



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

July 24, 2014

Mr. Dean Curtland, Site Vice President
c/o Michael Ossing
Seabrook Station
NextEra Energy Seabrook, LLC
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT
REGARDING THE RISK-INFORMED JUSTIFICATIONS FOR THE
RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS
TO A LICENSEE-CONTROLLED PROGRAM (TAC NO. MF1958)

Dear Mr. Curtland:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1 (Seabrook). This amendment consists of changes to the facility technical specifications (TSs) in response to your application dated May 28, 2013, as supplemented by letters dated July 31, 2013, January 29, and March 26, 2014.

The amendment modifies Seabrook's TSs by relocating specific surveillance frequencies to a licensee-controlled program with implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." The changes are consistent with NRC-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control-[Risk Informed Technical Specifications Task Force] RITSTF Initiative 5b," Revision 3. The Federal Register notice published on July 6, 2009 (74 FR 31996), announced the availability of this TS improvement.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

A handwritten signature in black ink, appearing to read "John G. Lamb".

John G. Lamb, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures:

1. Amendment No. 141 to NPF-86
2. Safety Evaluation

cc w/encls: Distribution via Listserv



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

NEXTERA ENERGY SEABROOK, LLC, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
License No. NPF-86

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by NextEra Energy Seabrook, LLC, et al., (the licensee) dated May 28, 2013, as supplemented July 31, 2013, January 29, 2014, and March 26, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

*NextEra Energy Seabrook, LLC is authorized to act as agent for the: Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, and Taunton Municipal Light Plant 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-86 is hereby amended to read as follows:

- (2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 141, and the Environmental Protection Plan contained in Appendix B are incorporated into the Facility License No. NPF-86. NextEra Energy Seabrook, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Robert G. Schaaf, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and TS

Date of Issuance: July 24, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following page of Facility Operating License No. NPF-86 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove
3

Insert
3

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

Remove

Insert

1-8
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3/4 9-11	3/4 9-11
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3/4 9-14	3/4 9-14
3/4 10-1	3/4 10-1
3/4 10-2	3/4 10-2
3/4 10-3	3/4 10-3
3/4 10-5	3/4 10-5
3/4 11-4	3/4 11-4
6-14b	6-14b
6-14c	6-14c

- (4) NextEra Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
- (5) NextEra Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;
- (6) NextEra Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein; and
- (7) DELETED

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NextEra Energy Seabrook, LLC, is authorized to operate the facility at reactor core power levels not in excess of 3648 megawatts thermal (100% of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 141*, and the Environmental Protection Plan contained in Appendix B are incorporated into the Facility License No. NPF-86. NextEra Energy Seabrook, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) License Transfer to FPL Energy Seabrook, LLC**

- a. On the closing date(s) of the transfer of any ownership interests in Seabrook Station covered by the Order approving the transfer, FPL Energy Seabrook, LLC**, shall obtain from each respective transferring owner all of the accumulated decommissioning trust funds for the facility, and ensure the deposit of such funds and additional funds, if necessary, into a decommissioning trust or trusts for Seabrook Station established by FPL Energy Seabrook, LLC**, such that the amount of such funds deposited meets or exceeds the amount required under 10 CFR 50.75 with respect to the interest in Seabrook Station FPL Energy Seabrook, LLC**, acquires on such dates(s).

* Implemented

** On April 16, 2009, the name "FPL Energy Seabrook, LLC" was changed to "NextEra Energy Seabrook, LLC".

TABLE 1.1

FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.
SFCP	In accordance with the Surveillance Frequency Control Program

TABLE 1.2

OPERATIONAL MODES

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

*Excluding decay heat.

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

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

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN for four-loop operation shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

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

ACTION:

With the SHUTDOWN MARGIN less than the limiting value, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limiting value:

- a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);
- b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1 in accordance with the Surveillance Frequency Control Program by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;
- c. When in MODE 2 with k_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;
- d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

REACTIVITY CONTROL SYSTEMS

BORATION CONTROL

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

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 (Continued)

- e. When in MODE 3 or 4, in accordance with the Surveillance Frequency Control Program by consideration of the following factors:
- 1) Reactor Coolant System boron concentration,
 - 2) Control rod position,
 - 3) Reactor Coolant System average temperature,
 - 4) Fuel burnup based on gross thermal energy generation,
 - 5) Xenon concentration, and
 - 6) Samarium concentration.

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

REACTIVITY CONTROL SYSTEMS

BORATION CONTROL

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR). Additionally, the Reactor Coolant System boron concentration shall be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than the limit specified in the COLR or the Reactor Coolant System boron concentration less than the limit specified in the COLR, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN and boron concentration are restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limit specified in the COLR and the Reactor Coolant System boron concentration shall be determined to be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition:

- a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and
- b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:
 - 1) Reactor Coolant System boron concentration,
 - 2) Control rod position,
 - 3) Reactor Coolant System average temperature,
 - 4) Fuel burnup based on gross thermal energy generation,
 - 5) Xenon concentration, and
 - 6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

ISOLATION OF UNBORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 Provisions to isolate the Reactor Coolant System from unborated water sources shall be OPERABLE with:

- a. The Boron Thermal Regeneration System (BTRS) isolated from the Reactor Coolant System, and
- b. The Reactor Makeup Systems inoperable except for the capability of delivering up to the capacity of one Reactor Makeup Water pump to the Reactor Coolant System.

APPLICABILITY: MODES 4, 5, and 6

ACTION:

With the requirements of the above specification not satisfied immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and, if within 1 hour the required SHUTDOWN MARGIN is not verified, initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limits specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN is restored and the isolation provisions are restored to OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The provisions to isolate the Reactor Coolant System from unborated water sources shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by:

- a. Verifying that at least the BTRS outlet valve, CS-V-302, or the BTRS moderating heat exchanger outlet valve, CS-V-305, or the manual outlet isolation valve for each demineralizer* not saturated with boron, CS-V-284, CS-V-295, CS-V-288, CS-V-290, CS-V-291, is closed and locked closed, and
- b. Verifying that power is removed from at least one of the Reactor Makeup Water pumps, RMW-P-16A or RMW-P-16B.

*A demineralizer may be unisolated to saturate a bed with boron provided the effluent is not directed back to the Reactor Coolant System.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 ACTION b.3 (Continued)

- c) A power distribution map is obtained from the Incore Detector System and $F_Q(Z)$ and $F_{\Delta H}^N$ 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 of Specification 3.1.3.6. 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 in accordance with the Surveillance Frequency Control Program, except during time intervals when the rod position deviation monitor is inoperable; then verify the group positions at least once per 4 hours.

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

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

- a. With a maximum of one digital rod position indicator per bank inoperable, either:
 1. Determine the position of the nonindicating rod(s) indirectly by the Incore Detector System at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

- b. With a maximum of one demand position indicator per bank inoperable, either:
 1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps in accordance with the Surveillance Frequency Control Program, except during time intervals when the rod position deviation monitor is inoperable; then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

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

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full range of rod travel in accordance with the Surveillance Frequency Control Program.

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

**See Special Test Exceptions Specification 3.10.5.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

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

- a. T_{avg} for each loop greater than or equal to 551°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 that could affect the drop time of those specific rods, and
- c. In accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

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[#] as specified in the COLR:

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

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

#The fully withdrawn position is defined as the interval within 225 to the mechanical fully withdrawn position, inclusive.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the insertion limits specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

- a. Restore the control banks to within the limits within 2 hours, or
- b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or
- c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

SURVEILLANCE REQUIREMENTS

- 4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:
- a. Monitoring the indicated AFD for each OPERABLE excore channel in accordance with the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE, and
 - 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 (THIS SPECIFICATION NUMBER IS NOT USED)

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

SURVEILLANCE REQUIREMENTS

- 4.2.2.1 The provisions of Specification 4.0.4 are not applicable.
- 4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits by:
- Using the incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.
 - Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% when using the moveable incore detectors or 5.21% when using the fixed incore detectors to account for measurement uncertainties.
 - Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \quad \text{for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \quad \text{for } P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and $W(Z)$ is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP} , $K(Z)$, and $W(Z)$ are specified in the COLR.

- Measuring $F_Q^M(Z)$ according to the following schedule:
 - Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or
 - In accordance with the Surveillance Frequency Control Program, whichever occurs first.

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

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}^N$ shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

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

- a. Within 2 hours reduce the THERMAL POWER to the level where the LIMITING CONDITION FOR OPERATION is satisfied.
- b. THERMAL POWER may be increased, provided $F_{\Delta H}^N$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F_{\Delta H}^N$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fuel loading and in accordance with the Surveillance Frequency Control Program thereafter by:

- a. Using the Incore Detector System to obtain a power distribution map at any THERMAL POWER greater than 5% RATED THERMAL POWER.
- b. Using the measured value of $F_{\Delta H}^N$ which does not include an allowance for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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 Setpoints within the next 4 hours.
- b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ and $F_{\Delta H}^N$ 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 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 using the Incore Detector System to confirm indicated QUADRANT POWER TILT RATIO in accordance with the Surveillance Frequency Control Program by either:

- a. Using the four pairs of symmetric detector locations or
- b. Using the Incore Detector System to monitor the QUADRANT POWER TILT RATIO subject to the requirements of Technical Requirement TR20-3.3.3.2.

*See Special Test Exceptions Specification 3.10.2

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

- a. Reactor Coolant System T_{avg} is less than or equal to the limit specified in the COLR,
- b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and
- c. Reactor Coolant System Flow shall be:
 1. $\geq 374,400$ gpm**; and,
 2. $\geq 383,800$ 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 in accordance with the Surveillance Frequency Control Program.

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

4.2.5.3 The RCS total flow rate shall be determined by an approved method to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

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

**Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

***Minimum measured flow used in the Revised Thermal Design Procedure.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified 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 in accordance with the Surveillance Frequency Control Program. Each verification shall include at least one train and one channel per function.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Manual Reactor Trip	N.A.	N.A.	N.A.	SFCP(13)	N.A.	1,2,3*,4*,5*
2. Power Range, Neutron Flux						
a. High Setpoint	SFCP	SFCP(2, 4), SFCP(3, 4), SFCP(4, 6), SFCP(4, 5)	SFCP	N.A.	N.A.	1, 2
b. Low Setpoint	SFCP	SFCP(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1, 2
4. (NOT USED)						
5. Intermediate Range, Neutron Flux	SFCP	SFCP(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	SFCP	SFCP(4, 5)	S/U(8), SFCP(9)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
8. Overpower ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	SFCP	SFCP(16)	SFCP	N.A.	N.A.	1
10. Pressurizer Pressure--High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	SFCP	SFCP	SFCP	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	SFCP	SFCP	SFCP	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
13. Steam Generator Water Level-- Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
15. Underfrequency - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	SFCP	N.A.	S/U(8, 10)	N.A.	1
b. Turbine Stop Valve	N.A.	SFCP	N.A.	S/U(8, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	SFCP(4)	SFCP	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
Reactor Trip System Interlocks (Continued)						
e. Power Range Neutron Flux, P-10	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	SFCP	SFCP	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	SFCP(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	SFCP(7)	1, 2, 3*, 4*, 5*
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	SFCP(7, 14), SFCP(15)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be 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 92 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 50% 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 purposes of this surveillance requirement, monthly shall mean at least once per 31 EFPD.
- (4) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (5) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. The plateau curves for the Intermediate Range and Power Range detectors are required to be measured or obtained within 24 hours after attaining 100% of RATED THERMAL POWER. 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.
- (7) Each train shall be tested in accordance with the Surveillance Frequency Control Program.
- (8) If not performed in previous 31 days.
- (9) 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.
- (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.

INSTRUMENTATION

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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.

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

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generator, Phase "A" Isolation, Containment Ventilation Isolation, and Emergency Feedwater, Service Water to Secondary Component Cooling Water Isolation, CBA Emergency Fan/Filter Actuation, and Latching Relay).								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
c. Containment Pressure-Hi-1	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	SFCP	SFCP(4)	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
c. Containment Pressure-Hi-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
3) Containment Pressure-Hi-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Ventilation Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) Containment On Line Purge Radioactivity- High	SFCP	SFCP	SFCP(2)	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
7. Emergency Feedwater (Continued)								
b. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3
c. Steam Generator Water Level-Low-Low, Start Motor-Driven Pump and Turbine-Driven Pump	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection, Start Motor-Driven Pump and Turbine-Driven Pump	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. for all Loss-of-Offsite Power Surveillance Requirements.							
8. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
b. RWST Level Low-Low Coincident With Safety Injection	N.A.	SFCP	SFCP	SFCP(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
9. Loss of Power (Start) Emergency Feedwater)								
a. 4.16 kV Bus E5 and E6 Loss of Voltage	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV Bus E5 and E6 Degraded Voltage Coincident With Safety Injection	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements							
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	1, 2, 3
c. Steam Generator Water Level, P-14	SFCP	SFCP	SFCP	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3
d. Cold Leg Injection, P-15	S	R	Q	N.A.	M(1)	M(1)	Q	1, 2, 3

TABLE NOTATION

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

(2) A DIGITAL CHANNEL OPERATIONAL TEST will be performed on this instrumentation.

(3) Setpoint verification is not applicable.

(4) CHANNEL CALIBRATION shall include verification that the time constants are adjusted to the prescribed values.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

- a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.
- b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program.

TABLE 4.3-3

(THIS TABLE NUMBER IS NOT USED)

INSTRUMENTATION

MONITORING INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.
- b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels as required by Table 3.3-9, within 60 days restore the inoperable channel(s) to OPERABLE status or, pursuant to Specification 6.8.2, submit a Special Report that defines the corrective action to be taken.
- c. With one or more Remote Shutdown System transfer switches, power, or control circuits inoperable, restore the inoperable switch(s) / circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel in Table 3.3-9 shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CHECK, and
- b. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CALIBRATION.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit listed in Table 3.3-9, including the actuated components, shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CHECK, and |
- b. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CALIBRATION. |

TABLE 4.3-6

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>SOURCE CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL OPERATIONAL TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. RADIOACTIVE GAS WASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
Oxygen Monitor (Process)	SFCP	N.A.	SFCP(4)	SFCP	**

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

HOT STANDBY

SURVEILLANCE REQUIREMENTS

- 4.4.1.2.1 At least the above required reactor coolant pumps, shall be determined OPERABLE 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 14% in accordance with the Surveillance Frequency Control Program.
- 4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

*Not required to be performed until 24 hours after a required pump is not in operation.

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE 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 14% in accordance with the Surveillance Frequency Control Program. |

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

- a. One additional RHR loop shall be OPERABLE^{**}, or
- b. The secondary-side water level of at least two steam generators shall be greater than 14%.

APPLICABILITY: MODE 5 with reactor coolant loops filled^{***}.

ACTION:

- a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.
- b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

- a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.
- b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

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

*One RHR loop 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 dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of pressurizer level (1656 cubic feet), and at least two groups of pressurizer heaters each having a capacity of at least 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the pressurizer otherwise inoperable, fully insert all rods, place the Control Rod Drive System in a condition incapable of rod withdrawal, and be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters from the emergency power supply and measuring circuit current in accordance with the Surveillance Frequency Control Program. |

REACTOR COOLANT SYSTEM

RELIEF VALVES

SURVEILLANCE REQUIREMENTS

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

- a. Performance of a CHANNEL CALIBRATION, and
- b. Operating the valve through one complete cycle of full travel during MODES 3 or 4.

4.4.4.2 Each block valve shall be demonstrated OPERABLE 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 with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

REACTOR COOLANT SYSTEM (RCS)

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3. Restore either the containment drainage sump level monitoring system or the containment atmosphere particulate monitor to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

- a. Required containment atmosphere radioactivity monitor:
 1. Performance of a CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,
 2. Performance of a DIGITAL CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program, and
 3. Performance of a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.
- b. Containment Drainage Sump Level Monitoring System - performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

3.4.6.2

ACTION: (Continued)

- c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

- a. Not Used
- b. Not Used
- c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig in accordance with the Surveillance Frequency Control Program with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;
- d. Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program during steady-state operation, except that not more than 96 hours shall elapse between any two successive inventory balances; ⁽¹⁾ ⁽²⁾
- e. Monitoring the Reactor Head Flange Leakoff System in accordance with the Surveillance Frequency Control Program, and
- f. Verifying primary to secondary leakage is ≤ 150 gallons per day through any one SG in accordance with the Surveillance Frequency Control Program. ⁽²⁾

(1) Not applicable to primary to secondary leakage.

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

- a. In accordance with the Surveillance Frequency Control Program,
- b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,
- c. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve, and
- d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.*
- e. Testing pursuant to Specification 4.0.5.

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

* Not applicable to RHR Pumps 8A and 8B suction isolation valves.

TABLE 4.4-3

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS FREQUENCY</u>	<u>MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED</u>
1. Gross Radioactivity Determination	In accordance with the Surveillance Frequency Control Program.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	In accordance with the Surveillance Frequency Control Program.	1
3. Radiochemical for \bar{E} Determination*	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 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT 1-131 or 100/ \bar{E} microCi/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

* A radiochemical analysis for \bar{E} shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of \bar{E} for the reactor coolant sample. Determination of the contributors to \bar{E} shall be based upon those energy peaks identifiable with a 95% confidence level.

** Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

GENERAL

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 REQUIRMENTS

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

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 200°F in any 1-hour period, and
- c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3

ACTION: (Continued)

- f) With more than one charging pump capable of injecting into the RCS, immediately initiate action to restore a maximum of one charging pump capable of injecting into the RCS.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORV(s) are being used for overpressure protection by:

- a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, in accordance with the Surveillance Frequency Control Program thereafter when the PORV is required OPERABLE; and
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel in accordance with the Surveillance Frequency Control Program; and
- c. Verifying the PORV isolation valve is open in accordance with the Surveillance Frequency Control Program.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valve(s) are being used for overpressure protection as follows:

- a. For RHR suction relief valve RC-V89 by verifying in accordance with the Surveillance Frequency Control Program that RHR suction isolation valves RC-V87 and RC-V88 are open.
- b. For RHR suction relief valve RC-V24 by verifying in accordance with the Surveillance Frequency Control Program that RHR suction isolation valves RC-V22 and RC-V23 are open.
- c. Testing pursuant to Specification 4.0.5.

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

**Except when the vent pathway is provided with a valve(s) or device(s) that is locked, sealed, or otherwise secured in the open position, then verify this valve(s) or device(s) open in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

OVERPRESSURE PROTECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.4 The reactor vessel water level shall be verified to be lower than 36 inches below the reactor vessel flange in accordance with the Surveillance Frequency Control Program when the reduced inventory condition is being used for overpressure protection.

4.4.9.3.5 All charging pumps, excluding one OPERABLE pump, shall be demonstrated inoperable^{***} by verifying that the motor circuit breakers are secured in the open position^{****} in accordance with the Surveillance Frequency Control Program, except when the reactor vessel head closure bolts are fully detensioned or the vessel head is removed.

^{***} An additional pump may be made capable of injecting under administrative control for up to 1 hour during pump-swap operation, except during RCS water-solid conditions. Additionally, an inoperable pump may be energized for testing 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.

^{****} An alternate method to assure pump inoperability may be used by placing the control room pump-control switch in the Pull-to-Lock position and isolating the discharge flow path of the pump from the RCS by a least one closed isolation valve. Use of the alternative method requires inoperability verification in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of one vent valve and one block valve powered from emergency busses shall be OPERABLE and closed* at each of the following locations:

- a. Reactor vessel head, and
- b. Pressurizer steam space.

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

ACTION:

- a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With both Reactor Coolant System vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per COLD SHUTDOWN, if not performed within the previous 92 days, by operating the valve through one complete cycle of full travel from the control room.

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

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

*For an OPERABLE vent path using a power-operated relief valve (PORV) as the vent path, the PORV block valve is not required to be closed.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

HOT STANDBY, STARTUP, AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

- a. The isolation valve open and power removed,
- b. A contained borated water volume of between 6121 and 6596 gallons,
- c. A boron concentration of between the limits specified in the COLR, and
- d. A nitrogen cover-pressure of between 585 and 664 psig.

APPLICABILITY: MODES 1, 2, and 3*

ACTION:

- a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.
- b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.
- c. With one pressure or water level channel inoperable per accumulator, return the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- d. With two pressure channels or two water level channels inoperable per accumulator, immediately declare the affected accumulator(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by:
 - 1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

ACCUMULATORS

HOT STANDBY, STARTUP, AND POWER OPERATION

SURVEILLANCE REQUIREMENTS

4.5.1.1 (Continued)

- 2) Verifying that each accumulator isolation valve is open.
- b. By verifying the boron concentration of the accumulator solution under the following conditions:
- 1) In accordance with the Surveillance Frequency Control Program,
 - 2) Within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume. This surveillance is not required when the volume increase makeup source is the RWST and the RWST has not been diluted since verifying that the RWST boron concentration is equal to or greater than the accumulator boron concentration limit.
- c. 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 disconnected.
- d. In accordance with the Surveillance Frequency Control Program by verifying that each accumulator isolation valve opens automatically under each of the following conditions:
- 1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and
 - 2) Upon receipt of a Safety Injection test signal.

EMERGENCY CORE COOLING SYSTEMS

ACCUMULATORS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each reactor coolant system accumulator isolation valve shall be shut with power removed from the valve operator.

APPLICABILITY: MODES 4* and 5**.

ACTION:

With one or more accumulator isolation valve(s) open and/or power available to the valve operator(s), immediately close the accumulator isolation valves and/or remove power from the valve operator(s).

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each accumulator isolation valve will be verified shut with power removed from the valve operator in accordance with the Surveillance Frequency Control Program.

*Within 12 hours prior to entry into MODE 3 from MODE 4 and if pressurizer pressure is greater than 1000 psig, each accumulator isolation valve shall be open as required by Specification 3.5.1.1.a.

**With accumulator pressure greater than 100 psig.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

- a. 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:

<u>Valve Number</u>	<u>Valve Function</u>	<u>Valve Position</u>
SI-V-3	Accumulator Isolation	Open*
SI-V-17	Accumulator Isolation	Open*
SI-V-32	Accumulator Isolation	Open*
SI-V-47	Accumulator Isolation	Open*
SI-V-114	SI Pump to Cold-Leg Isolation	Open
RH-V-14	RHR Pump to Cold-Leg Isolation	Open
RH-V-26	RHR Pump to Cold-Leg Isolation	Open
RH-V-32	RHR to Hot-Leg Isolation	Closed
RH-V-70	RHR to Hot-Leg Isolation	Closed
SI-V-77	SI to Hot-Leg Isolation	Closed
SI-V-102	SI to Hot-Leg Isolation	Closed

- b. In accordance with the Surveillance Frequency Control Program by:
- 1) Verifying that the ECCS piping is full of water, and
 - 2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- c. By 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 suction during LOCA conditions. This visual inspection shall be performed:
- 1) For all accessible areas of the containment prior to establishing primary CONTAINMENT INTEGRITY, and
 - 2) At least once daily of the areas affected within containment by containment entry and during the final entry when primary CONTAINMENT INTEGRITY is established.

*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

- d. In accordance with the Surveillance Frequency Control Program by:
 - 1) Verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 440 psig, the interlocks prevent the valves from being opened.
 - 2) 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. In accordance with the Surveillance Frequency Control Program, during shutdown, by:
 - 1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and
 - 2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:
 - a) Centrifugal charging pump,
 - b) Safety Injection pump, and
 - c) RHR pump.

- f. By verifying OPERABILITY of each pump when tested pursuant to Specification 4.0.5:
 - 1) Centrifugal charging pump;
 - 2) Safety Injection pump; and
 - 3) RHR pump.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

- g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:
- 1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and
 - 2) In accordance with the Surveillance Frequency Control Program.

High Head SI System
Valve Number

SI-V-143
SI-V-147
SI-V-151
SI-V-155

Intermediate Head SI System
Valve Number

SI-V-80
SI-V-85
SI-V-104
SI-V-109
SI-V-117
SI-V-121
SI-V-125
SI-V-129

- h. NOT USED

EMERGENCY CORE COOLING SYSTEMS

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

SURVEILLANCE REQUIREMENTS

- 4.5.3.1.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.
- 4.5.3.1.2 All centrifugal charging pumps and 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 in accordance with the Surveillance Frequency Control Program thereafter.

*An additional charging pump may be made capable of injecting under administrative control for up to 1 hour during pump-swap operation, except during RCS water-solid conditions. Additionally, 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.

**An alternate method to assure pump inoperability may be used by placing the control room pump-control switch(s) in the Pull-to-Lock position and isolating the discharge flow path of the pump(s) from the RCS by at least one closed isolation valve. Use of the alternate method requires inoperability verification in accordance with the Surveillance Frequency Control Program.

EMERGENCY CORE COOLING SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.5.3.2 As a minimum, the following number of Safety Injection pumps shall be inoperable*:

- a. Two when the RCS vent area is less than 18 square inches.
- b. One when the RCS vent area is equal to or greater than 18 square inches,
or
- c. One when the RCS is in a reduced inventory condition**.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.

ACTION:

With fewer than the required number of Safety Injection pumps inoperable, immediately restore all pumps required to inoperable status.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps required to be inoperable shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position 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.

** A reduced inventory condition exists whenever reactor vessel (RV) water level is lower than 36 inches below the RV flange.

*** An alternate method to assure pump inoperability may be used by placing the control room pump-control switch(s) in the Pull-to-Lock position and isolating the discharge flow path of the pump(s) from the RCS by at least one closed isolation valve. Use of the alternate method requires inoperability verification in accordance with the Surveillance Frequency Control Program.

BORON INJECTION SYSTEM

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

- 3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:
- a. A minimum contained borated water volume of 477,000 gallons,
 - b. A boron concentration between the limits specified in the COLR,
 - c. A minimum solution temperature of 50°F, and
 - d. A maximum solution temperature of 98°F.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

- 4.5.4 The RWST shall be demonstrated OPERABLE:
- a. In accordance with the Surveillance Frequency Control Program by:
 - 1) Verifying the contained borated water volume in the tank, and
 - 2) Verifying the boron concentration of the water.
 - b. In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature.

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. 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 for valves that are open under administrative control as permitted by Specification 3.6.3; and
- 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 once per 92 days.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

CONTAINMENT AIR-LOCKS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. With the leakage rate in accordance with the Containment Leakage Rate Testing Program.
- b. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between 14.6 and 16.2 psia.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

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

ACTION:

With the containment average air temperature greater than 120°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 the temperatures at the following locations and shall be determined in accordance with the Surveillance Frequency Control Program:

Location

- a. Elevation 45 feet
- b. Elevation 71 feet
- c. Elevation 110 feet
- d. Elevation 130 feet

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

CONTAINMENT VENTILATION SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is in accordance with the Containment Leakage Rate Testing Program.

4.6.1.7.2 Each 8-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7 in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST* and automatically transferring suction to the containment sump.

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;
- b. By verifying OPERABILITY of each pump when tested pursuant to Specification 4.0.5;
- c. In accordance with the Surveillance Frequency Control Program during shutdown, by:
 - 1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal, and
 - 2) Verifying that each spray pump starts automatically on a Containment Pressure-Hi-3 test signal.
- d. By verifying each spray nozzle is unobstructed following activities that could result in nozzle blockage.

*In MODE 4, when the Residual Heat Removal System is in operation, an OPERABLE flow path is one that is capable of taking suction from the refueling water storage tank upon being manually realigned.

CONTAINMENT SYSTEMS

DEPRESSURIZATION AND COOLING SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

- a. A spray additive tank containing a volume of between 9420 and 9650 gallons of between 19 and 21% by weight NaOH solution, and
- b. Two gravity feed paths each capable of adding NaOH solution from the chemical additive tank to the Refueling Water Storage Tank.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;
- b. In accordance with the Surveillance Frequency Control Program by:
 - 1) Verifying the contained solution volume in the tank, and
 - 2) Verifying the concentration of the NaOH solution by chemical analysis.
- c. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE*.

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

ACTION:

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

- a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or
- b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or
- c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or
- d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Not used

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE during shutdown in accordance with the Surveillance Frequency Control Program by:

- a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" Isolation valve actuates to its isolation position,
- b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" Isolation valve actuates to its isolation position, and

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

CONTAINMENT SYSTEMS

COMBUSTIBLE GAS CONTROL

HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 Two independent Containment Structure Recirculation Fan Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Containment Structure Recirculation Fan inoperable, restore the inoperable fan to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each Containment Structure Recirculation Fan System shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by starting each system from the control room and verifying that the system operates for at least 15 minutes, and
- b. In accordance with the Surveillance Frequency Control Program by verifying a system flow rate of at least 4000 cfm through the hydrogen mixing flow path.

CONTAINMENT SYSTEMS

3/4.6.5 CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent Containment Enclosure Emergency Air Cleanup System trains shall be OPERABLE.

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

ACTION:

With one Containment Enclosure Emergency Air Cleanup System train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each Containment Enclosure Emergency Air Cleanup System train shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;
- b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:
 - 1) Verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978*, and the system flow rate is 2100 cfm \pm 10%;
 - 2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or

* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Rev. 2, March 1978.

CONTAINMENT SYSTEMS

CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.5.1b.2 (Continued)

equal to 5% when tested at a temperature of 30°C, at a relative humidity of 95% and a face velocity of 46 fpm in accordance with ASTM-D3803-1989; and

- 3) Verifying a system flow rate of 2100 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1980.
- c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 5% when tested at a temperature of 30°C, at a relative humidity of 95% and a face velocity of 46 fpm in accordance with ASTM-D3803-1989.
- d. In accordance with the Surveillance Frequency Control Program by:
- 1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 2100 cfm \pm 10%,
 - 2) Verifying that the system starts on a Safety Injection test signal, and
 - 3) Verifying that the filter cross connect valves can be manually opened.
- e. After each complete or partial replacement of a high efficiency particulate air (HEPA) filter bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a dioctyl phthalate (DOP) test aerosol while operating the system at a flow rate of 2100 cfm \pm 10%; and
- f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 2100 cfm \pm 10%.

CONTAINMENT SYSTEMS

CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.2 Containment enclosure building integrity shall be maintained.

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

-----NOTE-----

Entry into ACTION is not required when the access opening is being used for normal transit entry or exit.

ACTION:

- a. Without containment enclosure building integrity for reasons other than Action b, restore containment enclosure building integrity within 12 hours. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. Without containment enclosure building integrity when equipment ingress and egress requires the access door to be maintained open, verify a dedicated individual, who is in continuous communication with the control room, is available to rapidly close the door; and restore containment enclosure building integrity within 24 hours. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Containment enclosure building integrity shall be demonstrated:

- a. In accordance with the Surveillance Frequency Control Program by verifying that the door in each access opening is closed except when the access opening is being used for normal transit entry and exit, and
- b. In accordance with the Surveillance Frequency Control Program by verifying the containment enclosure building can be maintained at a negative pressure greater than or equal to 0.25 inch water gauge by one train of the containment enclosure emergency air cleanup system within 4 minutes after a start signal.

PLANT SYSTEMS

TURBINE CYCLE

AUXILIARY FEEDWATER SYSTEM

SURVEILLANCE REQUIREMENTS

- 4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:
- a. In accordance with the Surveillance Frequency Control Program by:
 - 1) 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;
 - 2) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and
 - 3) Verifying that valves FW-156 and FW-163 are OPERABLE for alignment of the startup feedwater pump to the emergency feedwater header.
 - b. In accordance with the Surveillance Frequency Control Program by verifying the following pumps develop the required discharge pressure and flow as specified in the Technical Requirements Manual:
 - 1) The motor-driven emergency feedwater pump;
 - 2) The steam turbine-driven emergency feedwater pump when the secondary steam supply pressure is greater than 500 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;
 - 3) The startup feedwater pump.
 - c. In accordance with the Surveillance Frequency Control Program during shutdown by:
 - 1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Emergency Feedwater System Actuation test signal;
 - 2) Verifying that each emergency feedwater pump starts as designed automatically upon receipt of an Emergency Feedwater Actuation System test signal;

PLANT SYSTEMS

TURBINE CYCLE

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) system shall be OPERABLE with

- a. A volume of 212,000 gallons of water contained in the condensate storage tank, and
- b. A concrete CST enclosure that is capable of retaining 212,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST or the CST enclosure inoperable, within 4 hours restore the CST and the CST enclosure to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

- 4.7.1.3a. The CST shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the contained water volume in the CST is within its limits.
- b. The CST enclosure shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by an inspection to verify that CST enclosure integrity is maintained.

PLANT SYSTEMS

TURBINE CYCLE

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant shall be less than or equal to 0.1 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant greater than 0.1 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 In accordance with the Surveillance Frequency Control Program, verify the specific activity of the secondary coolant is less than or equal to 0.1 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.

*The provisions of Specification 4.0.4 are not applicable for entry into MODE 4, however, once steam generator pressure exceeds 100 psig, the requirements of Specification 4.7.1.4 must be met within 12 hours if not performed within the past 31 days.

PLANT SYSTEMS

TURBINE CYCLE

ATMOSPHERIC RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 At least four atmospheric relief valves and associated manual controls including the safety-related gas supply systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3[#], and 4^{**}.

ACTION:

- a. With one less than the required atmospheric relief valves OPERABLE, restore the required atmospheric relief valves to OPERABLE status within 7 days; or be in at least HOT STANDBY within the next 12 hours.
- b. With two less than the required atmospheric relief valves OPERABLE, restore at least three atmospheric relief valves to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric relief valve and associated manual controls including the safety-related gas supply systems shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that the nitrogen accumulator tank is at a pressure greater than or equal to 500 psig.
- b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, verify that all valves will open and close fully by operation of manual controls.

*When steam generators are being used for decay heat removal.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

PLANT SYSTEMS

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent primary component cooling water loops shall be OPERABLE, including one OPERABLE pump in each loop.

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

ACTION:

With one primary component cooling water (PCCW) loop inoperable, restore the required primary component cooling water loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and
- b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation signal.

PLANT SYSTEMS

3/4.7.4 SERVICE WATER SYSTEM/ULTIMATE HEAT SINK

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and
- b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal.

4.7.4.2 Each service water cooling tower loop shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and
- b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:
 - 1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal,
 - 2) Each automatic valve in the flowpath actuates to its correct position on a Tower Actuation (TA) test signal and
 - 3) Each service water cooling tower pump starts automatically on a TA signal.

4.7.4.3 The service water pumphouse shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the water level to be at or above 25.1' (-15.9' Mean Sea Level).

4.7.4.4 The mechanical draft cooling tower shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying the water in the mechanical draft cooling tower basin to be at a level of greater than or equal to 42.15* feet.
- b. In accordance with the Surveillance Frequency Control Program by verifying that the water in the cooling tower basin to be at a bulk average temperature of less than or equal to 70°F.

*With the cooling tower in operation with valves aligned for tunnel heat treatment, the tower basin level shall be maintained at greater than or equal to 40.55 feet.

PLANT SYSTEMS

3/4.7.4 SERVICE WATER SYSTEM/UTIMATE HEAT SINK

SURVEILLANCE REQUIREMENTS

- c. In accordance with the Surveillance Frequency Control Program by:
 - 1) Starting from the control room each cooling tower fan that is required to be OPERABLE and operating each of these fans for at least 15 minutes, and
 - 2) Verifying that the portable tower makeup pump system is stored in its design operational readiness state.

- d. In accordance with the Surveillance Frequency Control Program by verifying that the portable tower makeup pump develops a flow greater than or equal to 200 gpm.

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM SUBSYSTEM

EMERGENCY MAKEUP AIR AND FILTRATION

LIMITING CONDITION FOR OPERATION (Continued)

In MODE 5 or 6, or during movement of irradiated fuel assemblies:

- d. With one CREMAFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable system to OPERABLE status within 7 days or either immediately initiate and maintain operation of the remaining OPERABLE CREMAFS train in the filtration/recirculation mode or immediately suspend movement of irradiated fuel assemblies.
- e. With both CREMAFS trains inoperable, or with the OPERABLE CREMAFS train, required to be in the filtration/recirculation mode by ACTION d., not capable of being powered by an OPERABLE emergency power source, immediately suspend all movement of irradiated fuel assemblies.
- f. With one or both CREMAFS trains inoperable due to an inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each CREMAFS train shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

PLANT SYSTEMS

CONTROL ROOM SUBSYSTEMS

EMERGENCY MAKEUP AIR AND FILTRATION

SURVEILLANCE REQUIREMENTS (Continued)

- b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:
- 1) Verifying that the filtration system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% and uses the test procedure guidance in Regulatory Position 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 1100 cfm \pm 10%;
 - 2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 34.5 fpm (Train A) and 58.3 fpm (Train B) in accordance with ASTM-D-3803-1989;
 - 3) Verifying a system flow rate of 1100 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1980.
- c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 34.5 fpm (Train A) and 58.3 fpm (Train B) in accordance with ASTM-D-3803-1989;
- d. In accordance with the Surveillance Frequency Control Program by:
- 1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks, for filter CBA-F-38, is less than 2.8 inches Water Gauge while operating the system at a flow rate of 1100 cfm \pm 10%; and verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks, for filter CBA-F-8038, is less than 6.3 inches Water Gauge while operating the system at a flow rate of 1100 cfm \pm 10%.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 as referenced in Regulatory Guide 1.52, Revision 2, March 1978.

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM SUBSYSTEMS

AIR CONDITIONING

LIMITING CONDITION FOR OPERATION

3.7.6.2 Two independent Control Room Air Conditioning Subsystems shall be OPERABLE.

APPLICABILITY: All MODES

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Air Conditioning Subsystem inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

- a. With one Control Room Air Conditioning Subsystem inoperable, restore the inoperable system to OPERABLE status within 30 days or initiate and maintain operation of the remaining OPERABLE Control Room Air Conditioning Subsystem or immediately suspend all operations involving CORE ALTERATION.
- b. With both Control Room Air Conditioning Subsystems inoperable, or with the OPERABLE Control Room Air Conditioning Subsystem unable to maintain temperature below the limiting equipment qualification temperature in the control room area, suspend all operations involving CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each Control Room Air Conditioning Subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the ability to maintain temperature in the control room area below the limiting equipment qualification temperature for 24 hours.

PLANT SYSTEMS

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

- a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:
 1. Decontaminate and repair the sealed source, or
 2. Dispose of the sealed source in accordance with Commission Regulations.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

- a. The licensee, or
- b. Other persons specifically authorized by the Commission or an Agreement State.

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

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

- a. Sources in use - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive materials:
 1. With a half-life greater than 30 days (excluding Hydrogen 3), and
 2. In any form other than gas.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

SURVEILLANCE REQUIREMENTS

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

- a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability, and
- b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.*

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE.**

- a. In accordance with the Surveillance Frequency Control Program by:
 - 1) Verifying the fuel level in the day fuel tank;
 - 2) Verifying the fuel level in the fuel storage tank;
 - 3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank;
 - 4) Verifying the lubricating oil inventory in storage;
 - 5) Verifying the diesel starts from standby conditions and attains a steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.***

* This surveillance requirement shall not be performed in Mode 1 or 2.

** All planned starts for the purpose of these surveillances may be preceded by an engine pre-lube period.

*** A modified start involving idling and gradual acceleration to synchronous speed may be used for this surveillance. When modified start procedures are not used, the time, voltage, and frequency tolerances of Specification 4.8.1.1.2e must be met.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

- 6) Verifying the generator is synchronized, gradually loaded**** to greater than or equal to 5600 kW and less than or equal to 6100 kW, and operates within this load band for at least 60 minutes, and until stable engine operating temperature is attained; and
- 7) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.
 - b. In accordance with the Surveillance Frequency Control Program by checking for and removing accumulated water from the day fuel tank;
 - c. In accordance with the Surveillance Frequency Control Program by checking for and removing accumulated water from the fuel oil storage tanks;
 - d. By verifying fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program;
 - e. In accordance with the Surveillance Frequency Control Program# by verifying the diesel starts from standby condition and achieves:
 - 1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and
 - 2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

**** Diesel generator loading may be in accordance with manufacturers recommendations, including a warmup period. Momentary transients outside the load range, due to changing bus conditions, do not invalidate the test. In addition, this surveillance shall be preceded by and immediately follow without shutdown a successful performance of Specification 4.8.1.1.2a.5) or 4.8.1.1.2e.

Performance of Specification 4.8.1.1.2a.6) must immediately follow this surveillance. Additionally, performance of Specification 4.8.1.1.2e satisfies Specification 4.8.1.1.2a.5).

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

- f. In accordance with the Surveillance Frequency Control Program, during shutdown^{##}, by:
- 1) (NOT USED)
 - 2) Verifying the generator capability to reject a load of greater than or equal to 671 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 4.0 Hz;
 - 3) Verifying the generator capability to reject a load of 6083 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection;
 - 4) Simulating a loss-of-offsite power by itself, and:
 - a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and
 - b) Verifying the diesel starts from standby conditions^{###} on the loss of offsite power signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected shutdown loads through the emergency power 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 emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.
 - 5) Verifying that on an SI actuation test signal, without loss-of-offsite power, the diesel generator starts from standby conditions^{###} on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test;

^{##} Selected surveillance requirements, or portions thereof, may be performed during conditions or modes other than shutdown, provided an evaluation supports safe conduct of that surveillance in a condition or mode that is consistent with safe operation of the plant. (Ref. NRC GL 91-04)

^{###} Starting of the diesel for Specifications 4.8.1.1.2f.4) and 4.8.1.1.2f.5) may be performed with the engine at or near normal operating temperature.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

- 14) Simulating a Tower Actuation (TA) signal while the diesel generator is loaded with the permanently connected loads and auto-connected emergency (accident) loads, and verifying that the service water pump automatically trips, and that the cooling tower pump automatically starts. After energization the steady state voltage and frequency of the emergency buses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz; and
 - 15) While diesel generator 1A is loaded with the permanently connected loads and auto-connected emergency (accident) loads, manually connect the 1500 hp startup feedwater pump to 4160-volt bus E5. After energization the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz.
- g. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously from standby condition, during shutdown, and verifying that both diesel generators achieve:
- 1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and
 - 2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE and energized:

- a. Train A
 - 1) 125-volt Battery Banks 1A and 1C,
 - 2) One full-capacity battery charger on Bus #11A, and
 - 3) One full-capacity battery charger on Bus #11C.

- b. Train B
 - 1) 125-volt Battery Banks 1B and 1D,
 - 2) One full-capacity battery charger on Bus #11B, and
 - 3) One full-capacity battery charger on Bus #11D.

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

ACTION:

- a. With one of the required battery banks in one train inoperable, close the bus tie to connect the remaining operable battery bank to the D.C. bus supplied by the inoperable battery bank within 2 hours; restore the inoperable battery bank to OPERABLE status within 30 days* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

- b. With one of the full-capacity chargers inoperable, restore the inoperable charger 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. In accordance with the Surveillance Frequency Control Program by verifying that:
 - 1) The parameters in Table 4.8-2 meet the Category A limits, and
 - 2) The total battery terminal voltage is greater than or equal to 128 volts on float charge.

- b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

*No more than one battery at a time may be taken out of service for more than 30 days.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

OPERATING

SURVEILLANCE REQUIREMENTS

4.8.2.1b (Continued)

- 1) The parameters in Table 4.8-2 meet the Category B limits,
 - 2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm,* and
 - 3) The average electrolyte temperature of 16 connected cells (4 cells per row) is above 65°F.
- c. In accordance with the Surveillance Frequency Control Program by verifying that:
- 1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,
 - 2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,
 - 3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm,* and
 - 4) Each battery charger will supply at least 150 amperes at a minimum of 132 volts for at least 8 hours.
- d. 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 emergency loads for the design duty cycle when the battery is subjected to a battery service test;
- e. In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and
- f. At least once per 18 months by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

* Obtained by subtracting the normal resistance of: (1) the cross room rack connector (210×10^{-6} ohm, typical) and (2) the bi-level rack connector (35×10^{-6} ohm, typical) from the measured cell-to-cell connection resistance.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

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

*No more than one Battery Bank (1A, 1B, 1C, or 1D) at a time may be taken out of service for more than 30 days.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical buses shall be energized in the specified manner:

- a. One train of A.C. emergency buses consisting of the 4160-volt and the 480-volt A.C. emergency buses listed in 3.8.3.1a. and b. (excluding 480-volt Emergency Bus #E64);
- b. Two of the four 120-volt A.C. vital Panels 1A, 1B, 1C, and 1D energized from their associated inverters connected to their respective D.C. buses;
- c. One of the two 120-volt A.C. Vital Panels 1E or 1F energized from its associated inverter connected to the respective D.C. bus; and
- d. Two 125-volt D.C. buses (in the same train) energized from their associated battery banks.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical buses and panels not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical buses and panels in the specified manner as soon as possible, and within 8 hours, depressurize and vent the RCS through at least a 1.58-square-inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified buses and panels shall be determined energized in the required manner 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

TRIP CIRCUIT FOR INVERTER I-2A

LIMITING CONDITION FOR OPERATION

3.8.3.3 The safety-related trip circuit that trips the D.C. feed from D.C. Bus #11C to inverter #I-2A after 15 minutes of discharge from the battery shall be OPERABLE. Note that this LIMITING CONDITION FOR OPERATION is applicable only when D.C. Bus #11C is required to be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.

ACTION:

With this safety-related trip circuit inoperable, restore the trip circuit to OPERABLE status within 7 days or de-energize the D.C. feed to inverter #I-2A by tripping the D.C. circuit breaker in D.C. Bus #11C. Verify that this breaker is open once per 7 days thereafter.

SURVEILLANCE REQUIREMENTS

4.8.3.3 The safety-related trip circuit shall be demonstrated operable in accordance with the Surveillance Frequency Control Program.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 The circuit breakers feeding the following loads inside primary containment shall be padlocked in the open position:

<u>Loads</u>	<u>Circuit</u>	<u>Panel</u>
Refueling Canal Skimmer Pump	1-SF-P-272	1-ED-MCC-111
Polar Gantry Crane	1-MM-CR-3	1-ED-US-11
Distribution Panel	1-ED-PP-7A	1-ED-US-11
Distribution Panel	1-ED-PP-7B	1-ED-US-23
Rod Control Cluster Change Fixture	1-FH-RE 12	1-ED-MCC-111

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

EXCEPTION:

If any of the above-mentioned loads are required for brief durations (not to exceed 72 hours) during plant operation, the pertinent circuit breaker can be unlocked and closed for the required duration provided this change in breaker position becomes part of the applicable operating procedure used for the work inside containment.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Verify in accordance with the Surveillance Frequency Control Program that the circuit breakers listed above are locked in the open position.

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.3 Each thermal overload protection for safety-related motor-operated valves shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above-required valves inoperable, bypass the inoperable thermal overload within 8 hours, restore the inoperable thermal overload to OPERABLE status within 30 days, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program and following maintenance on the motor starter by selection of a representative sample of at least 25% of all thermal overloads for the above-required valves and replacing them with precalibrated devices that have been subjected to a CHANNEL CALIBRATION.

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 a boron concentration of greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the boron concentration is restored to greater than or equal to the limit specified in the COLR.

SURVEILLANCE REQUIREMENTS

4.9.1.1 Verify boron concentration is within the limits specified in the COLR prior to:

- a. Removing or unbolting the reactor vessel head, and
- b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

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

*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.

REFUELING OPERATIONS

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 positive reactivity changes.
- b. With both of the above required monitors inoperable or not operating, immediately initiate corrective action to restore one source range neutron flux monitor to OPERABLE status and 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 in accordance with the Surveillance Frequency Control Program,
 - b. A CHANNEL CALIBRATION* in accordance with the Surveillance Frequency Control Program.

*Neutron detectors may be excluded from CHANNEL CALIBRATION.

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

SURVEILLANCE REQUIREMENTS

- 4.9.4 For the above required containment building penetrations:
- a. Determine that each of the above required containment building penetrations shall be in its required condition within 100 hours prior to the start of, and in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel in the containment building, and
 - b. Demonstrate that the Containment Purge and Exhaust Isolation System is OPERABLE in accordance with the Surveillance Frequency Control Program and within 10 days prior to the start of movement of recently irradiated fuel in the containment building by verifying that containment purge and exhaust isolation occurs on manual initiation and on a High Radiation test signal from each of the manipulator crane radiation area monitoring instrumentation channels. **

** Not required for those valves complying with Specification 3.9.4.c.1 or Specification 3.9.4.c.3.

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2750 gpm 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

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

- a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.
- b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2750 gpm 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.

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

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 containment 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 in accordance with the Surveillance Frequency Control Program thereafter during movement of fuel assemblies or control rods.

REFUELING OPERATIONS

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 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 storage pool.

ACTION:

- a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

REFUELING OPERATIONS

3/4.9.12 FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent trains of the Fuel Storage Building Emergency Air Cleaning System shall be OPERABLE whenever irradiated fuel is in the storage pool and shall be OPERABLE with one train operating during fuel movement.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

- a. With one train of the Fuel Storage Building Emergency Air Cleaning System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE train of the Fuel Storage Building Emergency Air System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.
- b. With no trains of the Fuel Storage Building Emergency Air Cleaning System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one train of the Fuel Storage Building Emergency Air Cleaning System is restored to OPERABLE status and is in operation.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required trains of the Fuel Storage Building Emergency Air Cleaning System shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;
- b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

REFUELING OPERATIONS

FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.9.12b (Continued)

- 1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 16,450 cfm \pm 10%;
 - 2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 44 fpm in accordance with ASTM-D-3803-1989; and
 - 3) Verifying a system flow rate of 16,450 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1980.
- c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 44 fpm in accordance with ASTM-D-3803-1989.
- d. In accordance with the Surveillance Frequency Control Program by:
- 1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 16,450 cfm \pm 10%,
 - 2) Verifying that the system maintains the spent fuel storage pool area at a negative-pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation,

* ANSI N510-1980 shall be used in place of ANSI N510-1975 as referenced in Regulatory Guide 1.52, Rev. 2, March 1978.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

- a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.
- b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

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

SPECIAL TEST EXCEPTIONS

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

LIMITING CONDITION FOR OPERATION

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

- a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and
- b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

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

- a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or
- b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.10.2.2 The requirements of Specifications 4.2.2.2, 4.2.2.3, and 4.2.3.2 shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

- a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,
- b. The Reactor Trip Setpoints on the OPERABLE Intermediate Range 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 541°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 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Verify each OPERABLE Intermediate Range and Power Range* channel has been subjected to an ANALOG CHANNEL OPERATIONAL TEST per Specification Table 4.3-1 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 541°F in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

*Power Range Low Setpoint only.

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

- a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and
- b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and in accordance with the Surveillance Frequency Control Program thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps when the rods are stationary.

*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) k_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each temporary 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 temporary 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 6.8.1.4.
- 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 temporary unprotected outdoor tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

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

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

- e. The quantitative limits on unfiltered air in-leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air in-leakage measured by the testing described in paragraph c. The unfiltered air in-leakage limit for radiological challenges is the in-leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air in-leakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.
- f. The provisions of SR 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered in-leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

m. Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 at least once every 20 years. In lieu of Position C.4.b(1) and C.4.b(2), this inspection shall be by either of the following examinations:

- a. An in-place examination, utilizing ultrasonic testing, over the volume from the inner bore of the flywheel to the circle of one-half the outer radius; or
- b. A surface examination, utilizing magnetic particle testing and/or penetrant testing, of the exposed surfaces of the disassembled flywheel.

n. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

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

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
- c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

6.8 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.8.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

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

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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



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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141

TO FACILITY OPERATING LICENSE NO. NPF-86

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By application dated May 28, 2013, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13155A002), as supplemented by letters dated July 31, 2013 (ADAMS Accession No. ML13217A370), January 29, 2014 (ADAMS Accession No. ML14035A457), and March 26, 2014 (ADAMS Accession No. ML14092A360), NextEra Energy Seabrook, LLC (NextEra, the licensee) requested changes to the technical specifications (TSs) for Seabrook Station, Unit 1 (Seabrook). The supplements dated July 31, 2013, January 29, 2014, and March 26, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 20, 2013 (78 FR 51227).

The requested change is the adoption of NRC-approved TS Task Force (TSTF)-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – [Risk Informed Technical Specifications Task Force] RITSTF Initiative 5b" (ADAMS Accession No. ML090850642). When implemented, TSTF-425, Revision 3 (Rev. 3) relocates most periodic frequencies of TS surveillances to a licensee-controlled program, the Surveillance Frequency Control Program (SFCP), and provides requirements for the new program in the administrative controls section of the TS. All surveillance frequencies can be relocated except:

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

A new Administrative Controls Program is added to TS Section 6. The new program is called the SFCP and describes the requirements for the program to control changes to the relocated surveillance frequencies. The TS Bases for each affected surveillance are revised to state that the frequency is set in accordance with the SFCP. Various editorial changes have been made to the Bases to facilitate the addition of the Bases changes. Some surveillance Bases do not contain a discussion of the frequency. In these cases, Bases describing the current frequency were added to maintain consistency with the Bases for similar surveillances. These instances are noted in the markup along with the source of the text. The proposed licensee changes to the administrative controls of TS to incorporate the SFCP includes a specific reference to Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1 (Rev. 1), (ADAMS Accession No. ML071360456) as the basis for making any changes to the surveillance frequencies once they are relocated out of TS.

In a letter dated September 19, 2007, the NRC staff approved NEI Topical Report (TR) 04-10, Rev. 1, "Risk-Informed Technical Specification initiative 5B, Risk Informed Method for Control of Surveillance Frequencies" (ADAMS Accession No. ML072570267), as acceptable for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04-10, Rev. 1, and the final acceptance SE providing the basis for NRC acceptance of NEI 04-10, Rev 1.

2.0 REGULATORY EVALUATION

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

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

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

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

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, the NRC 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) SRs; (4) design features; and (5) administrative controls. As stated in 10 CFR 50.36(c)(3), "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

These categories will remain in TS. The new TS SFCP provides the necessary administrative controls to require that surveillances relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, Rev. 1, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of SSCs; and are required to be documented. Furthermore, 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 (ROP).

The licensee SFCP ensures that SRs 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, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," 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, Rev. 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 SRs are retained in TSs, but the related surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the In-Service Testing Program and the Primary Containment Leakage Rate Testing Program.

Thus, this proposed change complies with 10 CFR 50.36(c)(3) by retaining the requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met and meets the first key safety principle articulated in

Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998 (ADAMS Accession No. ML003740176) for plant-specific, risk-informed TS changes by complying with current regulations.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (ADAMS Accession No. ML090410014), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light water-reactors.

Licensees are required by TS to perform surveillance test, calibration, or inspection on specific safety-related system 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, Rev. 1, provides a way to establish risk-informed surveillance frequencies that complements the deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

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

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425, Rev. 3, provides for administrative relocation of applicable surveillance frequencies, and provides for the addition of the SFCP to the administrative controls of TS. TSTF-425, Rev. 3, also requires the application of NEI 04-10, Rev. 1, for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes proposed in TSTF-425, Rev. 3, included documentation regarding the PRA technical adequacy consistent with the requirements of RG 1.200, Section 4.2.

In accordance with NEI 04-10, Rev. 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 RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," August 1998 (ADAMS Accession No. ML003740133), and RG 1.177 in support of changes to surveillance test intervals.

3.1 RG 1.177 Five Key Safety Principles

RG 1.177 identifies five key safety principles required for risk-informed changes to TS. Each of these principles is addressed by the industry methodology document, NEI 04-10, Rev. 1. The second through the fifth principles, which relate to the technical aspects of the proposed change, are discussed below in Sections 3.1 through 3.4. The first principle requires the proposed change to meet the current regulations. The NRC staff finds that the change meets that requirement.

3.1.1 The Proposed Change Meets Current Regulations.

The regulations in 10 CFR 50.36(c)(3) provide 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 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 In-service 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.

3.1.2 The Proposed Change Is Consistent With The Defense-In-Depth Philosophy.

Consistency with the defense-in-depth philosophy, the second key safety principle of RG 1.177, is maintained if:

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

TSTF-425, Rev. 3, requires the application of NEI 04-10, Rev. 1, for any changes to surveillance frequencies within the SFCP. NEI 04-10, Rev. 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. 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 common cause failures. Defense-in-depth is also included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to increased likelihood of common cause failures.

Both the quantitative risk analysis and the qualitative considerations assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177.

3.1.3 The proposed change maintains sufficient safety margins.

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

The design, operation, testing methods, and acceptance criteria for SSCs, specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the Updated 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.

3.1.4 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, which requires identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. TSTF-425, Rev. 3, requires application of NEI 04-10, Rev. 1, in the SFCP. NEI 04-10, Rev. 1, satisfies the intent of RG 1.177 requirements for evaluation of the change in risk, and for assuring that such changes are small by providing the technical methodology to support risk informed technical specifications for control of surveillance frequencies.

3.1.4.1 Quality of the PRA

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

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

Review Process Guidelines,” (ADAMS Accession No. ML061510621); and NEI 05-04, “Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard.” Revision 1 of RG 1.200 (ADAMS Accession No. ML070240001), had endorsed the internal events PRA standard ASME RA-Sb-2005, “Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications.” For the internal events PRA, there are no significant technical differences in the standard requirements, and therefore assessments using the previously endorsed internal events standard are acceptable.

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

An Industry PSA Certification peer review of the Seabrook internal events and internal flood PRA was performed in 1999. All findings and observations (F&Os) from this peer review have been addressed. Focused scope peer reviews were conducted in 2005, 2009 and 2012. The 2009 peer review was for the internal flood PRA and the 2012 peer review was for the element LE (LERF analysis). In addition, the licensee performed self-assessments of the Seabrook internal events and internal flood PRA in 2010 and 2011, using the ASME PRA Standard, ASME RA-Sa-2009. Attachment A to Attachment 2 of the license amendment request application provides a list of all significant F&Os from the peer reviews. Attachment B to Attachment 2 of the license amendment request application provides a list of the open F&Os from the 2012 focused scope peer review which do not currently meet Capability Category II.

The NRC staff’s evaluation of important F&Os is summarized below:

F&O IE-2 (1999) for Supporting Requirement IE-C10. The F&O cites the common cause initiators L2CCA and L2CCB using 24 hours rather than the annual frequency. The licensee stated that it corrected this issue in the 1999 PRA update. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O IE-6 (1999) for Supporting Requirement IE-C14. The F&O recommended that the interfacing-systems loss-of-coolant accident (ISLOCA) analyses should be reviewed for consistency with a methodology such as NUREG/CR-5744. The licensee stated that it reviewed and updated its ISLOCA methodology against NUREG/CR-5744. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O AS-6 (1999) for Supporting Requirement DA-C15. The F&O is related to emergency diesel generator recovery time. The licensee stated that it used plant-specific experience for repair data. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O DA-4 (1999) for Supporting Requirement DA-D5. The F&O concerned estimating common cause failure (CCF) parameters for significant CCF events. The licensee stated that it recalculated the parameters with the correct equations and used appropriate distributions. The

NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O DA-6 (1999) for Supporting Requirement DA-C15. The F&O concerned identifying and evaluating the dependencies associated with Human Error Probabilities (HEPs) used for modeling recovery actions. The licensee stated that operator dependencies were examined, and that logic rules and HEP quantification changes which resulted from the review were included in the 1999 PRA update. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O QU-3 (1999) for Supporting Requirement QU-B2. The F&O relates to the limitations of saved sequences. The licensee stated that it addressed this issue of truncation in the PRA documentation. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O QU-9 (1999) for Supporting Requirement QU-E4. The F&O states that there is no parametric uncertainty analysis for the current version of the PRA model. The licensee stated that it performed the uncertainty analysis and quantified significant sequences for CDF and LERF. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O MU-2 (1999) for Supporting Requirement SY-A2. The F&O identified a single design change that did not appear to be completely incorporated in the PRA. The licensee stated that it verified the change had been incorporated in the PRA and updated the system notebook. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O AS-A9-1 (2005) for Supporting Requirement AS-A9. The F&O noted that the PRA could use more realistic analyses such as Modular Accident Analysis Program (MAAP). The licensee stated it used MAAP to add plant-specific, realistic support to its analyses for the 2005 update. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O HR-E3-1 (2005) for Supporting Requirement HR-E3. The F&O documented a concern that talk-throughs with operators had not been performed. The licensee stated it performed walkthroughs and talk-throughs with Operations and Training staff to confirm PRA modeling. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O HR-G4-1 (2005) for Supporting Requirement HR-G4. The F&O is related to using realistic codes to establish timeframes for operator actions. The licensee stated that it revised the Human Reliability Analysis (HRA) calculator quantification using Seabrook-specific MAAP runs. In letter dated January 29, 2014, the licensee response to a request for additional information (RAI) noted that HRA specifies the time of cues and points in time which operators are expected to receive relevant indications. The NRC staff concludes that, based on this information, the licensee has adequately dispositioned this F&O for the application.

F&O HR-G7-1 (2005) for Supporting Requirement HR-G7. The F&O concerns the identification of sequences with multiple operator actions for inclusion in the dependency

analysis. The licensee stated that it identified all dynamic actions in hardware top events, created new operator top events, added new top events to event trees, and modified logic rule to account for operator action dependencies. In letter dated January 29, 2014, the licensee response to an RAI confirmed that the Seabrook PRA accounts for influence factors that could lead to dependencies. The NRC staff concludes that, based on this information, the licensee has adequately dispositioned this F&O for the application.

F&O 4-9 (2009) for Supporting Requirement IFQU-B3. The F&O stated that certain internal flooding assumptions and uncertainties need to be reviewed against other industry studies. The licensee completed this review and updated the documentation. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O 5-2 (2009) for Supporting Requirement IFSO-B3. The F&O notes the need to expand uncertainty review with respect to other flood sources. The licensee revised the uncertainty review and performed a sensitivity evaluation for a postulated circulating water break flood. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O 5-3 (2009) for Supporting Requirement IFSN-A2. The F&O related to the crediting of "rugged" doors withstanding water higher than 3 feet. The licensee performed a structural evaluation of typical doors at Seabrook and included the results in the PRA documentation. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O LE-C3-01 (2012) for Supporting Requirement LE-C3. The F&O indicated that no credit was taken for repair of equipment other than the recovery of AC power and there was no evidence of a review for opportunities to credit repairs. The licensee stated this is a documentation issue for not taking credit for LERF sequences. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O LE-C5-01 (2012) for Supporting Requirement LE-C5. The F&O notes that there was no basis for auxiliary feedwater success criteria. The licensee stated this is a documentation issue and credit was based on realistic generic analyses (i.e., Pressurized-Water Reactor Owners Group (PWROG) analyses for severe accident management guidelines (SAMGs)). The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O LE-C10-01 (2012) for Supporting Requirement LE-C10 and LE-C12. The F&O cites a lack of evidence for review of significant accident sequences resulting in a large early release to determine if engineering analyses or operator actions during accident progression and after containment failure that could reduce LERF. The licensee stated that SAMG actions and associated equipment were credited where feasible. In letter dated January 29, 2014, the licensee's response to an RAI provided a summary of the review conducted of equipment and operator actions for significant accident progression sequences. The NRC staff concludes that, based on this information, the licensee has adequately dispositioned this F&O for the application.

F&O LE-D6-01 (2012) for Supporting Requirement LE-D6. The F&O states that the analysis for steam generator tube rupture (SGTR): (1) does not consider secondary side conditions;

and (2) conditional probabilities should be reassessed. The licensee stated these documentation issues: (1) the Seabrook PRA documents the failure probabilities for induced SGTR due to a bounding failure of main steam line safety valves; and (2) the licensee addressed the conditional probabilities of thermally-induced tube ruptures. The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

F&O LE-C5-01 (2012) for Supporting Requirement LE-C5. The F&O notes that there was no basis for auxiliary feedwater success criteria. The licensee stated this is a documentation issue and credit was based on realistic generic analyses (i.e., PWROG analyses for SAMGs). The NRC staff concludes that, based on this information, the licensee has dispositioned this F&O for the application.

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

3.1.4.2 Scope of the PRA

The licensee is required to evaluate each proposed change to a relocated surveillance frequency using the guidance contained in NEI 04-10, Rev. 1, to determine its potential impact on risk, due to impacts from internal events, fires, seismic, other external events, and from shutdown conditions. Consideration is made of both CDF and LERF metrics. In cases where a PRA of sufficient scope or where quantitative risk models were unavailable, the licensee uses bounding analyses, or other conservative quantitative evaluations. A qualitative screening analysis may be used when the surveillance frequency impact on plant risk is shown to be negligible or zero.

Seabrook has a full-scope PRA model, whose full-power internal events and internal flood portions have received a peer review, self-assessments, and focused scope peer reviews as discussed previously.

Seabrook's PRA models internal fire events and seismic events and contains an analysis for other external hazards; however, these portions of the PRA have not been subjected to a peer review or assessed to RG 1.200, Revision 2. In accordance with NEI 04-10, Revision 1 the licensee will use qualitative or bounding analyses to assess the impact of internal fire and seismic risk for surveillance test interval changes. The licensee provided additional information on their bounding analysis approach in response to a follow-up RAI in a letter dated March 26, 2014. The bounding analysis will be performed in accordance with NEI 04-10, Rev. 1, Step 10b, when the SSC is modeled explicitly or implicitly in fire and seismic portions of the Seabrook PRA. PRA personnel knowledgeable in the scope, level of detail, and assumptions of the plant-specific fire PRA and seismic PRA will make determinations for the bounding analysis with respect to the fire PRA and seismic PRA portions, respectively. Therefore, the NRC staff finds this approach to be consistent with NEI 04-10, Step 10b guidance in performing a bounding analysis.

The licensee's evaluation methodology is sufficient to ensure the scope of the risk contribution of each surveillance frequency change is properly identified for evaluation, and is consistent with Regulatory Position 2.3.2 of RG 1.177.

3.1.4.3 PRA Modeling

The licensee's methodology includes the determination of whether the SSCs affected by a proposed change to a surveillance frequency are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact may be carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted 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 the surveillance frequency. Potential impacts on the risk analyses due to screening criteria and truncation levels are addressed by the requirements for PRA technical adequacy consistent with guidance contained in RG 1.200, and by sensitivity studies identified in NEI 04-10.

3.1.4.4 Assumptions for Time-Related Failure Contributions

The failure probabilities of SSCs modeled in the licensee PRA include a standby time-related contribution and a cyclic demand-related contribution. NEI 04-10, Rev. 1, criteria adjust 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 SRs. 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 will be 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. Thus, through the application of NEI 04-10, Rev. 1, the licensee has employed reasonable assumptions with regard to extensions of surveillance test intervals, and is consistent with Regulatory Position 2.3.4 of RG 1.177.

3.1.4.5 Sensitivity and Uncertainty Analyses

NEI 04-10, Rev. 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 PRA Standard (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 will also be performed. Thus, through the application of NEI 04-10, Rev. 1, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations, consistently with Regulatory Position 2.3.5 of RG 1.177.

3.1.4.6 Acceptance Guidelines

The licensee will quantitatively evaluate the change in total risk (including internal and external events contributions) in terms of CDF and LERF for both the individual risk impact of a proposed change in surveillance frequency and the cumulative impact from all individual changes to surveillance frequencies using the guidance contained in NRC approved NEI 04-10, Rev. 1 in accordance with the TS SFCP. Each individual change to surveillance frequency must show 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 NRC 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 NRC staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with 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 supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history.

The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results compared to numerical acceptance guidelines. Post implementation performance monitoring and feedback are also required to assure continued reliability of the components. The licensee's application of NEI 04-10, Rev. 1, provides reasonable acceptance

guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177. Therefore, the proposed licensee methodology 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.

3.1.5 The Impact Of The Proposed Change Should Be Monitored Using Performance Measurement Strategies.

The licensee adoption of TSTF-425, Rev. 3, requires application of NEI 04-10, Rev. 1, in the SFCP. NEI 04-10, Rev. 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 will be reassessed in accordance with the methodology, in addition to any corrective actions which may apply as part of the maintenance rule requirements. The performance monitoring and feedback specified in NEI 04-10, Rev. 1, 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.

3.2 Addition of SFCP to TS Section 6

The licensee has included the SFCP and specific requirements into TS Section 6, administrative controls, as follows:

Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
- c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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

3.3 TSTF-425 Optional Changes and Variations

3.3.1 Revised Clean TS Pages

The licensee did not include revised clean TS pages in the submittal given the number of TS pages affected, the straightforward nature of the proposed changes, and several outstanding license amendment requests that may affect some of the same TS pages. The licensee provided only mark-ups of the proposed TS changes, which satisfies the requirements of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," in that the mark-ups fully describe the changes desired.

This is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the NRC staff's model safety evaluation published in the same Federal Register Notice. The NRC staff finds this acceptable.

3.3.2 NUREG-0452

The Seabrook TS were based on NUREG-0452, "Standard Technical Specifications - Westinghouse Pressurized Water Reactors." As a result, the Seabrook TS surveillance numbers and associated Bases numbers differ from the surveillance and Bases numbers in NUREG-1431 and TSTF-425, Revision 3. In addition, the Administrative Controls section of the TS is Section 6.0 for Seabrook versus Section 5.0 for NUREG-1431. In addition, there are surveillances contained in NUREG-1431 that are not contained in the Seabrook TS. Therefore, the NUREG-1431 mark-ups included in TSTF-425 for these surveillances are not applicable to Seabrook.

These differences are administrative deviations from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

3.3.3 Plant-Specific Surveillances

The Seabrook TS include plant-specific surveillances that are not contained in NUREG-1431 and, therefore, are not included in the NUREG-1431 mark-ups provided in TSTF-425. The NRC staff has determined that the relocation of the frequencies for these Seabrook-specific surveillances is consistent with TSTF-425, Revision 3, and with the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996), including the scope exclusions identified in Section 1.0, "Introduction," of the model safety evaluation, because the plant-specific surveillances involve fixed periodic frequencies. Changes to the frequencies for these plant-specific surveillances would be controlled under the SFCP. The SFCP provides the necessary administrative controls to require that surveillances related to testing, calibration, and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met. Changes to frequencies in the SFCP would be evaluated using the methodology and probabilistic risk guidelines contained in NEI 04-10, Revision 1, as approved by NRC letter dated September 19, 2007. The NEI 04-10, Revision 1, methodology includes qualitative considerations, risk analyses, sensitivity studies and bounding analyses, as necessary, and recommended monitoring of the performance of structures, systems, and components (SSCs) for which frequencies are changed to assure that

reduced testing does not adversely impact the SSCs. In addition, the NEI 04-10, Revision 1, methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, relative to changes in surveillance frequencies. Therefore, the proposed relocation of the Seabrook-specific surveillance frequencies is consistent with TSTF-425 and with the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

3.3.4 Definition of Staggered Test Basis

The definition of STAGGERED TEST BASIS is being retained in Seabrook TS Definition Section 1.0 since this terminology is used in Administrative TS Section 6.7.6.1, "Control Room Envelope Habitability Program," which is not the subject of this amendment request and is not proposed to be changed. This is an administrative deviation from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

3.3.5 Insert 2 for TS Bases in TSTF-425

The insert provided in TSTF-425 for the TS Bases (Insert 2) states: "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program." In a letter dated April 14, 2010, (ADAMS Accession No. ML100990099) the NRC staff agreed that the insert applies to surveillance frequencies that are relocated and subsequently evaluated and changed in accordance with the SFCP, but it does not apply to frequencies relocated to the SFCP but not changed. Therefore, the licensee revised the insert for the bases to: "The Surveillance Frequency is controlled under the Surveillance Frequency Control Program." This is an administrative deviation from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

3.3.6 NUREG-0452 Bases Versus NUREG-1431 Bases

The Seabrook TS, which were based on NUREG-0452, do not contain Bases as comprehensive as those in NUREG-1431, which discuss most all SRs. Therefore, many of the Bases mark-ups in TSTF-425 are not applicable to the Seabrook TS. The proposed Bases changes in the submittal revise only those Bases that currently discuss surveillance frequencies. This is an administrative deviation from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996). The existing Bases information describing the basis for the surveillance frequencies will be relocated to the Seabrook SFCP. The NRC staff finds this acceptable.

3.3.7 Tabular Format

The SR for reactor trip and engineered safety features actuation system instrumentation in Seabrook TS 3.3.1 and 3.3.2 are presented in a tabular format, which is different from the format of the SR for the same instrumentation in NUREG-1431. To accommodate this difference, the proposed change includes use of "SFCP" as a frequency notation in the tables that specify instrumentation SR. This is an administrative deviation from TSTF-425 due to differences in format between the Seabrook TS and NUREG-1431, which has no

impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996). The NRC staff finds this acceptable.

3.3.8 Deletion of Table 4.3-3

SR 4.3.3.1 requires that "Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3." The requirement to demonstrate operability of instrument channels "for the MODES and at the frequencies shown in Table 4.3-3" is being deleted from SR 4.3.3.1. This requirement is not a restriction on the MODES in which the SRs are required. The MODES shown in Table 4.3-3, "Radiation Monitoring Instrumentation for Plant Operations Surveillance Requirements," are identical to the applicable MODES for which the instrument channels are required to be OPERABLE as shown in Table 3.3-6, "Radiation Monitoring Instrumentation for Plant Operations." SR 4.0.1 requires SR to be met during the MODES or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the SR. Therefore, the reference to MODES in SR 4.3.3.1 is redundant to the requirements of SR 4.0.1. Further, with the proposed change to SR 4.3.3.1, listing the SR frequencies in Table 4.3-3 is redundant to SR 4.3.3.1. Therefore, the licensee proposed to delete Table 4.3-3 since all of the information provided in the table is included elsewhere within TS 3.3.3.1. The SR and frequencies in Table 4.3-3 are incorporated in the proposed change to SR 4.3.3.1. The information in Table 4.3-3 column "Modes for Which Surveillance is Required" including the associated footnotes, is also contained in Table 3.3-6 in the column "Applicable Modes" and in the TABLE NOTATIONS. Therefore, eliminating Table 4.3-3 does not remove any requirements but only eliminates duplicate information. The same change (deletion of Table 4.3-3) was approved for Salem, Units 1 and 2 in Amendments 299 and 282 (ADAMS Accession No. ML110410691), which adopted TSTF-425. This change aligns the Seabrook TS more closely with NUREG-1431. This is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the NRC staff's model safety evaluation published in the same Federal Register Notice. The NRC staff finds this acceptable.

3.3.9 Table 4.4-3

TSTF-425 excludes relocating frequencies that reference other approved programs for the specific interval (such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program). The approved programs for Seabrook are described in Section 6.0, "Administrative Controls," of the Seabrook TS. The title of TS Table 4.4-3 (Reactor Coolant Specific Activity Sample and Analysis Program) may be misconstrued as a program; however, Section 6.0 of the TS does not contain a program for sampling and analysis of reactor coolant specific activity. To avoid a misunderstanding of these SR, NextEra proposes to delete the word "Program" from the title of TS Table 4.4-3. Consistent with TSTF-425, the eligible frequencies in Table 4.4-3 are proposed for relocation to the Surveillance Frequency Control Program. This change aligns the Seabrook TS more closely with NUREG-1431. This is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the NRC staff's model safety evaluation published in the same Federal Register Notice. The NRC staff finds this acceptable.

3.4 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of some surveillance frequencies to a licensee controlled document, and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the administrative controls of TS. The SFCP and TS Section 6.0 references NEI 04-10, Rev. 1, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TS to a licensee-controlled document, provided those frequencies are changed in accordance with NEI 04-10, Rev. 1, which is specified in the administrative controls of the TS.

The proposed licensee adoption of TSTF-425, Rev. 3, and risk-informed methodology of NEI 04-10, Rev. 1, as referenced in the administrative controls of 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, in that:

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

The regulation in 10 CFR 50.36(c)(3) states "Technical specifications will include items in the following categories: Surveillance Requirements. Surveillance Requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The NRC staff finds that with the proposed relocation of surveillance frequencies to an owner-controlled document and administratively controlled in accordance with the TS SFCP, the licensee continues to meet the regulatory requirement of 10 CFR 50.36, and specifically, 10 CFR 50.36(c)(3), Surveillance Requirements.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Hampshire and Massachusetts State officials were notified of the proposed issuance of the amendment. The State officials provided no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or Surveillance Requirements. The NRC staff has determined that the amendment involves 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 NRC has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding published August 20, 2013 (78 FR 51227). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and c(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Evans, D. Gennardo, and J. Lamb

Date: July 24, 2014

July 24, 2014

Mr. Dean Curtland, Site Vice President
c/o Michael Ossing
Seabrook Station
NextEra Energy Seabrook, LLC
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT REGARDING THE RISK-INFORMED JUSTIFICATIONS FOR THE RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE-CONTROLLED PROGRAM (TAC NO. MF1958)

Dear Mr. Curtland:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1 (Seabrook). This amendment consists of changes to the facility technical specifications (TSs) in response to your application dated May 28, 2013, as supplemented by letters dated July 31, 2013, January 29, and March 26, 2014.

The amendment modifies Seabrook's TSs by relocating specific surveillance frequencies to a licensee-controlled program with implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." The changes are consistent with NRC-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control-[Risk Informed Technical Specifications Task Force] RITSTF Initiative 5b," Revision 3. The Federal Register notice published on July 6, 2009 (74 FR 31996), announced the availability of this TS improvement.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

John G. Lamb, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures:

1. Amendment No. 141 to NPF-86
2. Safety Evaluation

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ADAMS Accession No.: ML13212A069

***via memo**

OFFICE	LPL1-2/PM	LPL1-2/LA	APLA/BC	STSB/BC	OGC - NLO	LPL1-2/BC(A)	LPL1-2/PM
NAME	JLamb	ABaxter	HHamzehee*	RElliott	BMizuno	RSchaaf	JLamb
DATE	06/05/2014	07/09/14	05/15/2014	06/04/2014	06/23/2014	07/24/2014	07/24/2014

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