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

October 31, 2008

Mr. Edward D. Halpin
Chief Nuclear Officer
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO RELOCATE SURVEILLANCE TEST INTERVALS TO LICENSEE-CONTROLLED PROGRAM (RISK-INFORMED INITIATIVE 5-b) (TAC NOS. MD7058 AND MD7059)

Dear Mr. Halpin:

The Commission has issued the enclosed Amendment No. 188 to Facility Operating License No. NPF-76 and Amendment No. 175 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to the STP Nuclear Operating Company submittal dated October 23, 2007, as supplemented by letter dated May 20, 2008.

The amendments revise the TSs to relocate surveillance frequencies of most surveillance tests from the TS to a licensee-controlled surveillance frequency control program. Once relocated, the surveillance frequency changes are permitted based on the risk-informed methodology as specified in the Administrative Controls section of the TS.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 188 to NPF-76
2. Amendment No. 175 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

cc:

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)* acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated October 23, 2007, as supplemented by letter dated May 20, 2008, 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, as amended, 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 license 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.

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*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 188, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-76 and the Technical Specifications

Date of Issuance: October 31, 2008

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 175
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)* acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated October 23, 2007, as supplemented by letter dated May 20, 2008, 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, as amended, 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 license 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.

---

*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-80 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 175, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-80 and the Technical Specifications

Date of Issuance: October 31, 2008

ATTACHMENT TO LICENSE AMENDMENT NOS. 188 AND 175

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

Replace the following pages of the Facility Operating Licenses, Nos. NPF-76 and NPF-80, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-76

REMOVE	INSERT
-4-	-4-

Facility Operating License No. NPF-80

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT	REMOVE	INSERT	REMOVE	INSERT
vi	vi	3/4 4-6	3/4 4-6	3/4 7-13	3/4 7-13
3/4 1-17	3/4 1-17	3/4 4-9	3/4 4-9	3/4 7-14	3/4 7-14
3/4 1-21	3/4 1-21	3/4 4-11	3/4 4-11	3/4 7-16	3/4 7-16
3/4 1-22	3/4 1-22	3/4 4-19	3/4 4-19	3/4 7-17	3/4 7-17
3/4 2-2	3/4 2-2	3/4 4-21	3/4 4-21	3/4 7-33	3/4 7-33
3/4 2-10	3/4 2-10	3/4 4-29	3/4 4-29	3/4 8-3	3/4 8-3
3/4 2-11	3/4 2-11	3/4 4-31	3/4 4-31	3/4 8-4	3/4 8-4
3/4 3-1	3/4 3-1	3/4 4-38	3/4 4-38	3/4 8-6	3/4 8-6
3/4 3-11	3/4 3-11	3/4 4-39	3/4 4-39	3/4 8-7	3/4 8-7
3/4 3-12	3/4 3-12	3/4 5-1	3/4 5-1	3/4 8-11	3/4 8-11
3/4 3-13	3/4 3-13	3/4 5-4	3/4 5-4	3/4 8-13a	3/4 8-13a
3/4 3-14	3/4 3-14	3/4 5-5	3/4 5-5	3/4 8-13c	3/4 8-13c
3/4 3-15	3/4 3-15	3/4 5-7	3/4 5-7	3/4 8-15	3/4 8-15
3/4 3-17	3/4 3-17	3/4 5-8	3/4 5-8	3/4 8-16	3/4 8-16
3/4 3-42	3/4 3-42	3/4 5-10	3/4 5-10	3/4 9-1	3/4 9-1
3/4 3-43	3/4 3-43	3/4 5-11	3/4 5-11	3/4 9-2	3/4 9-2
3/4 3-44	3/4 3-44	3/4 6-1	3/4 6-1	3/4 9-8	3/4 9-8
3/4 3-45	3/4 3-45	3/4 6-6	3/4 6-6	3/4 9-9	3/4 9-9
3/4 3-46	3/4 3-46	3/4 6-7	3/4 6-7	3/4 9-11	3/4 9-11
3/4 3-47	3/4 3-47	3/4 6-8	3/4 6-8	3/4 9-12	3/4 9-12
3/4 3-48	3/4 3-48	3/4 6-13	3/4 6-13	3/4 9-13	3/4 9-13
3/4 3-49	3/4 3-49	3/4 6-14	3/4 6-14	3/4 9-17	3/4 9-17
3/4 3-61	3/4 3-61	3/4 6-15	3/4 6-15	3/4 10-1	3/4 10-1
3/4 3-67	3/4 3-67	3/4 6-16	3/4 6-16	3/4 10-2	3/4 10-2
3/4 3-73	3/4 3-73	3/4 6-17	3/4 6-17	3/4 10-3	3/4 10-3
3/4 3-85	3/4 3-85	3/4 7-5	3/4 7-5	3/4 10-4	3/4 10-4
3/4 4-1	3/4 4-1	3/4 7-6	3/4 7-6	3/4 11-1	3/4 11-1
3/4 4-2	3/4 4-2	3/4 7-8	3/4 7-8	3/4 11-3	3/4 11-3
3/4 4-4	3/4 4-4	3/4 7-12	3/4 7-12	--	6-12e
3/4 4-5	3/4 4-5				

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 188, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SER)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Safety Parameter Display System (Section 18, SSER No. 4)*

Before startup after the first refueling outage, HL&P[**] shall perform the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to issues as described in Section 18.2 of SER Supplement 4.

(6) Supplementary Containment Purge Isolation (Section 11.5, SSER No. 4)

HL&P shall provide, prior to startup from the first refueling outage, control room indication of the normal and supplemental containment purge sample line isolation valve position.

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

** The original licensee authorized to possess, use and operate the facility was HL&P. Consequently, historical references to certain obligations of HL&P remain in the license conditions.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 175 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SR)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) License Transfer

Texas Genco, LP shall provide decommissioning funding assurance, to be held in decommissioning trusts for South Texas Project, Unit 2 (Unit 2) upon the direct transfer of the Unit 2 license to Texas Genco, LP, in an amount equal to or greater than the balance in the Unit 2 decommissioning trust immediately prior to the transfer. In addition, Texas Genco, LP shall ensure that all contractual arrangements referred to in the application for approval of the transfer of the Unit 2 license to Texas Genco, LP to obtain necessary decommissioning funds for Unit 2 through a non-bypassable charge are executed and will be maintained until the decommissioning trusts are fully funded, or shall ensure that other mechanisms that provide equivalent assurance of decommissioning funding in accordance with the Commission's regulations are maintained.

(6) License Transfer

The master decommissioning trust agreement for Unit 2, at the time the direct transfer of Unit 2 to Texas Genco, LP is effected and thereafter, is subject to the following:

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

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ACTION (Continued)

c) A core power distribution measurement is obtained and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits as specified in the COLR. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours 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 a frequency in accordance with the Surveillance Frequency Control Program.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.8 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 561°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. At a frequency in accordance with the Surveillance Frequency Control Program.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn, as specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

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

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

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

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

b. At a frequency in accordance with the Surveillance Frequency Control Program.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
**With K_{eff} greater than or equal to 1.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

3. greater than 15%, but less than 50% of RATED THERMAL POWER:

   THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour cumulative penalty deviation during the previous 24 hours.

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the target band, and the indicated AFD has not been outside of the target band for more than 1 hour cumulative penalty deviation during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at a frequency in accordance with the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE.

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

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference

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: With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours 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 Setpoint within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ (by F_{xy} evaluation) and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

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 a frequency in accordance with the Surveillance Frequency Control Program when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the 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 measuring core power distribution to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by using:

a. The Power Distribution Monitoring System (PDMS), or

b. The movable incore detectors by either:

1. Using the four pairs of symmetric thimble locations, or

2. Using the movable incore detection system to monitor the QUADRANT POWER TILT RATIO with a full incore map.

* See Special Test Exceptions Specification 3.10.2.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System $T_{avg} \leq$ the limit as specified in the Core Operating Limits Report

b. Pressurizer Pressure, > the limit as specified in the Core Operating Limits Report

c. Thermal Design Reactor Coolant System Flow, $\geq$ 392,000 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 than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at a frequency in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable for verification that RCS flow is within its limit.

4.2.5.2 The RCS flow rate indicators shall be subjected to a channel calibration at a frequency in accordance with the Surveillance Frequency Control Program.

> NOTE
> SR 4.2.5.3 is required at beginning-of-cycle with reactor power $\geq$ 90% RTP.

4.2.5.3 The RCS total flow rate shall be determined by precision heat balance or elbow tap ΔP measurements at a frequency in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Chapter 16 in the Updated Final Safety Analysis Report (UFSAR).

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1 and at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit at a frequency in accordance with the Surveillance Frequency Control Program. Each verification shall include at least one train such that both trains are verified at a frequency in accordance with the Surveillance Frequency Control Program and one channel per function such that all channels are verified at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(20)	CHANNEL CALIBRATION(20)	ANALOG CHANNEL OPERATIONAL TEST (19)(20)	TRIP ACTUATING DEVICE OPERATIONAL TEST(20)	ACTUATION LOGIC TEST(20)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A	(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint		(2, 4), (3, 4), (4, 6), (4, 5)	(17)	N.A.	N.A.	1, 2
b. Low Setpoint		(4)	S/U (1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	(4)	(17)	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range, Neutron Flux		(4, 5)	S/U (1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux		(4, 5)	S/U (1), (9) (17)	N.A.	N.A.	2**, 3, 4, 5
7. Extended Range, Neutron Flux		(4)	(12, 17)	N.A.	N.A.	3, 4, 5
8. Overtemperature ΔT			(17)	N.A.	N.A.	1, 2
9. Overpower ΔT			(17)	N.A.	N.A.	1, 2
10. Pressurizer Pressure--Low			(17)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(20)	CHANNEL CALIBRATION(20)	ANALOG CHANNEL OPERATIONAL TEST (19)(20)	TRIP ACTUATING DEVICE OPERATIONAL TEST(20)	ACTUATION LOGIC TEST(20)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. Pressurizer Pressure --High			(17)	N.A.	N.A.	1, 2
12. Pressurizer Water Level --High			(17)	N.A.	N.A.	1
13. Reactor Coolant Flow --Low			(17, 18)	N.A.	N.A.	1
14. Steam Generator Water Level--Low-Low			(17, 18)	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.		N.A.	(17)	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.		N.A.	(17)	N.A.	1
17. Turbine Trip						
a. Low Emergency Trip Fluid Pressure	N.A.		N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.		N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESFAS	N.A.	N.A.	N.A.		N.A.	1, 2

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (20)	CHANNEL CALIBRATION(20)	ANALOG CHANNEL OPERATIONAL TEST (19)(20)	TRIP ACTUATING DEVICE OPERATIONAL TEST(20)	ACTUATION LOGIC TEST(20)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	(4)		N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	(4)		N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	(4)		N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	(4)		N.A.	N.A.	1
e. Power Range Neutron Flux, P-10	N.A.	(4)		N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-8	N.A.			N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	(7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	(7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	(15), R(16)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

*(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

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

(5) Detector plateau curves shall be obtained and evaluated. If a low noise preamplifier is used with the Source Range Detector, no plateau curve is obtained. Instead, with the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

*(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, quarterly shall mean at least once per 92 EFPD.

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

(8) (Not Used)

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) OPERABILITY shall be verified by a check of memory devices, input accuracies, Boron Dilution Alarm setpoints, output values, and software functions.

(13) (Not used)

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(15) Local manual shunt trip prior to placing breaker in service.

(16) Automatic undervoltage trip.

(17) Each channel shall be tested at a frequency in accordance with the Surveillance Frequency Control Program.

(18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

(19) For channels with bypass test instrumentation, input relays are tested at a frequency in accordance with the Surveillance Frequency Control Program.

(20) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2 and at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at a frequency in accordance with the Surveillance Frequency Control Program. Each verification shall include at least one train so that:

a. Each logic train is verified at a frequency in accordance with the Surveillance Frequency Control Program,

b. Each actuation train is verified at a frequency in accordance with the Surveillance Frequency Control Program*, and

c. One channel per function so that all channels are verified at least once per N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

*If an ESFAS instrumentation channel is inoperable due to response times exceeding the required limits, perform an engineering evaluation to determine if the verification failure is a result of degradation of the actuation relays. If degradation of the actuation relays is determined to be the cause, increase the ENGINEERED SAFETY FEATURES RESPONSE TIME surveillance frequency such that all trains are verified at a frequency specified in the Surveillance Frequency Control Program.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)								
a. Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
d. Containment Pressure-High-1				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
e. Pressurizer Pressure-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Compensated Steam Line Pressure-Low				N.A.	N.A.	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(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
d. Containment Pressure-High-3				N.A.	N.A.	N.A.	N.A.	1, 2, 3
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
3) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
4) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Containment Ventilation Isolation								
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) RCB Purge Radioactivity-High				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4, 5*, 6*
5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
6) Phase "A" Isolation-Manual Initiation	See Item 3. a. above for Phase "A" Isolation manual initiation Surveillance Requirements.							
c. Phase "B" Isolation								
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
3) Containment Pressure--High-3				N.A.	N.A.	N.A.	N.A.	1, 2, 3
4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
d. RCP Seal Injection Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.		(8)	1, 2, 3, 4
2) Charging Header Pressure - Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Coincident with Phase "A" Isolation	See Item 3.a. above for Phase "A" surveillance requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(1)	(6)	(8)	1, 2, 3
c. Steam Line Pressure-Negative Rate-High				N.A.	N.A.	N.A.	N.A.	3
d. Containment Pressure - High-2				N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Compensated Steam Line Pressure-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(1)	(6)	(8)	1, 2, 3
b. Steam Generator Water Level-High-High (P-14)				N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Deleted								
d. Deleted								
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Turbine Trip and Feedwater Isolation (Continued)								
f. Tavg -Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)				N.A.	N.A.	N.A.	N.A.	1, 2, 3
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3
d. Steam Generator Water Level--Low-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Loss of Power	See Item 8. below for all Loss of Power Surveillance Requirements.							
7. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(6)	(6)	(8)	1, 2, 3, 4
b. RWST Level--Low-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Coincident With: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Loss of Power								
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.		N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.		N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.		N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.			N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.			N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3
10. Control Room Ventilation								
a. Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Control Room Ventilation (Continued)								
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(6)	N.A.	N.A.	1, 2, 3, 4
d. Control Room Intake Air Radioactivity-High				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
e. Loss of Power	See Item 8. above for all Loss of Power Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(1) Each train shall be tested at the frequency specified in the Surveillance Frequency Control Program.

(2) Deleted

(3) Deleted

(4) Deleted

(5) Deleted

(6) Each actuation train shall be tested at a frequency in accordance with the Surveillance Frequency Control Program. Testing of each actuation train shall include master relay testing of both logic trains. If an ESFAS instrumentation channel is inoperable due to failure of the Actuation Logic Test and/or Master Relay Test, increase the surveillance frequency such that each train is tested at the frequency specified in the Surveillance Frequency Control Program unless the failure can be determined by performance of an engineering evaluation to be a single random failure.

(7) For channels with bypass test instrumentation, input relays are tested at a frequency in accordance with the Surveillance Frequency Control Program.

(8) The test interval is R for Potter & Brumfield MDR Series slave relays.

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

* During CORE ALTERATIONS or movement of irradiated fuel within containment.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System Functions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one or more required channels of one or more Remote Shutdown System Functions inoperable, restore the inoperable Function(s) to OPERABLE status within 30 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE: Separate condition entry is allowed for each Function.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each normally energized Remote Shutdown System monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit including the actuated components, shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.3.5.3 Each Remote Shutdown System required instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CALIBRATION at a frequency in accordance with the Surveillance Frequency Control Program. [NOTE: Neutron detectors and reactor trip breaker indication are excluded from CHANNEL CALIBRATION.]

NOTE: This corrected page was issued by letter dated February 8, 2005

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

As shown in Table 3.3-10.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at a frequency in accordance with the Surveillance Frequency Control Program.

TABLE 4.3-7 has been deleted.

Page 3/4 3-74 has been deleted.

3/4.3.5 ATMOSPHERIC STEAM RELIEF VALVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION:

3.3.5.1 The atmospheric steam relief valve instrumentation shown in Table 3.3-14 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-14

ACTION: As shown in Table 3.3-14

SURVEILLANCE REQUIREMENTS:

4.3.5.1 Perform a CHANNEL CHECK on each atmospheric steam relief valve automatic actuation channel at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.5.2 Perform a CHANNEL CALIBRATION on each atmospheric steam relief valve automatic actuation channel at a nominal setpoint of 1225 psig ± 7 psi at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.5.3 Perform an ANALOG CHANNEL OPERATIONAL TEST on each atmospheric steam relief valve automatic actuation channel at a nominal setpoint of 1225 psig ± 7 psi at a frequency in accordance with the Surveillance Frequency Control Program.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

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

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

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

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

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

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

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

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

4.4.1.3.3 At least one reactor coolant loop, or one RHR loop with valve CV0198 locked or pinned in position to limit flow to 125 gpm shall be verified in operation and circulating reactor coolant at a frequency in accordance with the Surveillance Frequency Control Program.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop with valve CV0198 locked or pinned in position to limit flow to 125 gpm shall be OPERABLE and in operation*, and either:

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

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

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

ACTION:

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

b. With no RHR loop in operation, suspend operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.1.2 At least one RHR loop with valve CV0198 locked or pinned in position to limit flow to 125 gpm shall be determined to be in operation and circulating reactor coolant at a frequency in accordance with the Surveillance Frequency Control Program.

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* The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**Two RHR loops may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2

a. At least two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation**, and

b. Each valve or mechanical joint used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

b. With no RHR loop in operation, suspend all operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required RHR loop to operation.

c. With a valve or mechanical joint used to isolate unborated water sources not secured in the closed position, immediately suspend all operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN specified in the Core Operating Limits Report (COLR) and initiate action to secure the valve(s) or joint(s) in the losed position and within 4 hours verify the SHUTDOWN MARGIN is within limits specified in the COLR. The required action to verify the SHUTDOWN MARGIN within limits must be completed whenever ACTION c is entered. A separate ACTION entry is allowed for each unsecured valve or mechanical joint.

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.2.2 Each valve or mechanical joint used to isolate unborated water sources shall be verified closed and secured in position at a frequency in accordance with the Surveillance Frequency Control Program.

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*Two RHR loops may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1816 cubic feet, and at least two groups of pressurizer heaters supplied by ESF power each having a capacity of at least 175 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters supplied by ESF power shall be verified by energizing the heaters and measuring circuit current at a frequency in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by:

a. Performing a CHANNEL CALIBRATION on the actuation channel, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed in accordance with the ACTIONS of Specification 3.4.4.

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 Instrumentation shall be OPERABLE:

a. One Containment Atmosphere Radioactivity Monitor (particulate channel), and

b. The Containment Normal Sump Level and Flow Monitoring System.

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

ACTION:

a. With the required containment atmosphere radioactivity monitor inoperable perform the following actions or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

1) Restore the containment atmosphere monitor (particulate channel) to OPERABLE status within 30 days and,

2) Obtain and analyze a grab sample of the containment atmosphere for particulate radioactivity at least once per 24 hours, or

3) Perform a Reactor Coolant System water inventory balance at least once per 24 hours.

b. With the required containment normal sump level and flow monitoring system inoperable perform the following actions or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

1) Restore the containment normal sump and flow monitoring system to OPERABLE status within 30 days and,

2) Perform a Reactor Coolant System water inventory balance at least once per 24 hours.

c. With both a. and b. inoperable, enter 3.0.3.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Monitoring (particulate channel) performance of the following:

1) CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program, and
2) CHANNEL CALIBRATION at a frequency in accordance with the Surveillance Frequency Control Program

b. Containment Normal Sump Level and Flow Monitoring System performance of CHANNEL CALIBRATION at a frequency in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.1 Note: this requirement is not applicable to primary-to-secondary leakage (refer to 4.4.6.2.3).

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

a. Monitoring the containment atmosphere particulate radioactivity channel at a frequency in accordance with the Surveillance Frequency Control Program;

b. Monitoring the containment normal sump inventory and discharge at a frequency in accordance with the Surveillance Frequency Control Program;

c. Performing a Reactor Coolant System water inventory balance at a frequency in accordance with the Surveillance Frequency Control Program; and [(1)]

d. Monitoring the Reactor Head Flange Leakoff System at a frequency in accordance with the Surveillance Frequency Control Program.

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

a. At a frequency in accordance with the Surveillance Frequency Control Program,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

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

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve except for valves XRH0060 A, B, C, and XRH0061 A, B, C.

4.4.6.2.3 Primary-to-secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator at a frequency in accordance with the Surveillance Frequency Control Program. [(1)]

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

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(1) Not required to be performed until 12 hours after establishment of steady state operation.

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. Gross Radioactivity Determination	At a frequency in accordance with the Surveillance Frequency Control Program.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At a frequency in accordance with the Surveillance Frequency Control Program.	1
3. Radiochemical for $\bar{E}$ Determination*	At a frequency in accordance with the Surveillance Frequency Control Program**	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	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

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 10°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 a frequency in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

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 a frequency in accordance with the Surveillance Frequency Control Program thereafter when the PORV is required OPERABLE;

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

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

4.4.9.3.2 The RCS vent(s) shall be verified to be open at a frequency in accordance with the Surveillance Frequency Control Program[4] when the vent(s) is being used for overpressure protection.

4.4.9.3.3 The positive displacement pump shall be demonstrated inoperable[5] at a frequency in accordance with the Surveillance Frequency Control Program, except when the reactor vessel head is removed or when both centrifugal charging pumps are inoperable and secured, by verifying that the motor circuit breakers are secured in the open position.[2]

4.4.9.3.4 Verify at a frequency in accordance with the Surveillance Frequency Control Program that only one centrifugal charging pump is capable of injecting into the RCS[5], except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.[2]

4.4.9.3.5 Verify at a frequency in accordance with the Surveillance Frequency Control Program that each ECCS accumulator is isolated.

SPECIFICATION NOTATIONS

[1] ECCS accumulator isolation is required only when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by Figures 3.4-2 and 3.4-3.

[2] An inoperable centrifugal charging pump(s) and/or positive displacement charging pump may be energized for testing or pump switching provided the discharge of the pump(s) has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position. Reactor coolant pump seal injection flow may be maintained during the RCS isolation process.

[3] This ACTION may be suspended for up to 7 days to allow functional testing to verify PORV operability. During this test period, operation of systems or components which could result in an RCS mass or temperature increase will be administratively controlled. During the ASME stroke testing of two inoperable PORVS, cold overpressurization mitigation will be provided by two RHR discharge relief valves associated with two OPERABLE and operating RHR loops which have the auto closure interlock bypassed [or deleted]. If one PORV is inoperable, cold overpressure mitigation will be provided by the OPERABLE PORV and one RHR discharge relief valve associated with an OPERABLE and operating RHR loop which has the auto closure interlock bypassed [or deleted].

[4] Except when the vent pathway is provided with a valve that is locked, sealed, or otherwise secured in the open position, then verify these valves open at a frequency in accordance with the Surveillance Frequency Control Program.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be ultrasonically examined over the volume from the inner bore of the flywheel to the circle of one-half the outer radius at a frequency in accordance with the Surveillance Frequency Control Program and shall comply with regulatory positions C.4.b (3), (4), and (5) of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Safety Injection System accumulator shall be OPERABLE

APPLICABILITY: MODES 1 and 2
MODE 3 with pressurizer pressure > 1000 psig

ACTION:

a. With one accumulator inoperable, except as a result of boron concentration outside the required limits, within 24 hours restore the inoperable accumulator to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With more than one accumulator inoperable, except as a result boron concentration outside the required limits, within 1 hour restore at least two accumulators to OPERABLE status or apply the requirements of the GRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With the boron concentration of one accumulator outside the required limit, within 72 hours restore the boron concentration to within the required limits or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

d. With the boron concentrations of more than one accumulator outside the required limit, within 1 hour restore the boron concentration of at least two accumulators to within the required limits or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 .l Each accumulator shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying the contained borated water volume is ≥ 8800 gallons and ≤ 9100 gallons and nitrogen cover-pressure is ≥ 590 psig and ≤ 670 psig, and

2) Verifying that each accumulator isolation valve is open.

b. At a frequency in accordance with the Surveillance Frequency Control Program and within 6 hours* after each solution volume increase of greater than or equal to 1% of tank volume that is not the result of addition from the RWST by verifying the boron concentration of the accumulator solution is ≥ 2700 ppm and ≤ 3000 ppm, and

c. At a frequency in accordance with the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is removed.

* The 6 hr. SR is only required to be performed for affected accumulators.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number		Valve Function	Valve Position
XSI0008	A, B, C	High Head Hot Leg Recirculation Isolation	Closed
XRH0019	A, B, C	Low Head Hot Leg Recirculation Isolation	Closed

b. At a frequency in accordance with the Surveillance Frequency Control Program by:

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

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

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At a frequency in accordance with the Surveillance Frequency Control Program by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At a frequency in accordance with the Surveillance Frequency Control Program, during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on an Automatic Switchover to Containment Sump test signal, and

   2) Verifying that each of the following pumps starts automatically upon receipt of a Safety Injection test signal:

      a) High Head Safety Injection pump, and

      b) Low Head Safety Injection pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

   1) High Head Safety Injection pump $\geq$ 1480 psid, and

   2) Low Head Safety Injection pump $\geq$ 286 psid.

g. By performing a flow test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

   1) For High Head Safety Injection pump lines, with the High Head Safety Injection pump running, the pump flow rate is greater than 1470 gpm and less than 1620 gpm.

   2) For Low Head Safety Injection pump lines, with the Low Head Safety Injection pump running, the pump flow rate is greater than 2550 gpm and less than 2800 gpm.

ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

SURVEILLANCE REQUIREMENTS

4.5.3.1.1 The ECCS components shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.1.2 All High Head Safety Injection pumps, except the above allowed OPERABLE pumps, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first, and at a frequency in accordance with the Surveillance Frequency Control Program thereafter.

* An inoperable pump may be energized for testing or for filling accumulators provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

ECCS SUBSYSTEMS - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.5.3.2 All High Head Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all High Head Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All High Head Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at a frequency in accordance with the Surveillance Frequency Control Program.

* An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 458,000 gallons, and

b. A boron concentration between 2800 ppm and 3000 ppm.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by:

a. Verifying the contained borated water volume in the tank, and

b. Verifying the boron concentration of the water.

3/4.5.6 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.6 Three independent Residual Heat Removal (RHR) loops shall be OPERABLE with each loop comprised of:

a. One OPERABLE RHR pump,

b. One OPERABLE RHR heat exchanger, and

c. One OPERABLE flowpath capable of taking suction from its associated RCS hot leg and discharging to its associated RCS cold leg.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one RHR loop inoperable, within 7 days restore the required loop to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two RHR loops inoperable, within 24 hours restore at least two RHR loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three RHR loops inoperable, immediately initiate corrective action to restore at least one RHR loop to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.5.6.1 Each RHR loop shall be demonstrated OPERABLE on a STAGGERED TEST BASIS pursuant to the requirements of Specification 4.0.5.

4.5.6.2 At a frequency in accordance with the Surveillance Frequency Control Program by verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that:

a. With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 350 psig, the interlocks prevent the valves from being opened.

*Valves MOV-0060 A, B, and C and MOV-0061 A, B, and C may have power removed to support the FHAR (Fire Hazard Analysis Report) assumptions.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Specification 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than at a frequency in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program.

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

c. By verifying at a frequency in accordance with the Surveillance Frequency Control Program that the instrument air pressure in the header to the personnel airlock seals is $\geq$ 90 psig.

d. By verifying the door seal pneumatic system OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by conducting a seal pneumatic system leak test and verifying one of the following:

   1) That system pressure does not decay more than 1.5 psi from 90 psig minimum within 24 hours, or

   2) That system pressure does not decay more than .50 psi from 90 psig minimum within 8 hours.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and +0.3 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 a frequency in accordance with the Surveillance Frequency Control Program.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 110°F.

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

ACTION:

With the containment average air temperature greater than 110°F, reduce the average air temperature to within the 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 a minimum of four RCFC inlet temperatures and shall be determined at a frequency in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed at a frequency in accordance with the Surveillance Frequency Control Program.

4.6.1.7.2 At a frequency in accordance with the Surveillance Frequency Control Program, the inboard and outboard isolation valves with resilient material seals in each sealed closed 48-inch containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.

4.6.1.7.3 At a frequency in accordance with the Surveillance Frequency Control Program each 18-inch supplementary containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.01 L_a when pressurized to P_a.

4.6.1.7.4 At a frequency in accordance with the Surveillance Frequency Control Program each 18-inch supplementary containment purge supply and exhaust isolation valve shall be verified to be closed or open in accordance with Specification 3.6.1.7.b.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Three independent Containment Spray Systems shall be OPERABLE with each Spray system capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

a. With one Containment Spray System inoperable, within 7 days restore the inoperable Spray System to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

b. With more than one Containment Spray System inoperable, within 1 hour restore at least two Spray Systems to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying on a STAGGERED TEST BASIS, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 283 psid when tested pursuant to Specification 4.0.5;

c. At a frequency in accordance with the Surveillance Frequency Control Program during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure High 3 test signal, and

2) Verifying that each spray pump starts automatically on a Containment Pressure High 3 test signal coincident with a sequencer start signal.

d. By verifying each spray nozzle is unobstructed following maintenance activities that could result in spray nozzle blockage.

RECIRCULATION FLUID PH CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The recirculation fluid pH control system shall be operable with between 11,500 lbs. (213 cu. ft.) and 15,100 lbs (252 cu. ft.) of trisodium phosphate (w/12 hydrates) available in the storage baskets in the containment.

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

ACTION:

With the amount of trisodium phosphate outside the specified range, restore the system to the correct amount within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the system to the correct amount within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 At a frequency in accordance with the Surveillance Frequency Control Program, the recirculation fluid pH control system shall be demonstrated operable by visually verifying that:

a. 6 trisodium phosphate storage baskets are in place, and

b. have maintained their integrity, and

c. are filled with trisodium phosphate such that the level is within the specified range.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Three independent groups of Reactor Containment Fan Coolers (RCFC) shall be OPERABLE with a minimum of two units in two groups and one unit in the third group.

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

ACTION:

A. With one group of the above required Reactor Containment Fan Coolers inoperable, within 7 days restore the inoperable group of RCFC to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one group of the above required Reactor Containment Fan Coolers inoperable, within 1 hour restore at least two groups of RCFC to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of Reactor Containment Fan Coolers shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Starting each non-operating fan group from the control room, and verifying that each fan group operates for at least 15 minutes, and

2) Verifying a component cooling water flow rate of greater than or equal to 1800 gpm to each cooler.

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each fan group starts automatically on a Safety Injection test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation barrier* OPERABLE in each affected penetration that is open and within 24 hours:

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

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or check valve with flow through the valve secured**, or

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

d. Apply the requirements of the CRMP.

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 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.

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at a frequency in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Containment Ventilation Isolation test signal, each purge and exhaust valve actuates to its isolation position; and

c. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position.

d. Verifying that on a Phase "A" Isolation test signal, coincident with a low charging header pressure signal, that each seal injection valve actuates to its isolation position.

4.6.3.3 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.

*An isolation barrier may either be a closed system (i.e., General Design Criteria 57 penetrations) or an isolation valve.
**A check valve may not be used to isolate an affected penetration flow path in which more than one isolation valve is inoperable or in which the isolation barrier is a closed system with a single isolation valve (i.e., General Design Criteria 57 penetration)

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying that the developed head of each motor-driven pump at the flow test point is greater than or equal to the required developed head;

2) Verifying that the developed head of the steam turbine-driven pump at the flow test point is greater than or equal to the required developed head when tested at a secondary steam supply pressure greater than 1000 psig within 72 hours after entry into MODE 3;

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that each automatic valve in the flow path is in the correct position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER.

b. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

3) Verifying that each auxiliary feedwater flow regulating valve limits the flow to each steam generator between 550 gpm and 675 gpm.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater that 30 days prior to entering MODE 2 by verifying normal flow to each steam generator.

PLANT SYSTEMS

AUXILIARY FEEDWATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The auxiliary feedwater storage tank (AFST) shall be OPERABLE with a contained water volume of at least 485,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the AFST inoperable, within 4 hours restore the AFST to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The AFST shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by verifying the contained water volume is within its limits.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	At a frequency in accordance with the Surveillance Frequency Control Program
2. Isotopic Analysis for DOSE EQUIVALENT I −131 Concentration	a) At a frequency in accordance with the Surveillance Frequency Control Program, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) At a frequency in accordance with the Surveillance Frequency Control Program, whenever the gross radioacitivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least three independent component cooling water loops shall be OPERABLE.

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

ACTION:

a. With only two component cooling water loops OPERABLE, within 7 days restore at least three loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more component cooling water loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least three component cooling water loops shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each valve outside containment (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 a frequency in accordance with the Surveillance Frequency Control Program by verifying that:

1) Each automatic valve servicing safety-related equipment or isolating the non-nuclear safety portion of the system actuates to its correct position on a Safety Injection, Loss of Offsite Power, Containment Phase "B" Isolation, or Low Surge Tank test signal, as applicable (performed during shutdown);

2) Each Component Cooling Water System pump starts automatically on a Safety Injection or Loss of Offsite Power test signal (performed during shutdown); and

3) The surge tank level instrumentation which provides automatic isolation of portions of the system is demonstrated OPERABLE by performance of a CHANNEL CALIBRATION test.

c. By verifying that each valve inside containment (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position prior to entering MODE 4 following each COLD SHUTDOWN of greater than 72 hours if not performed within the previous 31 days.

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least three independent essential cooling water loops shall be OPERABLE.

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

ACTION:

a. With only two essential cooling water loops OPERABLE, within 7 days restore at least three loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more essential cooling water loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least three essential cooling water loops shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position;

b. At a frequency in accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

   1) Each Essential Cooling Water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal, and

   2) Each Essential Cooling Water pump starts automatically on an actual or simulated signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 25.5 feet Mean Sea Level, USGS datum, and

b. An Essential Cooling Water intake temperature of less than or equal to 99°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 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION is applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by verifying the intake water temperature and water level to be within their limits.

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Three independent Control Room Makeup and Cleanup Filtration Systems shall be OPERABLE.

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

ACTION:

a. With one Control Room Makeup and Cleanup Filtration System inoperable for reasons other than condition d, within 7 days restore the inoperable system to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Control Room Makeup and Cleanup Filtration Systems inoperable for reasons other than condition d, within 72 hours restore at least two systems to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With three Control Room Makeup and Cleanup Filtration Systems inoperable for reasons other than condition d, within 12 hours restore at least one system to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. One or more Control Room Makeup and Cleanup Filtration Systems inoperable due to inoperable Control Room Envelope (CRE) boundary perform the following:

   1) immediately initiate action to implement mitigating actions, and
   2) within 24 hours verify mitigating actions ensure CRE occupant exposures to radiological, chemical and smoke hazards will not exceed limits, and
   3) within 90 days restore CRE boundary to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Makeup and Cleanup Filtration System shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 78°F;

b. At a frequency in accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers of the makeup and cleanupair filter units and verifying that the system operates for at least 10 continuous hours with the makeup filter unit heaters operating;

SURVEILLANCE REQUIREMENTS (Continued)

c. At a frequency in accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the makeup and cleanup systems satisfy the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% for HEPA filter banks and 0.10% for charcoal adsorber banks and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units;

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, meets the laboratory testing criteria of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%; and

3) Verifying a system flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%.

e. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.1 inches Water Gauge for the makeup units and 6.0 inches Water Gauge for the cleanup units while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units;

2) Verifying that on a control room emergency ventilation test signal (High Radiation and/or Safety Injection test signal), the system automatically switches into a recirculation and makeup air filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks of the cleanup and makeup units;

3/4.7.14 ESSENTIAL CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.14 At least three independent Essential Chilled Water System loops shall be OPERABLE.

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

ACTION:

a. With only two Essential Chilled Water System loops OPERABLE, within 7 days restore at least three loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more Essential Chilled Water System loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.14 The Essential Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by Specification 4.0.5, and

b. At a frequency in accordance with the Surveillance Frequency Control Program by demonstrating that the system starts automatically on a Safety Injection test signal.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program during shutdown by transferring the unit power supply from the normal circuit to each of the alternate circuits.

4.8.1.1.2 Each standby diesel generator shall be demonstrated OPERABLE: (2)(11)

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying the fuel level in its associated fuel tank,

2) Verifying the diesel starts from standby condition and accelerates to 600 rpm (nominal) in less than or equal to 10 seconds. (3) The generator voltage and frequency shall be 4160 $\pm$ 416 volts and 60 $\pm$ 1.2 Hz within 10 seconds(3) after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-off site power by itself, or

c) Simulated loss-of-offsite power in conjunction with a Safety Injection test signal, or

d) A Safety Injection test signal by itself.

3) Verifying the generator is synchronized, loaded to 5000 to 5500 kW, and operates with a load of 5000 to 5500 kW for at least 60 minutes, (4)(6) and

4) Verifying the standby diesel generator is aligned to provide standby power to the associated emergency busses.

b. At a frequency in accordance with the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from its associated fuel tank;

c. Maintain properties of new and stored fuel oil in accordance with the Fuel Oil Monitoring Program.

SURVEILLANCE REQUIREMENTS (Continued)

d. Deleted

e. At a frequency in accordance with the Surveillance Frequency Control Program, during shutdown, by:

1) Deleted

2) Verifying the generator capability to reject a load of greater than or equal to 785.3 kW while maintaining voltage at 4160 ± 416 volts and frequency at 60 4.5 Hz; (4)(5)

3) Verifying the generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 5262 volts during and following the load rejection; (4)(5)

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

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

b) Verifying the diesel starts on the auto-start signal within 10 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 shutdown loads. After energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on a Safety Injection 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 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the autostart 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 a Safety Injection test signal, and:

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

b) Verifying the diesel starts on the auto-start signal within 10 seconds, energizes the auto-connected ESF (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator

SURVEILLANCE REQUIREMENTS (Continued)

13) Demonstrating the OPERABILITY of the automatic load shed bypass and the manual load shed reinstatement features of the load sequencer.

f. At a frequency in accordance with the Surveillance Frequency Control Program or after any modifications which could affect standby diesel generator interdependence by starting all standby diesel generators simultaneously, during shutdown, and verifying that all standby diesel generators accelerate to at least 600 rpm in less than or equal to 10 seconds; and

g. At a frequency in accordance with the Surveillance Frequency Control Program by draining each fuel tank, removing the accumulated sediment and cleaning the tank.

4.8.1.1.3 (Not used)

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

(Not used)

SPECIFICATION NOTATIONS

(1) Loss of one 13.8 kV Standby Bus to 4.16 kV ESF bus line constitutes loss of one offsite source. Loss of two 13.8 kV Standby busses to 4.16 kV ESF bus lines constitutes loss of two offsite sources.

(2) All diesel generator starts for the purpose of these surveillances may be preceded by a prelube period.

(3) A diesel generator start in less than or equal to 10 seconds (fast start) shall be at a frequency in accordance with the Surveillance Frequency Control Program. All other diesel generator starts for the purpose of this surveillance may be modified starts involving reduced fuel (load limit) and/or idling and gradual acceleration to synchronous speed.

(4) Generator loading may be accomplished in accordance with vendor recommendations, including a warmup period prior to loading.

(5) The diesel generator start for this surveillance may be a modified start (see SR 4.8.1.1.2a.2)).

(6) Momentary transients outside this load range due to changing conditions on the grid shall not invalidate the test.

(7) If Specification 4.8.1.1.2a.2) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the standby diesel generator may be operated at 5000-5500 kW for a minimum of 2 hours or until operating temperature has stabilized.

(8) (Not used)

(9) (Not used)

(10) This test may be performed during power operation provided that the other two diesel generators are operable.

(11) Credit may be taken for events that satisfy any of these Surveillance Requirements.

(12) For the Unit 2 Train B standby diesel generator (SDG 22) failure of December 9, 2003, restore the inoperable standby diesel generator to OPERABLE status within 113 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS (Continued)

d. With more than one channel with no battery chargers OPERABLE,

1 Restore terminal voltage for at least three battery banks to greater than or equal to the minimum established float voltage within 1 hour, AND

2. Verify float current for the affected batteries does not exceed 2 amps once per 12 hours, AND

3. Restore one battery charger to OPERABLE status on at least three channels within 1 hour.

If the battery terminal voltage cannot be restored in the allowed time, float current is excessive, or a battery charger is not restored to operability in the time allowed, apply the requirements of the CRMP or the affected reactor unit is to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With one of the required channels inoperable for reasons other than (a), (b), (c), or (d) above, restore the channel to OPERABLE status within 2 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that:

The total battery terminal voltage is greater than or equal to the minimum established float voltage.

b. Not used.

c. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that:

1. The battery charger can supply at least 300 amperes at greater than or equal to the minimum established float voltage for at least 8 hours.

OR

Each battery charger can recharge the battery to the fully charged state within 12 hours while supplying the largest combined demands of the various continuous steady-state loads following a battery discharge to the bounding design-basis event discharge state.

2.

NOTE:	1. The modified performance discharge test in SR 4.8.2.3.f may be performed in lieu of Surveillance Requirement 4.8.2.1.c.2.
	2. Credit may be taken for unplanned events that satisfy this surveillance requirement.

The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated ESF loads for the design duty cycle when the battery is subjected to a battery service test.

d. Not used.

e. Not used.

f. Not used.

LIMITING CONDITION FOR OPERATION (continued)

- suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, or movement of irradiated fuel, AND

- Initiate corrective action to restore the required DC electrical power subsystems to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 Each 125-volt battery bank shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the total battery terminal voltage is greater than or equal to the minimum established float voltage.

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the battery charger can supply at least 300 amperes at greater than or equal to the minimum established float voltage for at least 8 hours.

OR:

Verify each battery charger can recharge the battery to the fully charged state within 12 hours while supplying the largest combined demands of the various continuous steady-state loads following a battery discharge to the bounding design-basis event discharge state.

c.

NOTE:	
	1. The modified performance discharge test in SR 4.8.2.3.f may be performed in lieu of Surveillance Requirement 4.8.2.2.c.
	2. Credit may be taken for unplanned events that satisfy this surveillance requirement.

At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated ESF loads for the design duty cycle when the battery is subjected to a battery service test.

LIMITING CONDITION FOR OPERATION (continued)

f. If a battery has one or more battery cells with float voltage < 2.07 volts and float current > 2 amps, declare the associated battery INOPERABLE immediately.

SURVEILLANCE REQUIREMENTS

4.8.2.3. Each 125-volt battery bank and charger shall be demonstrated operable:

a. NOTE: Performance of this surveillance is not required when battery terminal voltage is less than the minimum established float voltage of surveillance requirement 4.8.2.1.a.

At a frequency in accordance with the Surveillance Frequency Control Program, verify the float current for each battery is ≤ 2 amps.

b. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery pilot cell voltage is ≥ 2.07 V on float charge.

c. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery connected cell voltage is ≥ 2.07 V on float charge.

d. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.

e. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery pilot cell temperature is greater than or equal to minimum established design limits.

f. Battery capacity is tested under the following conditions:

1. At least once per 12 months by giving modified performance discharge tests of battery capacity to any battery that shows degradation or reaches 85% of the service life expected for the application with capacity less than 100% of the manufacturer's rating. Degradation is indicated when battery capacity drops more than 10% from its capacity on the previous performance/modified performance discharge test, or is below 90% of the manufacturer's rating; AND

2. At least once per 24 months by giving modified performance discharge tests of battery capacity to any battery reaching 85% of the service life with capacity greater than or equal to 100% of the manufacturer's rating; AND

3. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a modified performance discharge test.

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

a. With one of the required trains of A.C. ESF busses not fully energized, within 8 hours reenergize the train or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one of the required trains of A.C. ESF busses not fully energized, within 1 hour reenergize at least two trains or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. vital distribution panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) within 2 hours reenergize the A.C. distribution panel or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) within 24 hours reenergize the A.C. vital distribution panels from its associated inverter connected to its associated D.C. bus or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With more than one A.C. vital distribution panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) within 1 hour reenergize at lease five A.C. distribution panels or apply the of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) within I hour reenergize at least five A.C. vital distribution panels from their associated inverter connected to their associated D.C. bus or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With one D.C. bus not energized from its associated battery bank, within 2 hours reenergize the D.C. bus from its associated battery bank or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With more than one D.C. bus not energized from its associated battery bank, within 1 hour reenergize at least three D.C. buses from their associated battery banks or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at a frequency in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 The necessary portion of AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With one or more required AC, DC, or AC vital bus electrical power distribution subsystems inoperable, immediately declare associated supported required feature(s) inoperable OR immediately initiate action to suspend operations with a potential for draining the reactor vessel, suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, and immediately initiate corrective action to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status and declare associated required residual heat removal subsystem(s) inoperable and not in operation.

SURVEILLANCE REQUIREMENT

4.8.3.2 Verify correct breaker alignment and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems at a frequency in accordance with the Surveillance Frequency Control Program.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2800 ppm, and

c. Each valve or mechanical joint used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.*

ACTION:

a. With the requirements of LCO a. or b. not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2800 ppm, whichever is the more restrictive.

b. With a valve or mechanical joint used to isolate an unborated water source not secured in the closed position, immediately suspend CORE ALTERATIONS and initiate action to secure the valve(s) or mechanical joint(s) in the closed position and within 4 hours verify boron concentration is within limit. The required action to verify the boron concentration within limits must be completed whenever ACTION b. is entered. A separate ACTION entry is allowed for each unsecured valve or mechanical joint.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at a frequency in accordance with the Surveillance Frequency Control Program.

4.9.1.3 Each valve or mechanical joint used to isolate unborated water sources shall be verified closed and secured in position at a frequency in accordance with the Surveillance Frequency Control Program.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors* shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program,

b. A CHANNEL CALIBRATION, excluding the Neutron detectors, at a frequency in accordance with the Surveillance Frequency Control Program.

* An Extended Range Neutron Flux Monitor may be substituted for one of the Source Range Neutron Flux Monitors provided the OPERABLE Source Range Neutron Flux Monitor is capable of providing audible indication in the containment and control room.

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 operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit of LCO 3.9.1 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 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 a frequency in accordance with the Surveillance Frequency Control Program.

* The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

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 operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit of LCO 3.9.1 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 a frequency in accordance with the Surveillance Frequency Control Program.

*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.10 WATER LEVEL - REFUELING CAVITY

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods* within the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at a frequency in accordance with the Surveillance Frequency Control Program thereafter during movement of fuel assemblies or control rods.

* Water level requirements are not applicable when control rods are moved in conjunction with the head package during a rapid refueling.

3/4.9.11 WATER LEVEL - STORAGE POOLS

SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel 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.

SURVEILIANCE REQUIREMENTS

4.9.11.1 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at a frequency in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the spent fuel pool.

IN-CONTAINMENT STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11.2 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the in-containment 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

4.9.11.2 The water level in the in-containment storage pool shall be determined to be at least its minimum required depth at a frequency in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the in-containment storage pool.

3/4.9.13 SPENT FUEL POOL MINIMUM BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.13 The boron concentration of the spent fuel pool water shall be maintained greater than or equal to 2500 ppm.

APPLICABILITY: Whenever one or more fuel assemblies are stored in the spent fuel pool racks.

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of fuel assemblies in the spent fuel storage pool and initiate action to restore the boron concentration in the spent fuel pool to greater than or equal to 2500 ppm.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 The boron concentration of the spent fuel pool shall be determined by chemical analysis at a frequency in accordance with the Surveillance Frequency Control Program.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, initiate boration within 15 minutes and continue boration until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, initiate boration within 15 minutes and continue boration until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at a frequency in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

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 551°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 551°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 a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F at a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of STARTUP and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at a frequency in accordance with the Surveillance Frequency Control Program during STARTUP and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any unprotected outdoor 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

4.11.1.4 The quantity of radioactive material contained in each unprotected outdoor tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at a frequency in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

*Tanks included in this specification are those outdoor tanks that are either not surrounded by liners, dikes, or walls capable of holding the tank contents or that do not have tank overflows and surrounding area drains connected to the Liquid Waste Processing System.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 1.0×10^5 Curies of noble gases (considered as Xe -133 equivalent)

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding above limit, immediately suspend all additions of the 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

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at a frequency in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

6.8.3.r Surveillance Frequency Control Program

This program provides controls for surveillance frequencies. The program shall ensure that surveillance requirements specified in the technical specifications are performed at intervals sufficient to assure the associated limiting conditions for operations are met.

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

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

   STP takes the following exception to NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1:

   a. STP will use the Independent Decisionmaking Panel (IDP) described in the applications approved by the NRC for the Graded Quality Assurance Program and the Exemption from Certain Special Treatment Requirements, augmented by the Surveillance Test Coordinator and Subject Matter Expert(s), to perform the IDP function.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 188 AND 175 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By application dated October 23, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073050348), as supplemented by letter dated May 20, 2008 (ADAMS Accession No. ML081540471), STP Nuclear Operating Company (the licensee) requested changes to the Technical Specifications (TS) for South Texas Project (STP), Units 1 and 2, in accordance with Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR). The supplemental letter dated May 20, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 18, 2007 (72 FR 71716).

1.1 Proposed License Amendment

The proposed changes would relocate surveillance frequencies of most surveillance tests from the TS to a licensee-controlled document, the Surveillance Frequency Control Program (SFCP). Once relocated, changes to the surveillance frequencies may be made using the risk-informed methodology in Nuclear Energy Institute (NEI) document NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies" Revision 1 (ADAMS Accession No. ML071360456), and as specified in the Administrative Controls section of the TS. By letter dated September 19, 2007, the NRC staff issued its "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 04-10, Revision 1, 'Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies (TAC No. MD6111)'" (ADAMS Accession No. ML072570267), which approved NEI 04-10 Rev. 1, as acceptable for referencing by licensees proposing to amend their TS to establish a surveillance frequency control program.

A new TS Section 6.8.3.r, "Surveillance Frequency Control Program" is added to the Administrative Controls:

> This program provides controls for surveillance frequencies. The program shall ensure that surveillance requirements specified in the technical

specifications are performed at intervals sufficient to assure the associated limiting conditions for operation are met.

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

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

   STP takes the following exception to NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1:

   a. STP will use the Independent Decisionmaking Panel (IDP) described in the applications approved by the NRC for the Graded Quality Assurance Program and the Exemption from Certain Special Treatment Requirements, augmented by the Surveillance Test Coordinator and Subject Matter Expert(s), to perform the IDP function.

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

The following TS surveillance frequencies were proposed to be removed from the TS and relocated, based on the original submittal and response to the NRC staff's request for additional information:

4.1.3.1.2	4.4.1.4.2.1	4.5.2.a	4.7.3.a	4.8.2.3.f.3
4.1.3.4.c	4.4.1.4.2.2	4.5.2.b	4.7.3.b	4.8.3.1
4.1.3.5.b	4.4.3.1	4.5.2.d	4.7.4.a	4.8.3.2
4.2.1.1.a	4.4.3.2	4.5.2.e	4.7.4.b	4.9.1.2
4.2.4.1.a	4.4.4.1	4.5.3.1.2	4.7.5	4.9.1.3
4.2.5.1	4.4.4.2	4.5.3.2	4.7.7.a	4.9.2.a
4.2.5.2	4.4.6.1.a.1	4.5.5	4.7.7.b	4.9.2.b
4.2.5.3	4.4.6.1.a.2	4.5.6.2	4.7.7.c	4.9.4
4.3.1.1	4.4.6.1.b	4.6.1.1.a	4.7.7.e	4.9.8.1
4.3.1.2	4.4.6.2.1.a	4.6.1.3.b	4.7.8.a	4.9.8.2
4.3.2.1	4.4.6.2.1.b	4.6.1.3.c	4.7.8.b	4.9.9
4.3.2.2.a	4.4.6.2.1.c	4.6.1.3.d	4.7.8.d	4.9.10
4.3.2.2.b	4.4.6.2.1.d	4.6.1.4	4.7.14.b	4.9.11.1
4.3.2.2.c	4.4.6.2.2.a	4.6.1.5	4.8.1.1.1.a	4.9.11.2
4.3.3.5.1	4.4.6.2.3	4.6.1.7.1	4.8.1.1.1.b	4.9.12.a
4.3.3.5.2	4.4.8 Table 4.4-4 Item 1	4.6.1.7.2	4.8.1.1.2.a	4.9.12.b
4.3.3.5.3	4.4.8 Table 4.4-4 Item 2	4.6.1.7.3	4.8.1.1.2.b	4.9.12.d
4.3.3.6	4.4.8 Table 4.4-4 Item 3	4.6.1.7.4	4.8.1.1.2.e	4.9.13
4.3.5.1	4.4.9.1.1	4.6.2.1.a	4.8.1.1.2.f	4.10.1.1
4.3.5.2	4.4.9.3.1.a	4.6.2.1.c	4.8.1.1.2.g	4.10.2.1
4.3.5.3	4.4.9.3.1.b	4.6.2.2	4.8.2.1.a	4.10.2.2
4.4.1.1	4.4.9.3.1.c	4.6.2.3.a	4.8.2.1.c	4.10.3.1
4.4.1.2.1	4.4.9.3.2	4.6.2.3.b	4.8.2.2.a	4.10.3.3
4.4.1.2.2	4.4.9.3.3	4.6.3.2	4.8.2.2.b	4.10.4.1
4.4.1.2.3	4.4.9.3.4	4.7.1.2.1.a	4.8.2.2.c	4.11.1.4
4.4.1.3.1	4.4.9.3.5	4.7.1.2.1.b	4.8.2.3.a	4.11.2.6
4.4.1.3.2	4.4.10	4.7.1.3	4.8.2.3.b	
4.4.1.3.3	4.5.1.1.a	4.7.1.4 Table 4.7-1 Item 1	4.8.2.3.c	
4.4.1.4.1.1	4.5.1.1.b	4.7.1.4 Table 4.7-1 Item 2a	4.8.2.3.d	
4.4.1.4.1.2	4.5.1.1.c	4.7.1.4 Table 4.7-1 Item 2b	4.8.2.3.e	

1.2 Related NRC Actions

By letter dated September 28, 2006 (ADAMS Accession No. ML062420049), Richard V. Guzman, NRC, to Christopher M. Crane, Exelon, the NRC issued a Safety Evaluation related to Amendment Nos. 186 and 147 to Facility Operating License Nos. NPF-39 and NPF-85, Exelon Generation Company, LLC for Limerick Generating Station, Units 1 and 2, Docket Nos. 50-352 and 50-353. Limerick was the first pilot submittal in support of this risk-informed TS Initiative 5b.

2.0 REGULATORY EVALUATION

In the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" published in the *Federal Register* on July 22, 1993 (58 FR 39132), the Nuclear Regulatory Commission (the Commission) addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Analysis or PRA) in Standard Technical Specifications (STS). In this 1993 *FR* publication, the Commission states, in part:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of 10 CFR 50.36] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed.

Additionally, the Commission states in the 1993 Final Policy Statement:

> The Commission Policy in this regard is consistent with its Policy Statement on 'Safety Goals for the Operation of Nuclear Power Plants,' 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "... probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made ... about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This 'defense-in-depth' approach is expected to continue to ensure the protection of public health and safety."

The Commission further states in the 1993 publication:

> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

Approximately 2 years later, the Commission provided additional detail concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" published in the *Federal Register (FR)* (60 FR 42622, August 16, 1995). In this FR publication, the Commission's opening statement states, in part:

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential

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

The following excerpts are taken, in part, from the 1995 Commission Policy Statement:

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

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

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

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

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

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

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

In 10 CFR 50.36, "Technical specifications," the Commission established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. As stated in 10 CFR 50.36(c)(3), "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The SRs are required by 10 CFR 50.36(c)(3) to reside in TS and will remain in TS. The new TS SFCP will provide the necessary surveillance frequency programmatic controls and is located in the TS Administrative Controls Section (STS Section 5.0).

Changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, Revision 1, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of SSCs are required to be documented. These may be subject to regulatory review and oversight of the SFCP implementation.

Changes to frequencies are subject to regulatory review and oversight of the SFCP implementation through the rigorous NRC review of safety-related SSC performance provided by the reactor oversight program.

STP's SFCP ensures that surveillance requirements specified in the TS are performed at intervals sufficient to assure the above regulatory requirements are met. Existing regulatory requirements, such as 10 CFR 50.65 (maintenance rule) and 10 CFR 50 Appendix B (corrective action program), require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. One of these actions may be to consider increasing the frequency at which a surveillance test is performed. In addition, the SFCP implementation guidance in NEI 04-10, Revision 1, requires monitoring of the performance of structures, systems, and components (SSCs) for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs.

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

Regulatory Guide (RG) 1.177 (Reference 4) for plant-specific, risk-informed TS changes by complying with current regulations.

Licensees are required by TS to perform surveillance test, calibration, or inspection on specific safety-related equipment such as reactivity control, power distribution, electrical, instrumentation, and others to verify system operability. Surveillance frequencies, currently identified in TS, are based primarily upon deterministic methods such as engineering judgment, operating experience, and manufacturer's recommendations. The licensee's use of NRC-approved PRA methodologies identified in NEI 04-10, Revision 1, provides a way to establish risk-informed surveillance frequencies that complements the deterministic approach and is consistent with the NRC's defense-in-depth philosophy.

These regulatory requirements, and the monitoring required by NEI 04-10, Revision 1 ensure that surveillance frequencies that are insufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified and appropriate corrective actions taken.

3.0 TECHNICAL EVALUATION

The licensee proposed changes to the TS, which provide for administrative relocation of applicable surveillance frequencies into the SFCP as specified in the Administrative Controls section of the TS. In accordance with NEI 04-10, Revision 1, PRA methods are used, in combination with plant performance data and other considerations, to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is in accordance with guidance provided in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998 (RG 1.174), and Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998 (RG 1.177). The requirement to perform a surveillance frequency is retained in the TS, and the requirement for a frequency is retained in the TS; it is the specific frequency interval (number) that is relocated to a licensee-controlled document. Initial surveillance frequencies were taken from vendor/manufacturer's recommendations, industry codes and standards, engineering judgment, TS completion times (allowed outage times), historical equipment reliability data, and when (under what plant conditions) can the surveillance frequency be performed. Once a surveillance frequency is determined in accordance with the SFCP it will be recorded in a licensee-controlled document, and as far as the operator/maintainer is concerned it is the same as having the actual frequency number in TS.

Figure 1 in NEI 04-10 provides a process flow map for making changes to surveillance frequencies in the SFCP. In approving the Limerick license amendment, six cases/proposals for changing surveillance test intervals (STI) were evaluated (some acceptable for change, others not), thereby testing all the logic paths in NEI-04-10 (see flow chart in NEI-04-10) prior to approval. The process in NEI 04-10 includes three different flow paths to follow depending on how the PRA is modeled, as shown in steps 8, 9, and 10 of Figure 1:

- STI SSC Modeled in PRA
- STI SSC Not Modeled in PRA, but could be
- STI SSC Modeling is Impractical

RG 1.177 identifies five key safety principles to be met for risk-informed changes to TS. As noted above, the licensee has not proposed any variations or deviations from the guidance in NEI 04-10, Revision 1. As such, the evaluations provided in NRC safety evaluation dated September 19, 2007, for each of these principles, as addressed by the industry methodology document (NEI 04-10, Revision 1), are applicable to the licensee. In addition, the NRC staff has provided plant-specific discussions for several of the five key safety principles. The generic and plant-specific evaluations are provided below:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

   10 CFR 50.36(c)(3) provides that TSs will include surveillances which are "requirements relating to test, calibration, or inspection to assure that necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." NEI 04-10, Revision 1, provides guidance for relocating the surveillance frequencies from the TSs to a licensee-controlled program by providing an NRC-approved methodology for control of the surveillance frequencies. The surveillances themselves would remain in the TSs, as required by 10 CFR 50.36(c)(3).

   This change is consistent with other NRC-approved TS changes in which the surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program. Thus, this proposed change meets the first key safety principle of RG 1.177 by complying with current regulations.

2. The proposed change is consistent with the defense-in-depth philosophy.

   Consistency with the defense-in-depth philosophy is maintained if:

   - A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

   - Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

   - System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers). Because the scope of the proposed methodology is limited to revision of surveillance frequencies, the redundancy, independence, and diversity of plant systems are not impacted.

   - Defenses against potential common cause failures (CCFs) are preserved, and the potential for the introduction of new CCF mechanisms is assessed.

   - Independence of barriers is not degraded.

- Defenses against human errors are preserved.

- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A, are maintained.

NEI 04-10, Revision 1, uses both the core damage frequency (CDF) and the large early release frequency (LERF) metrics to evaluate the impact of proposed changes to surveillance frequencies. Compliance with the guidance of RG 1.174 and RG 1.177 for changes to CDF and LERF is achieved by evaluation using a comprehensive risk analysis, which assesses the impact of proposed changes including contributions from human errors and CCFs. Defense-in-depth is included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to increased likelihood of CCFs. Both the quantitative risk analysis and the qualitative considerations assure a reasonable balance that defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177.

3. The proposed change maintains sufficient safety margins.

   The design, operation, testing methods, and acceptance criteria for SSCs, specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. Thus, safety margins are maintained by the proposed methodology and the third key safety principle of RG 1.177 is satisfied.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

   RG 1.177 provides a framework for risk evaluation of proposed changes to surveillance frequencies. The framework requires identification of the risk contribution due to surveillance changes, determination of the risk impact, and performance of sensitivity and uncertainty evaluations. NEI 04-10, Revision 1, satisfies the intent of RG 1.177 requirements for evaluation of the change in risk, and for assuring that such changes are small.

<u>Quality of the PRA</u>

The quality of the PRA must be compatible with the safety implications of the proposed TS change, and the role the PRA assessments play in justifying the change. The NRC has developed Regulatory Guide 1.200 for Trial Use, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004 (RG 1.200), to address PRA technical adequacy. RG 1.200 uses the American Society of Mechanical Engineers (ASME) RA-Sa-2005, Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, and the NEI peer review process NEI 00-02, PRA Peer Review Process Guidance. NEI 04-10, Revision 1 requires an assessment of the PRA models used to support the SFCP against the requirements of RG 1.200. The assessments are made to assure that the PRA models are adequately determining the change in risk due to changes to surveillance frequencies of SSCs, using plant specific data and models. Capability category II of ASME RA-Sa-2005 is applied as the standard, and any identified deficiencies to those requirements are assessed further in sensitivity studies to determine any impacts to proposed changes to surveillance frequencies.

The STP PRA has been previously evaluated using RG 1.200 and the ASME standard, and the staff determined that the STP PRA internal events model satisfied the guidance of RG 1.200, Revision 1, and conformed to capability category II guidance of the ASME standard.

Based on the above, the NRC staff concludes that the STP PRA is acceptable for this application, because the level of PRA quality is sufficient to support the evaluation of changes to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1 of RG 1.177.

Scope of the PRA

NEI 04-10, Revision 1 requires evaluation of each proposed surveillance frequency change to determine its potential impact on risk due to impacts from internal events, fires, seismic events, other external events, and from shutdown conditions. Consideration is given to both CDF and LERF metrics. Where quantitative risk models are unavailable, bounding analyses or other conservative quantitative evaluations are performed. Per NEI 04-10, a qualitative screening analysis may be used when the surveillance frequency impact on plant risk can be shown to be negligible or zero. The STP PRA is a full-scope, at-power model, which has been found by the NRC staff to be acceptable for this application. The licensee does not have a shutdown PRA model, and will use the alternative evaluation methods consistent with NEI 04-10, Revision 1 guidance.

The NRC staff concludes that the scope of the STP PRA model and the shutdown evaluation method is sufficient to ensure the scope of the risk contribution of each surveillance is properly identified and evaluated, consistent with Regulatory Position 2.3.2 of RG 1.177.

PRA Modeling

NEI 04-10, Revision 1 provides guidance to determine if the SSCs affected by a surveillance are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact is carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted common cause failure modes, based on the proposed change to the surveillance frequency. Where the SSC is not modeled in the PRA, bounding analyses are performed to characterize the impact of the proposed change to surveillance frequency. Potential impacts on the risk analyses due to screening criteria and truncation levels are adequately addressed by the requirements for PRA technical adequacy addressed by RG 1.200, and by sensitivity studies identified in NEI 04-10, Revision 1. Guidance is provided for the quantitative evaluation of the impact of selected testing strategy (i.e., staggered testing or sequential testing) consistent with the guidance of NUREG/CR-6141 and NUREG/CR-5497.

The NRC staff concludes that the STP program conforms to the guidance in NEI 04-10, Revision 1 and is, therefore, acceptable because PRA modeling is sufficient to ensure an acceptable evaluation of risk due to the change in surveillance frequency, and is consistent with Regulatory Position 2.3.3 of RG 1.177.

Assumptions.

The failure probabilities of SSCs modeled in a PRA include a standby time-related contribution and a cyclic demand-related contribution. NEI 04-10, Revision 1 adjusts the time-related failure contribution of SSCs affected by the proposed change to surveillance frequency. This is consistent with RG 1.177 Section 2.3.3 which permits separation of the failure rate contributions into demand and standby for evaluation of surveillance requirements. If the available data do not support distinguishing between the time-related failures and demand failures, then the change to surveillance frequency is conservatively assumed to impact the total failure probability of the SSC, including both standby and demand contributions. The SSC failure rate (per unit time) is assumed to be unaffected by the change in test frequency, and this is confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented.

The process requires consideration of qualitative sources of information with regards to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus the process is not reliant upon risk analyses as the sole basis for the proposed changes.

The potential beneficial risk impacts of reduced surveillance frequency, including reduced downtime, lesser potential for restoration errors, reduction of potential for test-caused transients, and reduced test-caused wear of equipment, are identified qualitatively, but are conservatively not required to be quantitatively assessed.

The NRC staff concludes that the STP program conforms to the guidance in NEI 04-10, Revision 1 and is, therefore, acceptable because it employs reasonable assumptions with regard to extensions of surveillance test intervals, and is consistent with Regulatory Position 2.3.4 of RG 1.177.

Sensitivity and Uncertainty Analyses.

NEI 04-10, Revision 1 requires sensitivity studies to assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact to the frequency of initiating events, and of any identified deviations from capability category II of ASME RA-Sb-2005. Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered, along with any qualitative considerations that may bear on the results of such sensitivity studies. Required monitoring and feedback of SSC performance, once the revised surveillance frequencies are implemented, are also used.

The NRC staff concludes that the STP program conforms to the guidance in NEI 04-10, Revision 1 and is, therefore, acceptable because it appropriately considers the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations, consistent with Regulatory Position 2.3.5 of RG 1.177.

Acceptance Guidelines.

NEI 04-10, Revision 1 quantitatively evaluates the change in total risk (including internal and external events contributions) in terms of CDF and LERF for both the individual risk impact of a proposed change in surveillance frequency and the cumulative impact from all individual changes to surveillance frequencies. Each individual change to surveillance frequency must be shown to result in a risk impact below 1E-6 per year for change to CDF, and below 1E-7 per year for change to LERF. These are consistent with the limits of RG 1.174 for very small changes in risk. Where the RG 1.174 limits are not met, the process either considers revised surveillance frequencies which are consistent with RG 1.174, or the process terminates without permitting the proposed changes. Where quantitative results are unavailable to permit comparison to acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or zero. Otherwise, bounding quantitative analyses are required which demonstrate the risk impact is at least one order of magnitude lower than the RG 1.174 acceptance guidelines for very small changes in risk.

In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact below 1E-5 per year for change to CDF, and below 1E-6 per year for change to LERF, and the total CDF and total LERF must be reasonably shown to be less than 1E-4 per year and 1E-5 per year, respectively. These are consistent with the limits of RG 1.174 for acceptable changes in risk, as referenced by RG 1.177 for changes to surveillance frequencies. The staff interprets this assessment of cumulative risk as a requirement to calculate the change in risk from a baseline model utilizing failure probabilities based on the surveillance frequencies prior to implementation of the SFCP, compared to a revised model with failure probabilities based on changed surveillance frequencies. The staff further notes that NEI 04-10, Revision 1 includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with small risk increases (less than 5E-8 CDF and 5E-9 LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

The quantitative acceptance guidance of RG 1.174 is necessary but not sufficient to accept decreases in surveillance frequencies. The process also considers qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history. The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results compared to numerical acceptance guidelines. Performance monitoring and feedback are also required to assure that lessons learned from past experience are considered. These items are evaluated by an Integrated Decision-making Panel (IDP) in accordance with NEI 04-10, Revision 1. Per NEI 04-10, Revision 1, the IDP is composed of the site Maintenance Rule Expert Panel, supplemented by a Surveillance Test Coordinator, and a Subject Matter Expert.

STP proposed to take an exception to the composition of the IDP, to allow the use of a long standing independent decision making panel of highly qualified individuals with extensive experience in risk informed evaluations to determine the acceptability of the proposed surveillance frequency changes, in lieu of the site Maintenance Rule Expert Panel. This group is comprised of individuals whose experience is equal to or exceeds the requirements of those on the Maintenance Rule Expert Panel. The individuals who make up this panel are designated by the senior management team that provides process oversight. The designated individuals have expertise in the areas of probabilistic risk assessment, operations, maintenance, engineering, quality assurance, operating experience, and licensing. At least three individuals must have a minimum of 5 years experience at STP or similar nuclear plants, and at least one individual who has worked on the modeling and updating of the PRA for STP or similar plants for a minimum of 3 years. This level of experience and expertise will ensure the final decision is well-considered and safety-focused. When reviewing potential changes, the panel will be augmented by the Surveillance Test Coordinator and at least one subject matter expert.

The NRC staff has concluded the use of this panel is not a significant deviation from NEI 04-10, Revision 1, and that the STP program otherwise conforms to the guidance in NEI 04-10, Revision 1 and is, therefore, acceptable because it provides reasonable acceptance guidelines and methods for evaluating the risk increase due to proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177.

Therefore, based on the technical adequacy of the licensee's PRA models and on the licensee's conformance to the guidance of NEI 04-10, Revision 1, the NRC staff concludes that the licensee's proposed change satisfies the fourth key safety principle of RG 1.177 by assuring any increase in risk is small consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

The STP proposed change requires application of NEI 04-10, Revision 1 in the SFCP.

> NEI 04-10, Revision 1 requires performance monitoring of SSCs whose surveillance frequency has been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback includes consideration of Maintenance Rule monitoring of equipment performance. In the event of degradation of SSC performance, the surveillance frequency is reassessed in accordance with the methodology, in addition to any corrective actions which may apply as part of the Maintenance Rule requirements.

The NRC staff concludes that the STP program conforms to the guidance in NEI 04-10, Revision 1 and is, therefore, acceptable because the performance monitoring and feedback is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 of RG 1.177. Thus, the fifth key safety principle of RG 1.177 is satisfied.

4.0 SUMMARY AND CONCLUSIONS

The NRC staff has reviewed the licensee's proposed TS changes for relocation of surveillance frequencies to a licensee controlled document, and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the Administrative Controls section of the TS. The SFCP references NEI 04-10, Revision 1, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within an SFCP, allowing for licensee control of the surveillance frequencies.

The NRC staff finds that the proposed TS changes and implementation of the methodology of NEI 04-10, Revision 1, as referenced in the Administrative Controls section of the TS, satisfies the key principles of risk-informed decision making applied to changes to TS as delineated in RG 1.177 and RG 1.174. The NRC staff concludes that:

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

Based on the above evaluation, the NRC staff concludes the proposed changes meet the technical requirements in the regulations that are discussed in Section 2.0 of this safety evaluation. Based on this, the NRC staff further concludes that the proposed TS changes in the proposed amendment meet 10 CFR 50.36 and, therefore, the proposed amendments are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on December 18, 2007 (72 FR 71716). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Charles T. Bowman, STP Nuclear Operating Company, Letter to NRC, "Proposed Revision to Technical Specifications to Relocate Surveillance Test Intervals to a Licensee Controlled Program (Risk-Informed Initiative 5b)," October 23, 2007 (ADAMS Accession No. ML073050348).

2. Charles T. Bowman, STP Nuclear Operating Company, Letter to NRC, "Response to Request for Additional Information on Proposed Amendment Related to Risk-Informed Initiative 5b," May 20, 2008 (ADAMS Accession No. ML081540471).

3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," U.S. Nuclear Regulatory Commission, July 1998 (ADAMS Accession No. ML003740133).

4. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," U.S. Nuclear Regulatory Commission, August 1998 (ADAMS Accession No. ML003740176).

5. Regulatory Guide 1.200 for Trial Use, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004 (ADAMS Accession No. ML070240001).

6. American Society of Mechanical Engineers, ASME RA-Sb-2005, Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, December 2005.

7. NEI 00-02, Revision 3, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," 2000 (ADAMS Accession No. ML003728023).

8. NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1, April 2007 (ADAMS Accession No. ML071360456).

9. Ho K. Nieh, NRC, Letter to Biff Bradley, Nuclear Energy Institute, "Final Safety Evaluation For Nuclear Energy Institute (NEI) Topical Report (TR) 04-10, Revision 1, "Risk-Informed Technical Specification Initiative 5B, "Risk-Informed Method For Control Of Surveillance Frequencies (TAC No. MD6111)," September 19, 2007 (ADAMS Accession No. ML072570267).

10. Richard V. Guzman, NRC, Letter to Christopher M. Crane, Exelon Generation Company, LLC, "Limerick Generating Station, Units 1 and 2 - Issuance of Amendment Re: Relocate Surveillance Test Intervals to Licensee-Controlled Program (TAC Nos. MC3567 and MC3568)," September 28, 2006 (ADAMS Accession No. ML062420049).

11. American National Standards Institute/American Nuclear Society, ANSI/ANS-58.23-2007, "Fire PRA Methodology," November 20, 2007.

12. Kassawara, R.P., September 30, 2005, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 1: Summary and Overview, NUREG/CR-6850 (ADAMS Accession No. ML052580075).

Principal Contributor: A. Howe

Date: October 31, 2008

October 31, 2008

Mr. Edward D. Halpin
Chief Nuclear Officer
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO RELOCATE SURVEILLANCE TEST INTERVALS TO LICENSEE-CONTROLLED PROGRAM (RISK-INFORMED INITIATIVE 5-b) (TAC NOS. MD7058 AND MD7059)

Dear Mr. Halpin:

The Commission has issued the enclosed Amendment No. 188 to Facility Operating License No. NPF-76 and Amendment No. 175 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to the STP Nuclear Operating Company submittal dated October 23, 2007, as supplemented by letter dated May 20, 2008.

The amendments revise the TSs to relocate surveillance frequencies of most surveillance tests from the TS to a licensee-controlled surveillance frequency control program. Once relocated, the surveillance frequency changes are permitted based on the risk-informed methodology as specified in the Administrative Controls section of the TS.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499
Enclosures:
1. Amendment No. 188 to NPF-76
2. Amendment No. 175 to NPF-80
3. Safety Evaluation
cc w/encls: See next page

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RidsRgn4MailCenter Resource
AHowe, NRR/DRA/APLA
GHill, OIS (4)

ADAMS Accession No.: ML082830172 *SE input memo **See previous concurrence

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DRA/APLA/BC	OGC – NLO	NRR/LPL4/BC	NRR/LPL4/PM
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DATE	~~10/28/08~~ 10/31/08	10/28/08	8/19/08	10/30/08	10/31/08	10/31/08

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