is Consistent With the Defense-in-Depth Philosophy

The defense-in-depth philosophy (i.e., the second key safety principle of RG 1.177, Revision 1) is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation;
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided;
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers). (Because the scope of the proposed methodology is limited to revision of surveillance frequencies, the redundancy, independence, and diversity of plant systems are not impacted.);
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed;
- Independence of barriers is not degraded;
- Defenses against human errors are preserved; and
- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A, is maintained.

The changes to the Administrative Controls section of the TSs will require the application of NEI 04-10, Revision 1, for any changes to surveillance frequencies within the SFCP. NEI 04-10, Revision 1 uses both the CDF and the large early release frequency (LERF) metrics to evaluate the impact of proposed changes to surveillance frequencies. The guidance of RG 1.174, Revision 2, and RG 1.177, Revision 1, for changes to CDF and LERF is achieved by evaluation using a comprehensive risk analysis, which assesses the impact of proposed changes including contributions from human errors and common cause failures (CCFs). Defense-in-depth is also included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to an increased likelihood of common cause failures. The staff concludes that both the quantitative risk analysis and the qualitative considerations assure that a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety, thus satisfying the second key safety principle of RG 1.177, Revision 1.

3.1.3 The Proposed Change Maintains Sufficient Safety Margins

The engineering evaluation that will be conducted by the licensee under the SFCP when frequencies are revised will assess the impact of the proposed frequency change to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the proposed surveillance test frequency change is not in conflict with approved industry codes and standards or adversely affects any assumptions or inputs to the safety analysis, or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The design, operation, testing methods, and acceptance criteria for SSCs specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plants' licensing bases, including the Updated Final Safety Analysis Report (UFSAR) and TS Bases, because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plants' licensing bases. On this basis, the staff concludes that safety margins are maintained by the proposed methodology, and the third key safety principle of RG 1.177, Revision 1, is satisfied.

3.1.4 When Proposed Changes Result in an Increase in CDF or Risk, the Increases Should Be Small and Consistent with the Intent of the Commission's Safety Goal Policy Statement

RG 1.177, Revision 1, provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies, which requires identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. The changes to the Administrative Controls section of the TSs will require application of NEI 04-10, Revision 1, in the SFCP. NEI 04-10, Revision 1, satisfies the intent of RG 1.177, Revision 1, guidance for evaluating the change in risk and for assuring that such changes are small by providing the technical methodology to support risk-informed TSs for control of surveillance frequencies.

3.1.4.1 Quality of the PRA

The quality of the licensee's PRA must be commensurate with the safety significance of the proposed TS change and the role the PRA plays in justifying the change. That is, the higher change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the quality of the PRA.

RG 1.200 provides regulatory guidance for assessing the technical adequacy of a PRA. The current revision (i.e., Revision 2) of this RG endorses, with clarifications and qualifications, the use of (1) the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard, RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (i.e., the PRA Standard), (2) NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process

Guidelines,"[12] and (3) NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard."[13]

The licensee has performed an assessment of the PRA models used to support the SFCP using the guidance of RG 1.200, Revision 2, to assure that the PRA models are capable of determining the change in risk due to changes to surveillance frequencies of SSCs, using plant-specific data and models. Capability Category II of the NRC-endorsed PRA standard is the target capability level for supporting requirements for the internal events PRA for this application. Any identified deficiencies to those requirements are assessed further to determine any impacts to proposed decreases to surveillance frequencies, including the use of sensitivity studies where appropriate, in accordance with NEI 04-10, Revision 1.

An Industry PRA Certification peer review of the Turkey Point internal events PRA was performed in 2002. In its application dated April 9, 2014, the licensee stated that all peer review facts and observations (F&Os) findings from this peer review have been addressed in model updates. The licensee also performed an internal gap analysis after issuance of the PRA Standard and RG 1.200. The licensee stated that the current Turkey Point gap analysis uses the RA-Sa-2009 version of the standard as endorsed by RG 1.200, Revision 2. The NRC staff notes that the licensee's April 4, 2014, response[14] to NRC's PRA RAI 22.01 associated with the licensee's request to adopt National Fire Protection Association (NFPA) Standard 805 (NFPA 805) provides the results of the Turkey Point Gap Analysis performed to ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2. In its response to RAI 22.01, the licensee stated:

> The Industry Self-Assessment Actions and Regulatory Position from RG 1.200, Revision 2, for each SR were also considered for each of the Standard SRs. No new F&Os were created during this process, as the scope and requirements of the NEI 00-02 process and the Standard were very similar. Focused peer reviews were conducted in 2011 and 2013, and their findings were integrated into the overall gap analysis.

In Attachment 2 of its application dated April 9, 2014, the licensee describes the focused-scope peer reviews. The April 2011 focused-scope peer review was performed for the internal flooding portion of the PRA and for the internal events PRA elements for Human Reliability Analysis. The October 2013 focused-scope peer review was performed to review upgrades to the CCFs, Level 2 PRA, and Interfacing Systems Loss of Coolant Accident (ISLOCA) analyses. Both focused-scope peer reviews were performed using ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2. In enclosure to Attachment 2 of its application dated April 9, 2014, the licensee provides a list of the significant F&Os from the 2002 peer review (i.e., significance ranking "A" or "B") as well as the findings from the 2011 and 2013 peer reviews.

In its application dated April 9, 2014, the licensee noted that 13 of the F&Os from the 2013 peer review had not yet been resolved in the current PRA model. One of these F&Os is F&O

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[12] ADAMS Accession Nos. ML061510619 and ML063390588.
[13] ADAMS Accession No. ML083430462.
[14] ADAMS Accession No. ML14113A176.

LE-D2-01 from the 2013 peer review, which is associated with assessing the potential impact of electrical penetration assembly failure on LERF results. In Attachment 2 of its application dated April 9, 2014, the licensee explained in its disposition of this F&O that the current Level 2 PRA for Turkey Point uses existing containment analyses and NUREG references, and the licensee concluded in its disposition that the F&O finding is not expected to have a significant effect. Because this F&O finding would only potentially impact the LERF metrics associated with a small portion of the Level 2 PRA, the NRC staff concludes that the licensee's disposition of this F&O is acceptable for this application.

In its letter dated February 20, 2015, the licensee responded to NRC's APLA (NRC's PRA Licensing Branch in its Office of Nuclear Reactor Regulation) RAIs 3 and 4 and provided more information on the impact and the resolution of the 12 remaining 2013 peer review F&Os (i.e., DA-D5-01, DA-D6-01, DA-D6-02, IE-C14-01, IE-C14-02, IE-C14-03, IE-C14-04, IE-C14-05, IE-C14-06, IE-C14-07, LE-F1-01 and LE-G5-01). The licensee explained that F&O findings LE-F1-01 and LE-G5-01 are related to documentation associated with the categorization and uncertainty discussion of LERF results. The licensee also explained that F&O suggestion IE-C14-07 is related to documentation. Regarding F&O DA-D6-01, the licensee stated that no CCF events were found as part of the review of plant records for the data analysis used in the current model-of-record and that this will be documented in the CCF notebook. F&O IE-C14-06 is a suggestion only and is not required to be addressed for this application. These F&Os will not affect the risk calculations performed to support surveillance frequency changes, and the NRC staff concludes that the licensee's dispositions of these F&Os are acceptable for this application.

In its letter dated February 20, 2015, the licensee responded to APLA RAI 3 and explained that it has the capability to perform sensitivity studies to address F&O findings DA-D5-01, IE-C14-01, IE-C14-02, and IE-C14-04 for the application. NEI 04-10, Revision 1 allows for sensitivity studies to assess the impact of deviations from Capability Category II of the PRA Standard. NEI 04-10, Revision 1, Section 4.0, Step 14 also states that qualitative considerations can be used to support surveillance frequency changes even though the change may not be supported by the sensitivity study. The use of sensitivity studies is discussed further in Section 3.1.4.5 of this SE. The staff concludes that, based on this information, the licensee has adequately dispositioned these F&Os for this application. Furthermore, in its letter dated February 20, 2015, the licensee also stated that resolution of these F&Os is planned for the next internal events PRA model update such that sensitivities would no longer be required after the update.

F&O DA-D6-02 pertains to the treatment of CCFs in fault tree initiating event models for components with year-long mission times. In its application dated April 9, 2014, the licensee stated that CCFs are included for the components in the initiating event fault trees. The licensee also provided the following example of the treatment of CCFs for the component cooling water (CCW) pumps:

> ... where 2/3 pumps are normally running, there are AND gates with a single FTR [fail to run] event of one of the normally running pumps with an 8760-hour [or 1 year] mission time and CCF events for the other 2 running pumps with mission times equal to the MTR [meantime to recovery] of the pumps. There is

not a CCF for all 3 pumps with a mission time of 8760 hours, nor should there be; all 3 pumps are not normally running at the same time ...

Section 9.3.2 of the Turkey Point UFSAR confirms this by stating that the two standby pumps provide 100 percent backup during normal operation and that the design bases specify that the operation of one CCW pump and one residual heat removal heat exchanger is adequate for accident mitigation. Therefore, the NRC staff concludes that the licensee's disposition of F&O DA-D6-02 is acceptable for this application.

F&O IE-C14-03 pertains to reactor coolant pump thermal barrier ISLOCA initiating event frequency resulting from the CCF of two normally-open active check valves. The licensee performed a conservative sensitivity study that showed a delta CDF of 5.2E-9 per year. In its letter dated February 20, 2015, the licensee also stated that the associated penetrations would be examined further as part of the next internal events PRA model update. F&O IE-C14-05 pertains to the screening of penetrations that have three check valves in series. The licensee noted that three check valves may not be as secure as three locked-closed isolation valves, but the licensee still considered this to be adequate screening criteria. Given the minimal impact on CDF and LERF, the NRC staff concludes that the licensee's dispositions of F&Os IE-C14-03 and IE-C14-05 are acceptable for this application.

Based on the licensee's assessments using the currently applicable PRA standard and revision of RG 1.200, the NRC staff concludes that the level of PRA quality, combined with the proposed evaluation and disposition of gaps, is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP and is consistent with Regulatory Position 2.3.1 of RG 1.177, Revision 1.

3.1.4.2 Scope of the PRA

The changes to the Administrative Controls section of the TSs will require the licensee to evaluate each proposed change to a relocated surveillance frequency using NEI 04-10, Revision 1, to determine its potential impact on risk (CDF and LERF) from internal events, fires, seismic, other external events, and shutdown conditions. In cases where a PRA of sufficient scope or quantitative risk models were unavailable, the licensee uses bounding analyses, or other conservative quantitative evaluations. A qualitative screening analysis may be used when the surveillance frequency impact on plant risk is shown to be negligible or zero.

The licensee has an at-power internal events and internal flooding PRA model, as well as an at-power fire PRA, to support the adoption of NFPA 805. In accordance with NEI 04-10, Revision 1, the licensee will use these models to perform quantitative evaluations to support the development of changes to surveillance frequencies in the SFCP. In Attachment 2 to its application dated April 9, 2014, the licensee noted that the fire PRA produces a conservative estimate of core damage risk due to fire and that the Fire PRA model will be exercised to obtain quantitative fire risk insights, but refinements may need to be made on a case-by-case basis. This is acceptable because the NRC-approved methodology in NEI 04-10, Revision 1 allows for more refined analysis to be performed to support changes to surveillance frequencies in the SFCP.

In Attachment 2 to its application dated April 9, 2014, the licensee stated that Turkey Point does not have a seismic PRA and that Turkey Point is sited in an area of very low seismicity. The licensee also stated that it recently performed additional seismic walkdowns in response to Near-Term Task Force Recommendation 2.3 to identify and address plant degraded, non-conforming, or unanalyzed conditions, with respect to the current seismic licensing basis and that no operability concerns were identified. The licensee also referenced the NRC staff memorandum on Generic Issue 199,[15] which discusses recent updates to estimates of the seismic hazard in the central and eastern United States. This analysis reported a seismic CDF estimate of 5.9E-6 per year for Turkey Point using the "simple average" approach based on the 2008 U.S. Geologic Survey seismic hazard curves and a bounding estimate of 1E-5 per year. This supports the licensee's conclusion that seismic risk will not be a significant factor for the Turkey Point SFCP. Similarly, for other external hazards for which there is no PRA model, the licensee stated that a qualitative or a bounding approach will be used in most cases. This is an acceptable approach in accordance with NEI 04-10, Revision 1.

In its letter dated February 20, 2015, the licensee responded to NRC's APLA RAI 2 and explained that Turkey Point does not currently have a shutdown PRA model that meets the guidance in RG 1.200. However, the licensee explained that the guidance in NEI 04-10, Revision 1 will also be applied for shutdown events. This includes the use of the Turkey Point shutdown safety program developed in support of NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management,"[16] as applicable, consistent with NEI 04-10, Revision 1.

Thus, the staff concludes that through the application of NEI 04-10, Revision 1, the licensee's evaluation methodology is sufficient to ensure the scope of the risk contribution of each surveillance frequency change is properly identified for evaluation and is consistent with Regulatory Position 2.3.2 of RG 1.177, Revision 1.

3.1.4.3 PRA Modeling

The licensee's methodology includes the determination of whether the SSCs affected by a proposed change to a surveillance frequency are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact may be carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted CCF modes, based on the proposed change to the surveillance frequency. Where the SSC is not modeled in the PRA, bounding analyses are performed to characterize the impact of the proposed change to the surveillance frequency. Potential impacts on the risk analyses due to screening criteria and truncation levels are addressed by the requirements for PRA technical adequacy consistent with guidance contained in RG 1.200, Revision 2, and by sensitivity studies identified in NEI 04-10, Revision 1.

Thus, the staff concludes that through the application of NEI 04-10, Revision 1, the Turkey Point PRA modeling is sufficient to ensure an acceptable evaluation of risk for the proposed changes

[15] ADAMS Accession No. ML100270582.
[16] ADAMS Accession No. ML14365A203.

in surveillance frequency and is consistent with Regulatory Position 2.3.3 of RG 1.177, Revision 1.

3.1.4.4 Assumptions for Time Related Failure Contributions

The failure probabilities of SSCs modeled in PRAs may include a separate standby time-related contribution and a cyclic demand-related contribution. In its letter dated February 20, 2015, the licensee responded to APLA RAI 1 and explained that the Turkey Point SFCP plant procedures are consistent with NEI 04-10, Revision 1 and that the standby time-related contribution evaluation will be performed in accordance with NEI 04-10, Revision 1. NEI 04-10, Revision 1 criteria adjust the time-related failure contribution of SSCs affected by the proposed change to a surveillance frequency. This is consistent with RG 1.177, Revision 1, Section 2.3.3, which permits separation of the failure rate contributions into demand and standby for evaluation of surveillance requirements. If the available data do not support distinguishing between the time related failures and demand failures, then the change to surveillance frequency is conservatively assumed to impact the total failure probability of the SSC, including both standby and demand contributions. The SSC failure rate per unit time is assumed to be unaffected by the change in test frequency, such that the failure probability is assumed to increase linearly with time, and will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The NEI 04-10, Revision 1 process requires consideration of qualitative sources of information with regards to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus, the process is not reliant upon risk analyses as the sole basis for the proposed changes.

The potential benefits of a reduced surveillance frequency, including reduced downtime and reduced potential for restoration errors, test-caused transients, and test-caused wear of equipment, are identified qualitatively but not quantitatively assessed. Thus, the staff concludes that through the application of NEI 04-10, Revision 1, the licensee has employed reasonable assumptions with regard to extensions of surveillance test intervals and is consistent with Regulatory Position 2.3.4 of RG 1.177, Revision 1.

3.1.4.5 Sensitivity and Uncertainty Analyses

By having the TSs require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04 10, Revision 1, the licensee will be required to have sensitivity studies that assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact on the frequency of initiating events, and any identified deviations from Capability Category II of the PRA standard. Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered, along with any qualitative considerations that may bear on the results of such sensitivity studies. The licensee will also be required to perform monitoring and feedback of SSC performance, once the revised surveillance frequencies are implemented. Thus, the staff concludes that through the application of NEI 04-10, Revision 1, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations and is consistent with Regulatory Position 2.3.5 of RG 1.177, Revision 1.

3.1.4.6 Acceptance Guidelines

The licensee will be required to quantitatively evaluate the change in total risk (including internal and external events contributions) in terms of CDF and LERF for both the individual risk impact of a proposed change in surveillance frequency and the cumulative impact from all individual changes to surveillance frequencies using NEI 04-10, Revision 1, in accordance with the TS SFCP. Each individual change to a surveillance frequency must show a risk impact below 1E-6 per year for changes to CDF, and below 1E-7 per year for changes to LERF. These changes to CDF and LERF are consistent with the acceptance criteria of RG 1.174, Revision 2 for very small changes in risk. Where the RG 1.174, Revision 2 acceptance criteria are not met, the process in NEI 04-10, Revision 1, either considers revised surveillance frequencies, which are consistent with RG 1.174, Revision 2, or the process terminates without permitting the proposed changes. Where quantitative results are unavailable for comparison with the acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or zero. Otherwise, bounding quantitative analyses are required, which demonstrate the risk impact is at least one order of magnitude lower than the RG 1.174, Revision 2 acceptance guidelines for very small changes in risk. In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact of less than 1E-5 per year for changes to CDF and less than 1E-6 per year for changes to LERF. Further, the total CDF and total LERF must be reasonably shown to be less than 1E-4 per year and 1E-5 per year, respectively. These values are consistent with the acceptance criteria of RG 1.174, Revision 2, as referenced by RG 1.177, Revision 1, for changes to surveillance frequencies.

Consistent with the NRC's SE dated September 19, 2007, for NEI 04-10, Revision 1, the TS SFCP will require the licensee to calculate the total change in risk (i.e., the cumulative risk) by comparing a baseline model that uses failure probabilities based on surveillance frequencies prior to being changed per the SFCP to a revised model that uses failure probabilities based on the changed surveillance frequencies. The NRC staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with insignificant risk increases (i.e., less than 5E-8 CDF and 5E-9 LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

The quantitative acceptance guidance of RG 1.174, Revision 2 is supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history. The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results. Post implementation performance monitoring and feedback are also required to assure continued reliability of the components. The licensee's application of NEI 04-10, Revision 1 provides acceptable methods for evaluating the risk increase associated with proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177, Revision 1. Therefore, the staff concludes that the proposed methodology satisfies the fourth key safety principle of RG 1.177, Revision 1 by assuring any increase in risk is small consistent with the intent of the Commission's Safety Goal Policy Statement.

3.1.5 The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies

The licensee's adoption of TSTF-425, Revision 3 requires application of NEI 04-10, Revision 1 in the SFCP. NEI 04-10, Revision 1 requires performance monitoring of SSCs whose surveillance frequencies have been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback include consideration of Maintenance Rule monitoring of equipment performance. In the event of SSC performance degradation, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions that may be required by the Maintenance Rule. The performance monitoring and feedback specified in NEI 04-10, Revision 1 is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 of RG 1.177, Revision 1. Thus, the staff concludes that the fifth key safety principle of RG 1.177, Revision 1 is satisfied.

3.2 Addition of Surveillance Frequency Control Program to Administrative Controls

The licensee proposed including the SFCP and specific requirements in the Turkey Point 3 and 4 TSs, Section 6.0, "Administrative Controls," as follows:

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 that the associated Limiting Conditions for Operation are met.

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

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

The proposed program is consistent with the model application of TSTF-425, and therefore, the staff concludes that it is acceptable.

3.3 Deviations from TSTF-425 and Other Changes

3.3.1 Revised Clean TS Pages

In its application dated April 9, 2014, the licensee stated that revised (i.e., clean) TS pages are not included in the amendment request because of the number of TS pages affected, the straightforward nature of the proposed changes, and outstanding license amendment requests that may affect some of the same TS pages. The licensee stated that providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," in that the mark-ups fully describe the changes desired. The licensee stated that this is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996), with no impact on the NRC staff's model SE published in the same FR notice. The licensee stated that because of this deviation, the contents and numbering of the attachments for the amendment request differ from the attachments specified in the NRC staff's model application. The NRC staff finds this acceptable.

3.3.2 Differences between Turkey Point 3 and 4 TSs and NUREG-1431

In its application dated April 9, 2014, the licensee stated that the Turkey Point 3 and 4 TSs surveillance numbers and associated TSs Bases numbers differ from those in NUREG-1431, Revision 4, "Standard Technical Specifications – Westinghouse Plants," Volumes 1 and 2,[17] and TSTF-425, Revision 3. In addition, the Administrative Controls section in the Turkey Point 3 and 4 TSs is Section 6.0 versus Section 5.0 for NUREG-1431, Revision 4. There are also surveillances contained in NUREG-1431, Revision 4 that are not contained in the Turkey Point 3 and 4 TSs. These surveillances identified in TSTF-425, Revision 3 for NUREG-1431, Revision 4 are not applicable to Turkey Point. These differences are administrative deviations from TSTF-425, Revision 3 with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

In its letter dated April 3, 2015, the licensee stated that in some instances, a table notation or footnote in the Turkey Point TSs specifies the frequency for an SR, and the licensee proposed to relocate these frequencies to the SFCP. These differences are administrative deviations from TSTF-425, Revision 3 with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

In its application dated April 9, 2014, and in its letter dated April 3, 2015, the licensee noted that the Turkey Point 3 and 4 TSs include plant-specific surveillances that are not contained in NUREG-1431, Revision 4, and therefore, are not included in the NUREG-1431, Revision 4 surveillances provided in TSTF-425, Revision 3. The licensee requested that some plant-specific surveillance frequencies be relocated to the SFCP. The relocation of the plant-specific surveillance frequencies is consistent with TSTF-425, Revision 4 and with the NRC staff's model SE dated July 6, 2009 (74 FR 31996), including the scope exclusions identified in Section 1.0, "Introduction," of the model SE, because the plant-specific surveillance frequencies

[17] ADAMS Accession Nos. ML12100A222 and ML12100A228.

involve fixed period frequencies. Changes to the frequencies for these plant-specific surveillances would be controlled under the SFCP. The NRC staff finds this acceptable.

In its application dated April 9, 2014, the licensee stated that the SR for the reactor trip and engineered safety features actuation system instrumentation in Turkey Point 3 and 4 TSs 3.3.1 and 3.3.2 are presented in tabular format, which is different from the format for the same instrumentation in NUREG-1431, Revision 4. To accommodate this difference, the licensee proposed to include the use of "SFCP" as a frequency notation in the tables that specify instrumentation SRs. This is an administrative deviation from TSTF-425, Revision 3, resulting from differences in format between Turkey Point 3 and 4 TSs and NUREG-1431, Revision 4, which has no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

3.3.3 Definition of Staggered Test Basis

In its application dated April 9, 2014, the licensee stated that the definition of STAGGERED TEST BASIS is being retained in Section 1.0, "Definitions," of the Turkey Point 3 and 4 TSs because the terminology is used in TS Sections 3/4.3, "Instrumentation," 3/4.7, "Plant Systems," 3/4.8, "Electrical Power Systems," and 6.8.4.k, "Control Room Envelop Habitability Program." This is an administrative deviation from TSTF-425, Revision 3, with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

3.3.4 Changes to TS SR 4.5.2

Turkey Point 3 and 4 TS SR 4.5.2 contains the surveillance requirements for emergency core cooling system (ECCS) components and flow paths, which include the residual heat removal (RHR) and safety injection (SI) pumps. TS SR 4.0.5 requires that inservice testing (IST) 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, and SR 4.0.5b specifies the frequencies for IST activities. SR 4.0.5b requires that when the ASME OM Code specifies testing activities be done quarterly or every three months, the licensee is to perform those testing activities every 92 days. SR 4.5.2b.3 requires that each ECCS component and flow path be demonstrated OPERABLE at least once per 31 days by verifying that each RHR pump develops the indicated differential pressure applicable to the operating conditions in accordance with TS Figure 3.5-1 when tested pursuant to SR 4.0.5. SR 4.5.2c.1 requires that each ECCS component and flow path be demonstrated OPERABLE at least once per 92 days by verifying that each SI pump develops the indicated differential pressure applicable to the operating conditions specified in the SR when tested pursuant to SR 4.0.5.

In its letter dated April 3, 2015, the licensee stated that SR 4.5.2b.3 contains conflicting frequencies because this SR requires that the RHR pumps be tested every 31 days whereas the ASME OM Code specifies a quarterly frequency (every 92 days) in accordance with SR 4.0.5. The licensee therefore expanded the scope of its original application and requested that the TS 3/4.5.2 SRs for the RHR and SI pump differential pressure tests be revised to eliminate the 31-day and 92-day frequencies within these SRs and retain the references to SR 4.0.5, which already establishes the frequency for the RHR and SI pump differential tests (every 92 days). This change would eliminate conflicting and any potential for conflicting test

frequency requirements. The licensee stated that this change would be consistent with other SRs (e.g., the containment spray pumps in SR 4.6.2.1b and the containment isolation valves in SR 4.6.4.3) and with the STS SR 3.5.2.4 in NUREG-1431, Revision 4.

The licensee also proposed to move SR 4.5.2b.3 to SR 4.5.2c and to rewrite SR 4.5.2c to state:

c. By verifying that each SI and RHR pump develops the indicated differential pressure applicable to the operating conditions when testing pursuant to Specification 4.0.5:

   1) SI pump ≥ 1083 psid at a metered flowrate ≥ 300 gpm (normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST [refueling water storage tank]), or

   ≥ 1113 psid at a metered flowrate ≥ 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.

The NRC staff determined that the proposed change to SR 4.5.2c.1 for the SI pump testing is acceptable because it removes redundant requirements and does not relax any requirements. Regarding the proposed change to SR 4.5.2b.3 for the RHR pump testing, the NRC reviewed the licensing history for this SR to determine how the apparent inconsistency was introduced. The original TSs for this SR required a monthly frequency. By letter dated June 5, 1989,[18] the licensee requested to convert its custom TSs to a set of Revised TSs (RTSs) based on STSs for Westinghouse plants. In its application dated June 5, 1989, the licensee requested that the SI and RHR pump surveillance frequencies be relaxed from monthly to quarterly per the requirements of the IST program based on the ASME Code. The licensee stated that the relaxation was justified based on the high reliability of the pumps, reduced probability of system failure that could be introduced by human error at the time of the surveillance test, and reduced wear on the affected components. However, the proposed TS page that the licensee submitted in its application for the RTS did not remove the monthly frequency for the RHR pump SR and, therefore, contained the inconsistent frequencies.

By letter dated August 28, 1990,[19] the NRC approved the RTSs conversion in Amendments 137 and 132. The NRC acknowledged the relaxation in the "Categorization of Changes to the Current Tech Specs" table and generically in Section 3.4 of the SE for those amendments. However, it issued the TS page as submitted by the licensee. Therefore, the staff's SE was not consistent with the issued TSs. With regard to the current application, the NRC staff has determined that the relaxed frequency was previously evaluated and accepted in the NRC staff's SE for Amendments 137 and 132 and that the licensee's proposed correction to the frequency inconsistencies in its letter dated April 3, 2015, is acceptable. The proposed change

[18] ADAMS Legacy Library Accession No. 8906070246.
[19] ADAMS Accession Nos. ML013370513 and ML013440606, and Legacy Library Accession Nos. 9009050244, 9009050245, and 9009050246.

is also consistent with the STSs in NUREG-1431, Revision 4 in that the STS SR 3.5.2.4 (for ECCS pumps) refers to the IST program, which establishes the test frequency.

3.3.5 Deletion of "Program" Terminology

In its letter dated April 3, 2015, the licensee stated that TSTF-425, Revision 3 excludes relocating frequencies that reference other approved programs for the specific interval, such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program. The approved programs for Turkey Point 3 and 4 are described in Section 6.0, "Administrative Controls," of the Turkey Point 3 and 4 TSs. The licensee stated that the titles of TS Table 4.4-4, "Primary Coolant Specific Activity Sample and Analysis Program," and Table 4.7-1, "Secondary Coolant System Specific Activity Sample and Analysis Program," may be misconstrued as programs. However, the licensee stated that Section 6.0 of the TSs does not contain programs for sampling and analysis of reactor coolant or secondary coolant specific activity. To avoid a misunderstanding of these SRs, the licensee proposed to delete the word "Program" from the titles of TS Table 4.4-4, and Table 4.7-1. The licensee also proposed deleting the word "program" from SR 4.4.8 and SR 4.7.1.4, which refer to the tables. The NRC staff determined that this is an administrative deviation from TSTF-425, Revision 3 with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). The NRC staff finds the proposed changes acceptable.

3.3.6 Editorial Changes

In its application dated April 9, 2014, and in its letters dated April 3, and July 7, 2015, the licensee proposed editorial changes to the TSs. In its application dated April 9, 2014, the licensee proposed to revise TS Table 3.3-4, "Action Statements," Action 27 by changing "expect" to "except." In its application dated April 9, 2014, the licensee also proposed revising TS 3.7.1.2 by changing "APPLICA8ILITY" to "APPLICABILITY." In its letters dated April 3, and July 7, 2015, the licensee proposed to revise SR 4.8.1.1.2a.4 by changing "fro m" to "from" and SR 4.8.1.1.2a.6 by changing "busses" to "buses." In its letter dated July 7, 2015, the licensee proposed renumbering SR 4.8.1.1.2l.1 and SR 4.8.1.1.2l.2 to SR 4.8.1.1.2i and SR 4.8.1.1.2j, respectively. These are administrative deviations from TSTF-425, Revision 3 with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). The NRC staff determined that these changes are editorial or correct typographical errors and are acceptable.

3.3.7 Conforming Changes

In its letters dated April 3, and July 7, 2015, the licensee proposed conforming changes to the Turkey Point 3 and 4 TSs Index, which include consolidation of Index pages xvii and xvi, and Section 6 that reflect new page numbering resulting from the changes associated with this application. The NRC staff finds these changes acceptable.

3.4 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of some surveillance frequencies to a licensee-controlled document, and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the Administrative Controls of TSs. The NRC staff confirmed that this amendment does not relocate surveillance

frequencies that reference other approved programs for the specific interval, are purely event-driven, are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs, or are related to specific conditions. The SFCP and TSs Section 6.0, Subsection 6.8.4, references NEI 04-10, Revision 1, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TSs to a licensee-controlled document, provided those frequencies are changed in accordance with NEI 04-10, Revision 1, which is specified in the Administrative Controls of the TSs.

The licensee's proposed adoption of TSTF-425, Revision 3, and risk-informed methodology of NEI 04-10, Revision 1, as referenced in the Administrative Controls of TSs, satisfies the key principles of risk-informed decision making applied to changes to TSs as delineated in RG 1.177, Revision 1, and RG 1.174, Revision 2, in that:

- The proposed change meets current regulations;
- The proposed change is consistent with defense-in-depth philosophy;
- The proposed change maintains sufficient safety margins;
- Increases in risk resulting from the proposed change are small and consistent with the Commission's Safety Goal Policy Statement; and
- The impact of the proposed change is monitored with performance measurement strategies.

Paragraph 50.36(c) of 10 CFR discusses the categories that will be included in TSs. Paragraph 50.36(c)(3) of 10 CFR discusses the specific category of SRs and states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The NRC staff finds that with the proposed relocation of surveillance frequencies to a licensee-controlled document and administratively controlled in accordance with the TS SFCP, the licensee continues to meet the requirements in 10 CFR 50.36.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official (Ms. Cynthia Becker, M.P.H., Chief of the Bureau of Radiation Control, Florida Department of Health) on June 2, 2015,[20] of the proposed issuance of the amendments. By e-mail dated June 2, 2015,[21] the State official acknowledged receipt of the notification, and the State official did not provide any comments.

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[20] The NRC staff notified the State official by telephone and by e-mail. The e-mail is in ADAMS under Accession No. ML15153B131.
[21] ADAMS Accession No. ML15195A696.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (80 FR 27199, dated May 12, 2015). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

Based on the aforementioned considerations, the NRC staff concluded that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Jonathan E. Evans
David J. Gennardo
Audrey L. Klett

Date: July 16, 2015

The NRC staff's safety evaluation of the amendments is enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Audrey L. Klett, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 263 to DPR-31
2. Amendment No. 258 to DPR-41
3. Safety Evaluation

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