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

August 16, 2022

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Dominion Energy South Carolina
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 222 TO RELOCATE SELECTED SURVEILLANCE FREQUENCIES TO A RISK-INFORMED LICENSEE CONTROLLED PROGRAM (EPID L-2021-LLA-0064)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (NRC, or the Commission) has issued the enclosed Amendment No. 222 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit 1. The amendment revises the Technical Specifications (TSs) in response to your application dated April 8, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21102A127), as supplemented by letter dated January 20, 2022 (ADAMS Accession No. ML22020A226).

The amendment revises the TSs by relocating selected TS Surveillance Frequencies to a licensee-controlled document. This change is consistent with Technical Specification Task Force (TSTF) traveler TSTF-425, Revision 3 (ADAMS Accession No. ML090850642).

A copy of the related safety evaluation and notice and environmental findings are also enclosed. The Commission's monthly *Federal Register* notice will include the notice of issuance.

Sincerely,

/RA/

G. Edward Miller, Project Manager
Plant Licensing Branch II-I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 222 to NPF-12
2. Safety Evaluation

cc: Listserv

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

DOMINION ENERGY SOUTH CAROLINA, INC.

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
Renewed License No. NPF-12

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

   A. The application for amendment to the Virgil C. Summer Nuclear Station, Unit No. 1 (the facility), Renewed Facility Operating License No. NPF-12, filed by the Dominion Energy South Carolina, Inc. (the licensee), dated April 8, 2021, as supplemented by letter dated January 20, 2022, 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 public health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations as set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by a page change to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-12 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 222, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. Dominion Energy South Carolina, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Shawn A. Williams
Digitally signed by Shawn A. Williams
Date: 2022.08.16 14:47:24 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License and Technical Specifications

Date of Issuance: August 16, 2022

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

ATTACHMENT TO LICENSE AMENDMENT NO. 222

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the renewed facility operating license with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Page	Insert Page
License	License
Page 3	Page 3
Technical Specifications	Technical Specifications
1-8	1-8
3/4 1-1	3/4 1-1
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3/4 3-35	3/4 3-35

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3/4 4-6	3/4 4-6
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3/4 4-10b	3/4 4-10b
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Technical Specifications	Technical Specifications
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3/4 10-5	3/4 10-5
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6-12i	6-12i

(3) SCE&G, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage amounts required for reactor operation, as described in the Final Safety Analysis Report, as amended through Amendment No. 33;

(4) SCE&G, pursuant to the Act and 10 CFR Part 30, 40 and 70 to receive, possess and use at any time byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE&G, pursuant to the Act and 10 CFR Parts 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 of components; and

(6) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as m[a]y be produced by the operation of the facility.

C. This renewed 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; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

SCE&G is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this renewed license. The preoccupation tests, startup tests and other items identified in Attachment 1 to this renewed license shall be completed as specified. Attachment 1 is hereby incorporated into this renewed license.

(2) <u>Technical Specifications and Environmental Protection Plant</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 222, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
V	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.
P	Completed prior to each release
N.A.	Not applicable.
SFCP	In accordance with the Surveillance Frequency Control Program.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% delta k/k for 3 loop operation.

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, 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 equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be demonstrated to be greater than or equal to 1.77% delta k/k:

a. Within one 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 1.0, 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 1.0, 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 Surveillance Requirement 4.1.1.1.2 with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least 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 production,
5. Xenon concentration, and
6. Samarium.

The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SHUTDOWN MARGIN - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal the limits shown in Figure 3.1-3.

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value, 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 equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be demonstrated to be greater than or equal to the required value:

a. Within one 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 operable 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).

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.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A flow path from the boric acid tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

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

SURVEILLANCE REQUIREMENTS

4.1.2.1.1 At least one of the above required flow paths shall be demonstrated OPERABLE 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.

4.1.2.1.2 Demonstrate operability of the required charging pump per Surveillance 4.5.2.f.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths 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 verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300° F.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system with:

1. A minimum contained borated water volume of 2700 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 51,500 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40° F.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

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

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

3. Verifying the boric acid storage system solution temperature when it is the source of borated water.

b. In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is less than 40°F.

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

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

   d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER

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

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control 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 rod position indicator per bank inoperable either:

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

    2. 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 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 rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the 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 rod position indication system at least once per 4 hours.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One 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 rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program.

* With the reactor trip system breakers in the closed position.
# See Special Test Exception 3.10.5.

REACTIVITY CONTROL SYSTEMS
ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} 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 which could affect the drop time of those specific rods, and

c. In accordance with the Surveillance Frequency Control Program.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR) .

APPLICABILITY: MODES 1 * and 2 * #

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Restore the rod to within the limit specified in the COLR, 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 within the insertion limit specified in the COLR.

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

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

* See Special Test Exceptions 3.10.2 and 3.10.3.
# With K_{eff} greater than or equal to 1.0

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) figure entitled Rod Group Insertion Limits versus Thermal Power For Three Loop Operation.

APPLICABILITY: MODES 1 * and 2 * #.

ACTION:

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

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

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position 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 3.10.2 and 3.10.3
# With K_{eff} greater than or equal to 1.0.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

4.2.1.4 When in Base Load operation, the target flux difference shall be updated in accordance with the Surveillance Frequency Control Program by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map

1. When THERMAL POWER is $\leq$ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

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

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

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR, 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 CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

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

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or

2. 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 the core power distribution measurement is obtained.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND} when the Power Distribution Monitoring System (PDMS) is inoperable; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS at any THERMAL POWER greater than APL^{ND}; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \text{ x } K(z)}{P \text{ x } W(z)_{BL}} \quad \text{for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing and measurement uncertainties as specified in the COLR. The F_Q limit is F_Q^{RTP}. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a core power distribution measurement has been obtained in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to measurement, and

2. In accordance with the Surveillance Frequency Control Program.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through a core power distribution measurement and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,

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

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation specified in the COLR.

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

b. In accordance with the Surveillance Frequency Control Program.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR in accordance with the Surveillance Frequency Control Program, when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

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

4.2.3.5 The RCS total flow rate shall be determined by heat balance measurement at ≥90% RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio in accordance with the Surveillance Frequency Control Program when the alarm is OPERABLE.

b. Calculating the ratio in accordance with the Surveillance Frequency Control Program 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 percent RATED THERMAL POWER with one Power Range Channel inoperable in accordance with the Surveillance Frequency Control Program by using the PDMS or movable incore detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QUADRANT POWER TILT RATIO. The incore detector monitoring shall be done with 2 sets of 4 symmetric thimbles or a full incore flux map.

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 limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg^*}

b. Pressurizer Pressure

APPLICABILITY: MODE 1*

ACTION:

With any of the above parameters exceeding its limits, 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 Each of the parameters of Table 3.2-1 shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

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 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.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A	N.A.	SFCP (11)	N.A.	1, 2, 3* ,4*, 5*
2. Power Range, Neutron Flux						
High Setpoint	SFCP	SFCP(2, 4), SFCP (3, 4), SFCP 4, 6), SFCP (4, 5)	SFCP	N.A.	N.A.	1, 2
Low Setpoint	SFCP	SFCP (4)	S/U(18), (16)	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. Deleted						
5. Intermediate Range, Neutron Flux	SFCP	SFCP (4)	S/U(18), (16)	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	SFCP	SFCP (4)	S/U(18), (17), (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	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. Loss of Flow	SFCP	SFCP	SFCP	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	1,2
14. Steam Generator Water Level - Low Coincident with Steam/ Feedwater Flow Mismatch	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
17. Turbine Trip						
A. Low Fluid Oil Pressure	N.A.	SFCP	N.A.	(1, 8, 10)	N.A.	1
B. Turbine Stop Valve Closure	N.A.	SFCP	N.A.	(1, 8, 10)	N.A.	1
19. 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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D	Low Setpoint Power Range Neutron Flux, P-10	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1,2
E.	Turbine Impulse Chamber Pressure, P-13	N.A.	SFCP	SFCP	N.A.	N.A.	1
F.	Low Power Range Neutron Flux, P-9	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1
20.	Reactor Trip Breaker	N.A.	N.A.	N.A.	(7, 12)	N.A.	1, 2, 3,* 4*, 5*
21.	Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	SFCP (15)	1, 2, 3*, 4*, 5*
22.	Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	(7, 13), SFCP (14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

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

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

(1) - If not performed in previous 31 days.

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

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

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

(5) - Detector plateau curves shall be obtained evaluated and compared to manufacturer's data. For the 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) - Prior to entering MODE 1 whenever the unit has been in MODE 3.

(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 required.

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

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

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

(14) - Automatic undervoltage trip.

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

(16) 12 hours after reducing power below P-10 and in accordance with the Surveillance Frequency Control Program thereafter.

(17) 4 hours after reducing power below P-6 and 4 hours after entering MODE 3 from MODE 2 and in accordance with the Surveillance Frequency Control Program thereafter.

(18) If not performed in previous 184 days.

3/4 3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint Column but more conservative than the value shown in the Allowable Value Column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to its OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable take the ACTION shown in Table 3.3-3.

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 engineered safety feature actuation system 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.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
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 (3)	1, 2, 3, 4
c. Reactor Building Pressure-High-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. Differential Pressure Between Steam Lines--High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING 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 (3)	1, 2, 3, 4
c. Reactor Building Pressure-High-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continue)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONTAINMENT ISOLATION								
a. Phase “A” Isolation								
1) Manual	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
3) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP1)	SFCP (3)	1, 2, 3, 4
b. Phase “B” Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2, 3, 4
2) Reactor Building Pressure-High-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (2,3)	1, 2, 3, 4
2) Containment Radioactivity-High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. STEAM LINE ISOLATION								
a. Manual	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 (3)	1, 2, 3
c. Reactor Building Pressure-High-2	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION								
a. Steam Generator Water Level--High-High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2
6. EMERGENCY FEEDWATER								
a. Manual	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 (3)	1, 2, 3
c. Steam Generator Water Level--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
EMERGENCY FEEDWATER (Continued)								
d. Undervoltage - Both ESF Busses	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
f. Undervoltage - One ESF Bus	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2
h. Suction transfer on low pressure	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. LOSS OF POWER								
a. 7.2 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 7.2 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP								
a. RWST level low-low	SFCP	SFCP	SFCP	N.A.	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 (3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS								
a. Pressurizer Pressure, P-11	N.A.	SFCP	SFCP	N.A.	N.A	N.A.	N.A.	1, 2, 3
b. Low, Low Tavg, P-12	N.A.	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

(2) The 36 inch containment purge supply and exhaust isolation valves are sealed closed during Modes 1 through 4, as required by TS 3.6.1.7. With these valves sealed closed, their ability to open is defeated; therefore, they are excluded from the quarterly slave relay test.

(3) Slave Relay Testing will be conducted in accordance with the Surveillance Frequency Control Program for Westinghouse type AR relays and preferably during a refueling outage to preclude the risk of actuation. Replacement relays other than Westinghouse type AR or reconciled Cutler-Hammer relays will require further analysis and NRC approval to maintain the established frequency.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	SFCP	SFCP	SFCP	*
b. Deleted				
2. PROCESS MONITORS				
a. Deleted				
b. Containment				
i. Deleted				
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	SFCP	SFCP	SFCP	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	SFCP	SFCP	SFCP	ALL MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	SFCP	SFCP	SFCP	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	SFCP	SFCP	SFCP	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	SFCP	SFCP	SFCP	1, 2, 3 & 4

* With fuel in the storage pool or building

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

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

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO using a full-core flux map per Specification 4.2.4.2, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program, by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Wind Speed Lower 10m	SFCP	SFCP
b. Wind Speed Upper 61m	SFCP	SFCP
2. Wind Direction		
a. Wind Direction Lower 10m	SFCP	SFCP
b. Wind Direction Upper 61m	SFCP	SFCP
3. Atmospheric Stability		
a. Delta T 1 10-61m	SFCP	SFCP
b. Delta T 2 10-40m	SFCP	SFCP

Elevations nominal above grade elevation

TABLE 4.3-6
REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Reactor Trip Breaker Indication	SFCP	N.A.
2.	Pressurizer Pressure	SFCP	SFCP
3.	Pressurizer Level	SFCP	SFCP
4.	Steam Generator Pressure	SFCP	SFCP
5.	Steam Generator Level	SFCP	SFCP
6.	Condensate Storage Tank Level	SFCP	SFCP
7.	Reactor Coolant System Hot Leg Temperature	SFCP	SFCP
8.	Reactor Coolant System Cold Leg Temperature	SFCP	SFCP
9.	Reactor Coolant System Pressure	SFCP	SFCP
10.	Pressurizer Relief Tank Level	SFCP	SFCP
11.	Reactor Building Temperature	SFCP	SFCP
12.	Boric Acid Tank Level	SFCP	SFCP

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown on Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

i) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

ii) Submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b.2 Deleted

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a CHANNEL CHECK and a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	SFCP	SFCP (1)	SFCP	**
b. Oxygen Monitor	SFCP	SFCP (2)	SFCP	**

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to 10 CFR 50.4 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

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

b. An ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program, and

c. A CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, or 3.3.3.11.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11.1 The operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, and 3.3.3.11.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.11.2 Calibration of the PDMS is required:

a. In accordance with the Surveillance Frequency Control Program when the minimum number and core coverage criteria as defined in 3.3.3.11.b.1 and 3.3.3.11.b.2 are satisfied, or

b. In accordance with the Surveillance Frequency Control Program when only the minimum number criterion as defined in 3.3.3.11.b.3 is satisfied.

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 1 hour.

SURVEILLANCE REQUIREMENTS

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

* See Special Test Exceptions 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the Reactor Coolant loops listed below shall be OPERABLE and at least one of these Reactor Coolant loops shall be in operation.*

a. Reactor Coolant Loop A and its associated steam generator and Reactor Coolant Pump,

b. Reactor Coolant Loop B and its associated steam generator and Reactor Coolant Pump,

c. Reactor Coolant Loop C and its associated steam generator and Reactor Coolant Pump,

APPLICABILITY: MODE 3

ACTIONS:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 At least one cooling loops shall be verified to operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

COLD SHUTDOWN – LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

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

b. The secondary side water level of at least two steam generators shall be greater than 10 percent of wide range indication.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE and/or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no residual heat removal 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 residual heat removal loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

One residual heat removal 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 with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1288 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

* The RHR pump may be de-energized 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.

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 loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

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

SURVEILLANCE REQUIREMENTS

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

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

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

* The RHR pump may be de-energized 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 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 kW.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

ACTIONS: (Continued)

g. With three block valves inoperable:
   1) within 1 hour:
      a) restore the block valves to OPERABLE status, or
      b) place the associated PORVs in manual control and
   2) within the next 2 hours restore at least one of the three block valves to OPERABLE status and
   3) within the next 72 hours:
      a) restore at least two of the three block valves to OPERABLE status and
      b) ensure that the remaining inoperable block valve is closed and the power is removed;

   otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT STUTDOWN within the following 6 hours.

h. The provisions of Specification 3.0.4 are not applicable.

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 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 the power removed in order to meet the requirements of 3.4.4.b, 3.4.4.c, or 3.4.4.d.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

detection monitor, analyze grab samples of the containment atmosphere at least once per 12 hours and restore the required reactor building sump level monitor or the reactor building cooling unit condensate flow rate monitor to OPERABLE status within 7 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the required reactor building atmosphere radioactivity monitor and the reactor building cooling unit condensate flow rate monitor inoperable, restore the required reactor building atmosphere radioactivity monitor or the reactor building air cooler condensate flow rate monitor to OPERABLE status within 30 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With all required monitoring systems inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Reactor building atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Reactor building sump level-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program,

c. Reactor building atmosphere gaseous radioactivity monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION, AND ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified is Table 4.3-3,

d. Reactor building cooling unit condensate flow detector-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

---

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

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

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

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

e. 33 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The leakage rate specified for each Reactor Coolant System Pressure Isolation Valve in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

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

c. With any Reactor Coolant System Pressure Isolation Valve Leakage greater than the 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 The Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the reactor building atmosphere (gaseous or particulate) radioactivity monitor in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory in accordance with the Surveillance Frequency Control Program.

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.(1) This requirement is not applicable to primary-to-secondary leakage.

e. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program.

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

a. During startup in accordance with the Surveillance Frequency Control Program.

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

c. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve for valves denoted on Table 3.4-1 by an asterisk*.

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

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

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

TABLE 4.4-3

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETERS	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	SFCP
CHLORIDE	SFCP
FLUORIDE	SFCP

*Not required with $T_{avg} \leq 150^0F$

TABLE 4.4-4
PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	SFCP	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	SFCP	1
3. Radiochemical for $\overline{E}$ Determination	SFCP*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

Until the specific activity of the primary coolant system is restored within its limits.

* 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.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. A maximum heatup of 100^{0}F in any one hour period,

b. A maximum cooldown of 100^{0}F in any one hour period, and

c. A maximum temperature change of less than or equal to 10^{0}F in any one 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-of0limit condition on the fracture toughness properties 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^{0}F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperatures shall be limited to:

a. A maximum heatup of 100^{0}F in any one hour period,

b. A maximum cooldown of 200^{0}F in any one hour period, and

c. A maximum auxiliary spray water temperature differential of 625^{0}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 fracture toughness properties 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.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each RHR relief valve shall be demonstrated OPERABLE by:

a. Verifying the RHR relief valve isolation valves (8701A, 8701B, 8702A, and 8702B) are open in accordance with the Surveillance Frequency Control Program when the RHR relief valve is being used for overpressure protection.

b. Testing pursuant the Specification 4.0.5.

c. Verification of the RHR relief valve setpoint of at least one RHR relief valve, in accordance with the Surveillance Frequency Control Program on a rotating basis.

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

4.4.9.3.3 At least two charging pumps shall be verified incapable of injecting into the RCS in accordance with the Surveillance Frequency Control Program, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7489 and 7685 gallons,

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 600 and 656 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 one hour or be in at least HOT STANDBY within the next 6 hours and ion HOT SHUTDOWN 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 one hour and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE.

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

1. Verify the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. In accordance with the Surveillance Frequency Control Program when the RCS pressure is above 2000 psig by verifying that the isolation valve operator breaker opened at the motor control center and locked in the open position.

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,

2. Upon receipt of a safety injection test signal.

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:

	Valve Number	Valve Function	Valve Position
1.	8884	HHSI Hot Leg Injection	Closed
2.	8886	HHSI Hot Leg Injection	Closed
3.	8888A	LHSI Cold Leg Injection	Open
4.	8888B	LHSI Cold Leg Injection	Open
5.	8889	LHSI Hot Leg Injection	Closed
6.	8701A	RHR Inlet	Closed
7.	8701B	RHR Inlet	Closed
8.	8702A	RHR Inlet	Closed
9.	8702B	RHR Inlet	Closed
10.	8133A	Charging/HHSI Cross-Connect	Open
11.	8133B	Charging/HHSI Cross-Connect	Open
12.	8106	Charging Mini-Flow Header Isolation	Open

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

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

2. Verify ECCS locations susceptible to gas accumulation are sufficiently filled with water.

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

1. For all accessible areas of the reactor building prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected with the reactor building at the completion of each reactor building entry when CONTAINMENT INTEGRITY is established.

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

1. Verifying automatic interlock action of the RHR system from the Reactor Coolant System by ensuring that, with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig, the interlocks prevent the valves from being opened.

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* Not required to be met for system vent flow paths opened under administrative control.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump component (trash, racks, screens, etc.) show no evident 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 a safety injection actuation and containment sump recirculation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a) Centrifugal charging pump

b) Residual heat removal pump

f. By verifying each ECCS pump's developed head at the test flow point for the pump is greater than or equal to the required developed head in accordance with Specification 4.0.5

g. By verifying the correct position of each mechanical position stope 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.

2. In accordance with the Surveillance Frequency Control Program.

	HPSI System Valve Number
a.	8886A
b.	8996B
c.	8996C
d.	8994A
e.	8994B
f.	8994C
g.	8989A
h.	8989B
i.	8989C
j.	8991A
k.	8991B
l.	8991C

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2

4.5.3.2 All charging pumps except the above required OPERABLE pumps, shall be demonstrated inoperable in accordance with the Surveillance Frequency Control Program whenever the temperature of one or more of the RCS Cold legs is less than or equal to 300^0F by verifying that the motor circuit breakers have been secured in the open position.

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 453,800 gallons,

b. A boron concentration of between 2300 and 2500 ppm of boron, and

c. A minimum water temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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 when the outside air temperature is less than 40°F.

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 one 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 3.6.4.

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

c. Deleted.

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* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building air lock shall be demonstrated OPERABLE:

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

b. Deleted.

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

d. Deleted.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Reactor building internal pressure shall be maintained between -0.1 and 1.5 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 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 or above the following locations and shall be determined in accordance with the Surveillance Frequency Control Program:

a. Elevation 412' - 3 locations

b. Elevation 436' - 3 locations

c. Elevation 463' - 3 locations

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 36-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 6-inch containment purge supply and exhaust isolation valves may be open for less than or equal to 1000 hours per 365 days.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed close, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a 6-inch containment purge supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, close the open 6-inch valve(s) or isolate the penetration within 4 hours otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status within 24 hours; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment purge supply and exhaust isolation valve shall be verified to be:

a. Closed in accordance with the Surveillance Frequency Control Program.

b. Sealed closed in accordance with the Surveillance Frequency Control Program.

4.6.1.7.2 The cumulative time that the 6-inch purge supply and exhaust isolation valves have been open during the past 365 days shall be determined in accordance with the Surveillance Frequency Control Program.

4.6.1.7.3 In accordance with the Surveillance Frequency Control Program each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent reactor building spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and automatically transferring suction to the spray sump.

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

ACTION:

With one reactor building 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 reactor building spray system shall be demonstrated OPERABLE:

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

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

2. Verifying Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 195 psig 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 each of the following test signals a Phase 'A', Reactor Building Spray Actuation, and Containment Sump Recirculation.

2. Verifying that each spray pump starts automatically on a Reactor Building Spray Actuation test signal.

d. At least once per 10 years by performing an air or smoke or equivalent flow test through each spray header and verifying each spray nozzle is unobstructed.

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* Not required to be met for system vent flow paths opened under administrative control.

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 3140 and 3230 gallons of between 20.0 and 22.0 percent by weight NaOH solution, and

b. A flow path capable of adding NaOH solution from the spray additive tank to the suction of each reactor building spray pump.

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 Phase 'A' signal.

d. In accordance with the Surveillance Frequency Control Program by verifying each solution flow rate from the following drain connections in the spray additive system:

   1. NaOH Tank to Loop A $\geq$ 15 gpm

   2. NaOH Tank to Loop B $\geq$ 15 gpm

REACTOR BUILDING COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 Two independent groups of reactor building cooling units shall be OPERABLE with at least one of two cooling units OPERABLE in slow speed in each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required reactor building cooling units inoperable and both reactor building spray systems OPERABLE, restore the inoperable group of cooling units 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.

b. With two groups of the above required reactor building cooling units inoperable, and both reactor building spray systems OPERABLE, restore at least one group of cooling units to OPERABLE status within 72 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required reactor building cooling units inoperable and one reactor building spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of reactor building cooling units shall be demonstrated OPERABLE:

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

   1. Starting each cooling unit group from the control room, and verifying that each cooling unit group operates for at least 15 minutes in the slow speed mode.

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

   1. Verifying that each fan group starts automatically on a safety injection test signal.

   2. Verifying a cooling water flow rate of greater than or equal to 2,000 gpm to each cooling unit group.

3. 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 a simulated SI test signal or on an ESFLS, as applicable.

4. In accordance with the Surveillance Frequency Control Program, by verifying that each service water system booster pump starts automatically on a safety injection signal.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent groups of HEPA filter banks (associated with the OPERABLE reactor building cooling units of Specification 3.6.2.3) with at least one filter bank in each group, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one group of HEPA filter banks OPERABLE, restore one of the inoperable banks in the other group 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

4.6.3 The two groups of HEPA filter banks 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 verifying that the system operates for at least 15 minutes.

b. By performing required filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

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

1. Verifying that the filter bypass damper can be opened by operator action.

2. Verifying that the filter bypass damper closes on a Safety Injection Test Signal.

CONTAINMENT SYSTEMS

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 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 barrier OPERABLE in each affected penetration flow path and:

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

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

c. Isolate each affected penetration flow path 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.

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

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

4.6.4.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE in accordance with the Surveillance Frequency Control Program by:

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

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Reactor Building Purge and Exhaust isolation test signal, each Purge and Exhaust valve actuates to its isolation position.

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

PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator emergency feedwater pumps and flow paths shall be OPERABLE with:

a. Two motor-driven emergency feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine driven emergency feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one emergency feedwater pump inoperable, restore the required emergency feedwater pumps 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 two emergency feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three emergency feedwater pumps inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

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

1. Not used

2. Not used

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

4. Verifying that each automatic valve in the flow path from the condensate storage tank to the steam generators is in the fully open position whenever the emergency feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

5. Verifying that valves 1010-EF and 1007-EF are locked in the open position.

b. In accordance with the Surveillance Frequency Control Program by verifying that the check valve in the instrument air supply line to the six emergency feedwater control valve air accumulators closes when the normal instrument air supply is not available.

c. In accordance with the Surveillance Frequency Control Program during shutdown by verifying that:

1. Each emergency feed pump starts as designed automatically upon receipt of an emergency feedwater actuation test signal.

2. The six emergency feedwater control valves can be closed and held closed for three hours with air from the accumulators when the normal instrument air supply is not available.

3. The turbine driven emergency feedwater pump can be manually stopped from the main control board by closing the steam supply valve with air from the accumulator when the normal instrument air supply is not available.

4. Each automatic valve in the flow path actuates to its correct position on receipt of an emergency feedwater actuation test signal.

d. In accordance with the Inservice Testing Program as required by Specification 4.0.5 by verifying:

1. The developed head of each emergency feedwater pump at the flow test point is greater than or equal to the required developed head. Notes: 1) Not required to be performed for the turbine driven emergency feedwater pump until secondary steam supply pressure is greater than 865 psig. 2) The provisions of Specification 4.0.4 are not applicable for the turbine driven emergency feedwater pump.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 179,850 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying service water system pressure whenever the service water system is the supply source for the emergency feedwater pumps.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	In accordance with the Surveillance Frequency Control Program.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of the primary coolant and the steam generator shells shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig in accordance with the Surveillance Frequency Control Program when the temperature of either the primary coolant or the steam generator shell is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops 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 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.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops 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

4.7.4 At least two service 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.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The service water pond (ultimate heat sink) shall be OPERABLE with:

a. A minimum water level at or above elevation 416.5 Mean Sea Level, USGS datum, and

b. A water temperature of less than or equal to 90.5°F at the discharge of the service water pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The service water pond shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the water temperature and water level to be within their limits.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS)

LIMITING CONDITION FOR OPERATION

3.7.6 Two CREFS trains shall be OPERABLE.*

APPLICABILITY: ALL MODES

ACTION:

a. MODES 1, 2, 3 and 4:

1. With one CREFS train inoperable for reasons other than ACTION 3.7.6.a.2, 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.

2. With one or more CREFS trains inoperable due to an inoperable control room envelope (CRE) boundary, immediately initiate action to implement mitigating actions and verify within 24 hours that the mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits and restore CRE boundary to OPERABLE status within 90 days. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With both CREFS trains inoperable for reasons other than ACTION 3.7.6.a.2, immediately enter LCO 3.0.3.

b. MODES 5 and 6:

1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable train to OPERABLE status within 7 days, or immediately place the OPERABLE CREFS train in the emergency mode of operation or immediately suspend movement of irradiated fuel assemblies.

2. With both CREFS trains inoperable or one or more CREFS trains inoperable due to an inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies.

3. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6 Each CREFS train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 85ºF.

b. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the CREFS train operates for at least 15 minutes.

c. By performing required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

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* The control room envelope (CRE) boundary may be opened intermittently under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

d. In accordance with the Surveillance Frequency Control Program by verifying that on a simulated SI or high radiation test signal, each CREFS train automatically switches into an emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks.

e. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Habitability Program.

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 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, 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 Specifications 3.0.3 and 3.0.4 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 microcuries 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.

3/4.7.9 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.9 The temperature of each area shown in Table 3.7-7 shall be maintained below the limits indicated in Table 3.7-7.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-7:

a. For more than eight hours, prepare and submit a Special Report to the Commission pursuant to 10 CFR 50.4 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to below its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.9 The temperature in each of the areas of Table 3.7-7 shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

PLANT SYSTEMS

3/4.7.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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.

SURVEILLANCE REQUIREMENTS

4.7.10 The water level in the spent fuel 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 spent fuel pool.

PLANT SYSTEMS

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.13 The boron concentration in the spent fuel pool, the fuel transfer canal, and the cask loading pit shall be maintained at a boron concentration greater than or equal to 500 ppm.

APPLICABILITY: Whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit.

ACTION:

With the requirements of the above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool, the fuel transfer canal, and the cask loading pit until the boron concentration in the area where fuel is being moved shall be verified greater than or equal to 500 ppm.

SURVEILLANCE REQUIREMENTS

4.7.13 The boron concentration of the spent fuel pool, fuel transfer canal, or cask loading pit shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program when moving new or irradiated fuel in the spent fuel pool, transfer canal, or cask loading pit.

ACTION: (Continued)

d. With two of the required offsite A. C. Circuits inoperable:

1. Demonstrate the OPERABILITY of the two EDG's by sequentially performing Surveillance Requirement 4.8.1.1.2.a.3 on both within 8 hours, unless the EDG's are already operating, and

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours.

3. Following restoration of one offsite source, follow Action Statement a. with the time requirements of that Action Statement based on the time of initial loss of the remaining inoperable offsite A. C. circuit.

e. With two of the above required EDG's inoperable:

1. Demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirements 4.8.1.1.1 within one hour and at least once per 8 hours thereafter, and

2. Restore one of the inoperable EDG's to OPERABLE status within 2 hours or bin in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. Following restoration of one EDG, follow Action Statement b. wit the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator.

SURVEILLANCE REQUIRMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indication of power availability for each Class 1E bus and its preferred offsite power source.

4.8.1.1.2 Each EDG shall be demonstrated OPERABLE:

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

1. Verifying the fuel level in the day tank and fuel storage tank.

2. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

3. Verifying the diesel generator can start* and accelerate to synchronous speed (504 rpm) with generator steady state voltage greater than or equal to 6840 volts and less than or equal to 7445 volts and generator steady state frequency at ± 0.6 Hz.

4. Verifying the generator is synchronized, gradually loaded* to an indicated 4150-4250 kW** and operates for at least 60 minutes.

b. In accordance with the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the day 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 sampling new fuel oil based on the applicable ASTM standard prior to addition to storage tanks and:

1. By verifying based on the tests specified in the applicable ASTM standard prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

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* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance when tested based on the applicable ASTM standard.

2. By verifying within 30 days of obtaining the sample that the specified properties are met when tested based on the applicable ASTM standard.

e. In accordance with the Surveillance Frequency Control Program by obtaining a sample of fuel oil based on the applicable ASTM standard, and verifying that total contamination is less than 10 mg/liter when checked based on the applicable ASTM standard.

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

1. Verify each EDG starts from standby conditions and:
   a) In less than or equal to 10 seconds, achieves a voltage greater than 6480 volts (7200 - 720 volts) and a frequency greater than 58.8 Hz (60 - 1.2 Hz).
   b) Achieve a steady state voltage greater than 6480 volts but less than 7920 volts and a steady state frequency greater than 58.8 Hz but less than 61.2 Hz.

   The EDG shall be started for this test by using one of the following signals:
   a) Simulated loss of offsite power by itself.
   b) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
   c) An ESF actuation test signal by itself.
   d) Simulated degraded offsite power by itself.
   e) Manual.

2. The generator shall be manually synchronized, loaded to an indicated 4150-4250 kW** in less than or equal to 60 seconds, and operate for at least 60 minutes.

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

1. Deleted

2. Verifying that on rejection of a load of greater than or equal to 729 kW, the voltage and frequency are maintained at 7200 ± 720 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 4250 kW without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 7200 $\pm$ 720 volts and 60 $\pm$ 1.2 Hz.

h. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 504 rpm in less than or equal to 10 seconds.

i. In accordance with the Surveillance Frequency Control Program, by Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent.

j. At least once per 10 years, by performing a pressure test of those portions of the diesel fuel oil system designed to Section III subsection ND of the ASME Code in accordance with Specification 4.0.5.

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:

a. 125-volt Battery bank No. 1A and its associated full capacity charger.

b. 125-volt Battery bank No. 1B and its associated full capacity charger.

APPLICABILITY: Modes 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank 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.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIRMENTS

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 129 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:

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 battery connection resistance is less than or equal to the individual connection resistance for the connection types listed below or total battery resistance is less than or equal to 2890 μΩ:

Maximum Individual Battery Connection Resistances		
Connection Type	Number of Connections	Individual Connection Resistance (μΩ)
Inter-cell	56	45
Jumper	3	100
Terminal Plate	2	35

, and

3. The average electrolyte temperature of 10 of the connected cells is ≥ 60°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 anti-corrosion material,

3. The battery connection resistance is less than or equal to the individual connection resistance for the connection types listed below or total battery resistance is less than or equal to 2890 μΩ:

Maximum Individual Battery Connection Resistances		
Connection Type	Number of Connections	Individual Connection Resistance (μΩ)
Inter-cell	56	45
Jumper	3	100
Terminal Plate	2	35

, and

4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

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

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1.d.

f. Annual performance discharge tests of battery capacity shall be given 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.

ACTION: (Continued)

d. With one D.C. bus not energized from its associated Battery Bank, re-energize the D.C. bus from its associated Battery Bank 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

4.8.3.1 The specified busses 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

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One train of A.C. Emergency Busses consisting of two 7200 volt and three 480 volt A.C. Emergency Busses.

b. Three 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. One 125 volt D.C. Bus energized from its associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, and initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses 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.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 For each containment penetration provided with a penetration conductor overcurrent protective device(s), each device(s) shall be operable.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 Protective devices required to be operable as containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE.

a. In accordance with the Surveillance Frequency Control Program:

   1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

      (a) A CHANNEL CALIBRATION of the associated protective relays and

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

CIRCUIT PROTECTION DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.3 Circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that if faulted could cause failure in both adjacent, redundant Class 1E cables shall be OPERABLE.

APPLICABILITY: All modes

ACTION:

a. With one or more of the above required non-Class 1E circuit breaker(s) inoperable, within 72 hours, either:

    1. Restore the circuit breaker(s) to OPERABLE status; or
    2. De-energize the circuit breaker(s); or
    3. Establish a one (1) hour roving fire watch for those areas in which redundant systems or components could be damaged.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above required circuit breakers shall be demonstrated OPERABLE.

a. In accordance with the Surveillance Frequency Control Program:

    1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis) at least 10% of the circuit breakers and performing the following:

        (a) A CHANNEL CALIBRATION of the associated protective relays, and

        (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

        (c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least ten percent (10%) of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturers data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least ten percent (10%) of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

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

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6 * with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

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

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

4.9.1.3 The following valves shall be verified locked closed ** in accordance with the Surveillance Frequency Control Program: 8430, 8454, 8441 and 8439.

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

** Valves may be opened under administrative control to add borated makeup.

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, 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. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program.

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical a period of time within the acceptable domain of Figure 3.9-1, but not less than 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, immediately suspend all movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for greater than 72 hours but not within the acceptable domain of Figure 3.9-1, immediately suspend movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for a period of time within the acceptable domain of Figure 3.9-1 by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.2 Prior to moving irradiated fuel from the reactor pressure vessel, and in accordance with the Surveillance Frequency Control Program during movement of irradiated fuel, verify the CCW temperature at the inlet to the Spent Fuel Pool Cooling System heat exchanger is within the acceptable domain of Figure 3.9-1.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.1 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 pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no residual heat removal 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

4.9.7.1.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm in accordance with the Surveillance Frequency Control Program.

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

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* The residual heat removal 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 pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 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 pressure 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 pressure 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

4.9.7.2.1 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm in accordance with the Surveillance Frequency Control Program.

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

* Prior to initial criticality the residual heat removal 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 pressure vessel hot legs.

3/4.9.9 WATER LEVEL - REFUELING CAVITY AND FUEL TRANSFER CANAL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.9 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.

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 rod either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

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

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirement 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 requirement 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 Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Either Specifications 4.2.2.2 or 4.2.2.4 and Specification 4.2.2.5.

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

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

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear 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 the 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 Each Intermediate and Power Range Channel shall be subjected to an ANALOG CHANNEL OPERATION TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.4.1.1 may be suspended during the performance of start up and PHYSICS TESTS provided:

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 interlock Setpoint in accordance with the Surveillance Frequency Control Program during start up and PHYSICS TESTS.

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

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specifications 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 system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to 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 rod position indication systems agree:

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

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of 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.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted by Amendment 104.
3.11.1.2 Deleted by Amendment 104 .
3.11.1.3 Deleted by Amendment 104.

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Condensate Storage Tank
b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1 Deleted by Amendment 104.
4.11.1.2 Deleted by Amendment 104.
4.11.1.3 Deleted by Amendment 104.

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks 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.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

n. Snubber Testing Program

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

1) This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

2) The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.

3) The program shall, as allowed by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements," and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," or meet authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

4) The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)) subject to the limitations and modifications listed therein.

o. 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.

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOMINION ENERGY SOUTH CAROLINA, INC.

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By letter dated April 8, 2021 (Reference 1), as supplemented by letter dated January 20, 2022 (Reference 2) Dominion Energy South Carolina (DESC, the licensee), submitted a license amendment request (LAR) for the Virgil C. Summer Nuclear Station, Unit 1 (VCSNS). The proposed changes would revise the Technical Specifications (TS) to relocate specific surveillance requirement (SR) frequencies to a licensee-controlled program. The proposed changes are based on Technical Specification Task Force (TSTF) traveler TSTF-425, Revision 3, “Relocate Surveillance Frequencies to Licensee Control – RITSTF [Risk-Informed TSTF] Initiative 5b,” (Reference 3).

The supplemental letter dated January 20, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on June 15, 2021 (86 FR 31741).

2.0 REGULATORY EVALUATION

2.1 Background

The licensee proposed to modify the VCSNS TS by relocating specific surveillance frequencies from the TS to a new surveillance frequency control program (SFCP) in accordance with Nuclear Energy Institute (NEI) technical report NEI 04-10, Revision 1, “Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies,” dated April 2007 (Reference 4). In a letter dated September 19, 2007 (Reference 5), the NRC staff approved NEI 04-10, Revision 1, as acceptable for referencing in licensing actions, to the extent specified and under the limitations delineated in NEI 04-10, Revision 1, and in the NRC staff’s safety evaluation for NEI 04-10, Revision 1.

The NRC published a notice in the *Federal Register* on July 6, 2009 (74 FR 31996) announcing the availability of TSTF-425, Revision 3, for adoption by licensees and provided a model safety evaluation (SE) for the NRC staff to use to more efficiently process LARs to adopt TSTF-425, Revision 3.

When implemented, TSTF-425, Revision 3, relocates most periodic frequencies of TS SRs to the SFCP, and provides requirements for the new SFCP in the Administrative Controls section of the TS. All surveillance frequencies can be relocated except the following:

- 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 ≥ 95% RTP [rated thermal power]”); and

- Frequencies that are related to specific conditions (e.g., battery degradation, age and capacity) or conditions for the performance of a SR (e.g., “drywell to suppression chamber differential pressure decrease”).

The requirements for the SFCP will be added to the administrative controls section of the TS as TS 6.8.4.o, “Surveillance Frequency Control Program.” Once implemented, the SFCP will allow the licensee to make changes to the relocated surveillance frequencies in accordance with the guidance in NEI 04-10, Revision 1.

The licensee proposed TS 6.8.4.o reads as follows:

> This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.
>
> 1) The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
>
> 2) Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1.
>
> 3) The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

In LAR Attachment 5, a markup of the TS Bases for each affected surveillance was included indicating that the surveillance frequency will be controlled under the SFCP.

The LAR proposed other changes and deviations from TSTF-425, Revision 3, which are discussed in SE Section 3.3, “Deviations from TSTF-425 and Other Changes.”

2.2 Applicable Commission Policy Statements

In the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors,” dated July 22, 1993 (58 FR 39132), the NRC addressed the use of probabilistic safety analysis (PSA, currently referred to as probabilistic risk assessment or PRA) in standard technical specifications (STS). In this 1993 publication, the NRC stated, 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 Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed…
>
> The Commission Policy in this regard is consistent with its Policy Statement on “Safety Goals for the Operation of Nuclear Power Plants,” 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, “. . . probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made . . . about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety.”
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

Approximately two years later, the NRC provided additional detail concerning the use of PRA in the “Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities; Final Policy Statement,” dated August 16, 1995 (60 FR 42622). In this PRA Policy Statement, the NRC stated, 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 for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

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

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

2.3 Regulatory Requirements

The regulations at 10 CFR 50.36, "Technical specifications," establish the regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. These categories will remain in the VCSNS TS.

Paragraph 50.36(c)(3) of 10 CFR states: "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The *Federal Register* notice published on July 6, 2009 (74 FR 31996), which announced the availability of TSTF-425, Revision 3, states that the addition of the SFCP to the TS provides the necessary administrative controls to require that surveillance frequencies relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), and 10 CFR Part 50, "Domestic Licensing of Production and Utilization," Appendix B, Criterion XVI, "Corrective Action," require licensees to monitor surveillance test failures and implement corrective actions to address such failures. In addition, the licensee proposed TS 6.8.4.o would require the licensee to monitor the performance of systems, structures, and components (SSCs) for surveillance frequencies that are decreased under the SFCP to assure reduced testing does not adversely impact the SSCs.

2.4 Regulatory Guidance

Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 6), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

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

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

NUREG-0800, Chapter 16, Section 16.1, Revision 1, "Risk-Informed Decisionmaking: Technical Specifications" (Reference 9), provides guidance for changes to surveillance test intervals (STIs) (i.e., surveillance frequencies) as part of risk-informed decision making.

NUREG-0800, Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load" (Reference 10), provides guidance on evaluating PRA technical adequacy.

NUREG-0800, Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference 11), provides general guidance for evaluating the technical basis for proposed risk-informed changes.

NUREG-0800, Chapter 19, Section 19.2 references the same criteria as RG 1.174, Revision 2, and RG 1.177, Revision 1, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

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

NUREG-1431, Revision 4.0, “Standard Technical Specifications, Westinghouse Plants,” Volume 1, “Specifications,” and Volume 2, “Bases” (Reference 12 and Reference 13, respectively), contain the improved STS for Westinghouse plants. The improved STS were developed based on the criteria in the “Final Policy Statement of Technical Specifications Improvements for Nuclear Power Reactors,” dated July 22, 1993 (58 FR 39132), which was subsequently codified by changes to 10 CFR 50.36 (60 FR 36953).

3.0 TECHNICAL EVALUATION

The proposed amendment would revise the TS by adding TS 6.8.4.o to the administrative controls section, which would specify the requirements for the SFCP, and relocating specific surveillance frequencies to the SFCP. The proposed TS 6.8.4.o would require the licensee to use NEI 04-10, Revision 1, to make changes to surveillance frequencies within the SFCP. LAR Attachment 2, “Documentation of PRA Acceptability,” and the licensee’s January 20, 2022, letter provided documentation regarding the technical adequacy of the PRA following the guidance in RG 1.200, Revision 2. NEI 04-10, Revision 1, states that PRA methods are used with plant performance data and other considerations to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is consistent with guidance in RG 1.174, Revision 2, and RG 1.177, Revision 1, that supports changes to surveillance test intervals (STIs).

3.1 Key Principles

RG 1.177, Revision 1, identifies the five key safety principles required for risk-informed changes to TSs. Each of these principles is also addressed within NEI 04-10, Revision 1. Sections 3.1.1 through 3.1.5 of this SE discuss the five key principles and provide the NRC staff’s evaluation of how each principle is satisfied to support the proposed change.

3.1.1 Key Principle 1: The Proposed Change Meets Current Regulations

The licensee’s use of the NRC-approved methodologies in NEI 04-10, Revision 1, provides an approach to establish risk-informed surveillance frequencies that complements the deterministic approach and supports the NRC’s traditional DID philosophy.

With the proposed changes, the SRs will remain in the TS as required by 10 CFR 50.36(c)(3). However, the relocated surveillance frequencies will be specified in the TSs by reference to the SFCP. The proposed TS 6.8.4.o requires the applicable SRs to be performed at intervals sufficient to assure the associated limiting conditions for operation (LCOs) are met. This change is analogous to other TS requirements in which the SRs are retained in TSs, but the related surveillance frequencies are specified in licensee-controlled documents. Thus, the proposed change complies with 10 CFR 50.36(c)(3) because the TS will retain 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 LCOs will be met.

The regulatory requirements in 10 CFR 50.65 and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," and the proposed adoption of the monitoring requirements in NEI 04-10, Revision 1, will ensure that surveillance frequencies are sufficient to satisfy 10 CFR 50.36(c)(3), any performance deficiencies will be identified, and that appropriate corrective actions will be taken. The proposed SFCP will ensure that SRs specified in the TSs are performed at a frequency sufficient to assure that the above regulatory requirements are met.

Based on the above, the NRC staff concludes that the proposed changes meet the first key safety principle of RG 1.177, Revision 1, by complying with current regulations.

3.1.2 Key Principle 2: The Change is Consistent with the Defense-in-Depth Philosophy

The defense-in-depth philosophy is the second key safety principle of RG 1.177, Revision 1, and it 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 as compensatory measures associated with the change in licensing basis is avoided.
- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.
- Defenses against potential common-cause failures are maintained and the potential for the introduction of new common-cause failure mechanisms is assessed.
- Independence of physical barriers is not degraded.
- Defenses against human errors are maintained.
- The intent of the plant's design criteria is maintained.

The proposed TS 6.8.4.o will require the licensee to use NEI 04-10, Revision 1, to make changes to surveillance frequencies within the SFCP. NEI 04-10, Revision 1, uses both Core Damage Frequency (CDF) and large early release frequency (LERF) metrics to evaluate the impact of proposed changes to surveillance frequencies. In accordance with RG 1.174, Revision 2, and RG 1.177, Revision 1, changes to CDF and LERF are evaluated using a comprehensive risk analysis that assesses the impact of proposed changes, including

contributions from human errors and common-cause failures. Defense-in-depth is explicitly included in the methodology as a qualitative consideration outside of the risk analysis and so is the potential impact on detection of component degradation that could lead to an increased likelihood of common-cause failures. Because the scope of the proposed methodology is limited to revision of surveillance frequencies, the redundancy, independence, and diversity of plant systems are not affected.

The NRC staff concludes that both the quantitative risk analysis and the qualitative considerations provide reasonable assurance that defense-in-depth will be maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177, Revision 1.

3.1.3 Key Principle 3: The Change Maintains Sufficient Safety Margins

The engineering evaluation that the licensee will conduct 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 licensing basis, including the updated final safety analysis report and TS Bases, because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. Therefore, the NRC staff concludes that safety margins are maintained by the proposed methodology and the third key safety principle of RG 1.177, Revision 1, is satisfied.

3.1.4 Key Principle 4: Increases in Risk Should be Small and Consistent with the Intent of the Commission's Safety Goal Policy Statement

RG 1.177, Revision 1, provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies, which includes identification of the risk contribution from affected surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. The proposed TS 6.8.4.o requires the licensee to use NEI 04-10, Revision 1, to change surveillance frequencies listed in the SFCP. NEI 04-10, Revision 1, satisfies the intent of the RG 1.177, Revision 1, guidelines for evaluating the change in risk and ensures that changes in risk are small by providing the methodology to support risk-informed TS for control of surveillance frequencies.

3.1.4.1 Technical Acceptability of PRAs

The licensee used RG 1.200, Revision 2, to address the plant PRA technical acceptability for this application. RG 1.200, Revision 2, provides regulatory guidance for assessing the technical acceptability of a PRA and endorses (with clarifications and qualifications) the use of the following:

1. American Society of Mechanical Engineers/American Nuclear Society (ASME/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" (hereafter referred to as the ASME/ANS PRA Standard) (Reference 14);

2. NEI 00-02, Revision 1, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance" (Reference 15); and

3. NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard" (Reference 16).

The licensee has performed an assessment of the PRA models used to support the SFCP against the guidance provided in RG 1.200, Revision 2, to assure that the PRA models, using plant specific data and models, are capable of determining the change in risk due to changes to surveillance frequencies of SSCs. NEI 04-10 states that Capability Category II (CC-II) of the ASME/ANS PRA Standard should be met, and any identified deficiencies to the CC-II Supporting Requirements of the ASME/ANS PRA Standard are assessed further to determine any impacts to proposed decreases to surveillance frequencies, including the use of sensitivity studies, as appropriate. As described in the NRC staff approval of NEI 04-10 (Reference 5), this level of PRA acceptability is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP and is consistent with Regulatory Position 2.3.1, "Technical Adequacy of the PRA," of RG 1.177, Revision 1.

3.1.4.2 Scope of the PRA

The proposed TS 6.8.4.o requires changes to surveillance frequencies listed within the SFCP to be made using NEI 04-10, Revision 1, which requires the licensee to determine the potential impact of each surveillance frequency change on CDF and LERF from internal events, internal fires, seismic, other external events, and shutdown conditions. In cases where a PRA of sufficient scope or quantitative risk models are unavailable, the licensee may use 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 insignificant.

The VCSNS has full-power internal events, internal flood, fire, and seismic PRA models. These models received peer reviews as discussed in Section 3.1.4.1 of this SE. In accordance with NEI 04-10, Revision 1, the licensee will use these models to perform quantitative evaluations to support the development of changes to surveillance frequencies in the SFCP. The NRC staff finds that the use of these models is acceptable because the NRC-approved methodology in NEI 04-10, Revision 1, allows for more refined analysis to be performed to support changes to surveillance frequencies in the SFCP.

The VCSNS does not have PRA models for high winds, external flooding events, and other external hazards. These events were assessed in the licensee's response to Generic Letter (GL) 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10CFR 50.54(f)" (Reference 17). In Section 2.5, "Other External Events," of LAR Attachment 2, the licensee stated that insights from the IPEEE evaluation will be used to qualitatively assess the impacts on STI evaluations. The NRC staff notes that several updates to external hazards have been subsequently performed at nuclear power plants.

Section 2.5 of LAR Attachment 2, states that the VCSNS hurricane, tornado and high winds analyses show that that the plant is adequately designed, or procedures exist to cope with the effects of these natural events. In the licensee's response to the NRC's request for additional

information (RAI) (Reference 2), the licensee stated that high winds information from the IPEEE continues to be appropriate, as no new updated information is available with respect to high winds hazards. The licensee stated that since the IPEEE, it has augmented its ability to respond to high wind events that are beyond the design basis as a part of industry response to the Fukushima event. The licensee will use this information to qualitatively screen the proposed STI changes for adverse risk impact with respect to high winds, per NEI 04-10, step 10. The NRC staff finds the licensee's response to the RAI acceptable because the licensee's justification of high winds and tornados, including tornado missiles, is consistent with the guidance in NEI 04- 0, Revision 1.

Section 2.5 of LAR Attachment 2, states that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities. In the licensee's Flood Hazard Reevaluation Report (Reference 18), the licensee determined that the local intense precipitation (LIP) was not bounded by the current design basis. The licensee elected to remediate the plant design to correct for the LIP, as confirmed by the Focused Evaluation for External Flooding Report (Reference 19). In the NRC staff assessment of the Focused Evaluation for External Flooding (Reference 20), the NRC staff concluded that the licensee has demonstrated that effective flood protection, if appropriately implemented, exists for the unbounded flooding mechanisms during a beyond-design-basis external flooding event at VCSNS. The licensee confirmed that the plant design has been remediated to correct for LIP. Therefore, the NRC staff's review finds that the justification of external flooding consideration provided in the licensee's supplement is consistent with the guidance in NEI 04-10, Revision 1.

The VCSNS IPEEE hurricane, tornado and high winds analyses show that the plant is adequately designed, or procedures exist, to cope with these natural events. The licensee stated that transportation and nearby facility accidents were not considered to be significant vulnerabilities at VCSNS. The licensee further stated that it will use a qualitative or a bounding approach, as applicable, for assessing the risk impact of non-PRA modeled hazards, such as, high winds and tornados, external flooding, and other external hazards on extending the surveillance frequency of SSCs, and that this approach is consistent with the methodology of NEI 04-10, Revision 1, as endorsed by the NRC.

The licensee stated that it operates under a shutdown risk management program to support implementation of Nuclear Management and Resources Council 91-06 (Reference 21), and that it has an implementing procedure that provides guidelines for outage risk management. The licensee stated that it will use its shutdown risk management program procedures to assess the potential impact of shutdown risk for proposed STI extensions, consistent with the guidance in NEI 04-10. This is an acceptable approach in accordance with NEI 04-10, Revision 1.

The licensee stated that it will address the impact of all external hazards on a specific STI extension using a qualitative approach that follows the methodology of NEI 04-10, Revision 1, as endorsed by the NRC staff.

The NRC staff notes that in accordance with NEI 04-10, Revision 1, the licensee can perform an initial qualitative screening analysis, and, if the qualitative information is insufficient to provide confidence that the net impact of the STI change would be negligible, a bounding analysis will be performed. The licensee stated that its approach will use a qualitative process, and bounding analyses where appropriate, for assessing the risk impact of extending the surveillance frequency on SSCs for non-PRA modeled hazards and shutdown events.

The NRC staff finds that high winds and tornadoes, external flooding, and other external hazards are considered and do not significantly impact this application. This is because of (1) the plant's robust design, procedures, and enhancements, and (2) the licensee's approach for considering any impact is consistent with the guidance in NEI 04-10, Revision 1, and the NRC staff's SE on the guidance. Further, the NRC staff finds that in cases where either a PRA of sufficient scope or a quantitative risk model is unavailable, the licensee will use bounding analyses, or other conservative quantitative evaluations consistent with NEI 04-10, Revision 1.

The proposed TS 6.8.4.o will require the licensee to comply with the NRC-approved methodology in NEI 04-10, Revision 1. Therefore, the NRC staff concludes that the licensee's evaluation methodology will ensure the risk contribution of each surveillance frequency change is properly identified for evaluation and is consistent with Regulatory Position 2.3.2, "Scope of the Probabilistic Risk Assessment for Technical Specification Change Evaluations," of RG 1.177, Revision 1.

<u>Internal Events and Internal Flooding PRA</u>

The licensee's evaluation of the technical acceptability of its internal events and internal flooding PRA models consisted of a full-scope peer review performed in March 2016. The licensee updated its internal events model in July 2020 which addressed or partially addressed some of the findings, including many of the most technically complex and consequential findings. The licensee stated that following the July 2020 update, the internal events model underwent a formal finding closure process as described in Appendix X to NEI 05-04 (Reference 22), which addressed the findings that were resolved in the update. The licensee also stated that a focused scope peer review of PRA upgrades were subsequently performed, and new findings related to the focused scope peer review were issued by the peer review team. In LAR Attachment 2, Section 4.0, "F&O Tables", the licensee provided the finding-level F&Os that remain open for the internal events and internal flooding PRA. The NRC staff's evaluation of those finding-level F&Os is provided later in this section.

The F&O 06-19, associated with Supporting Requirement SY-A13, was generated because it appeared that several flow divergence pathways were not adequately addressed. In the LAR, the licensee stated that system model screening was re-reviewed to identify additional components to be incorporated in the model. In its RAI response, the licensee clarified that the process of identifying and incorporating flow divergence paths into the PRA model is very similar to the process of screening components in the systems analysis. The licensee clarified the disposition of F&O 06-19 by stating that system model screening was re-reviewed to identify additional components that have the function of precluding flow divergence and therefore need to be added to the PRA systems analysis, and that the failure of these components has been added to the PRA model. The NRC finds the licensee's response to the RAI acceptable because the licensee provided clarifying statements that confirmed that the disposition provided is correctly stated.

The F&O 02-06, associated with Supporting Requirement IE-D3, was generated because not all the assumptions related to initiating events development are adequately documented. In the LAR, the licensee stated that this issue does not impact quantification of the PRA models The NRC staff notes that Step 5 of the NEI 04-10 states that identified sources of key uncertainties will provide input to determine additional sensitivity studies to appropriately assess their impact. In its RAI response, the licensee stated that assumptions and sources of uncertainty have been identified and sufficiently documented to support the SFCP and that these assumptions are discussed "in-line" with various analyses they support. The licensee further stated that the

resolution of finding 02-06 involves compiling all assumptions into a designated assumptions section of relevant documents and that the intent of this documentation finding is to facilitate future peer reviews. The NRC staff finds the licensee's response to the RAI acceptable because the licensee's process is consistent with Step 5 of NEI 04-10.

The F&O 04-32, associated with Supporting Requirement LE-G5, was generated because the documentation of the limits of the large early release frequency (LERF) for applications appears to be minimal. In the LAR, the licensee stated that this issue does not impact quantification of the PRA models. It was unclear to the NRC staff that the LERF analysis limitations have been adequately addressed for this application. In its RAI response the licensee stated that while some progress has been made to address this finding, the consensus from the F&O closeout team was that some topics on limitations require additional discussion and documentation to fully resolve this issue. The licensee further stated that they fully understand the extent of these limitations and is tracking this issue in its PRA configuration control process. Further, the LERF modeling limitations must be formally documented in order to close this finding and that areas where LERF assessment methodology is limited are handled with simplifying conservative treatments in its PRA, and therefore, the scope of the sensitivity analysis described in its response does not need to be expanded to include undocumented LERF modeling limitations. The NRC staff finds the licensee's response to the RAI acceptable because until the finding is closed, the licensee will address the LERF limitations by applying conservative treatments in its PRA.

Fire PRA

A Fire PRA Peer Review was performed in December 2010 against ASME/ANS PRA Standard RA-Sa-2009 and RG 1.200, Revision 2. A follow-on fire PRA Peer Review was performed in July 2011, also against RA-Sa-2009 and RG 1.200, Revision 2. The licensee stated that these two peer reviews were focused scope, but when the two are combined they encompass all Supporting Requirements in Part 4 of RA-Sa-2009.

F&Os CF-A1-01, ES-B1-01, ES-B1-03, and PRM-B9-02, associated with Supporting Requirements CF-A1, ES-B1, and PRM-B9, were generated because of issues associated with circuit failure mode likelihood analysis and data mapping fidelity. In the LAR, the licensee stated that these issues will be evaluated in accordance with Steps 5 and 14 of NEI 04-10, Revision 1, which details the performance of additional sensitivity studies. It appeared to the NRC staff that these issues are model completeness issues and therefore it was unclear how the sensitivity process will address these issues. In its RAI response, the license stated that as part of STI evaluations it will generate a change-in-risk cutset solution and will use the change-in-risk cutset solution to qualitatively review the Circuit Failure Mode Likelihood Analysis (CFMLA) for anomalies and NUREG/CR-7150, Volume 2 [4.9] (Reference 23) implementation issues relevant to the proposed STI change. The licensee further stated that it will qualitatively screen each of these four findings on a STI specific basis because in many cases the relevant issues are not affecting quantification results (I.e., documentation related), not affecting risk significant or STI relevant sequences, or are a conservative treatment. If it is not able to qualitatively screen all four of these findings in this manner, then a sensitivity analysis would be performed either by resolving the relevant issue in the CFMLA or applying a conservatively bounded approximation of the resolution into the CFMLA and re-quantifying the analysis. The NRC staff finds the licensee's response to the RAI acceptable because the licensee demonstrated that its sensitivity analysis process will address these issues.

Seismic PRA

Section 3.4.3 of LAR Attachment 2 discusses the seismic PRA (SPRA) peer review, SPRA F&O independent assessment closure review, and status of SPRA peer review finding-level F&Os. During the review of this LAR, the NRC staff utilized docketed information from the licensee's submittal in response to the 10 CFR 50.54(f) information request arising from Near Term Task Force recommendation 2.1 (NTTF 2.1) (Reference 24), and the corresponding NRC staff assessment dated September 6, 2019 (Reference 25), because the same SPRA was used to support this application.

Section 3.4.3 of LAR Attachment 2, states that the SPRA was reviewed against the requirements in the 2009 version of the ASME/ANS PRA Standard (ASME/ANS RA-Sa-2009) (Reference 14) for the seismic hazard (SHA) element, and 2013 version of the ASME/ANS PRA Standard (ASME/AND Ra-Sb-2013) (Reference 26), for the seismic fragility analysis (SFR) element and the seismic plant response (SPR) element.

In its letter dated January 20, 2022, the licensee stated that the NRC staff previously concluded that the differences between the Supporting Requirements in the 2009 and 2013 versions of the PRA Standard for the seismic hazard technical element were not significant with respect to the review and decision for VCSNS SPRA NTTF Recommendation 2.1: Seismic. In addition, the licensee indicated that the DESC's experience with implementing SFCP at other sites has demonstrated that surveillance frequency evaluations generally have very low sensitivity to seismic hazards because SPRAs are dominated by seismic fragilities and have low dependence on marginal changes in non-seismic equipment failure rates postulated by the SFCP. The NRC staff's review finds that the licensee's response is acceptable because the differences in the seismic hazard technical element in the 2009 and 2013 versions of the seismic PRA Standard does not significantly impact on the seismic PRA acceptability to be used in this application, as a low sensitivity of Surveillance Frequency Evaluations to seismic hazards.

The NRC staff reviewed the table provided in Section 4.3, "Seismic PRA F&Os and SFCP Dispositions," of LAR Attachment 2, which summarized the peer review team assessment of the VCSNS PRA models that do not conform to CC-II of the ASME/ANS PRA Standard Supporting Requirements. The NRC staff's assessment of these open finding level facts and observations (F&Os), the impacted Supporting Requirements, and the licensee's resolutions concluded that they are addressed and dispositioned for this application per the NEI 04-10 guidance, as discussed below.

The F&O 19-10, associated with Supporting Requirement SPR-B1, was generated because the peer review team was not able to assess the collective impact on the SPRA model because of the extensive number of open internal events F&Os. In the LAR, the licensee stated that these issues will be evaluated in accordance with Steps 5 and 14 of NEI 04-10, which details the performance of additional sensitivity studies. In its RAI response, the licensee explained that because external hazards PRAs are dependent on the internal events PRA, internal events resolutions will also be propagated to external hazard PRAs for sensitivity analysis using the external hazards models. The licensee stated that in order to account for the cumulative impact, a single cumulative sensitivity analysis combining all of the applicable F&O resolutions will be performed for each hazard group. The licensee further clarified its intent is to perform this evaluation, given the internal events PRA F&Os, on an STI-specific basis. The NRC staff finds the licensee's response to the RAi acceptable because the licensee will follow the guidance in Steps 5 and 14 of NEI 04-10, Revision 1, to evaluate the impact of F&O 19-10 on an STI-specific basis.

The F&O 24-07, associated with SFR-D1, states that the liquefaction potential was not considered in identification of failure modes that could affect the Service Water System Pumphouse. In its RAI response, the licensee stated that in the SPRA, failures of the Service Water System Pumphouse were conservatively assumed to result directly in core damage due to an unrecoverable loss of the ultimate heat sink. Therefore, a liquefaction induced failure mode would have little or no impact on the relevant core damage sequences. The NRC staff finds the licensee's response to the RAI acceptable because F&O 24-07 has a very small impact on the SFCP.

The F&O 20-01, associated with SHA-H1, states that the probabilistic seismic hazard analysis (PSHA) for the VCSNS site was performed using the seismic source model described in NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," (Reference 27). In its RAI response the licensee cited an evaluation report developed by independent consultants that stated the more recent earthquake data does not increase rates or estimates of the seismic hazard at VCSNS from the existing model and no updates to the VCSNS PSHA are needed. The NRC staff finds licensee's response to the RAI acceptable because the disposition demonstrates no impact on the SFCP.

Based on its review, the NRC staff concludes that the licensee's SPRA is technically acceptable to support the evaluation of changes proposed to surveillance frequencies within the SFCP using the process in NEI 04-10, Revision 1, and is consistent with regulatory position 2.3.1 of RG 1.177, Revision 1.

All PRA Modeled Hazards

In the LAR, several of the dispositions to open F&Os did not provide a specific evaluation of the finding but stated that, if it is determined during the SFCP process that the impact is non-trivial, then additional sensitivity studies will be performed. In its RAI response, the licensee described the criteria that constitutes a trivial impact to STI calculations. The NRC staff finds the licensee's response to the RAI acceptable because it considers PRA acceptability in a STI specific manner that considers cumulative impact, and the thresholds identified by the licensee are sufficiently low to avoid prematurely screening out issues of cumulative significance.

In the LAR, several of the dispositions to open F&Os regarding the use of conservative treatments, stated these items will be dispositioned in accordance with Steps 9 and 11 of the NEI 04-10 process when an STI evaluation fails to meet acceptance guidelines. Steps 9 and 11 involve PRA model updates to appropriately evaluate STIs. The NRC staff questioned how the cumulative impact of all conservative treatments would be considered in the SFCP. In its RAI response, the licensee stated that in order to account for the cumulative impact of PRA acceptability issues, a single cumulative sensitivity analysis combining all of the applicable resolutions will be performed for each hazard group and that the sensitivity analysis will sum the delta CDF and delta LERF from each hazard group for comparison to success criteria. The NRC staff finds the licensee's response to the RAI acceptable because the licensee's approach will ensure that the sensitivity analysis addresses the cumulative impact of the findings on all hazard groups.

Conclusion

Based on its review, the NRC staff concludes that the licensee's internal events PRA, including internal flooding, fire PRA and seismic PRA are technically acceptable to support the evaluation of changes proposed to surveillance frequencies within the SFCP using the process

in NEI 04-10, Revision 1, and is consistent with Regulatory Position 2.3.1 of RG 1.177, Revision 1.

3.1.4.3 Application of the PRA Models

Consistent with NEI 04-10, Revision 1, upon implementation of the SFCP, the licensee stated that it will determine 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 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 the guidance provided in RG 1.200, Revision 2, and by sensitivity studies identified in NEI 04-10, Revision 1.

The NRC staff finds that through the application of NEI 04-10, Revision 1, the licensee's PRA modeling is sufficient to ensure an acceptable evaluation of risk for the proposed changes in surveillance frequency and is consistent with Regulatory Position 2.3.3, "Probabilistic Risk Assessment Modeling," of RG 1.177, Revision 1.

3.1.4.4 Assumptions for Time-Related Failure Contributions

The licensee stated that failure probabilities of SSCs modeled in the licensee's PRAs assume all failures to be time-related because the breakdown between standby time-related contribution and a cyclic demand-related contribution is unknown. The NEI 04-10, Revision 1, criteria adjust the time-related failure contribution of SSCs affected by the proposed change to a surveillance frequency. This is consistent with the guidance in RG 1.177, Revision 1, Section 2.3.3, "Probabilistic Risk Assessment Modeling," which permits separation of the failure rate contributions into demand and standby for evaluation of Supporting Requirements. According to the guidance in NEI 04-10, Revision 1, if the available data does 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. Consistent with RG 1.177, Revision 1, 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 licensee 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.

Therefore, the NRC staff finds that through the application of NEI 04-10, Revision 1, the licensee has employed reasonable assumptions with regard to extensions of STIs and is

consistent with Regulatory Position 2.3.4, “Assumptions in Completion Time and Surveillance Frequency Evaluations,” of RG 1.177, Revision 1.

3.1.4.5 Sensitivity and Uncertainty Analyses

NEI 04-10, Revision 1, provides that sensitivity studies be performed 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 CC-II of the ASME/ANS PRA Standard, as endorsed in RG 1.200, Revision 2. 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. Guidance in Step 5 of NEI 04-10, Revision 1, specifies risk sensitivity studies to be conducted by changing the unavailability terms for PRA basic events that correspond to SSCs being evaluated.

Consistent with NEI 04-10, Revision 1, the licensee will implement monitoring and feedback of SSC performance once the revised surveillance frequencies are utilized. Therefore, the NRC staff finds that through the application of NEI 04-10, Revision 1, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations, consistent with Regulatory Position 2.3.5, “Sensitivity and Uncertainty Analyses Relating to Assumptions in Technical Specification Change Evaluations,” of RG 1.177, Revision 1.

3.1.4.6 PRA Acceptability Conclusions

The licensee (1) demonstrated the technical acceptability of the PRAs for this application following the guidance in RG 1.200, Revision 2, (2) established a periodic update and review process to update the PRA model to incorporate changes made to the plant consistent with the SFCP, (3) assessed key assumptions and sources of uncertainty for impact on the application, and (4) will calculate extended STIs using NRC-accepted PRA methods consistent with NEI 04-10, Revision 1. Additionally, the licensee's approach for considering the impact of non-seismic external hazards and other hazards for the extension of STIs is acceptable and consistent with the guidance in NEI 04-10, Revision 1.

Based on the above conclusions further discussed in Sections 3.1.4.1 through 3.1.4.5, the NRC staff finds that the licensee internal events PRA, which include internal floods, fire RPA, and seismic PRA are acceptable for use in the SCFP consistent with NEI 04-10.

3.1.4.7 Acceptance Guidelines

The licensee proposed TS 6.8.4.o would require changes to surveillance frequencies listed within the SFCP to be made using NEI 04-10, Revision 1. The NEI 04-10, Revision 1, methodology requires a quantitative evaluation of the change in total risk (including contributions from internal and external events) in terms of CDF and LERF for both the individual risk impact of a proposed surveillance frequency change and the cumulative impact from all individual changes to surveillance frequencies. Each individual change to a surveillance frequency must show a risk-increase below 10^{-6} per year for CDF and below 10^{-7} per year for LERF. These changes to CDF and LERF are consistent with the acceptance guidelines of RG 1.174, Revision 2, for very small changes in risk. If the RG 1.174, Revision 2, acceptance guidelines are not met, then the surveillance frequencies must either be revised to be consistent with RG 1.174, Revision 2, or the process terminates without permitting the proposed changes.

If quantitative results are unavailable for comparison with the acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to a surveillance frequency is negligible or insignificant. Otherwise, bounding quantitative analyses are required that demonstrate the risk impact is at least one order of magnitude lower than the RG 1.174, Revision 2, acceptance guidelines for very small changes in risk. In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in an increase in CDF and LERF of less than 10^{-5} per year and 10^{-6} per year, respectively. The total CDF and total LERF must be reasonably shown to be less than 10^{-4} per year and 10^{-5} per year, respectively. These values are consistent with the risk acceptance guidelines of RG 1.174, Revision 2, as referenced by RG 1.177, Revision 1, for changes to surveillance frequencies.

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

3.1.5 Key Principle 5: The Impact of the Proposed Change Should be Monitored Using Performance Measurement Strategies

The proposed TS 6.8.4.o requires changes to surveillance frequencies listed within the SFCP to be made using NEI 04-10, Revision 1. The methodology in NEI 04-10, Revision 1, includes performance monitoring of SSCs whose surveillance frequencies have been revised as part of a feedback process to ensure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback include consideration of Maintenance Rule monitoring of equipment performance. In the event of SSC performance degradation, the surveillance frequency would be reassessed in accordance with the methodology in NEI 04-10, Revision 1, in addition to any corrective actions that may be required by the Maintenance Rule.

The performance monitoring and feedback process specified in NEI 04-10, Revision 1, which would be required by proposed TS 6.8.4.o, provides reasonable assurance of acceptable SSC performance and is consistent with Regulatory Position 3.2, "Maintenance Rule Control," in RG 1.177, Revision 1. Therefore, the NRC staff concludes that the fifth key safety principle of RG 1.177, Revision 1, is satisfied.

3.1.6 NRC Conditions for NEI 04-10, Revision 1

Section 4.0 of the NRC staff's SE for NEI 04-10, Revision 1, states:

> The NRC staff finds that the methodology in NEI 04-10, Revision 1 is acceptable for referencing by licensees proposing to amend their TS to establish a SFCP provided the following conditions are satisfied:
>
> 1. The licensee submits documentation with regards to PRA technical adequacy consistent with the requirements of RG 1.200, Section 4.2.
>
> 2. When a licensee proposes to use PRA models for which NRC-endorsed standards do not exist, the licensee submits documentation which identifies the quality characteristics of those models, consistent with RG 1.200, Sections 1.2 and 1.3. Otherwise, the licensee identifies and justifies the methods to be applied for assessing the risk contribution for those sources of risk not addressed by PRA models.

Section 3.1.4.1 of this SE discusses the technical acceptability of the licensee's PRA model and finds it to be consistent with NRC-endorsed guidance. As discussed in SE Section 3.1.4.1, the NRC staff finds the information provided in the LAR, supports the licensee's proposed PRA and, therefore, the conditions in the NRC staff's SE for NEI 04-10, Revision 1, have been met.

3.2 Addition of Surveillance Frequency Control Program to Administrative Controls

As discussed in SE Section 2.1, the requirements for the SFCP will be added to the administrative controls section of the TS as TS 6.8.4.o. Based on its review in SE Section 3.1, the NRC staff finds that the proposed SFCP is an acceptable program for controlling changes to surveillance frequencies. The NRC staff finds that the proposed TS 6.8.4.o includes the necessary program and applicability requirements, that those requirements are consistent with the TSTF-425, Revision 3, and that those requirements will ensure that LCOs are met. Therefore, the NRC staff finds proposed TS 6.8.4.o acceptable.

3.3 Deviations From TSTF-425

In LAR Sections 2.2.1 and 2.2.2, as supplemented, the licensee identified variations and technical changes from the TSTF-425 template. The variations and technical changes as well as the NRC staff's evaluation of those changes is discussed below.

- The licensee stated that the definition of staggered test basis is being retained in TS Section 1.0, Definitions, since this terminology is used in TS, Procedures and Programs, Section 6.8.4.m, "Control Room Envelope Habitability Program," which is not the subject of this amendment request and is not proposed to be changed.

  The licensee stated that it has test scheduling strategies for logic trains, channels and other components within systems that are also being relocated, consistent with the guidance of NEI 04-10, Rev. 1. The licensee further stated that Revision 1 to NEI 04-10 addresses test strategy (e.g., staggered test basis), in addition to frequency and under the proposed change, the frequencies of all SRs (except those that reference other programs for the specific interval or that are event driven) are relocated. The licensee further stated that similar to a staggered test basis requirement, these SRs require at

least one logic train, channel or component to be tested within one interval, and all logic trains, channels or components to be tested with N intervals, where N is the total number of logic trains, channels or components subject to the test requirement. The licensee further stated that the following SRs contain test scheduling requirements proposed for relocation:

- o SR 4.3.1.2, Reactor Trip System Instrumentation Response Time
- o SR 4.3.2.2, Engineered Safety Features Response Time

The NRC staff finds that retaining the staggered test basis definition is acceptable because it applies to an approved program that is not subject to the SFCP, retaining the definition does not affect the applicability of TSTF-425, Revision 3, and does not impact the conclusions reached in the NRC's model SE for TSTF-425, Revision 3.

- The licensee stated that TS 4.8.1.1.2.i.1 and 4.8.1.1.2.i.2 were renumbered as TS 4.8.1.1.2.i and 4.8.1.1.2.j respectively to allow separate surveillance frequencies and that the frequency for TS 4.8.1.1.2.i would be in accordance with the SFCP. The frequency for TS 4.8.1.1.2.j would be at least once per 10 years. The NRC staff finds that the renumbering for different frequencies and the different frequencies do not affect the applicability of TSTF-425, Revision 3, and do not impact the conclusions reached in the NRC's model SE for TSTF-425, Revision 3.

- The licensee stated that NRC letter dated April 14, 2010, provides a change to an optioned insert (INSERT #2) to the existing TS Bases to facilitate adoption of the traveler. The licensee further stated that its TSTF-425 TS Bases insert states the following:

  The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

  The licensee stated that the above statement only applies to frequencies that have been changed in accordance with the SFCP and does not apply to frequencies that are relocated but not changed and that it has replaced the TSTF-425 TS Bases Insert #2 with the following:

  The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

  The NRC staff finds the licensee's proposed variation is consistent with the NRC letter dated April 14, 2010, and it does not impact the conclusions reached in the NRC's model SE for TSTF-425, Revision 3.

- The licensee stated that Table 1.2, "Frequency Notation", of its TS, was revised to include the definition for "SFCP", and SFCP is defined as "In accordance with the Surveillance Frequency Control Program." The NRC staff finds the licensee's proposed variation acceptable because it is editorial in nature and does not affect the applicability of TSTF-425, Revision 3, and does not impact the conclusions reached in the NRC's model SE for TSTF-425, Revision 3.

- The licensee stated that Section 4.6.2.1.d of its TS is not being changed even though it is eligible and that it may request a change to both the frequency and the method of testing in Section 4.6.2.1.d from at least once per 10 years to "following activities that could cause nozzle blockage" in a future LAR. The NRC staff finds that the licensee not moving a TS to the SFCP that is eligible and not changing the frequency is acceptable because it does not affect the applicability of TSTF-425, Revision 3, and does not impact the conclusions reached in the NRC's model SE for TSTF-425, Revision 3.

- The licensee stated that its TS varies significantly, especially in terms of format, from the Standard Technical Specifications as described in NUREG-1431 and in TSTF-425, and that there are SRs contained in TSTF-425 but not applicable or not contained in its TS. The licensee further stated that there are also SRs specific to (contained in) its TS but not in NUREG-1431 or in TSTF-425 and that for some of these SRs specific to its TS, it has determined that the relocation of the associated SR frequencies is consistent with the intent of TSTF-425, Revision 3 and with the NRC'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. The licensee further stated that the subject TS specific SRs involve fixed periodic frequencies and that in accordance with TSTF-425, changes to the frequencies for these SRs would be controlled under the SFCP. The licensee further stated that the SFCP provides the necessary administrative controls to require that SRs 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. The licensee confirmed that changes to frequencies in the SFCP would be evaluated using the NRC approved methodology and probabilistic risk guidelines contained in NEI 04-10, Revision 1.

  The licensee stated that LAR Attachment 7 provides a cross-reference between the proposed changes in its TS versus corresponding NUREG-1431 SRs and the corresponding SR location in TSTF-425, and that the first part (Part 1) of the Table provides comparisons only on its TS SR frequencies proposed to be relocated. The second part (Part 2) of the table provides a list of the SRs included in NUREG-1431 but not applicable to VCSNS TS.

  The NRC staff confirmed that the TS SRs proposed for inclusion in the SFCP either correspond to an equivalent SR in NUREG-1431 or are otherwise appropriated for inclusion in the program. Therefore, the NRC staff finds the licensee's proposed variations acceptable because they are considered editorial in nature and do not affect the applicability of TSTF-425, Revision 3, and do not impact the conclusions reached in the NRC's model SE for TSTF-425, Revision 3.

3.4 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of certain surveillance frequencies to a licensee-controlled document and its proposed control of changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the administrative controls of the TS. The SFCP in the new TS Section 6.8.4.o references NEI 04-10, Revision 1, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from the TS to a licensee-controlled

document, provided those frequencies are changed in accordance with NEI 04-10, Revision 1, which is specified in the administrative controls of the TS.

The licensee's proposed adoption of TSTF-425, Revision 3, and the risk-informed methodology of NEI 04-10, Revision 1, as referenced in the Administrative Controls section of the TS, satisfies the key principles of risk-informed decisionmaking applied to changes to TS as delineated in RG 1.177, Revision 1, and RG 1.174, Revision 3, 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 performance monitoring using measurement strategies.

The regulations in 10 CFR 50.36(c) require, in part, that the TS include SRs. The regulation at 10 CFR 50.36(c)(3) states that: "Surveillance Requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The NRC staff finds that with the proposed relocation of surveillance frequencies to a licensee-controlled document that will be administratively controlled in accordance with the SFCP, as specified in the proposed TS 6.8.4.o, the licensee will continue to meet the requirements in 10 CFR 50.36.

The NRC staff notes that the Technical Specification changes requested in the licensee's submittal do not modify the requirements for testing pumps, valves, and dynamic restraints and their frequencies specified in the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) as incorporated by reference in 10 CFR 50.55a, "Codes and standards," for the Inservice Testing (IST) Program at VCSNS Unit 1. According to current NRC information, the OM Code of record for the IST Program at VCSNS Unit 1 is the 2004 Edition through the 2006 Addenda of the ASME OM Code. Alternatives to the IST Program requirements specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a must be requested in accordance with 10 CFR 50.55a(z), "Alternatives to codes and standards requirements."

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the South Carolina State official of the proposed issuance of the amendment on June 15, 2022. On June 29, 2022 the State official confirmed the State of South Carolina had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance

requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on June 15, 2021 (86 FR 31741). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 222 TO RELOCATE SELECTED SURVEILLANCE FREQUENCIES TO A RISK-INFORMED LICENSEE CONTROLLED PROGRAM (EPID L-2021-LLA-0064) DATE AUGUST 16, 2022

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