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

June 13, 2014

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 OPERATING LICENSE CONDITIONS AND TECHNICAL SPECIFICATIONS (TAC NOS. MF0602 AND MF0603)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 260 to Renewed Facility Operating License No. DPR-31 and Amendment No. 255 to Renewed Facility Operating License No. DPR-41 for the Turkey Point Nuclear Generating Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Operating Licenses and the Technical Specifications (TSs) in response to your application dated September 14, 2012, as supplemented by letters dated January 29, February 14, May 30, and October 22, 2013, and March 11, 2014. Notably, the letter dated March 11, 2014, requested a reduction in scope of the license amendment request by removing the proposed change to TS Figure 3.1-2, "Boric Acid Tank Minimum Volume."

The amendments revise TSs, remove completed and satisfied license conditions, revise TS 5.5.1, "Criticality," to remove related conditions, correct inadvertent errors in the TSs, and update the references to the physical security plan in the operating licenses. The amendments also include editorial changes to the TSs. The NRC staff's safety evaluation of the amendments is enclosed.

The 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. 260 to DPR-31
2. Amendment No. 255 to DPR-41
3. Safety Evaluation

cc w/encls.: 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. 260
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 September 14, 2012, as supplemented by letters dated January 29, February 14, May 30, and October 22, 2013, and March 11, 2014, 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.

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. 260 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Farideh E. Saba

for Lisa M. Regner, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: June 13, 2014

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. 255
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 September 14, 2012, as supplemented by letters dated January 29, February 14, May 30, and October 22, 2013, and March 11, 2014, 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.

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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 255 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Farideh E. Saba

for Lisa M. Regner, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: June 13, 2014

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 260 RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 255 RENEWED FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace Pages 3, 4, 5, 6, and 7 of Renewed Facility Operating License DPR-31 with the attached Pages 3, 4, 5, and 6.

Replace Pages 3, 4, 5, 6, 7, and 8, of Renewed Facility Operating License DPR-41 with the attached Pages 3, 4, 5, and 6.

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 pages	Insert pages	Remove pages	Insert pages	Remove pages	Insert pages
i	i	3/4 1-3	3/4 1-3	3/4 2-9a	3/4 2-11
ii	ii	3/4 1-4	3/4 1-4	3/4 2-10	3/4 2-12
iii	iii	3/4 1-5	3/4 1-5	3/4 2-10a	3/4 2-13
iv	iv	3/4 1-6	3/4 1-6	3/4 2-11	3/4 2-14
v	v	3/4 1-7	3/4 1-7	3/4 2-12	------
vi	vi	3/4 1-8	3/4 1-8	3/4 2-12a	------
vii	vii	3/4 1-9	3/4 1-9	3/4 2-13	------
viii	viii	3/4 1-10	3/4 1-10	3/4 2-14	------
ix	ix	3/4 1-11	3/4 1-11	3/4 2-15	------
x	x	3/4 1-12	3/4 1-12	3/4 2-16	------
xi	xi	3/4 1-13	3/4 1-13	3/4 3-31a	3/4 3-32
xii	xii	3/4 1-14	3/4 1-14	3/4 3-32	3/4 3-33
xiii	xiii	3/4 1-14a	------	3/4 3-32a	3/4 3-34
xiv	xiv	3/4 1-16	3/4 1-16	3/4 3-33	3/4 3-35
xv	xv	3/4 1-17	3/4 1-17	3/4 3-33a	3/4 3-36
xvi	xvi	3/4 1-18	3/4 1-18	3/4 3-34	3/4 3-37
xvii	xvii	3/4 1-19	3/4 1-19	3/4 3-34a	3/4 3-38
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2-10	------	3/4 2-9	3/4 2-10	3/4 3-45	3/4 3-50

(Continued on next page.)

Remove pages	Insert pages	Remove pages	Insert pages	Remove pages	Insert pages
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3/4 4-15	3/4 4-16	3/4 7-15	3/4 7-17	5-10	------
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3/4 4-18	3/4 4-19	3/4 7-16b	3/4 7-20	5-13	------
3/4 4-19	3/4 4-20	3/4 7-17	3/4 7-21	5-14	------
3/4 4-20	3/4 4-21	3/4 7-18	3/4 7-22	5-15	------
3/4 4-21	3/4 4-22	3/4 7-18a	3/4 7-23	5-16	------
3/4 4-22	3/4 4-23	3/4 7-18b	3/4 7-24	6-3	6-3
3/4 4-23	3/4 4-24	3/4 7-19	3/4 7-25	6-4	6-4
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3/4 4-33	------	3/4 8-7	3/4 8-8	6-14	6-14
3/4 4-34	------	3/4 8-8	3/4 8-9	6-15	6-15
3/4 4-35	------	3/4 8-9	3/4 8-10	6-16	6-16
3/4 4-36	------	3/4 8-10	------	6-17	6-17
3/4 4-37	------	3/4 9-6	3/4 9-6	6-18	6-18
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3/4 6-7	3/4 6-7	3/4 9-13	3/4 9-13	6-21a	------
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3/4 6-9	3/4 6-9	3/4 9-15	------	6-22a	------
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3/4 6-15	3/4 6-15	3/4 10-4	3/4 10-4	6-25	------
3/4 6-16	3/4 6-16	3/4 10-5	------	6-26	------
3/4 6-17	3/4 6-17			6-27	------

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. 260 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

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

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

D. Fire Protection

FPL shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report (UFSAR) for Turkey Point Units 3 and 4 and as approved in the Safety Evaluation Report (SER) dated March 21, 1979 and supplemented by NRC letters dated April 3, 1980, July 9, 1980, December 8, 1980, January 26, 1981, May 10, 1982, March 27, 1984, April 16, 1984, August 12, 1987, and by Safety Evaluations dated February 25, 1994, February 24, 1998, October 8, 1998, December 22, 1998, May 4, 1999, and May 5, 1999, subject to the following provision:

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

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provision of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Florida Power and Light Turkey Point Nuclear Plant Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program - Revision 15" submitted by letter dated August 3, 2012.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Turkey Point Nuclear Generating Station CSP was approved by License Amendment No. 245 as supplemented by a change approved by Amendment No. 256.

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

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

G. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

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

(b) Operations to mitigate fuel damage considering the following

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

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

H. PAD TCD Safety Analyses

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

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: June 6, 2002

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. 255 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.

D. Fire Protection

FPL shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report (UFSAR) for Turkey Point Units 3 and 4 and as approved in the Safety Evaluation Report (SER) dated March 21, 1979 and supplemented by NRC letters dated April 3, 1980, July 9, 1980, December 8, 1980, January 26, 1981, May 10, 1982, March 27, 1984, April 16, 1984, August 12, 1987, and by Safety Evaluations dated February 25, 1994, February 24, 1998, October 8, 1998, December 22, 1998, May 4, 1999, and May 5, 1999, subject to the following provision:

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

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provision of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Florida Power and Light Turkey Point Nuclear Plant Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program - Revision 15" submitted by letter dated August 3, 2012.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Turkey Point Nuclear Generating Station CSP was approved by License Amendment No. 241 as supplemented by a change approved by Amendment No. 252.

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

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

G. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
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5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following
1. Protection and use of personnel assets
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3. Minimizing fire spread
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(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

H. PAD TCD Safety Analyses

1. PAD 4.0 TCD has been specifically approved for use for the Turkey Point licensing basis analyses. Upon NRC's approval of a revised generic version of PAD that accounts for Thermal Conductivity Degradation (TCD), FPL will within six months:
   a. Demonstrate that PAD 4.0 TCD remains conservatively bounding in licensing basis analyses when compared to the new generically approved version of PAD w/TCD, or
   b. Provide a schedule for the re-analysis using the new generically approved version of PAD w/TCD for any of the affected licensing basis analyses.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight April 10, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachments:
Appendix A – Technical Specifications for Unit 4
Appendix B – Environmental Protection Plan

Date of Issuance: June 6, 2002

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ADMINISTRATIVE CONTROLS

DOSE EQUIVALENT I-131

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

DOSE EQUIVALENT XE - 133

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

FREQUENCY NOTATION

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

GAS DECAY TANK SYSTEM

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

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.27 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTING

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

Action:

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

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

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

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

* Loop design flow = 86,900 gpm

** RTP = Rated Thermal Power

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

** RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

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

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

Where: ΔT	=	Measured ΔT by RTD Instrumentation
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead/Lag compensator on measured ΔT; τ_1 = [*]s, τ_2 = [*]s
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT; τ_3 = [*]s
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
K_1	=	[*];
K_2	=	[*]/°F;
$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4 , τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = [*]s, τ_5 = [*]s;
T	=	Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg}; τ_6 = [*]s
T'	≤	[*]°F (Indicated Loop T_{avg} at RATED THERMAL POWER);
K_3	=	[*]/psi;
P	=	Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

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

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

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

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

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

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

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

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

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

Where:

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

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

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

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. At least once per 24 hours by consideration of the following factors:

    1) Reactor Coolant System boron concentration,

    2) Control rod position,

    3) Reactor Coolant System average temperature,

    4) Fuel burnup based on gross thermal energy generation,

    5) Xenon concentration, and

    6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

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

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

** See Special Test Exceptions Specification 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

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

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

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

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2* **.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

** See Special Test Exceptions Specification 3.10.3.

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. At least once per 7 days 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. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

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

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

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

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

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days 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. At least once per 31 days 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. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. and c. delivers at least 16 gpm to the RCS.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least two charging pumps shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUI REMENTS

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

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.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the indicated borated water volume, and

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.5 The following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

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

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

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

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

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

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

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

4) A maximum solution temperature of 100°F.

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

ACTION:

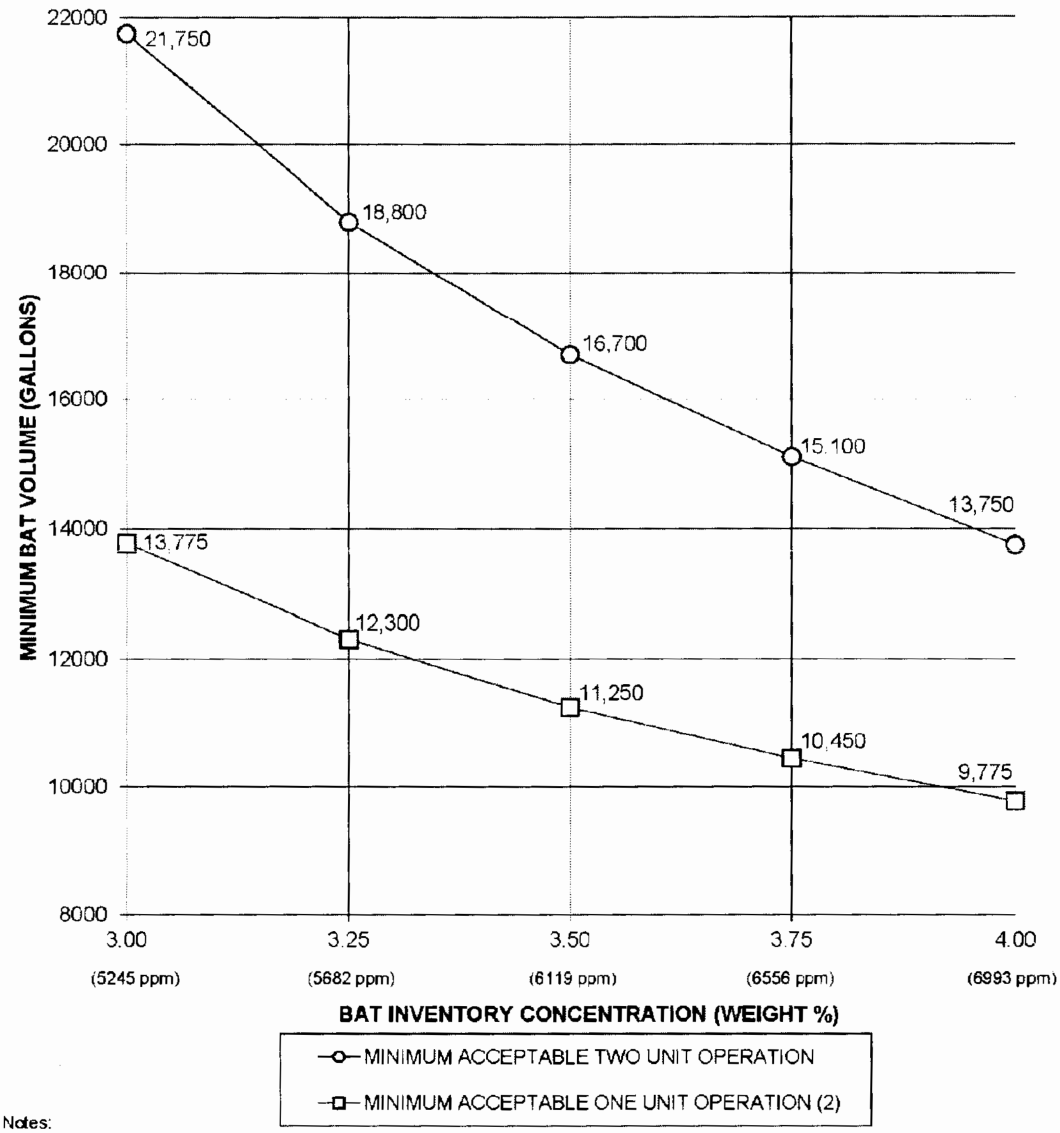
a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 70 hours or be in at least HOT STANDBY within the next 8 hours* and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN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

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

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

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

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



Notes:

(1) Combined volume of all available boric acid tanks assuming RWST boron concentration between 2400 ppm and 2600 ppm.
(2) Includes 2900 gallons for the shutdown unit.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1* and 2*

ACTION:

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

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

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

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

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

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

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

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

---

* See Special Test Exceptions 3.10.2 and 3.10.3.

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

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

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the Allowed Rod Misalignment of the group step counter demand position at least once per 12 hours (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 at least once per 92 days.

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

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

Single Rod Cluster Control Assembly Withdrawal at Full Power

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

Major Secondary Coolant System Pipe Rupture

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

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

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

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

b. Group demand counters; ± 2 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued):

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

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

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

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

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

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

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

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

APPLICABILITY: MODE 1

ACTION:

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

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

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

POWER DISTRIBUTION LIMITS

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. At least once per 31 Effective Full Power Days.

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:

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

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

---

* 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. At least once every 24 hours, and

2. Immediately following and as a minimum at 2, 4 and 8 hours following the events listed below and every 24 hours thereafter.

1) Raising the thermal power above P_T, or

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

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

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

---

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

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

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 at least once per 31 effective full power days.

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,

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.

b. Base Load operation is permitted provided:

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

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

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

P_{BL} and P_T are defined as:

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

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

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

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

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

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

4.2.2.4 RADIAL BURNDOWN

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

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

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

   $[F_Q^L(Z)]$ is equal to $[F_Q^L]$ x K(Z).

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

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

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

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

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

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours either:

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

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

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

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

1. A nominal 50% of RATED THERMAL POWER,

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

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

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

b. At least once per 31 Effective Full Power Days.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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

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

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

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

   2. Within 2 hours either:

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

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

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

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

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*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

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

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

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

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

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

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

d. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at least once per 7 days 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 at least once per 12 hours.

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.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 RCS flow rate shall be monitored for degradation at least once per 12 hours.

4.2.5.3 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.4 After each fuel loading, and at least once per 18 months, 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 4.3-2

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.	R	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3(3)
c. Containment Pressure--High	N.A.	R	N.A.	N.A.	M(1)	1, 2, 3
d. Pressurizer Pressure--Low	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
e. High Differential Pressure Between the Steam Line Header and any Steam Line	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
f. Steam Line Flow--High Coincident with:	S	R(a)(b)	Q(5)(a)(b)	N.A	N.A	1, 2, 3(3)
Steam Generator Pressure--Low	S	R(a)(b)	Q(5)(a)(b)	N.A.	N.A.	1, 2, 3(3)
or						
Tavg--Low	S	R	Q(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.	M(1)	1, 2, 3, 4
b. Containment Pressure--High-High	N.A.	R	N.A.	R	M(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
3. Containment Isolation						
a. Phase "A" Isolation						
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(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.	R	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(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.	R	N.A.	R	M(1)	1, 2, 3
with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
c. Containment Ventilation Isolation						
1) Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	R	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	S	R	M	N.A.	N.A.	1, 2, 3, 4
4. Steam Line Isolation						
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(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.	R	N.A.	R	M(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R	N.A	R	M(1)	1, 2, 3
d. Steam Line Flow--High	S(3)	R[(a)(b)]	Q(5)[(a)(b)]	N.A.	N.A.	1, 2, 3
Coincident with: Steam Generator Pressure--Low	S(3)	R[(a)(b)]	Q(5)[(a)(b)]	N.A.	N.A.	1, 2, 3
or Tavg--Low	S(3)	R	Q(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.	R	1, 2, 3
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Steam Generator Water Level--High-High	S	R[(a)(b)]	Q[(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.	R	1, 2, 3
b. Steam Generator Water Level--Low-Low	S	R[(a)(b)]	Q[(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.	R	N.A.	R	N.A.	1, 2, 3
e. Trip of All Main Feedwater Pump Breakers.	N.A.	N.A.	N.A.	R	N.A.	1. 2
7. Loss of Power						
a. 4.16 kV busses A and B (Loss of Voltage)	N.A.	R	N.A.	R	N.A.	1, 2, 3, 4
b. 480V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	S	R	N.A.	M(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	S	R	N.A.	M(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.	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
b. Tavg--Low	N.A.	R	Q(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	S	R	M	N.A.	N.A.	(4)
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
e. Control Room Air Intake Radiation Level	S	R	M	N.A.	N.A.	All

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

TABLE NOTATIONS

At least once per 18 months 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 at least every 62 days on a STAGGERED TEST BASIS.

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

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

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

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

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

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

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

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

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

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

** With irradiated fuel in the spent fuel pits.

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

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

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

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

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

ACTION STATEMENTS

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

1) A Containment sump level monitoring system is OPERABLE,

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

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

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

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

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

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

TABLE 3.3-4 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - In MODES 5 or 6 (expect 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	S	R	Q	All
b. RCS Leakage Detection				
1) Particulate Radio-activity	S	R	Q	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	Q	1, 2, 3, 4
2. Spent Fuel Pool Areas				
a. Unit 3 Radioactivity--High Gaseous	S	R	Q	*
b. Unit 4 (Plant Vent) Radioactivity--High Gaseous# (SPING and PRMS)	S	R	Q	*

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 at least once per 24 hours 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.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

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

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

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

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

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

TABLE NOTATIONS

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

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

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

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

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

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

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

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

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

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

ACTION 35 DELETED

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

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

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

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

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

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

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

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

TABLE 4.3-4

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

INSTRUMENTATION

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-8

ACTION:

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

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-8
EXPLOSIVE GAS MONITORING INSTRUMENTATION

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

TABLE NOTATION

* During GAS DECAY TANK SYSTEM operation.

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

TABLE 4.3-6

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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.

SURVEILLANCE REQUIREMENTS

4.4.4 Each block valve shall be demonstrated OPERABLE at least once per 92 days 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.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION*:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

*Separate Action entry is allowed for each SG tube.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

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

b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

1) A Containment Sump Level Monitoring System is OPERABLE;

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

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

b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

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

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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATING

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

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

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

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the containment atmosphere gaseous or particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump level at least once per 12 hours.

c.** Performance of a Reactor Coolant System water inventory balance at least once per 72*** hours; and

d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours; and

e. Verifying primary-to-secondary leakage is ≤ 150 gallons per day through any one SG at least once per 72*** hours.

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. At least once per 18 months.

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.

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

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

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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION (Continued)

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

1. Within 24 hours by verifying valve closure, and

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

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

ACCEPTABLE LEAKAGE LIMITS

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

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

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

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

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and 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 prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
Chloride**	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
Fluoride**	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

* Limit not applicable with average reactor coolant temperature less than or equal to 250°F.

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

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least 5 times per week not to exceed 72 hours between samples
Chloride**	At least 5 times per week not to exceed 72 hours between samples
Fluoride**	At least 5 times per week not to exceed 72 hours between samples

* 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 performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. NOT USED		
2. Tritium Activity Determination	1 per 7 days.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131	a) 1 per 14 days. 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	Monthly	1, 2, 3, 4
5. Isotopic Analysis for DOSE EQUIVALENT XE-133	1 per 7 days	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 at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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

MATERIAL PROPERTY BASIS

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

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

FPL/FLA 48 EFPY HEATUP CURVES

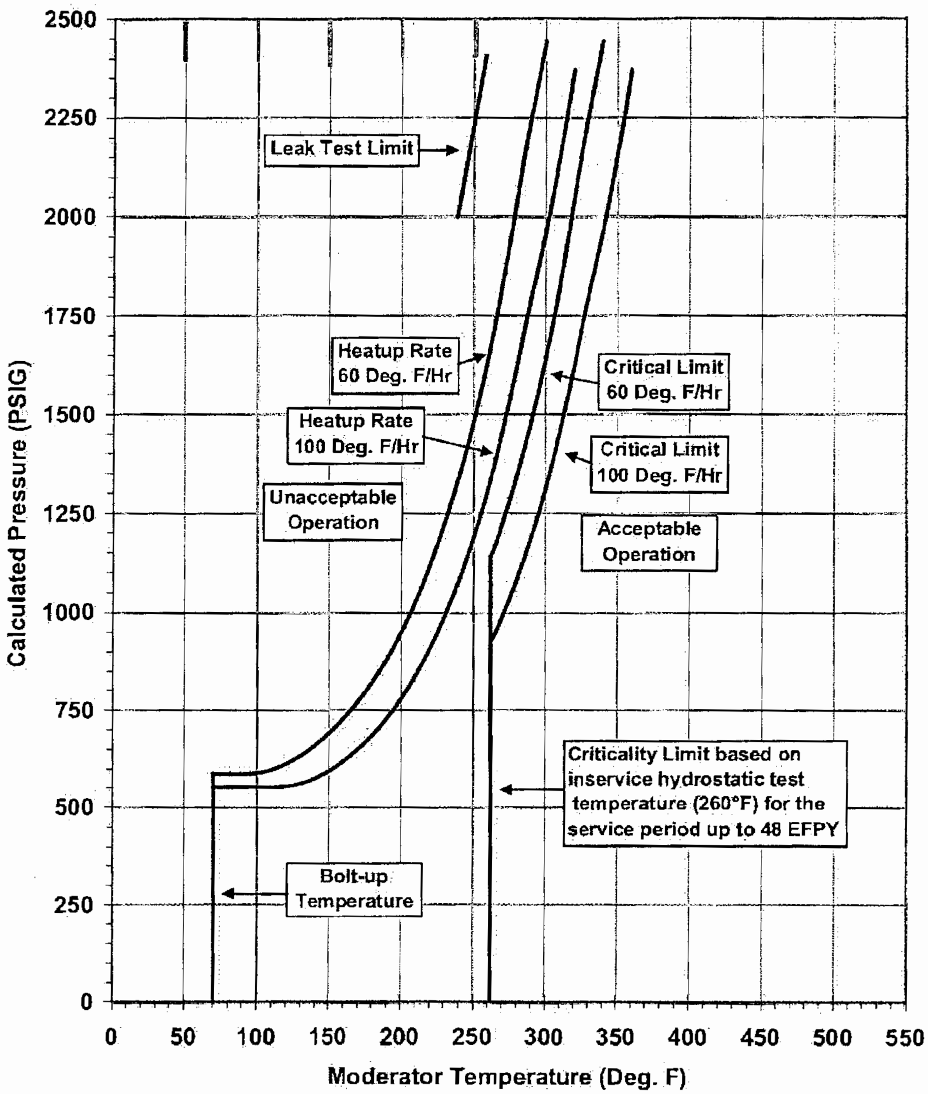

FIGURE 3.4-2 Turkey Point Units 3 & 4 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 48 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

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

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

FPL 48 EFPY COOLDOWN CURVES

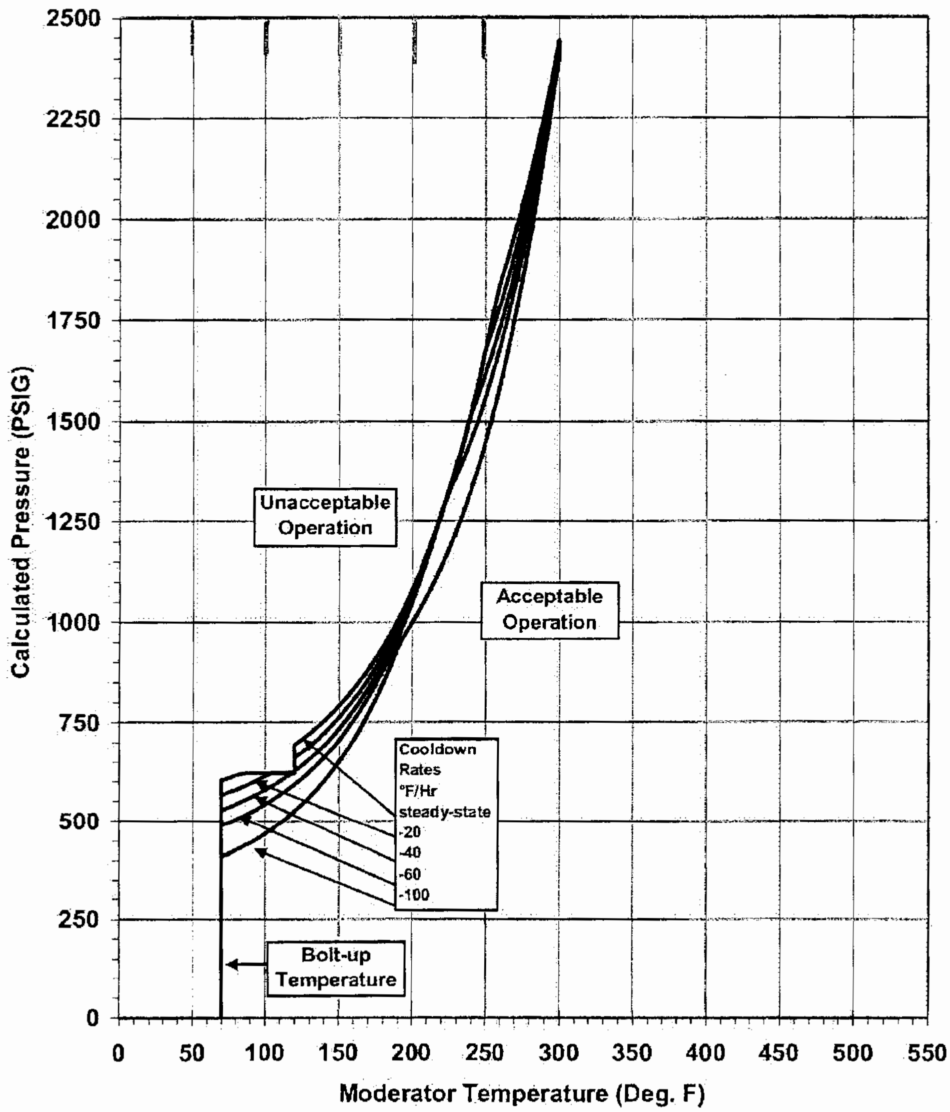

FIGURE 3.4-3 Turkey Point Units 3 & 4 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for 48 EFPY (Without Margins for Instrumentation Errors)

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

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

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

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

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

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

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

f. The provisions of Specification 3.0.4 are not applicable.

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 at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV block valve is open at least once per 72 hours 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 at least once per 24 hours.*

4.4.9.3.2 The 2.20 square inch vent shall be verified to be open at least once per 12 hours** 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 at least once per 24 hours 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 at least once per 31 days.

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 at least once per 18 months 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.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

a. With one containment air lock door inoperable:

   1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

   2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

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

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS

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. At least once per 24 months 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 at least once per 12 hours.

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 at least once per 24 hours:

Approximate Location

a.	0° Azimuth	58 feet elevation
b.	120° Azimuth	58 feet elevation
c.	240° Azimuth	58 feet elevation

* Equivalent hours are determined from actual hours using the time-temperature relationships that support the environmental qualification requirements of 10 CFR 50.49.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY MODES 1, 2, 3, and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between 90% and 95% of the predicted force, or with one tendon below 90% of the predicted force, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the average of all measured tendon forces for each type of tendon (dome, vertical, and hoop), including those measured in ACTION a., less than the predicted force, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any abnormal degradation of the structural integrity other than ACTION a. and ACTION b., at a level below the acceptance criteria of Specifications 4.6.1.6.1, 4.6.1.6.2 and 4.6.1.6.3, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQURECTS

4.6.1 .6.1 Containment Tendons. The containment tendons and the containment exterior surfaces shall be examined in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants," and the modifications presented in 10 CFR 50.55a(b)(2)(viii), "Examination of concrete containments," as modified by approved exemptions. The containment structural integrity shall be demonstrated during the inspection periods specified in IWL-2410 and IWL-2420. The tendons' structural integrity shall be demonstrated by:

a. Determining that tendons, selected in accordance with IWL-2521, have the average of all measured tendon forces for each type of tendon (dome, vertical and hoop) equal to or greater than the minimum required prestress specified at the anchorage for that type of tendon.

b. Assuring that the measured force in each individual tendon is not less than 95% of the predicted force unless the following conditions are satisfied:

   1) The measured force in no tendon is below 90% of the predicted force and the measured force in no more than one tendon is between 90% and 95% of the predicted force;
   2) The measured force in two tendons located adjacent to the tendon in 1) are not less than 95% of the predicted forces; and
   3) The measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

   The predicted force for each tendon shall be calculated individually for each inspection prior to the beginning of each inspection, and should consider such factors as:

   - Prestressing history;
   - Friction losses; and
   - Time-dependent losses (creep, shrinkage, relaxation), considering time elapsed from prestressing.

   When evaluation of consecutive surveillances of prestressing forces for the same tendon or tendons in a group indicates a trend of prestress loss such that the tendon force(s) would be less than the minimum design prestress requirements before the next inspection interval, an evaluation shall be performed and reported in the Engineering Evaluation Report as prescribed in IWL-3300.

c. Performing tendon detensioning, examinations, and testing on a sample tendon of each type (dome, vertical, and hoop). A single wire or strand shall be removed from each detensioned tendon. Each removed wire or strand shall be examined over its entire length for corrosion and mechanical damage. Tension tests shall be performed on each removed wire or strand: one at each end, one at mid-length, and one in the location of the most corroded area, if any. The following information shall be obtained from each test:

   1) Yield strength;
   2) Ultimate tensile strength;
   3) Elongation.

The condition of wire or strand is acceptable if:

1) Samples are free of physical damage;
2) Sample ultimate tensile strength and elongation are not less than minimum specified values.

d. Performing tendon retensioning of those tendons that have been detensioned to at least the force predicted for the tendon at the time of the test. However, the retensioning force shall not exceed 70% of the specified minimum ultimate tensile strength of the tendon based on the number of effective wires or strands in the tendon at the time of retensioning. During retensioning of these tendons, if the elongation corresponding to a specific load (adjusted for effective wires or strands) differs by more than 10% from that recorded during the last measurement, an evaluation must be performed to determine whether the difference is related to wire failures or slip of wires in anchorage. A difference of more than 10% must be identified in the ISI Summary Report required by IWA-6000.

e. Performing examination of corrosion protection medium and free water in accordance with IWL-2525, with acceptance standards prescribed in IWL-3221.4. The following conditions, if they occur, shall be reported in the ISI Summary Report required by IWA-6000:

1) The sheathing filler grease contains chemically combined water exceeding 10% by weight or the presence of free water;
2) The absolute difference between the amount removed and the amount replaced exceeds 10% of the tendon net duct volume.
3) Grease leakage is detected during general visual examination of the containment surface.

4.6.1.6.2 End Anchorages and Containment Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the containment concrete surfaces shall be demonstrated by performing examination of tendon anchorage areas and containment concrete surfaces in accordance with IWL-2000, with acceptance standards prescribed in IWL-3000. Acceptability of inaccessible areas shall be evaluated when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the following shall be provided in the ISI Summary Report required by IWA-6000:

1) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
2) An evaluation of each area, and the result of the evaluation; and
3) A description of necessary corrective actions.

4.6.1.6.3. Containment Surfaces Inspection for Containment Leakage Rate Testing Program. In accordance with the Containment Leakage Rate Testing Program, a visual inspection of the accessible interior and exterior surfaces of the containment, including the liner plate, shall be performed. The purpose of this inspection shall be to identify any evidence of structural deterioration which may affect containment structural integrity or leaktightness. The visual inspection shall be general in nature; its intent shall be to detect gross areas of widespread cracking, spalling, gouging, rust, weld degradation, or grease leakage. The visual examination may include the utilization of binoculars or other optical devices. Corrective actions taken, and recording of structural deterioration and corrective actions, shall be in accordance with the Containment Leakage Rate Testing Program. Records of previous inspections shall be reviewed to verify no apparent changes in appearance. The first inspection performed will form the baseline for future surveillances.

CONTAINMENT SYSTEMS

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. At least once per 18 months by:

   1. Verifying that the buffering agent baskets are in place and intact;

   2. Collectively contain $\geq$ 7500 pounds (154 cubic feet) of sodium tetraborate decahydrate, or equivalent.

3/4.6.3 DELETED

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE with isolation times less than or equal to required isolation times.

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

ACTION:

*With one or more isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic containment isolation valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

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

SURVEILLANCE REQUIREMENTS

4.6.4.1 The isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months 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

PLANT SYSTEMS

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. At least once every 31 days, by testing with the associated standby steam generator feedwater pump in recirculation.

b. At least once per 18 months, 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. At least every 18 months 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.

*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. At least once per 12 hours, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

b. At least once per 31 days by: (1) 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 (2) verifying by a performance test the heat exchanger surveillance curves.*

c. At least once per 18 months 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. At least once per 31 days 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. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Intake Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for system operability are OPERABLE.

PLANT SYSTEMS

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 The ultimate heat sink shall be OPERABLE with an average supply water temperature less than or equal to 100°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 at least once per 24 hours by verifying the average supply water temperature* to be within its limit.

*Portable monitors may be used to measure the temperature.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5 The Control Room Emergency Ventilation System shall be OPERABLE* with:

a. Three air handling units,

b. Two of three condensing units,

c. Two control room recirculation fans,

d. Two recirculation dampers,

e. One filter train,

f. Two isolation dampers in the normal outside air intake duct,

g. Two isolation dampers in the emergency outside air intake duct,

h. Two isolation dampers in the kitchen area exhaust duct, and

i. Two isolation dampers in the toilet area exhaust duct.

j. Control Room Envelope.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

a.1 With one air handling unit inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable air handling unit to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.2 With two condensing units inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore at least one of the inoperable condensing units to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.3 With one recirculation fan inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable fan to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

*The Control Room Envelope (CRE) boundary may be opened intermittently under administrative control.

LIMITING CONDITION FOR OPERATION (continued)

a.4 With one recirculation damper inoperable, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the recirculation dampers in the open position and place the system in recirculation mode** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.5 With the filter train inoperable, e.g., an inoperable filter, and/or two inoperable recirculation fans, and/or two inoperable recirculation dampers, immediately suspend all movement of irradiated fuel, and, immediately, initiate action to implement mitigating actions, [e.g., use of the compensatory filtration unit is required to be immediately initiated], and, within 24 hours, verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits and, within 7 days, restore the filter train to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.6 With an inoperable damper in the normal outside air intake, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the normal outside air intake isolation dampers in the closed position and place the system in recirculation mode** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.7 With an inoperable damper in the emergency outside air intake, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or place and maintain at least one of the emergency outside air intake isolation dampers in the open position** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.8 With an isolation damper inoperable in the kitchen area exhaust duct, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or isolate the flow path** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

a.9 With an isolation damper inoperable in the toilet area exhaust duct, immediately suspend all movement of irradiated fuel and, within 7 days, restore the inoperable damper to OPERABLE status or isolate the flow path** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

**If action is taken such that indefinite operation is permitted and the system is placed in recirculation mode, then movement of irradiated fuel may resume.

LIMITING CONDITION FOR OPERATION (continued)

b. With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary, immediately suspend all movement of irradiated fuel in the spent fuel pool, and immediately initiate action to implement mitigating actions, and within 24 hours, verify mitigating actions ensure CRE occupant radiological and chemical hazards will not exceed limits, and CRE occupants are protected from smoke hazards, and restore CRE boundary to OPERABLE status within 90 days, or:

   1) With the requirements not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   2) If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

c. With the Control Room Emergency Ventilation System inoperable**++**, immediately suspend all operations involving CORE ALTERATIONS, movement of irradiated fuel in the spent fuel pool, or positive reactivity changes. This ACTION shall apply to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The Control Room Emergency Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F;

b. At least once per 31 days 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. At least once per 18 months 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 At least once per 12 months 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 On staggered test basis every 36 months, test the supply fans (trains A and B) and measure CRE pressure relative to external areas adjacent to the CRE boundary.***

e. At least once per 18 months by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation,

f. At least once per 18 months 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.

PLANT SYSTEMS

3/4.7.6 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.6 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.6.f on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-2 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment 151 and 146.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual

TABLE 4.7-2

SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extended Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less that the number in Column A, the next inspection interval may be twice the previous interval but not greater that 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.6e. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Functional Tests

For each unit during refueling shutdown, a representative sample of snubbers shall be tested using the following sample plan:

1) At least 10% of the total number of safety related snubbers for the respective unit identified by site records shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.6e, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested;

2) The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following categories;

   A. Snubbers within 5 feet of heavy equipment (ex. valves, pumps, turbines, motors, etc.)

   B. Snubbers within 10 feet of the discharge from a safety relief valve.

3) Snubbers identified by site records as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

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*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber OPERABILITY for all design conditions at either the completion of their fabrication or at a subsequent date.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved with the specified range of velocity or acceleration in both tension and compression;

2) Snubber release rate, where required, is within the specified range in tension and compression,

3) The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

f. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated under the provisions of 10 CFR Part 21.

Should the results of the evaluation indicate that the failure was caused by either manufacturer or design deficiency, further action shall be taken, if needed, based on manufacturer or engineering recommendations.

For the snubber(s) found inoperable, an evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this evaluation shall be to determine if the components to which the inoperable snubber(s) are attached were adversely affected by the inoperability of the snubber(s) in order to ensure that the component remains capable of meeting the designed service.

SURVEILLANCE REQUIREMENTS (Continued)

g. Snubber Service Life Monitoring Program

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained.

Concurrent with the first inservice visual inspection and during refueling shutdown thereafter, the installation and maintenance records for each safety related snubber as identified by site records shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

PLANT SYSTEMS

3/4.7.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.7 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.7.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.7.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

4.7.7.4 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

PLANT SYSTEMS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.7.8 The concentration of oxygen in the GAS DECAY TANK SYSTEM (as measured in the inservice gas decay tank) shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the inservice gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the inservice gas decay tank greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the gas decay tanks and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8 The concentrations of hydrogen and oxygen in the inservice gas decay tanks shall be determined to be within the above limits by continuously* monitoring the waste gases in the inservice gas decay tank with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-8 of Specification 3.3.3.6.

*When continuous monitoring capability is inoperable, Table 3.3-8 allows the use of grab samples.

PLANT SYSTEMS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.7.9 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases (DOSE EQUIVALENT Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank and the Reactor Coolant System total activity exceeds the limit of Specification 3.4.8.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required startup transformers and their associated circuits shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months while shutting down, by transferring manually unit power supply from the auxiliary transformer to the startup transformer.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE*:

a. At least once per 31 days on a STAGGERED TEST BASIS 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. Once per 184 days, these conditions shall be reached within 15 seconds after the start signal fro m 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 busses.

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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 at least once per 92 days that a fuel transfer pump starts automatically and transfers fuel from the storage system to the day tank,

c. At least once per 31 days 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. At least once per 31 days 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. At least once per 18 months, during shutdown (applicable to only the two diesel generators associated with the unit):

1) Deleted

2)* Verifying the generator capability to reject a load of greater than or equal to 380 kw while maintaining voltage at 3950-4350 volts and frequency at 60 ± 0.6 Hz;

3)* Verifying the generator capability to reject a load of greater than or equal to 2500 kW (Unit 3), 2874 kW (Unit 4) without tripping. The generator voltage shall return to less than or equal to 4784 volts within 2 seconds following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b. Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently

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* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

connected loads within 15 seconds, energizes the auto–connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the auto-connected shutdown loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at 3950-4350 volts and 60 ± 0.6 Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently connected loads within 15 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at 3950-4350 volts and 60 ± 0.6 Hz during this test; and

c) Verifying that diesel generator trips that are made operable during the test mode of diesel operation are inoperable.

7)* # Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 2550-2750 kW (Unit 3), 2950-3150 kW (Unit 4)** and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2300-2500 kW (Unit 3), 2650-2850 kW (Unit 4)**. The generator voltage and frequency shall be 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds after the start signal; the steady-state generator voltage and frequency

* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

** Momentary transients outside these load bands do not invalidate this test.

This test may be performed during POWER OPERATION

shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, verify the diesel starts and accelerates to reach a generator voltage and frequency of 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds after the start signal.**

8) Verifying that the auto-connected loads to each diesel generator do not exceed 2500 kW (Unit 3), 2874 kW (Unit 4);

9) Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its standby status.

10) Verifying that the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from the fuel storage tank (Unit 3), fuel storage tanks (Unit 4) to the day tanks of each diesel associated with the unit via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

13) Verifying that the diesel generator lockout relay prevents the diesel generator from starting;

** If verification of the diesel's ability to restart and accelerate to a generator voltage and frequency of 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds following the 24 hour operation test of Specification 4.8.1.1.2.g.7) is not satisfactorily completed, it is not necessary to repeat the 24-hour test. Instead, the diesel generator may be operated between 2300-2500 kW Unit 3, 2650-2850 kW (Unit 4) for 2 hours or until operating temperature has stabilized (whichever is greater). Following the 2 hours/operating temperature stabilization run, the EDG is to be secured and restarted within 5 minutes to confirm its ability to achieve the required voltage and frequency within 15 seconds.

h. At least once per 10 years 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. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.*

2) For Unit 4 only, 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.1 for the Unit 3 storage tank while Unit 3 is in Modes 5, 6, or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days, Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1) A minimum capacity of 2750 pounds, and

   2) An overload cutoff limit less than or equal to 2700 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1) A minimum capacity of 610 pounds, and

   2) A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 At least once each refueling, each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 2750 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2700 pounds.

4.9.6.2 At least once each refueling, each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 610 pounds.

3/4.9.7 DELETED

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

4.9.8.1.2 The RHR flow indicator shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

*The required RHR loop may be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 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 at least once per 12 hours.

* 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 REQUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that Containment Ventilation Isolation occurs on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

REFUELING OPERATIONS

3/4.9.10 REFUELING CAVITY WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.10 Refueling cavity water level shall be maintained $\geq$ 23 feet above the top of the reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTION:

With the refueling cavity water level not within limit, suspend movement of irradiated fuel assemblies within containment immediately.

SURVEILLANCE REQUIREMENTS

4.9.10 Verify refueling cavity water level is $\geq$ 23 feet above the top of the reactor vessel flange within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of irradiated fuel assemblies within containment.

REFUELING OPERATIONS

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 at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

*The requirements of this specification may be suspended for more than 4 hours to perform maintenance provided a 10 CFR 50.59 evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

3/4.9.12 DELETED

3/4.9.13 RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.9.13 The Containment Radiation monitors which initiate containment and control room ventilation isolation shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a) With one or both radiation monitors inoperable, operation may continue provided the containment ventilation isolation valves are maintained closed.

b) With one or both radiation monitors inoperable, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

SURVEILLANCE REQUIREMENTS

4.9.13 Each Containment Radiation monitor shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-3.

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. The minimum boron concentration in the Spent Fuel Pit shall be 2300 ppm.

b. The combination of initial enrichment, burnup, and cooling time of each fuel assembly stored in the Spent Fuel Pit shall be in accordance with Specification 5.5.1.

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With boron concentration in the Spent Fuel Pit less than 2300 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 2300 ppm or greater.

b. With condition b not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14.1 The boron concentration of the Spent Fuel Pit shall be verified to be 2300 ppm or greater at least once per month.

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.

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 at least once per hour 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 at least once per 30 minutes 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 at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Analog Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

5.0 DESIGN FEATURES

5.1 SITE

5.1.1 The site is approximately 25 miles south of Miami, 8 miles east of Florida City, and 9 miles southeast of Homestead, Florida

5.2 DELETED

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4, ZIRLO®, or Optimized ZIRLO™ except that replacement of fuel rods by filler rods consisting of stainless steel, or by vacant rod positions, may be made in fuel assemblies if justified by cycle-specific reload analysis using NRC-approved methodology. The reactor core contains approximately 71 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. Each fuel rod shall have a nominal active fuel length of 144 inches. Should more than 30 individual rods in the core, or 10 fuel rods in any fuel assembly, be replaced per refueling, a Special Report discussing the rod replacements shall be submitted to the Commission within 30 days after cycle startup.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be silver, indium, and cadmium. All control rods shall be clad with stainless steel tubing.

5.4 DELETED

5.5 FUEL STORAGE

5.5.1 CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} less than 1.0 when flooded with unborated water, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

b. A k_{eff} less than or equal to 0.95 when flooded with water borated to 500 ppm, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

c. A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for the two region spent fuel pool storage racks. A nominal 10.1 inch center-to-center distance in the east-west direction and a nominal 10.7 inch center-to-center distance in the north-south direction for the cask area storage rack.

d. A maximum enrichment loading for fuel assemblies of 5.0 weight percent of U-235.

e. No restriction on storage of fresh or irradiated fuel assemblies in the cask area storage rack.

f. Fresh or irradiated fuel assemblies not stored in the cask area storage rack shall be stored in accordance with Specification 5.5.1.3.

g. The Metamic neutron absorber inserts shall have a minimum certified ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

5.5.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than a nominal 4.5 weight percent of U-235 if the assembly contains no burnable absorber rods and no more than 5.0 weight percent of U-235 if the assembly contains at least 16 IFBA rods.

5.5.1.3 Credit for burnup and cooling time is taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. Fresh or irradiated fuel assemblies in the Region I or Region II racks shall be stored in compliance with the following:

a. Any 2x2 array of Region I storage cells containing fuel shall comply with the storage patterns in Figure 5.5-1 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array.

b. Any 2x2 array of Region II storage cells containing fuel shall:

i. Comply with the storage patterns in Figure 5.5-2 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array,

ii. Have the same directional orientation for Metamic inserts in a contiguous group of 2x2 arrays where Metamic inserts are required, and

iii. Comply with the requirements of 5.5.1.3.c for cells adjacent to Region I racks.

c. Any 2x2 array of Region II storage cells that interface with Region I storage cells shall comply with the rules of Figure 5.5-3.

d. Any fuel assembly may be replaced with a fuel rod storage basket or non-fuel hardware.

e. Storage of Metamic inserts or RCCAs is acceptable in locations designated as empty (water-filled) cells.

DRAINAGE

5.5.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1535 fuel assemblies.

Table 5.5-1

<u>Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)</u>

See notes 1-6 for use of Table 5.5-1

Coeff.	Fuel Category						
	I-3	**I-4**	**II-1**	**II-2**	**II-3**	**II-4**	**II-5**
A1	5.66439153	-14.7363682	-7.74060457	-7.63345029	24.4656526	8.5452608	26.2860949
A2	-7.22610116	11.0284547	5.13978237	10.7798957	-20.3141124	-4.47257395	-18.0738662
A3	2.98646188	-1.80672781	-0.360186309	-2.81231555	6.53101471	2.09078914	5.8330891
A4	-0.287945644	0.119516492	0.0021681285	0.29284474	-0.581826027	-0.188280562	-0.517434342
A5	-0.558098618	0.0620559676	-0.0304713673	0.0795058096	-0.16567492	0.157548739	-0.0614152031
A6	0.476169245	0.0236575787	0.098844889	-0.0676341983	0.243843226	-0.0593584027	0.134626308
A7	-0.117591963	-0.0088144551	-0.0277584786	0.0335130877	-0.0712130368	0.0154678626	-0.0383060399
A8	0.0095165354	0.0008957348	0.0024057185	-0.0040803875	0.0063998706	-0.0014068318	0.0033419846
A9	-47.1782783	-20.2890089	-21.424984	14.6716317	-41.1150	-0.881964768	-12.1780
A10	33.4270029	14.7485847	16.255208	-10.0312224	43.9149156	9.69128392	23.6179517
A11	-6.11257501	-1.22889103	-1.77941882	5.62580894	-9.6599923	-0.18740168	-4.10815592
A12	0.490064351	0.0807808548	0.127321203	-0.539361868	0.836931842	0.0123398618	0.363908736

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [- (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal central zone U-235 enrichment. Axial blanket material is not considered when determining enrichment. Any enrichment between 2.0 and 5.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 0 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. DELETED

5. DELETED

6. This Table applies for any blanketed fuel assembly.

Table 5.5-2

Non-Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-4 for use of Table 5.5-2

Coeff.	Fuel Category						
	I-3	**I-4**	**II-1**	**II-2**	**II-3**	**II-4**	**II-5**
A1	2.04088171	-27.6637884	-11.2686777	20.7284208	29.8862876	-83.5409405	35.5058622
A2	-4.83684164	26.1997193	2.0659501	11.9673275	-37.0771132	94.7973724	-30.1986997
A3	2.59801889	-7.2982252	2.66204924	-14.4072388	16.3986049	-31.9583373	11.0102438
A4	-0.300597247	0.723731768	-0.513334362	2.83623963	-2.1571669	3.55898487	-1.27269125
A5	-0.610041808	0.401332891	-0.0987986108	-1.49118695	1.02330848	0.299948492	1.34723758
A6	0.640497159	-0.418616707	-0.0724198633	1.75361041	-1.21889631	-0.312341996	-1.19871392
A7	-0.219000712	0.144304039	0.106248806	-0.659046438	0.467440882	0.107463895	0.352920811
A8	0.0252870451	-0.0154239536	-0.0197359109	0.080884618	-0.0560129443	-0.0108814287	-0.0325155213
A9	-4.48207836	-5.54507376	-1.34620551	-245.825283	12.1549	39.4975573	-5.2576
A10	-2.12118634	-5.76555416	-10.1728821	243.59979	-22.7755385	-50.5818253	10.1733379
A11	2.91619317	6.29118025	8.71968815	-75.7805818	14.3755458	23.3093829	0.369083041
A12	-0.196645176	-0.732079719	-1.14461356	8.10936356	-1.80803352	-2.69466612	0.0443577624

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ -\ (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal U-235 enrichment. Any enrichment between 1.8 and 4.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 15 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. This Table applies only for pre-EPU non-blanketed fuel assemblies. If a non-blanketed assembly is depleted at EPU conditions, none of the burnup accrued at EPU conditions can be credited (i.e., only burnup accrued at pre-EPU conditions may be used as burnup credit).

Table 5.5-3

<u>Fuel Categories Ranked by Reactivity</u>
See notes 1-5 for use of Table 5.5-3

Region I	I-1	High Reactivity
	I-2	
	I-3	
	I-4	Low Reactivity
Region II	II-1	High Reactivity
	II-2	
	II-3	
	II-4	
	II-5	Low Reactivity

<u>Notes</u>:

1. Fuel Category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category I-2 is less reactive than Category I-1, etc. The more reactive fuel categories require compensatory measures to be placed in Regions I and II of the SFP, e.g., use of water filled cells, Metamic inserts, or full length RCCAs.

2. Any higher numbered fuel category can be used in place of a lower numbered fuel category from the same Region.

3. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.

4. Category I-2 is fresh unburned fuel that obeys the IFBA requirements of Table 5.5-4.

5. All Categories except I-1 and I-2 are determined from Tables 5.5-1 and 5.5-2.

Table 5.5-4

<u>IFBA Requirements for Fuel Category I-2</u>

Nominal Enrichment (wt% U-235)	Minimum Required Number of IFBA Pins
Enr. ≤ 4.3	0
4.3 < Enr. ≤ 4.4	32
4.4 < Enr. ≤ 4.7	64
4.7 < Enr. ≤ 5.0	80

FIGURE 5.5-1

ALLOWABLE REGION I STORAGE ARRAYS
See notes 1-8 for use of Figure 5.5-1

DEFINITION | **ILLUSTRATION**

Array I-A
Checkerboard pattern of Category I-1 assemblies and empty (water-filled) cells.

I-1	X
X	I-1

Array I-B
Category I-4 assembly in every cell.

I-4	I-4
I-4	I-4

Array I-C
Combination of Category I-2 and I-4 assemblies. Each Category I-2 assembly shall contain a full length RCCA.

I-2	I-4
I-4	I-4

I-2	I-2
I-4	I-4

I-2	I-2
I-2	I-4

I-2	I-2
I-2	I-2

Array I-D
Category I-3 assembly in every cell. One of every four assemblies contains a full length RCCA.

I-3	I-3
I-3	I-3

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.
3. Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 5.5-4.
4. Categories I-3 and I-4 are determined from Tables 5.5-1 and 5.5-2.
5. Shaded cells indicate that the fuel assembly contains a full length RCCA.
6. **X** indicates an empty (water-filled) cell.
7. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-2

ALLOWABLE REGION II STORAGE ARRAYS

See notes 1-6 for use of Figure 5.5-2

DEFINITION	ILLUSTRATION

Array II-A

Category II-1 assembly in three of every four cells;
one of every four cells is empty (water-filled);
the cell diagonal from the empty cell contains a Metamic insert or full length RCCA.

II-1	II-1
X	II-1

X	II-1
II-1	II-1

Array II-B

Checkerboard pattern of Category II-3 and II-5 assemblies with two of every four cells containing a Metamic insert or full length RCCA.

II-3	II-5
II-5	II-3

II-3	II-5
II-5	II-3

Array II-C

Category II-4 assembly in every cell with two of every four cells containing a Metamic insert or full length RCCA.

II-4	II-4
II-4	II-4

II-4	II-4
II-4	II-4

Array II-D

Category II-2 assembly in every cell with three of every four cells containing a Metamic insert or full length RCCA.

II-2	II-2
II-2	II-2

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
6. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-3

INTERFACE RESTRICTIONS BETWEEN REGION I AND REGION II ARRAYS

See notes 1-8 for use of Figure 5.5-3

DEFINITION

Array II-A, as defined in Figure 5.5-2, when placed on the interface with Region I shall have the empty cell in the row adjacent to the Region I rack.

ILLUSTRATION

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-1	X	II-1	X
II-1	II-1	II-1	II-1

Array II-A

Arrays II-B, II-C and II-D, as defined in Figure 5.5-2, when placed on the interface with Region I shall have an insert in every cell in the row adjacent to the Region I rack.

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Array II-B

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-C

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2

Array II-D

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only. Region I Array I-B is depicted as the example; however, any Region I array is allowed provided that
   a. For Array I-D, the RCCA shall be in the row adjacent to the Region II rack, and
   b. Array I-A shall not interface with Array II-D.
6. If no fuel is stored adjacent to Region II in Region I, then the interface restrictions are not applicable.
7. Figure 5.5-3 is applicable only to the Region I - Region II interface. There are no restrictions for the interfaces with the cask area rack.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
NPS	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

NPS - Nuclear Plant Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Nuclear Plant Supervisor from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear Plant Supervisor from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position may be filled by the Nuclear Plant Supervisor or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION

6.2.3.1 An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit and the opposite unit. This individual shall meet the qualifications specified by the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for

6.3.1.1 The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.1.2 The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i.

6.3.1.3 The licensed Operators and Senior Operators who shall also meet or exceed the minimum qualifications of the supplemental requirements specified in 10 CFR Part 55, and ANSI 3.1, 1981.

6.3.1.4 The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant

c. Training: Complete the Multi-Discipline Supervisor training program

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

6.3.3 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1.3, perform the functions described in 10 CFR 50.54(m)

6.4 DELETED

6.5 DELETED

6.6 DELETED

6.7 DELETED

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures required by the Quality Assurance Topical Report.

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Process Control Program implementation;

d. Offsite Dose Calculation Manual implementation;

e. Quality Control Program for effluent monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974;

f. Facility Fire Protection Program;

g. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975; and

h. Diesel Fuel Oil Testing Program implementation.

6.8.2 DELETED

6.8.3 DELETED

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at every 18 months.

The provisions of Specification 4.0.2 are applicable.

b. DELETED

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

PROCEDURES AND PROGRAMS (Continued)

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. DELETED

e. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API Gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for Grade No. 2-D fuel oil per ASTM D975, and

3. a clear and bright appearance with proper color;

b. Other properties for Grade No. 2-D fuel oil per ASTM D975 are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/liter when tested every 31 days in accordance with either ASTM D-2276 or ASTM D-5452.

PROCEDURES AND PROGRAMS (Continued)

f. Radioactive Effluent Controls Program

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

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

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

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

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

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

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

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

   a. For noble gases: a dose rate less or equal to 500 mrems/yr to the whole body and a dose rate less than or equal to 3000 mrems/yr to the skin, and

   b. For iodine 131, iodine 133 tritium and all radionuclides in particulate form with half live greater than 8 days: a dose rate less than or equal to 1500 mrems/yr to any organ.

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

PROCEDURES AND PROGRAMS (Continued)

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

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

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. DELETED

h. Containment Leakage Rate Testing Program

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

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) Type A testing frequency in accordance with NEI 94-01, Revision 0, Section 9.2.3, except:

   a) For Unit 3, the first Type A test performed after the November 1992 Type A test shall be performed no later than November 2007.

   b) For Unit 4, the first Type A test performed after October 1991 shall be performed no later than October 2006.

3) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is defined here as the containment design pressure of 55 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding 1.0 L_a, the As-left leakage rate acceptance criterion is $\leq$0.75 L_a, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than 0.60 L_a, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

- The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than 0.6 L_a, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criteria is $\leq 0.05\ L_a$, when pressurized to P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

i. Technical Specifications (TS) Bases Control Program

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

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

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

1. Change in the TS incorporated in the license or

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

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

d. Proposed changes that meet the criteria of Specification 6.8.4 i.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

j. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met..

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located greater than 18.11 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.11 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The portion of the tube below 18.11 inches from the top of the tubesheet is excluded from inspection. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period beings at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;

   b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

   c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

3. If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-secondary leakage.

k. Control Room Envelope Habitability Program

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

The program shall include the following elements:

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

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

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

d. Measurement, at designated locations, of the CRE pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the CREVS, operating at the flow rate required by Surveillance Requirement 4.7.5.d, at a Frequency of 18 months. Additionally, the supply fans (trains A and B) will be tested on a staggered test basis (defined in Technical Specification definition 1.29 every 36 months). The results shall be trended and the CRE boundary assessed every 18 months.

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

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

6.8.5 DELETED

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1 .1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

Reports required on an annual basis shall include:

The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded 0.25 microcuries per gram DOSE EQUIVALENT I-131.

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the Offsite Dose Calculation Manual (ODCM), and in (2) 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT**

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

6.9.1.5 DELETED

*A single submittal may be made for a multiple unit station.

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

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The W(Z) function(s) for Base-Load Operation corresponding to a ± 2% band about the target flux difference and/or a ± 3% band about the target flux difference, the Load-Follow function $F_z(Z)$ and the augmented surveillance turnon power fraction P_T shall be provided to the U.S. Nuclear Regulatory Commission, whenever P_T is <1.0. In the event, the option of Baseload Operation (as defined in Section 4.2.2.3) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.

The analytical methods used to generate the Peaking Factor limits shall be those previously reviewed and approved by the NRC. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1. Reactor Core Safety Limits for Specification 2.1.1.
2. Overtemperature ΔT, Note 1 of Table 2.2-1 for Specification 2.2.1, determination of values K_1, K_2, K_3, T', P', τ_1, τ_2, τ_3, τ_4, τ_5, τ_6, and the breakpoint and slope values for the f_1 (ΔI).
3. Overpower ΔT, Note 3 of Table 2.2-1 for Specification 2.2.1, determination of values for K_4, K_5, K_6, T'', τ_7 and f_2 (ΔI).
4. Shutdown Margin - T_{avg} >200°F for Specification 3/4.1.1.1.
5. Shutdown Margin - T_{avg} $\leq$200°F for Specification 3/4.1.1.2.
6. Moderator Temperature Coefficient for Specification 3/4.1.1.3.
7. Axial Flux Difference for Specification 3.2.1.
8. Control Rod Insertion Limits for Specification 3.1.3.6.
9. Heat Flux Hot Channel Factor - $F_Q(Z)$ for Specification 3/4.2.2.
10. All Rods Out position for Specification 3.1.3.2.
11. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.
12. DNB Parameters for Specification 3.2.5, determination of values for Reactor Coolant System T_{avg} and Pressurizer Pressure.

The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983.

2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974.

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine F_Q (Z), $F_{\Delta}H$ and the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

3. WCAP-10054-P-A, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

4. WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", January 2005.

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,' " June 28, 1996.**

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."**

7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

8. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006.

The analytical methods used to determine Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, "Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions, " September 1986

2. WCAP-9272-P-A, " Westinghouse Reload Safety Evaluation Methodology," July 1985

The analytical methods used to determine Safety Limits, Shutdown Margin - $T_{avg} > 200°F$, Shutdown Margin - $T_{avg} \leq 200°F$, Moderator Temperature Coefficient, DNB Parameters, Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR

AMENDMENT NO. 260 RENEWED FACILITY OPERATING LICENSE NO. DPR-31 AND

AMENDMENT NO. 255 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 September 14, 2012, as supplemented by letters dated January 29, February 14, May 30, and October 22, 2013, and March 11, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12262A290, ML13037A387, ML13050A028, ML13179A351, ML13318A008, and ML14087A127, respectively), Florida Power & Light Company (FPL, the licensee) requested changes to the Renewed Facility Operating Licenses (RFOLs) DPR-31 and DPR-41 and the Technical Specifications (TSs) for the Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Turkey Point 3 and 4). The proposed changes would remove completed and satisfied license conditions, revise TSs and the RFOLs to remove redundancy, correct inadvertent errors in the TSs, update the references to the physical security plan in the RFOLs, and make editorial changes to the TSs.

By correspondence dated January 24 and April 13, 2013 (ADAMS Accession Nos. ML13308C496 and ML13106A137, respectively), the U.S. Nuclear Regulatory Commission (NRC or the Commission) requested additional information from the licensee. By letters dated February 14 and May 30, 2013, the licensee responded to these requests.

By letter dated March 11, 2014, the licensee requested a reduction in scope of the license amendment request (LAR). Specifically, the licensee deleted the proposed change to TS Figure 3.1-2, "Boric Acid Tank Minimum Volume," from the LAR. The licensee stated that it will address changes to TS Figure 3.1-2 at a later date via a separate correspondence.

The NRC staff's original proposed no significant hazards consideration determination was published in the *Federal Register* (FR) on January 8, 2013 (78 FR 1271). By letter dated January 29, 2013, the licensee supplemented its application dated September 14, 2012, by requesting removal of an additional license condition, additional formatting and spelling corrections, and providing a replacement submittal. Therefore, the NRC published another proposed no significant hazards consideration determination in the *Federal Register* on

April 16, 2013 (78 FR 22569). The supplements dated February 14, May 30, and October 22, 2013, and March 11, 2014, provided clarifying information that did not expand the scope of the submittal dated January 29, 2013, and did not change the NRC staff's proposed no significant hazards consideration determinations, as published in the *Federal Register* on January 8, 2013, and April 16, 2013. Specifically, the supplement dated March 11, 2014, limited the scope of the submittals dated September 14, 2012, and January 29, 2013.

2.0 REGULATORY EVALUATION

The NRC staff reviewed the licensee's application to ensure 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) activities proposed will be conducted in compliance with the Commission's Regulations and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public. The NRC staff considered the following regulatory requirements, guidance, and licensing and design basis information during its review of the proposed change.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50) provides the regulatory requirements for the licensing of production and utilization facilities.

Section 10 CFR 50.92, "Issuance of amendment," paragraph 50.92(a) states that in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate.

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical Specifications." The TSs requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

Section 10 CFR 50.68, "Criticality accident requirements," paragraph 50.68(b)(4) states:

> If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50 establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety. The GDC used during the licensing of Turkey Point 3 and 4, which were based on the 1967 Atomic Energy Commission Proposed GDC, predate those provided in Appendix A of 10 CFR Part 50. The Turkey Point UFSAR [Updated Final Safety Analysis Report] describes the GDC applicable to Turkey Point 3 and 4. The Turkey Point UFSAR, Section 1.3, "General Design Criteria," states the following:

> The general design criteria define or describe safety objectives and approaches incorporated in the design. These general design criteria are addressed explicitly in the pertinent sections in this report. The remainder of this section, 1.3, presents a brief description of related features which are provided to meet the design objectives reflected in the criteria. The description is developed more fully in those succeeding sections of the report indicated by the references.
>
> The parenthetical numbers following the section headings indicate the numbers of the 1967 proposed draft General Design Criteria (GDC).

In addition, Attachment 4 to FPL's license amendment request (LAR) dated December 13, 2010 (ADAMS Accession No. ML103560177), for an extended power uprate (EPU) contained a comparison of the UFSAR GDC to the GDC in 10 CFR Part 50, Appendix A. The Turkey Point UFSAR GDC are denoted as "PTN GDC." The licensee's acronym for Turkey Point is "PTN."

GDC 62, "Prevention of criticality in fuel storage and handling," of 10 CFR Part 50, Appendix A states, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." Section 9.5.2.1 of the Turkey Point 3 and 4 UFSAR states, "Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls [...] (1967 Proposed GDC 66)."

The NRC staff evaluated the licensee's adoption of the Alternate Source Term (AST) as allowed pursuant to 10 CFR 50.67 and the licensee's assessment of the impact of the proposed changes to the Turkey Point TSs on design basis analysis. The staff's review also ensures continued compliance with the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, 3, 4, 5, and 19, which apply to control room envelope (CRE) habitability. A summary of these GDC and the corresponding Turkey Point UFSAR criteria (PTN GDC) follows.

- GDC 1, "Quality Standards and Records," requires that SSCs important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions performed. The corresponding Turkey Point UFSAR general design criterion is PTN GDC-1.

- GDC 2, "Design Basis for Protection Against Natural Phenomena," requires that SSCs important to safety be designed to withstand the effects of earthquakes and other natural hazards. The corresponding Turkey Point UFSAR general design criterion is PTN GDC-2.

- GDC 3, "Fire Protection," requires SSCs important to safety be designed and located to minimize the effects of fires and explosions. The corresponding Turkey Point UFSAR general design criterion is PTN GDC-3.

- GDC 4, "Environmental and Dynamic Effects Design Bases," requires SSCs important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). Attachment 4 (ADAMS Accession No. ML103560177) to FPL's LAR dated December 13, 2010, for an EPU, states, "PTN has no plant specific GDC analogous to 10 CFR 50, Appendix A, GDC-4 requiring that SSCs important to safety be designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents but such requirements are addressed under PTN's Environmental Qualification (EQ) program [...]."

- GDC 5, "Sharing of Structures, Systems, and Components," requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including in the event of an accident in one unit, the orderly shutdown and cooldown of the remaining units. The corresponding Turkey Point UFSAR general design criterion is PTN GDC-4.

- GDC 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and to maintain the reactor in a safe condition under accident conditions, including a loss of coolant accident (LOCA). Adequate radiation protection is to be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of specified values. The corresponding Turkey Point UFSAR general design criterion is PTN GDC-11.

Prior to incorporation of Revision 3 of Technical Specifications Task Force (TSTF) 448 (TSTF-448), "Control Room Habitability [CRH]" (ADAMS Accession No. ML062210095), the Standard Technical Specifications (STSs) addressing CRE boundary operability resided only in the following CRE ventilation system specifications:

- NUREG 1430, TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)";
- NUREG 1431, TS 3.7.10, "Control Room Emergency Filtration System (CREFS)";
- NUREG 1432, TS 3.7.11, "Control Room Emergency Air Cleanup System (CREACS)";
- NUREG 1433, TS 3.7.4, "Main Control Room Environmental Control (MCREC) System";
- NUREG 1434, TS 3.7.3, "Control Room Fresh Air (CRFA) System."

In these specifications, the SR associated with demonstrating the operability of the CRE boundary requires verifying that one CREVS train can maintain a positive pressure relative to the areas adjacent to the CRE during the pressurization mode of operation at a makeup flow rate. Facilities that pressurize the CRE during the emergency mode of operation of the CREVS have similar SRs. Other facilities that do not pressurize the CRE have only a system flow rate criterion for the emergency mode of operation. Regardless, the results of the American Society for Testing and Materials (ASTM) standard, ASTM E 741-00, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," 2000 (ASTM E741), tracer gas tests to measure CRE unfiltered inleakage at facilities indicated that the differential pressure surveillance (or the alternative surveillance at nonpressurization facilities) is not a reliable method for demonstrating CRE boundary operability. That is, licensees were able to obtain differential pressure and flow measurements satisfying the SR limit even though unfiltered inleakage was determined to exceed the value assumed in the safety analyses.

In addition to an inadequate SR, the action requirements of these specifications were ambiguous regarding CRE boundary operability in the event CRE unfiltered inleakage is found to exceed the analysis assumption. The ambiguity stemmed from the view that the CRE boundary may be considered operable but degraded in this condition, and that it would be deemed inoperable only if calculated radiological exposure limits for CRE occupants exceeded a licensing basis limit (e.g., as stated in GDC 19 of 10 CFR Part 50, Appendix A), even while crediting compensatory measures.

NRC Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," dated December 29, 1998, and which is available from the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/gen-comm/admin-letters/1998/al98010.html, states that "the discovery of an improper or inadequate TS value or required action is considered a degraded or nonconforming condition." Additional guidance is in NRC Regulatory Issue Summary (RIS) 2005-20, dated April 16, 2008 (ADAMS Accession No. ML073531473). Imposing administrative controls in response to the improper or inadequate TS is considered an acceptable short-term corrective action. The NRC staff expects that, following the imposition of administrative controls, an amendment to the inadequate TS, with appropriate justification and schedule, will be submitted in a timely fashion.

Licensees that have found unfiltered inleakage in excess of the limit assumed in the safety analyses and have yet to either reduce the inleakage below the limit or establish a higher bounding limit through re-analysis, have implemented compensatory actions to ensure the safety of CRE occupants, pending final resolution of the condition, consistent with RIS 2005-20. However, based on NRC Generic Letter 2003-01, "Control Room Habitability," dated June 12, 2003 (ADAMS Accession No. ML031620248), and AL 98-10, the NRC staff expects each licensee to propose TS changes that include a surveillance to periodically measure CRE unfiltered inleakage in order to satisfy 10 CFR 50.36(c)(3), which requires a facility's TSs to include SRs, which it defines as "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 limiting conditions for operation will be met*." (Emphasis added.)

The NRC staff also expects facilities to propose unambiguous remedial actions, consistent with 10 CFR 50.36(c)(2), for the condition of not meeting the LCO because of an inoperable CRE boundary. The action requirements should specify a reasonable completion time to restore conformance to the LCO before requiring a facility to be shut down. This completion time should be based on the benefits of implementing mitigating actions to ensure CRE occupant safety and sufficient time to resolve most problems anticipated with the CRE boundary, while minimizing the chance that operators in the CRE will need to use mitigating actions during accident conditions.

The NRC staff published its notice of the availability of Revision 3 of TSTF-448 and an associated model safety evaluation for use in license amendment requests in the *Federal Register* on January 17, 2007 (72 FR 2022). On March 30, 2012, the NRC approved Turkey Point 3 and 4 to adopt Revision 3 of TSTF-448 by issuance of Amendments 248 and 244 for Units 3 and 4, respectively (ADAMS Accession No. ML12067A176). The adoption of TSTF-448, Revision 3, assures that the facility's TS LCO for the CREVS is met by demonstrating unfiltered leakage into the CRE is within limits (i.e., the operability of the CRE boundary). In support of this surveillance, which specifies a test interval (frequency) described in NRC Regulatory Guide (RG) 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," dated May 2003 (ADAMS Accession No. ML031490664), TSTF-448 also adds TS administrative controls to assure the habitability of the CRE between performances of the ASTM E741 test. In addition, adoption of TSTF-448 establishes clear and reasonable required actions in the event CRE unfiltered inleakage is found to exceed the analysis assumption.

The changes made by TSTF-448 to the STS requirements for the CREVS and the CRE boundary conform to 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3). Their adoption better assures that the Turkey Point 3 and 4 CRE will remain habitable during normal operation and design basis accident conditions.

3.0 TECHNICAL EVALUATION

3.1 Changes to License Conditions and Technical Specifications Regarding AST Modifications, CRH, and CREVS

3.1.a Deletion of DPR-31 Condition 3.H and DPR-41 Condition 3.I on AST Modifications

In Sections 4.2.2 and 4.2.9 of its application dated January 29, 2013, the licensee proposed to delete RFOL DPR-31, Condition 3.H, "Alternative Source Term (AST) Modifications," and RFOL DPR-41, Condition 3.I, "Alternative Source Term Modifications." Both license conditions have the same requirements for their respective units. These license conditions state:

1. FPL will relocate the CR Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR 50 Appendix A, GDC 19. The new intakes will be located near the ground level extending out from the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. FPL will perform post-modification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes. In addition, FPL will provide to the NRC a confirmatory assessment which demonstrates that the requirements of 10 CFR 50 Appendix A, GDC 19 will be met. The confirmatory assessment will follow the methodology in Amendment 244 [244] [the alternative source term

amendment] including the methods used for the establishment of the atmospheric dispersion factors (X/Q values).

2. FPL will install ten (two large an eight small) stainless steel wire mesh baskets containing NaTB [sodium tetraborate] located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

3. The CREVS compensatory filtration unit, which is being installed by FPL as part of the AST methodology implementation at Turkey point, will be designed in accordance with the Class I Structures, Systems, and Equipment Design Requirements defined in Appendix 5A of the Turkey Point UFSAR. As such, the compensatory filtration unit will be designed so that the stress limits found in Table 5A-1 of the Turkey Point UFSAR will not be exceeded due to the loadings imposed by a maximum hypothetical earthquake. FPL shall ensure that the design of the compensatory filtration unit satisfies these stress limits prior to the implementation of the proposed AST methodology at Turkey Point.

On June 23, 2011, the NRC issued License Amendments 244/240 (ADAMS Accession No. ML110800666) to Turkey Point 3 and 4, which included a supporting safety evaluation regarding the AST. In its submittal dated January 29, 2013, the licensee stated that license conditions 3.H.1 for Unit 3 and 3.I.1 for Unit 4 were satisfied with the relocation of the control room emergency air intakes and post-modification testing that included flow balancing, control room pressurization testing, and tracer gas testing. The licensee provided the NRC with the required confirmatory assessment by letter dated August 11, 2011 (ADAMS Accession No. ML11228A011). The licensee stated that license conditions 3.H.2 for Unit 3 and 3.I.2 for Unit 4 were satisfied with the installation of the ten baskets loaded with the specified quantity of sodium tetraborate. The licensee stated that license conditions 3.H.3 for Unit 3 and 3.I.3 for Unit 4 were satisfied with the installation and testing of compensatory filtration units on July 25, 2012. The licensee provided the test results in its letter dated February 14, 2013. The licensee implemented Amendments 244/240, including TS 3/4.7.5 on CREVS, and EPU Amendments 249/245 (ADAMS Accession No. ML11293A359) at Turkey Point 3 and 4 during the EPU refueling outages for each unit. Based on the information provided, the NRC staff concludes that the licensee satisfied these license conditions; therefore, the NRC staff finds it acceptable to remove these conditions from the RFOLs.

3.1.b Deletion of DPR-31 Condition 3.I and DPR-41 Condition 3.J on CRH

In Sections 4.2.3 and 4.2.10 of its application dated January 29, 2013, the licensee proposed to delete RFOL DPR-31, Condition 3.I for Unit 3 and RFOL DPR-41, Condition 3.J for Unit 4. Both license conditions are titled, "Control Room Habitability," and contain the same requirements for their respective units. These license conditions state:

> Upon implementation of Amendment No. 248 [244 for Unit 4] adopting TSTF-448 Revision 3, the determination of [CRE] unfiltered air inleakage as required by Surveillance Requirement (SR) 4.7.5.g, in accordance with [TS] 6.8.4.k.c.(i), the assessment of CRE habitability as required by [TS] 6.8.4.k.c.(ii), and the

measurement of CRE pressure as required by [TS] 6.8.4.k.d, shall be considered met. Following implementation:

(a) The first performance of SR 4.7.5.g, in accordance with [TS] 6.8.4.k.c.(i), shall be within the specified Frequency of 3 years, plus the 9-month allowance of SR 4.0.2, as measured from July 31, 2009, the date of the most recent tracer gas test.[...]

(b) The first performance of the periodic assessment of CRE habitability, [TS] 6.8.4.k.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from July 31, 2009, the date of the most recent tracer gas test.

(c) The first performance of the periodic measurement of CRE pressure, [TS] 6.8.4.k.d, shall be within 36 months on a STAGGERED TEST BASIS, plus the 138 days allowed by SR 4.0.2, as measured from the date of the most recent successful pressure measurement test, or within 138 days of license amendment implementation if not performed previously.

On March 30, 2012, the NRC issued Amendments 248 and 244 (ADAMS Accession No. ML12125A103) to the licensee, which included a supporting safety evaluation, regarding CRH. These license conditions addressed CRE testing and CRH assessment. In its submittal dated January 29, 2013, the licensee stated that License Conditions 3.I.(a) and (c) for Unit 3 and 3.J.(a) and (c) for Unit 4 were satisfied with completion of the tracer gas testing and pressurization testing of the control room using the main CREVS filter train on July 18, 2012, and using the compensatory filtration unit on July 25, 2012. The licensee stated that License Conditions 3.I.(b) for Unit 3 and 3.J.(b) for Unit 4 were satisfied with completion of the required CRE Habitability Assessment.

By e-mail dated January 24, 2013, the NRC staff requested the licensee to provide the results of the tracer gas and pressurization testing and to discuss how the control room test configuration is consistent with the AST assumptions. By letter dated February 14, 2013, the licensee responded to the NRC's request, as follows.

> On September 14, 2012, FPL requested via letter L-2012-130 [...] deletion of license conditions 3.I for DPR-31 and 3.J for DPR-41 regarding (a) CRE unfiltered inleakage test (i.e., tracer gas test), (b) CRE habitability assessment, and (c) CRE differential pressure test. It was noted that license conditions 3.I.(a) & 3.I.(c) were satisfied with completion of the tracer gas testing and pressurization testing of the CRE using the normal control room emergency ventilation system (CREVS) filter train on July 18, 2012 and using the compensatory filtration unit on July 25, 2012 while license condition 3.I.(b) was satisfied with completion of the required CRE habitability assessment on September 13, 2012. These specific license conditions were established with the issuance of Amendments 248 and 244 for TSTF-448 on Control Room Habitability dated March 30, 2012 [...].

On July 18, 2012, the measured unfiltered CRE inleakage was approximately 37±5 scfm [standard cubic feet per minute] plus 10 scfm for CR [control room] ingress/egress using the normal CREVS filter train. On July 25, 2012, the measured unfiltered CRE inleakage was approximately 44+6 scfm plus 10 scfm for CR ingress/egress using the compensatory filtration train. The acceptance criterion for unfiltered CRE inleakage is ≤ 90 scfm plus 10 scfm for CR ingress/egress based on Alternative Source Term (AST) and associated radiological dose consequence analyses which were approved by the NRC under Amendments 244 and 240 on June 23, 2011 [...]. Therefore, the measured unfiltered CRE inleakage was found acceptable.

On July 18, 2012, the CRE differential pressure for the normal filter train, using the ['A'] supply fan, was measured at 0.26 inch H_2O [water] with regard to the battery room, 0.04 inch H_2O with regard to the cable spreading room, and 0.04 inch H_2O with regard to the outside environs. For the normal filter train using the ['B'] supply fan, the CRE differential pressure was measured at 0.28 inch H_2O with regard to the battery room, 0.04 inch H_2O with regard to the cable spreading room, and 0.04 inch H_2O with regard to the outside environs.

On July 25, 2012, the CRE differential pressure for the compensatory filter train was measured at 0.40 inch H_2O with regard to the battery room, 0.04 inch H_2O with regard to the cable spreading room, and 0.07 inch H_2O with regard to the outside environs. Therefore, as all of the CRE differential pressure measurements were positive, the test results were found acceptable.

This testing followed two major plant modifications involving CREVS: one to relocate the CREVS emergency air intakes away from the Auxiliary Building and another to install a compensatory filtration unit to serve as a qualified backup to the normal filter train. Both of these modifications required temporary breaching of the CRE to implement the required design changes. In addition, the AST changes to CREVS TS 3/4.7.5 explicitly identified CREVS components required to be operable and corresponding surveillance requirements.

The LCO lists the two isolation dampers in the kitchen area exhaust duct and two isolation dampers in the toilet area exhaust duct with associated action statements and surveillance requirements. However, subsequent inspections revealed that these gravity backdraft dampers would require substantial corrective maintenance to be considered operable. Therefore, the motor-operated dampers in the exhaust ducts were closed and the ducts were blanked off at the CRE boundary pending further design review by Engineering. In the interim, portable air filtration units have been installed in the control room kitchen and toilet areas. With an isolation damper inoperable in the kitchen area and toilet area exhaust dampers, TS 3/4.7.5 requires entry into Actions a.8 and a.9 that permit indefinite operation with the associated flow path isolated. This condition was documented in LAR No. 223 via FPL letter dated August 7, 2012 [...] where an additional footnote was proposed for Modes 5 and 6. The LAR "was approved with the NRC's issuance of Amendments 253 and 249 on November 5, 2012 [...].

> As a result of these actions, the CRE was tested with the exhaust ducts blanked off [...] and one of the two in-series normal air intake dampers [...] failed open [...] as the worst single failure for the unfiltered inleakage and pressurization testing of both normal and compensatory filter trains. The measured unfiltered CRE inleakage and differential pressure results are as stated above for both the normal and compensatory filter trains [...].

Based on a review of Amendments 244/240, 249/245, 248/244, and 253/249 dated November 5, 2012 (ADAMS Accession No. ML12291A730), and the licensee's RAI response dated February 14, 2013, the NRC staff concludes that the licensee satisfied parts (a) and (c) of these license conditions with completion of the tracer gas testing and pressurization testing of the control room using the main control room emergency ventilation system filter train on July 18, 2012, and using the compensatory filtration unit on July 25, 2012. The NRC staff also concluded that the licensee satisfied part (b) of these license conditions with completion of the CRE Habitability Assessment. Therefore, the NRC staff finds the proposed deletions acceptable.

3.1.c Changes to TS 3.7.5, "Control Room Emergency Ventilation System"

In its application dated January 29, 2013, the licensee proposed amendments to correct three inadvertent errors in TS 3/4.7.5. Specifically, the licensee proposed the following changes: TS LCO 3.7.5 be revised to add the CRE to the LCO listing of required operable components, SR 4.7.5.c.1 be revised to cite the proper test acceptance criterion in Revision 3 to NRC RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (ADAMS Accession No. ML011710176) for an assumed 99 percent filter efficiency (i.e., 99.95 percent at 95/95 confidence level), and SR 4.7.5.d.2 and SR 4.7.5.g be annotated with the existing footnote to indicate that the subject testing applies to the both the normal filter train and the compensatory filtration unit. The licensee also proposed that the footnote be modified slightly to clarify that "use of the compensatory filtration unit" is part of the mitigating actions specified in ACTION a.5.

3.1.c.i TS LCO 3.7.5

In Section 4.1.10 of its application dated January 29, 2013, the licensee proposed to add the CRE to the TS LCO 3.7.5 listing of required operable components. The licensee stated that the CRE is considered a component of the CREVS that is required to be operable in order for the CREVS to be considered operable. The licensee stated that addition of the CRE to the LCO is consistent with this requirement and ACTION b that addresses the inoperability of CREVS caused by an inoperable CRE. The NRC staff finds that addition of CRE to TS LCO 3.7.5 is consistent with Amendments 248/244 on TSTF-448 changes to control room habitability requirements and is therefore acceptable.

3.1.c.ii TS SR 4.7.5.c.1

In Section 4.1.11 of its application dated January 29, 2013, the licensee proposed to revise SR 4.7.5.c.1 to cite the proper test acceptance criterion in Revision 3 of NRC RG 1.52 for an assumed 99 percent filter efficiency (i.e., 99.95 percent at 95/95 confidence level). The licensee stated that the AST radiological dose consequence analyses assumed 99 percent control room filter efficiency for particulates, which requires an acceptance criterion for the Dioctyl Phthalate (DOP) test of 99.95 percent in accordance with Revision 3 of NRC RG 1.52 rather than 99 percent as currently stated. The licensee stated that the required change to the DOP test acceptance criterion was inadvertently overlooked when the dose consequence analyses were revised to account for new meteorological data, increased filter efficiency, and decreased unfiltered air inleakage and the summary report resubmitted to the NRC by letter dated June 25, 2010 (ADAMS Accession No. ML101800220). The licensee stated that correction of the acceptance criterion for the DOP test assures that the control room filter will satisfy the 99 percent particulate filter efficiency assumed in the design basis dose analyses, the associated surveillance procedures have been revised, and past surveillance results were found to have met this revised acceptance criterion. The NRC staff finds this change to be consistent with the guidance discussed in the NRC safety evaluation issued with Amendments 244/240 and therefore acceptable.

3.1.c.iii TS SR 4.7.5.d.2 Footnote

In Section 4.1.12 of its application dated January 29, 2013, the licensee proposed to modify Footnote *** and apply it to SR 4.7.5.d.2. The licensee stated that Amendments 244/240 annotated SR 4.7.5.d with the footnote regarding the applicability of the SR to the compensatory filtration unit. Amendments 248/244, which implemented TSTF-448 at Turkey Point 3 and 4, inadvertently omitted the footnote when SR 4.7.5.d was split into SR 4.7.5.d.1 and SR 4.7.5.d.2. This resulted in leaving the applicability of the SR for the compensatory filtration unit unspecified. The annotation of SR 4.7.5.d.2 with the footnote would restore the intent of SR as it applies to the compensatory filtration unit.

The NRC staff finds that the proposed change to the footnote clarifies its intent as indicated in the licensee's TSTF-448 request for additional information response letter dated October 27, 2011 (ADAMS Accession No. ML11304A184). TS 3.7.5 ACTION a.5 was amended to clarify that in the event of the filter train becoming inoperable, the required mitigating actions included the immediate initiation of the compensatory filtration unit. The footnote needs to reflect this requirement by deleting "may" and including SR 4.7.5.g in the footnote's applicability. Therefore, the NRC staff finds this change acceptable.

3.1.c.iv TS SR 4.7.5.g Footnote

In Section 4.1.13 of its application dated January 29, 2013, the licensee proposed to apply Footnote *** to SR 4.7.5.g. The licensee stated that Amendments 244/240 annotated SR 4.7.5.b, c, and d with the footnote regarding the applicability of SRs to the compensatory filtration unit. Amendments 248/244 inadvertently omitted the footnote on TS 4.7.5.g, which left the applicability of the SR for the compensatory filtration unit unspecified (ADAMS Accession No. ML12067A176).

The NRC staff finds that the annotation of SR 4.7.5.g with the footnote will restore the intent of the SR as it applies to the compensatory filtration unit, as stated in the licensee's letter dated September 15, 2010 (ADAMS Accession No. ML102630160). The addition of the footnote is appropriate, as the maintenance of the compensatory filtration unit must satisfy the applicable

SRs imposed on the normal CREVS filter train (TS 4.7.5.b, c, and d) in order to assure its operational readiness. Therefore, the NRC staff finds the change acceptable.

3.2 Changes to Operating Licenses and TSs Regarding Fuel Storage Criticality

3.2.a Changes to Operating Licenses and TSs Regarding Spent Fuel Pool Burnable Absorber Credit

On June 15, 2012, the NRC issued Amendments 249/245 to the licensee. These amendments established License Conditions 3.L for Unit 3 and 3.M for Unit 4, which addressed burnable absorber credit in the spent fuel pool. These conditions state, "With respect to Technical Specification 5.5.1.3, FPL shall not credit any burnable absorber other than Integral Fuel Burnable Absorber (IFBA) rods for the storage of fuel assemblies in the Region I spent fuel racks." In Sections 4.2.6 and 4.2.13 of its application dated January 29, 2013, the licensee proposed to delete these license conditions.

To support the deletion of these license conditions, the licensee proposed to modify the TSs to maintain the same level of requirement. Specifically, in Sections 4.1.16 and 4.1.17 of its application dated January 29, 2013, the licensee proposed removing notes from two tables and one figure in Section 5.5.1 of the TSs that allow the crediting of burnable absorbers other than IFBAs in the spent fuel pool. The licensee proposed to delete the phrase, "or contains an equivalent amount of another burnable absorber," from Note 4 in TS Table 5.5-3 and from Note 3 in TS Figure 5.5-1. The licensee stated in its application that removing the provision in the notes would effectively prohibit crediting any burnable absorber other than the IFBA rods for the storage of fuel assemblies in the Region I spent fuel racks. Thus the same prohibition established by License Conditions 3.L for Unit 3 and 3.M for Unit 4 could be deleted. The NRC staff concludes that removal of this phrase from the TS Notes ensures that only those absorbers that have been explicitly analyzed – in this case, IFBAs – may be credited for criticality control and thus, the intent of the license conditions will be maintained in the TSs. Therefore, the NRC staff finds that the proposed changes to the TSs are acceptable.

In Section 4.1.15 of its application dated January 29, 2013, the licensee proposed to delete Notes 4 and 5 of TS Table 5.5-1 because Table 5.5-1 is not applicable to either Category I-1 or I-2 fuel. Note 4 states, "Category I-1 is fresh unburned fuel up to 5.0 wt% [weight percent] U-235 enrichment," and Note 5 states, "Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 5.5-4 or contains an equivalent amount of another burnable absorber." The NRC staff finds that removal of these notes is acceptable because Table 5.5-1 does not contain any information applicable to either Category I-1 or I-2 fuel.

The NRC staff finds that deletion of Unit 3 License Condition 3.L.1 and Unit 4 License Condition 3.M.1 is acceptable because the TSs maintain the intent of the license conditions.

3.2.b Changes to TS 5.5.1 on Fuel Storage – Criticality

In Section 4.1.14 of its application dated January 29, 2013, the licensee proposed to delete the conditional phrase, "Unless otherwise specified in accordance with Specification 5.5.1.1.f," from TS 5.5.1.3. This conditional phrase was added to TS 5.5.1.3 by Amendments 246/242 dated October 31, 2011 (ADAMS Accession No. ML11216A057). This amendment also revised

TS 5.5.1.f by deleting the provision for alternate configurations other than those allowed by TS 5.5.1.3. The NRC staff determined that removal of the conditional phrase from TS 5.5.1.3 makes TS 5.5.1.3 consistent with TS 5.5.1.1.f and is therefore acceptable. The licensee also proposed clarifying the applicability of TS 5.5.1.3 by adding "in the Region I or Region II racks" after "Fresh or irradiated fuel assemblies." The NRC staff finds that this change clarifies current requirements and is acceptable.

3.2.c Deletion of Unit 4 License Condition 3.H Regarding the Boraflex Remedy

In Section 4.2.8 of its application dated January 29, 2013, the licensee requested the NRC to remove RFOL DPR-41 License Conditions 3.H.(a), (b), (c), and (d) for Unit 4 because these conditions have been met or are no longer applicable. These license conditions state:

> H. FPL will implement the following measures as part of the request for a change in the implementation date for Amendment 229 for Unit 4 [dated July 17, 2007; ADAMS Accession No. ML071800411]. These measures will remain in place until Amendment No. 229 is implemented or until the NRC approves the license amendment request discussed in Item (b) below but not later than February 28, 2011.
>
> (a) The Unit 4 Spent Fuel Pool (SFP) boron concentration will be increased to and maintained no less than 2100 ppm. This measure will be implemented within 72 hours of installing the transfer tube gate isolating the SFP from the reactor cavity during the current Unit 4 refueling outage.
>
> (b) FPL will complete Boraflex panel surveillance using EPRI BADGER neutron attenuation methodology in the Unit 4 SFP no later than May 30, 2010. The report documenting the results of the EPRI BADGER testing campaign and the license amendment request updating the SFP licensing basis will be submitted to the NRC no later than 90 days after completion of the BADGER testing.
>
> (c) FPL will increase the current MWD/MTU burnup requirements for SFP Region II storage by 10% [percent] and will configure the SFP to comply with these requirements or insert an RCCA [rod control cluster assembly] in any fuel assembly not in compliance with these requirements. These measures will be completed by February 28, 2010.
>
> (d) FPL will not move any fuel assemblies into the Unit 4 SFP subsequent to the successful completion of startup physics tests for Unit 4 Cycle 25.

The licensee stated that it implemented Amendment No. 229 in September 2010. As required by condition H.(b), the licensee provided the results of the EPRI Badger testing for Unit 4 to the NRC by letter dated August 5, 2010 (ADAMS Accession No. ML102250419). The licensee submitted a license amendment request updating the SFP licensing basis to the NRC on August 5, 2010 (ADAMS Accession No. ML102220022). The NRC approved the request by issuance of Amendments 246/242 dated October 31, 2012 (ADAMS Accession No. ML11216A057). Amendments 246/242 supersede the fuel storage requirements specified in Amendments 234/229. The license conditions have been met and are no longer applicable. Therefore, the NRC staff finds that the deletion of these license conditions is acceptable.

3.3 Other Changes to TSs

3.3.a Revisions to Various TSs Index Listings

In Section 4.1.1 of its application dated January 29, 2013, the licensee proposed to change the TSs' index title for Definition 1.13 from "Ē AVERAGE DISINTEGRATION ENERGY" to "DOSE EQUIVALENT XE-133." This change would correct an omission from Amendments 244/240. Amendments 244/240 revised the definition and its title but failed to include the corresponding revision in the index.

In the revised Section 4.1.18 of its application supplement dated May 30, 2013, the licensee proposed to remove the page numbers from the following index listings. The corresponding pages are deleted from the TSs or the text noting the deleted sections is moved to different pages with the proposed amendments.

- "3/4.4.10 DELETED"
- "3/4.9.7 DELETED"
- "3/4.9.12 DELETED"
- "3/4.10.4 (This specification number is not used)"
- "6.4 DELETED"
- "6.5 DELETED"
- "6.6 DELETED"
- "6.7 DELETED"
- "6.10 DELETED"
- "6.11 DELETED"
- "6.13 DELETED"

In the revised Section 4.1.18 of its application supplement dated May 30, 2013, the licensee proposed to revise the title for index listing,"3/4.6.3 Not Used," to "3/4.6.3 DELETED," and to remove the page number from this index listing.

In the revised Section 4.1.18 of its application supplement dated May 30, 2013, the licensee proposed to revise the titles of the following index listings. The NRC staff finds that the changes would correct discrepancies between the index and the TSs.

- On page viii, the index listing title, "FIGURE 3.4-2 TURKEY POINT UNITS 3&4 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 32 EFPY [effective full power years]," would be changed to, "FIGURE 3.4-2 TURKEY POINT UNITS 3 AND 4 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (Heatup Rates of 60 and 100°F/hr) APPLICABLE for 48 EFPY (Without Margins for Instrument Errors)."

- On page viii, the index listing title, "FIGURE 3.4-3 TURKEY POINT UNITS 3&4 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS (100°F/hr) APPLICABLE

TO 32 EFPY" would be changed to, "FIGURE 3.4-3 TURKEY POINT UNITS 3 AND 4 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS (Cooldown Rates of 0, 20, 40, 60, and 100°F/hr) APPLICABLE for 48 EFPY (Without Margins for Instrumentation Errors)."

- On page x, the index listing title, "TABLE 3.7-1 MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION," would be changed to, "TABLE 3.7-1 MAXIMUM ALLOWABLE POWER LEVEL WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION."

- On page x, under the index listing title, "Standby Steam Generator Feedwater System," would be changed to, "Standby Feedwater System."

- On page x, under the index listing title, "Feedwater Line Isolation Valves," would be changed to, "Feedwater Isolation."

In Section 4.1.5 of its application dated January 29, 2013, the licensee proposed to delete the index listing for Figure 3.4-1, "DOSE EQUIVALENT I-131 REACTOR COOLANT ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1 μCi [micro Curie]/gram DOSE EQUIVALENT I-131." This change would correct an omission from Amendments 244/240, which deleted the figure but failed to include the corresponding change to the index.

In Section 4.1.6 of its application dated January 29, 2013, the licensee proposed to delete the index listing for Table 3.4.5, "REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM – WITHDRAWAL SCHEDULE." This change would correct an omission from Amendments 208/202 (ADAMS Accession No. ML003765956), which deleted the table but failed to include the corresponding change to the index.

In Sections 4.1.7 and 4.1.8 of its application dated January 29, 2013, the licensee proposed to delete the index listings for TS 3/4.6.5, "COMBUSTIBLE GAS CONTROL," and TS 3/4.6.6, "POST ACCIDENT CONTAINMENT VENT SYSTEM." The changes would correct omissions from Amendments 217/211 (ADAMS Accession No. ML013550500), which deleted these specifications but omitted the corresponding changes to the index.

In the revised Section 4.1.18 of its application supplement dated May 30, 2013, the licensee proposed to delete the following index listings. The corresponding TS pages, figures, and tables for these index listings have already been removed, deleted, or designated as not used from the TSs by previously issued amendments.

- "FIGURE 2.1.1 DELETED .[...].2-2," on page iii (TS page 2-2 was listed as deleted from the TSs by Amendment Nos. 247 and 243 dated February 24, 2012 (ADAMS Accession No. ML12003A133.)

- "FIGURE 3.1-1 DELETED .[...].3/4 1-3," on page iv (TS page 3/4 1-3 was listed as deleted from the TSs by Amendment Nos. 247 and 243.)

- "FIGURE 3.2-1 DELETED .[…].3/4 2-3," on page v (TS page 3/4 2-3 was listed as deleted from the TSs by Amendment Nos. 156 and 150 dated November 12, 1993 (ADAMS Accession No. ML013380219).)

- "Fire Detection Instrumentation - .[…]. (DELETED)," on page vi (The corresponding TS requirements were deleted from TSs by Amendment Nos. 159 and 153, dated February 25, 1994 (ADAMS Accession No. ML013380447.)

- "Radioactive Liquid effluent Monitoring Instrumentation (DELETED)," on page vi (The corresponding TS requirements were deleted from TSs by Amendment Nos. 188 and 182 dated July 31, 1996 (ADAMS Accession No. ML013390265.)

- "TABLE 3.3-7 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION .[…]. (DELETED)," on page vi (The corresponding TS requirements were deleted from TSs by Amendment Nos. 188 and 182.)

- "TABLE 4.3-5 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS .[…]. (DELETED)," on page vi (The corresponding TS requirements were deleted from TSs by Amendment Nos. 188 and 182.)

- "TABLE 4.8-1 NOT USED .[…]. 3/4 8-10," on page xi (TS page 3/4 8-10 was designated as not used by Amendment Nos. 181 and 175 dated December 28, 1995 (ADAMS Accession No. ML013390070.)

The NRC staff determined that these changes are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content or operation requirements. These changes do not approve any design changes and do not affect current requirements for Turkey Point 3 and 4. Therefore, the NRC staff finds that the changes are acceptable.

3.3.b Changes to TSs Definitions and Corresponding Index Listings

In Section 4.1.2 of its application dated January 29, 2013, the licensee proposed to delete Definition 1.24, "REPORTABLE EVENT," and its corresponding index listing because the term is no longer used in the TSs. The NRC staff's review determined that the term, "REPORTABLE EVENT," is not referenced in the TSs, other than in Section 1, "DEFINITIONS." Section 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," of 10 CFR Part 50 and 10 CFR 50.73, "Licensee Event Report System," require the licensee to make reports to the NRC whenever the conditions in 10 CFR 50.72 and 10 CFR 50.73 are met. Because these regulations require the licensee to notify the NRC of reportable events, the NRC staff finds that it is not necessary for the licensee to have Definition 1.24, "REPORTABLE EVENT," in its TSs.

In Section 4.1.3 of its application dated January 29, 2013, the licensee proposed to delete Definition 1.27, "SOLIDIFICATION," and its corresponding index listing because the applicable section for which it provided a definition was removed from the TSs by Amendments 201/195

(ADAMS Accession No. ML013390567). The NRC staff's review determined that Definition 1.27, "SOLIDIFICATION," is not referenced in the TSs other than in Section 1, "DEFINITIONS," and that this change would remove information that is no longer referenced in TSs.

In Section 4.1.4 of its application dated January 29, 2013, the licensee proposed to delete Definition 1.34, "VENTILATION EXHAUST TREATMENT SYSTEM," and its corresponding index listing because the section for which it provided a definition was removed from the TSs by License Amendments 188 and 182 dated July 31, 1996 (ADAMS Accession No. ML013390265). The NRC determined that Definition 1.34, "VENTILATION EXHAUST TREATMENT SYSTEM," is not referenced in the TSs other than in Section 1, "DEFINITIONS," and that this change would remove information that is no longer referenced in TSs.

As a result of these proposed deletions, the licensee proposed to sequentially renumber Definitions 1.25, 1.26, 1.28 through 1.33, and 1.35 in the index and Section 1, "DEFINITIONS," as shown in Attachment 1 of its application dated January 29, 2013. The NRC determined that these conforming changes are editorial and do not affect current requirements. In Attachment 1 of its application dated January 29, 2013, the licensee proposed to relocate the title, "FREQUENCY NOTATION," within Section 1 of the TSs, to correspond to Definition 1.14, which defines FREQUENCY NOTATION. The title is currently located before Definition 1.12 for DOSE EQUIVALENT I-131. The NRC staff's review determined that this change would correct an editorial error.

The NRC staff determined that these changes are editorial, corrective in nature, or involve the reorganization or reformatting of requirements without affecting technical content or operation requirements. These changes do not approve any design changes and do not affect current requirements for Turkey Point 3 and 4. Therefore, the NRC staff finds that the changes are acceptable.

3.3.c Revisions to TSs Pages

In the revised Section 4.1.19 of its application supplement dated May 30, 2013, the licensee proposed to move the following text within the TSs.

- The text, "3/4.4.10 DELETED," from page 3/4 4-38 to page 3/4 4-37 [Because of the proposed deletion of various TS pages as described in the following paragraphs, the licensee proposed to move the subject text to new TS page 3/4 4-28.]

- The text, "3/4.6.3 Not Used," from page 3/4 6-16 to page 3/4 6-15 [The licensee proposed to change the title of this section to "Deleted" in the TS index. Therefore, the text within the TS would change from "3/4.6.3 Not Used" to "3/4.6.3 DELETED."]

- The text, "3/4.9.7 DELETED," from page 3/4 9-7 to page 3/4 9-6

- The text, "3/4.9.12 DELETED," from page 3/4 9-13 to page 3/4 9-12 [Because of the proposed deletion of various TS pages as described in the following paragraphs, the licensee proposed to move the subject text to new TS page 3/4 9-11.]

- The text, "3/4.10.4 (This specification number is not used)" from page 3/4 10-4 to page 3/4 10-3
- The text in Sections 5.3 and 5.4 from page 5-4 to page 5-1
- The text, "6.6 DELETED," and "6.7 DELETED," from page 6-12 to page 6-5 [Because of the proposed deletion of various TS pages as described in the following paragraphs, the licensee proposed the subject text to be moved to new TS page 6-4.]
- The text, "6.10 DELETED," from page 6-23 to page 6-22a [Because of the proposed deletion of various TS pages as described in the following paragraphs, the licensee proposed the subject text to be moved to new TS page 6-19.]
- The text in Sections 6.12.2 and 6.13 from page 6-25 to page 6-24 [Because of the proposed deletion of various TS pages as described in the following paragraphs, the licensee proposed the subject text to be moved to new TS page 6-20.]

In its application supplements dated May 30, 2013, and October 22, 2013, the licensee proposed to delete the following TS pages. The NRC staff notes these pages are marked as deleted, not used, or left intentionally blank; contain text that the licensee proposed to be moved to other pages; or contain obsolete requirements related to Amendment 154 to DPR-31 and its administrative correction, dated June 15, 1993, and June 23, 1995, respectively (ADAMS Accession Nos. ML013380295 and ML013380239, respectively).

2-2	3/4 3-40a	3/4 4-34	3/4 10-4	6-6
3/4 1-3	3/4 4-12	3/4 4-38	5-2	6-7
3/4 1-16	3/4 4-13	3/4 6-3	5-3	6-8
3/4 1-27	3/4 4-14	3/4 6-16	5-4	6-9
3/4 2-3	3/4 4-15	3/4 6-19	5-8	6-10
3/4 2-5	3/4 4-16	3/4 6-20	5-9	6-11
3/4 2-6a	3/4 4-17	3/4 6-21	5-11	6-12
3/4 2-9a	3/4 4-27	3/4 8-10	5-12	6-23
3/4 2-10a	3/4 4-29	3/4 9-7	5-17	6-25
3/4 2-12a	3/4 4-33	3/4 9-13	6-3	6-27

In Section 4.1.20 of its application supplement dated May 30, 2013, the licensee proposed to renumber various TS pages. The NRC staff notes that the proposed page numbering reflects the deletion of blank pages and obsolete requirements, and the shift of current requirements to other pages. The NRC staff finds that the page renumbering does not change any requirements.

The NRC staff determined that these changes are editorial, corrective in nature, or involve the reorganization or reformatting of requirements without affecting technical content or operation requirements. These changes do not approve any design changes and do not affect current requirements for Turkey Point 3 and 4. The NRC staff finds that the changes are acceptable.

3.4 Miscellaneous Changes to the Renewed Facility Operating Licenses

3.4.a Revised References to the Physical Security Plan

In Sections 4.2.1 and 4.2.7 of its application dated January 29, 2013, the licensee proposed to modify Condition 3.E of the Turkey Point 3 and 4 RFOLs. The licensee proposed to update the title of its physical security plan referenced in Condition 3.E from "Florida Power and Light & FPL Energy Seabrook Physical Security Plan, Training and Qualification Plan and Safeguards Contingency Plan – Revision 3" to "Florida Power & Light Turkey Point Nuclear Plant Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program – Revision 15." The licensee proposed to modify the physical security plan submittal date referenced in Condition 3.E from May 18, 2006, to August 3, 2012. The NRC staff finds that the proposed revisions reflect the title and date of the physical security plan submitted to the NRC on August 3, 2012 (ADAMS Accession No. ML12220A353), the enclosures to which contained safeguards information; therefore, the NRC withheld these enclosures from public disclosure. In its letter dated July 16, 2013 (ADAMS Accession No. ML13191A085), the NRC staff stated that it found that the licensee properly concluded that the reported changes to its physical security plan did not result in a decrease in safeguards effectiveness. The NRC staff determined that the change in title of the physical security plan and its submittal date in the license conditions updates the conditions to the newer version of the security plan. Therefore, the NRC staff finds the proposed changes to the license conditions acceptable.

3.4.b Deletion of DPR-31 Condition 3.J and DPR-41 Condition 3.K on EPU Modifications

In Sections 4.2.4 and 4.2.11 of its application dated January 29, 2013, the licensee proposed to delete RFOL DPR-31 License Condition 3.J and RFOL DPR-41 License Condition 3.K. Both license conditions, which are titled, "Extended Power Uprate Modifications," require the licensee to provide confirmation to the NRC staff that the design and structural integrity evaluations associated with the spent fuel pool supplemental heat exchanger modifications are complete and that the results demonstrate compliance with appropriate UFSAR and code requirements. The licensee stated that it satisfied these license conditions when it submitted the requested design information via letters L-2012-143, L-2012-179, and L-2012-318 dated June 19, July 13, and August 10, 2012, respectively (ADAMS Accession Nos. ML12177A054, ML12199A010, and ML12227A684, respectively). The licensee stated that the NRC closed these items via letters dated August 2, 2012 (ADAMS Accession No. ML12214A303), for Unit 3, and September 13, 2012 (ADAMS Accession No. ML12248A082), for Unit 4. The NRC staff finds that these license conditions were met because the licensee submitted the requested information. Therefore, the NRC staff finds the proposed deletions acceptable.

3.4.c Renumbering of License Conditions

In Sections 4.2.5 and 4.2.12 of its application dated January 29, 2013, the licensee proposed to renumber RFOL DPR-31, License Condition 3.K, and RFOL DPR-41, License Condition 3.L, "Pad TCD Safety Analysis," to 3.H in both RFOLs to account for the proposed deletions of License Conditions 3.H, 3.I, and 3.J for Unit 3 and License Conditions 3.H, 3.I, 3.J, and 3.K for Unit 4, for which NRC's evaluations were previously documented in this safety evaluation. The

NRC staff determined that these formatting changes are editorial and do not change requirements. Therefore, the NRC staff finds the proposed changes acceptable.

3.4.d RFOL Page Deletions

In Section 4.2.14 of its application dated January 29, 2013, the licensee proposed that for RFOL DPR-31, the remaining text on page 7 be moved to pages 5 and 6 and that page 7 be deleted. For RFOL FPR-41, the licensee proposed that the remaining text on pages 7 and 8 be moved to pages 5 and 6 and that pages 7 and 8 be deleted. The NRC staff determined these formatting changes are editorial and do not change requirements. Therefore, the NRC staff finds the proposed changes acceptable.

4.0 STATE CONSULTATION

By letter dated May 2, 2003 (ADAMS Accession No. ML032470912), from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, NRC, the State of Florida indicated it does not desire notification of issuance of license amendments. By electronic mail dated July 25, 2012 (ADAMS Accession No. ML12208A014), from Cynthia Becker of the Florida Department of Health, Bureau of Radiation Control, to Farideh E. Saba, Senior Project Manager, NRC, the State of Florida confirmed that the May 2003 letter continues to reflect the State's position on notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff 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. By *Federal Register* notices dated January 8, 2013 (78 FR 1271), and April 16, 2013 (78 FR 22569), the Commission previously issued proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on these findings. The amendments also make editorial, corrective, or other minor revisions to the TSs and renewed facility operating licenses. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10)(v). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

Based on the aforementioned considerations, the NRC staff has 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Kristy Bucholtz
Roberto Torres
Tony Nakanishi

Date: June 13, 2014

The 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. 260 to DPR-31
2. Amendment No. 255 to DPR-41
3. Safety Evaluation

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ADAMS Accession No.: ML13329A092 ***Via Email**

OFFICE	LPLII-2/PM	LPLII-2/LA	DSS/SBPB/BC*	DSS/SCVB/BC*	DSS/SRXB/BC*
NAME	AKlett	BClayton	GCasto	RDennig	CJackson
DATE	03/21/14	04/11/14	03/18/14	01/24/14	03/18/14
OFFICE	DSS/STSB/BC	OGC - NLO	LPLII-2/BC (A)	LPLII-2/PM	
NAME	RElliott	MYoung	LRegner (FSaba for)	AKlett	
DATE	03/20/14	6/12/14	6/12/14	6/13/14	

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