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

November 29, 2016

Ms. Tanya Hamilton
Site Vice President
Shearon Harris Nuclear Power Plant
Duke Energy
5413 Shearon Harris Road
New Hill, NC 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING RISK-INFORMED JUSTIFICATIONS FOR THE RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE-CONTROLLED PROGRAM (CAC NO. MF6583)

Dear Ms. Hamilton:

The Nuclear Regulatory Commission (NRC) has issued Amendment No. 154 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment changes the Technical Specifications (TSs) in response to your application dated August 18, 2015, as supplemented by letters dated September 29, 2015, and February 5, April 28, and May 19, 2016.

The amendment revises the TSs by relocating specfic surveillance frequencies to a licensee-controlled program. The TSs are revised to require that changes to such surveillance frequencies will be made in accordance with Nuclear Energy Institute document NEI 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," Revision 1. The changes are consistent with NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specifications change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control – RITSTF [Risk Informed TSTF] Initiative 5b," Revision 3. The *Federal Regsiter* (FR) notice published on July 6, 2009 (74 FR 31996), announced the availability of the TSTF.

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

Sincerely,

Martha Barillas, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 154 to NPF-63
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 154
License No. NPF-63

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee) (previously Duke Energy Progress, Inc.), August 18, 2015, as supplemented by letters dated September 29, 2015, and February 5, April 28, and May 19, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 154, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeanne A. Dion, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License No. NPF-63
and the Technical Specifications

Date of Issuance: November 29, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 154

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the renewed facility operating license with the revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

Remove	Insert
Page 4	Page 4

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

Remove	Insert	Remove	Insert	Remove	Insert	Remove	Insert
1-8	1-8	3/4 3-45	3/4 3-45	3/4 5-3	3/4 5-3	3/4 8-6	3/4 8-6
3/4 1-1	3/4 1-1	3/4 3-46	3/4 3-46	3/4 5-4	3/4 5-4	3/4 8-9	3/4 8-9
3/4 1-2	3/4 1-2	3/4 3-47	3/4 3-47	3/4 5-5	3/4 5-5	3/4 8-12	3/4 8-12
3/4 1-3	3/4 1-3	3/4 3-48	3/4 3-48	3/4 5-9	3/4 5-9	3/4 8-13	3/4 8-13
3/4 1-7	3/4 1-7	3/4 3-49	3/4 3-49	3/4 6-1	3/4 6-1	3/4 8-17	3/4 8-17
3/4 1-8	3/4 1-8	3/4 3-54	3/4 3-54	3/4 6-5	3/4 6-5	3/4 8-18	3/4 8-18
3/4 1-11	3/4 1-11	3/4 3-55	3/4 3-55	3/4 6-6	3/4 6-6	3/4 8-19	3/4 8-19
3/4 1-13	3/4 1-13	3/4 3-63	3/4 3-63	3/4 6-7	3/4 6-7	3/4 8-20	3/4 8-20
3/4 1-15	3/4 1-15	3/4 3-65	3/4 3-65	3/4 6-10	3/4 6-10	3/4 8-39	3/4 8-39
3/4 1-17	3/4 1-17	3/4 3-70	3/4 3-70	3/4 6-11	3/4 6-11	3/4 9-1	3/4 9-1
3/4 1-18	3/4 1-18	3/4 3-71	3/4 3-71	3/4 6-12	3/4 6-12	3/4 9-3	3/4 9-3
3/4 1-19	3/4 1-19	3/4 4-1	3/4 4-1	3/4 6-13	3/4 6-13	3/4 9-5	3/4 9-5
3/4 1-20	3/4 1-20	3/4 4-2	3/4 4-2	3/4 6-15	3/4 6-15	3/4 9-9	3/4 9-9
3/4 1-21	3/4 1-21	3/4 4-3	3/4 4-3	3/4 7-4	3/4 7-4	3/4 9-10	3/4 9-10
3/4 2-2	3/4 2-2	3/4 4-5	3/4 4-5	3/4 7-5	3/4 7-5	3/4 9-11	3/4 9-11
3/4 2-6	3/4 2-6	3/4 4-6	3/4 4-6	3/4 7-6	3/4 7-6	3/4 9-12	3/4 9-12
3/4 2-10a	3/4 2-10a	3/4 4-7	3/4 4-7	3/4 7-8	3/4 7-8	3/4 9-13	3/4 9-13
3/4 2-13	3/4 2-13	3/4 4-10	3/4 4-10	3/4 7-10	3/4 7-10	3/4 9-14	3/4 9-14
3/4 2-14	3/4 2-14	3/4 4-12	3/4 4-12	3/4 7-11	3/4 7-11	3/4 9-15	3/4 9-15
3/4 3-1	3/4 3-1	3/4 4-22	3/4 4-22	3/4 7-12	3/4 7-12	3/4 10-1	3/4 10-1
3/4 3-11	3/4 3-11	3/4 4-24	3/4 4-24	3/4 7-13	3/4 7-13	3/4 10-2	3/4 10-2
3/4 3-12	3/4 3-12	3/4 4-28	3/4 4-28	3/4 7-15	3/4 7-15	3/4 10-3	3/4 10-3
3/4 3-13	3/4 3-13	3/4 4-31	3/4 4-31	3/4 7-16	3/4 7-16	3/4 10-4	3/4 10-4
3/4 3-14	3/4 3-14	3/4 4-33	3/4 4-33	3/4 7-17	3/4 7-17	3/4 10-5	3/4 10-5
3/4 3-17	3/4 3-17	3/4 4-34	3/4 4-34	3/4 7-18	3/4 7-18	3/4 11-15	3/4 11-15
3/4 3-41	3/4 3-41	3/4 4-42	3/4 4-42	3/4 7-30	3/4 7-30	- - -	6-19j
3/4 3-42	3/4 3-42	3/4 4-44	3/4 4-44	3/4 7-31	3/4 7-31		
3/4 3-43	3/4 3-43	3/4 5-1	3/4 5-1	3/4 8-4	3/4 8-4		
3/4 3-44	3/4 3-44	3/4 5-2	3/4 5-2	3/4 8-5	3/4 8-5		

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

(1) Maximum Power Level

Duke Energy Progress, LLC, is authorized to operate the facility at reactor Core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 154, are hereby incorporated into this license. Duke Energy Progress, LLC. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

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

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company* shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II (1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company* will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

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

* On April 29, 2013, the name "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc." On August 1, 2015, the name "Duke Energy Progress, Inc." was changed to "Duke Energy Progress, LLC."

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.
SFCP	At the frequency specified in 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 1770 pcm for 3-loop operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1770 pcm, 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 determined to be greater than or equal to 1770 pcm:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at the frequency specified in the Surveillance Frequency Control Program by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. 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; and

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors below, with the control banks at the maximum insertion limit of Specification 3.1.3.6:

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (CONTINUED)

1) Reactor Coolant System boron concentration,
2) Control rod position,
3) Reactor Coolant System average temperature,
4) Fuel burnup based on gross thermal energy generation,
5) Xenon concentration, and
6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ±1000 pcm at the frequency specified in the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1d., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If later experience shows adjustment is desirable at approximately 60 EFPD, the adjustment is permissible.

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106.

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 determined to be greater than or equal to the required value:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At the frequency specified in 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 tank via either a boric acid transfer pump or a gravity feed connection and a charging/safety injection pump to the Reactor Coolant System if the boric acid tank in Specification 3.1.2.5a. or 3.1.2.6a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging/safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. or 3.1.2.6b. is OPERABLE.

APPLICABILITY: MODES 4*, 5*, and 6*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header is greater than or equal to 65°F when a flow path from the boric acid tank is used, and

b. At the frequencies specified in 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.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

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 tank via a boric acid transfer pump and a charging/safety injection pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging/safety injection pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106 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 HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header tank is greater than or equal to 65°F when a flow path from the boric acid tank is used;

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

c. At the frequencies specified in the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal; and

d. At the frequencies specified in the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

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. The boric acid tank with:

1. A minimum contained borated water volume of 7150 gallons which is ensured by maintaining indicated level of greater than or equal to 23%,
2. A boron concentration of between 7000 and 7750 ppm, and
3. A minimum solution temperature of 65°F.

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

1. A minimum contained borated water volume of 106,000 gallons, which is equivalent to 12% indicated level,
2. A boron concentration of between 2400 and 2600 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. At the frequency specified in 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 tank solution temperature when it is the source of borated water.

b. At the frequency specified in 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. At the frequency specified in 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 tank solution temperature when it is the source of borated water.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is either less than 40°F or greater than 125°F.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

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 power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F_{\Delta H}^{N}$ are verified to be within their limits within 72 hours; and

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

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each rod shall be determined to be within the group demand limit by verifying the individual rod positions at the frequency specified in 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 rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at the frequency specified in the Surveillance Frequency Control Program.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

    2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

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

    1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

    2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

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

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

ACTION:

a. With one of the above required position indicator(s) inoperable, either restore the indicator to OPERABLE within 8 hours or open the Reactor Trip System breakers.

b. With more than one of the above required position indicators inoperable, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

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

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

**See Special Test Exceptions Specification 3.10.5.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual 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:

a. With the drop time of any 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.

b. With the rod drop times within limits but determined with two reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of shutdown and control rods shall be demonstrated through measurement prior to reactor criticality:

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

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

c. At the frequency specified in the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn as specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

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

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn as specified in the COLR:

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

b. At the frequency specified in the Surveillance Frequency Control Program thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

REACTIVITY CONTROL SYSTEMS

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), plant procedure PLP-106.

APPLICABILITY: MODES 1* and 2* **

ACTION:

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

a. Restore the control banks to within the insertion limit specified in the COLR within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limit specified in the COLR at the frequency specified in 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.

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*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

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 at the frequency specified in the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE, and

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

4.2.1.2 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 The target AFD of each OPERABLE excore channel shall be determined by excore measurement at the frequency specified in the Surveillance Frequency Control Program in conjunction with the requirements of Specification 4.2.2.2. The target AFD may be updated between measurements by adding the most recently measured value and the change in the predicted value since the measurement. 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 $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

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

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

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

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. At the frequency specified in the Surveillance Frequency Control Program, whichever occurs first.

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

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F_{\Delta H}$ shall be determined to be within acceptable limits:

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

b. At the frequency specified in the Surveillance Frequency Control Program thereafter.

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; and

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

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at the frequency specified in the Surveillance Frequency Control Program when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at the frequency specified in the Surveillance Frequency Control Program.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg} ≤ 594.8°F after addition for instrument uncertainty, and

b. Pressurizer Pressure ≥ 2185 psig* after subtraction for instrument uncertainty, and

c. RCS total flow rate ≥ 293,540 gpm after subtraction for instrument uncertainty.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at the frequency specified in the Surveillance Frequency Control Program.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at the frequency specified in the Surveillance Frequency Control Program.**

* This limit is not applicable during either a THERMAL POWER Ramp in excess of ±5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of ±10% RATED THERMAL POWER.

** Required to be performed within 24 hours after ≥ 95% RATED THERMAL POWER.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channels and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit, specified in the Technical Specification Equipment List Program, plant procedure PLP-106, at the frequency specified in 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(12)	N.A.	1, 2, 3*, 4*, 5
2.	Power Range, Neutron Flux						
	a. High Setpoint	SFCP	SFCP (2,4), SFCP (3,4), SFCP (4,6), SFCP (4,5)	SFCP	N.A.	N.A.	1, 2
	b. Low Setpoint	SFCP	SFCP (4)	S/U(1)	N.A	N.A	1***, 2
3.	Power Range, Neutron Flux, High Positive Rate	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1, 2
4.	Power Range, Neutron Flux, High Negative Rate	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1, 2
5.	Intermediate Range, Neutron Flux	SFCP	SFCP (4,5)	S/U(1)	N.A.	N.A.	1***, 2
6.	Source Range, Neutron Flux	SFCP	SFCP (4,5)	S/U(1), SFCP(8)	N.A.	N.A.	2**, 3, 4, 5
7.	Overtemperature ΔT	SFCP	SFCP (11)	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 (16)
10.	Pressurizer Pressure -- High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2

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
11. Pressurizer Water Level--High	SFCP	SFCP	SFCP	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	SFCP	SFCP	SFCP	N.A.	N.A.	1
13. Steam Generator Water Level--Low Low	SFCP	SFCP	SFCP(16)	N.A.	N.A.	1, 2 (16)
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(9)	N.A.	1
16. Underfrequency -- Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP(9)	N.A.	1
17. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	SFCP	N.A.	S/U(1,9)	N.A.	1
b. Turbine Throttle Valve Closure	N.A.	SFCP	N.A.	S/U(1,9)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	SFCP(4)	SFCP	N.A.	N.A.	2**

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
19. Reactor Trip System Interlocks (Continued)						
b. Low Power Reactor Trips Block, P-7	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
d. Power Range Neutron Flux P-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
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	SFCP(7, 9,10)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	SFCP (7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	SFCP (7, 13) SFCP (14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

(1) If not performed in previous 31 days.

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

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

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

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(8) 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.

(9) Setpoint verification is not applicable.

(10) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within its limit specified in the Technical Specification Equipment List Program, plant procedure PLP-106, at the frequency specified in the Surveillance Frequency Control Program

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	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 Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)								
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. Containment Pressure -- High 1	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure -- Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure -- Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3. 4
c. Containment Pressure--High-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP(3)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Containment Spray Initiation	See Item 2.a. above for Manual Containment Spray Surveillance Requirements.							
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
3) Containment Pressure --High-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Ventilation Isolation								
1) Manual Containment Spray Initiation	See Item 2.a. above for Manual Containment Spray Surveillance Requirements.							
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1, 2)	SFCP(1, 2)	SFCP(2)	1, 2, 3, 4, 6#
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) Containment Radioactivity								
a) Area Monitors (both preentry and normal purges)	See Table 4.3-3, Item 1a, for surveillance requirements.							
b) Airborne Gaseous Radioactivity								
(1) RCS Leak Detection (normal purge)	See Table 4.3-3, Item 1b1, for surveillance requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
(2) Preentry Purge Detector	See Table 4.3-3, Item 1b2, for surveillance requirements.							
c) Airborne Particulate Radioactivity								
(1) RCS Leak Detection (normal purge)	See Table 4.3-3, Item 1C1, for surveillance requirements.							
(2) Preentry Purge Detector	See Table 4.3-3, Item 1C2, for surveillance requirements.							
5) Manual Phase A Isolation	See Item 3.a.1) above for Manual Phase A Isolation Surveillance Requirements.							
4. Main Steam Line Isolation								
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)(4)	SFCP(1)	SFCP	1, 2, 3, 4
c. Containment Pressure--High-2	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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. Main Steam Line Isolation (Continued)								
d. Steam Line Pressure --Low	See Item 1.e. above for Steam Line Pressure --Low Surveillance Requirements.							
e. Negative Steam Line Pressure Rate--High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	3**, 4**
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2
b. Steam Generator Water Level--High -High (P-14)	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3
c. Steam Generator Water Level--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss of Offsite Power Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2
g. Steam Line Differential Pressure--High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	SFCP(3)	1, 2, 3
Coincident With Main Steam Line Isolation (Causes AFW Isolation)	See Item 4. above for all Main Steam Line Isolation Surveillance Requirements.							
7. Safety Injection Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP(3)	1, 2, 3, 4
b. RWST Level --Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	SFCP(3)	1, 2, 3, 4
Coincident With Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Containment Spray Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP(3)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Containment Spray Switchover to Containment Sump (Continued)								
b. RWST Level--Low-Low	See Item 7.b. above for RWST Level--Low-Low Surveillance Requirements.							
Coincident with Containment Spray	See Item 2. above for Containment Spray Surveillance Requirements.							
9. Loss-of-Offsite Power								
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	SFCP	N.A.	SFCP*	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 6.9 kV Emergency Bus Undervoltage --Secondary	N.A.	SFCP	N.A.	SFCP*	N.A.	N.A.	N.A.	1, 2, 3, 4
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure,								
P-11	N.A.	SFCP	SFCP	N.A.	SFCP(1)	SFCP(1)	N.A.	1, 2, 3
Not P-11	N.A.	SFCP	SFCP	N.A.	SFCP(1)	SFCP(1)	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	SFCP	SFCP	N.A.	SFCP(1)	SFCP(1)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Engineered Safety Features Actuation System Interlocks (Continued)								
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level, P-14	See Item 5.b., above for P-14 Surveillance Requirements.							

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

(2) The Surveillance Requirements of Specification 4.9.9 apply during CORE ALTERATIONS or movement of irradiated fuel in containment.

(3) Except for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741 which shall be tested at the frequency specified in the Surveillance Frequency Control Program and during each COLD SHUTDOWN exceeding 72 hours unless they have been tested within the previous 92 days.

(4) The Steam Line Isolation-Safety Injection (Block-Reset) switches enable the Negative Steam Line Pressure Rate--High signal (item 4.e) when used below the P-11 setpoint. Verify proper operation of these switches each time they are used.

* Setpoint verification not required.

\# During CORE ALTERATIONS or movement of irradiated fuel in containment.

** Trip Function automatically blocked above P-11 and may be blocked below P-11 when safety injection or low steamline pressure is not blocked.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment Radioactivity--				
a. Containment Ventilation Isolation Signal Area Monitors	SFCP	SFCP	SFCP	1, 2, 3, 4, 6
b. Airborne Gaseous Radioactivity				
1) RCS Leakage Detection	SFCP	SFCP	SFCP	1, 2, 3, 4
2) Pre-entry Purge	SFCP	SFCP	SFCP##	#
c. Airborne Particulate Radioactivity				
1) RCS Leakage Detection	SFCP	SFCP	SFCP	1, 2, 3, 4
2) Pre-entry Purge	SFCP	SFCP	SFCP##	#
2. Spent Fuel Pool Area -- Fuel Handling Building Emergency Exhaust Actuation				
a. Fuel Handling Building Operating Floor--South Network	SFCP	SFCP	SFCP	**
b. Fuel Handling Building Operating Floor--North Network	SFCP	SFCP	SFCP	*

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Control Room Outside Air Intakes				
a. Normal Outside Air Intake Isolation	SFCP	SFCP	SFCP	1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools.
b. Emergency Outside Air Intake Isolation--South Intake	SFCP	SFCP	SFCP	1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools.
c. Emergency Outside Air Intake Isolation--North Intake	SFCP	SFCP	SFCP	1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools.

TABLE NOTATIONS

* With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.

** With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.

\# Whenever pre-entry purge system is to be used.

\#\# Prior to operation of pre-entry purge unless performed within the last 92 days.

INSTRUMENTATION
REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.a The Remote Shutdown System monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

3.3.3.5.b All transfer switches, Auxiliary Control Panel Controls and Auxiliary Transfer Panel Controls for the OPERABILITY of those components required by the SHNPP Safe Shutdown Analysis to (1) remove decay heat via auxiliary feedwater flow and steam generator power-operated relief valve flow from steam generators A and B, (2) control RCS inventory through the normal charging flow path, (3) control RCS pressure, (4) control reactivity, and (5) remove decay heat via the RHR system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels required by Table 3.3-9, restore the inoperable channels to OPERABLE status within 60 days or submit a Special Report in accordance with Specification 6.9.2 within 14 additional days.

c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit and control switch required by 3.3.3.5.b, shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Reactor Coolant System Hot-Leg Temperature	SFCP	SFCP
2.	Reactor Coolant System Cold-Leg Temperature	SFCP	SFCP
3.	Pressurizer Pressure	SFCP	SFCP
4.	Pressurizer Level	SFCP	SFCP
5.	Steam Generator Pressure	SFCP	SFCP
6.	Steam Generator Water Level--Wide Range	SFCP	SFCP
7.	Residual Heat Removal Flow	SFCP	SFCP
8.	Auxiliary Feedwater Flow	SFCP	SFCP
9.	Condensate Storage Tank Level	SFCP	SFCP
10.	Reactor Coolant System Pressure--Wide Range	SFCP	SFCP
11.	Wide-Range Flux Monitor (SR Indicator)	SFCP	SFCP
12.	Charging Header Flow	SFCP	SFCP
13.	a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge ΔP	SFCP	SFCP
	b. Auxiliary Feedwater Turbine Speed	SFCP	SFCP
14.	Boric Acid Tank Level	SFCP	SFCP

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure		
	a. Narrow Range	SFCP	SFCP
	b. Wide Range	SFCP	SFCP
2.	Reactor Coolant Hot-Leg Temperature--Wide Range	SFCP	SFCP
3.	Reactor Coolant Cold-Leg Temperature--Wide Range	SFCP	SFCP
4.	Reactor Coolant Pressure--Wide Range	SFCP	SFCP
5.	Pressurizer Water Level	SFCP	SFCP
6.	Steam Line Pressure	SFCP	SFCP
7.	Steam Generator Water Level--Narrow Range	SFCP	SFCP
8.	Steam Generator Water Level--Wide Range	SFCP	SFCP
9.	Refueling Water Storage Tank Water Level	SFCP	SFCP
10.	Auxiliary Feedwater Flow Rate	SFCP	SFCP
11.	Reactor Coolant System Subcooling Margin Monitor	SFCP	SFCP
12.	PORV Position Indicator	SFCP	SFCP
13.	PORV Block Valve Position Indicator	SFCP	SFCP
14.	Pressurizer Safety Valve Position Indicator	SFCP	SFCP
15.	Containment Water Level (ECCS Sump)--Narrow Range	SFCP	SFCP
16.	Containment Water Level--Wide Range	SFCP	SFCP

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
17. In Core Thermocouples	SFCP	SFCP
18. Plant Vent Stack--High Range Noble Gas Monitor	SFCP	SFCP
19. Main Steam Line Radiation Monitors	SFCP	SFCP
20. Containment--High Range Radiation Monitor	SFCP	SFCP*
21. Reactor Vessel Level	SFCP	SFCP
22. Containment Spray NaOH Tank Level	SFCP	SFCP
23. Turbine Building Vent Stack High Range Noble Gas Monitor	SFCP	SFCP
24. Waste Processing Building Vent Stack High Range Noble Gas Monitors		
a. Vent Stack 5	SFCP	SFCP
b. Vent Stack 5A	SFCP	SFCP
25. Condensate Storage Tank Level	SFCP	SFCP

* CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions Specification 3.10.4.

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: MODE 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

*All reactor coolant pumps may be deenergized for up to 1 hour provided:

(1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying narrow range secondary side water level to be greater than or equal to 30% at the frequency specified in the Surveillance Frequency Control Program.

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

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE at the frequency specified in 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 wide range (WR) secondary side water level is greater than 74% or narrow range (NR) secondary side water level is greater than 30% at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at the frequency specified in the Surveillance Frequency Control Program .

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

---

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

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

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

b. The secondary side water level of at least two steam generators shall be greater than 74% wide range (WR) or greater than 30% narrow range (NR).

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

---

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

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

*** A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

---

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

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

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water level of less than or equal to 75% 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 only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water level shall be determined to be within its limit at the frequency specified in 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 power (kW) at the frequency specified in the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of the Inservice Testing Program, each PORV shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Performing a CHANNEL CALIBRATION of the actuation instrumentation, and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4, prior to going to 325°F.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The accumulator for the safety-related PORVs shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by isolating the normal air and nitrogen supplies and operating the valves through a complete cycle of full travel.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. For Containment Airborne Gaseous and Particulate Monitoring Systems, performance of CHANNEL CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3.

b. For Reactor Cavity Sump Level and Flow Monitoring System, performance of CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated, at the frequency specified in the Surveillance Frequency Control Program, to be within each of the above limits by:

a. Monitoring the containment Airborne Gaseous or Particulate Radioactivity Monitor;

b. Monitoring the containment sump inventory and Flow Monitoring System;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig 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*; and

e. Monitoring the Reactor Head Flange Leakoff System.

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

a. At the frequency specified in the Surveillance Frequency Control Program,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

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

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

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 to be ≤ 150 gallons per day through any one steam generator at the frequency specified in the Surveillance Frequency Control Program**.

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* Not required to be performed until 12 hours after establishment of steady-state operation. Not applicable to primary-to-secondary leakage.

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

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At the frequency specified in the Surveillance Frequency Control Program
Chloride	At the frequency specified in the Surveillance Frequency Control Program
Fluoride	At the frequency specified in the Surveillance Frequency Control Program

* Not required with T_{avg} less than or equal to 250°F

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At the frequency specified in the Surveillance Frequency Control Program.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At the frequency specified in the Surveillance Frequency Control Program.	1
3. Radiochemical for $\bar{E}$ Determination	At the frequency specified in the Surveillance Frequency Control Program**.	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a. Once per 4 hours, whenever the specific activity exceeds 0.35 µCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b. One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITION FOR OPERATION

3.4.9.2 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, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1 hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the pressure limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

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

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Specification Equipment List Program, plant procedure PLP-106.

SURVEILLANCE REQUIREMENTS (Continued)

entering a condition in which the PORV is required OPERABLE and at the frequency specified in the Surveillance Frequency Control Program when the PORV is required OPERABLE;

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

c. Verifying the PORV isolation valve is open at the frequency specified in the Surveillance Frequency Control Program when the PORV is being used for overpressure protection.

4.4.9.4.2 The RCS vent(s) shall be verified to be open at the frequency specified in the Surveillance Frequency Control Program* when the vent(s) is being used for overpressure protection.

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

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

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

a. Reactor vessel head, and

b. Pressurizer steam space

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

ACTION:

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

b. With both Reactor Coolant System vent paths inoperable, due to causes other than the removal of power to both block valves pursuant to Action a, maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 (Section deleted)

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

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

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open with power supply circuit breaker open,

b. A contained borated water volume of between 66 and 96% indicated level,

c. A boron concentration of between 2400 and 2600 ppm, and

d. A nitrogen cover pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With one accumulator inoperable due to boron concentration not within limits, restore the boron concentration within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

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

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and
2. Verifying that each accumulator isolation valve is open.

---

*RCS pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At the frequency specified in the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of greater than or equal to 76 gallons, which is equivalent to an indicated level change of 9%, by verifying the boron concentration of the accumulator solution#; and

c. At the frequency specified in the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the respective isolation valve operator is open.

This surveillance is not required when the volume increase makeup source is the Refueling Water Storage Tank (RWST) and the RWST has not been diluted since verifying that the RWST boron concentration is equal to or greater than the accumulator boron concentration limit.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem 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. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

-- NOTE --

*The 'A' Train ECCS subsystem is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

1. Verifying that the following valves are in the indicated positions with the control power disconnect switch in the "OFF" position, and the valve control switch in the "PULL TO LOCK" position:

CP&L Valve No.	EBASCO Valve No.	Valve Function	Valve Position
1SI-107	2SI-V500SA-1	High Head Safety Injection to Reactor Coolant System Hot Legs	Closed
1SI-86	2SI-V501SB-1	High Head Safety Injection to Reactor Coolant System Hot Legs	Closed
1SI-52	2SI-V502SA-1	High Head Safety Injection to Reactor Coolant System Cold Legs	Closed
1SI-340	2SI-V579SA-1	Low Head Safety Injection to Reactor Coolant System Cold Legs	Open
1SI-341	2SI-V578SB-1	Low Head Safety Injection to Reactor Coolant System Cold Legs	Open
1SI-359	2SI-V587SA-1	Low Head Safety Injection to Reactor Coolant System Hot Legs	Closed

b. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that the ECCS locations susceptible to gas accumulation are sufficiently filled with water, and

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

d. At the frequency specified in 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.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on safety injection actuation test signal and on safety injection switchover to containment sump from an RWST Lo-Lo level test signal, and

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

      a) Charging/safety injection pump,

      b) RHR pump.

f. By verifying that each of the following pumps develops the required differential pressure when tested pursuant to the Inservice Testing Program:

   1. Charging/safety injection pump (Refer to Specification 4.1.2.4)

   2. RHR pump ≥ 100 psid at a flow rate of at least 3663 gpm.

g. By verifying that the locking mechanism is in place and locked for the following High Head ECCS throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

   2. At the frequency specified in the Surveillance Frequency Control Program.

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 436,000 gallons, which is equivalent to 92% indicated level.

b. A boron concentration of between 2400 and 2600 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 125°F.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At the frequency specified in 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. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 125°F.

* Except that while performing surveillance 4.4.6.2.2, the tank must be returned to OPERABLE status within 12 hours.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that all penetrations*# not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3;

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

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2a. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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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.

\# Valves CP-B3, CP-B7, and CM-B5 may be verified at the frequency specified in the Surveillance Frequency Control Program by manual remote keylock switch position.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE by:

a. Performing required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions###. The acceptance criteria for air lock testing are:

1. Overall air lock leakage rate is $\leq .05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq .01\ L_a$ when tested at $\geq P_a$.

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

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1. An inoperable air lock door does not invalidate the previous successful performance of the overall airlock leakage test.

2. Results shall be evaluated against Specification 3.6.1.2.a in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

** Only required to be performed upon entry or exit through the containment air lock. (If Surveillance Requirement 4.6.1.3.b has not been performed in the interval specified by the Surveillance Frequency Control Program, then perform Surveillance Requirement 4.6.1.3.b during the next containment entry through the associated air lock.)

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -1.0 inches water gauge and 1.6 psig.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at the frequency specified in 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 the following locations and shall be determined at the frequency specified in the Surveillance Frequency Control Program, to be within the limit:

Location

a. Elevation 290 ft

b. Elevation 240 ft

c. Elevation 230 ft

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 42-inch containment preentry purge makeup and exhaust isolation valve shall be verified to be sealed closed and closed at the frequency specified in the Surveillance Frequency Control Program.

4.6.1.7.2 At the frequency specified in the Surveillance Frequency Control Program, the inboard and outboard valves in each makeup and exhaust penetration (2-42 inch valves and 2-8 inch valves) shall be demonstrated OPERABLE by verifying that the measured penetration leakage rate is less than 0.06 L_a when pressurized to P_a.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. Refer also to Specification 3.6.2.3 Action.

-- NOTE --

**The 'A' Train Containment Spray System is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

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

b. By verifying that, on an indicated recirculation flow of at least 1832 gpm, each pump develops a differential pressure of greater than or equal to 186 psi when tested pursuant to the Inservice Testing Program;

c. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal and

   2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

   3. Verifying that, coincident with an indication of containment spray pump running, each automatic valve from the sump and RWST actuates to its appropriate position following an RWST Lo-Lo test signal.

d. At the frequency specified in the Surveillance Frequency Control Program by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

e. At the frequency specified in the Surveillance Frequency Control Program by verifying that containment spray locations susceptible to gas accumulation are sufficiently filled with water.

---

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

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 3268 and 3768 gallons of between 27 and 29 weight % of NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

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.

-- NOTE --

*The Spray Additive System is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At the frequency specified in 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. At the frequency specified in 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. At the frequency specified in the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a containment spray or containment isolation phase A test signal as applicable; and

d. At the frequency specified in the Surveillance Frequency Control Program by verifying each eductor flow rate is between 17.2 and 22.2 gpm, using the RWST as the test source containing at least 436,000 gallons of water.

LIMITING CONDITION FOR OPERATION

3.6.2.3 Four containment fan coolers (AH-1, AH-2, AH-3, and AH-4) shall be OPERABLE with one of two fans in each cooler capable of operation at low speed. Train SA consists of AH-2 and AH-3. Train SB consists of AH-1 and AH- 4.

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

ACTION:

a. With one train of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore the inoperable train of fan coolers 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 both trains of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore at least one train of fan coolers 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 both above required trains of fan coolers 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 train of the above required containment fan coolers inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable train of containment fan coolers 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.

-- NOTE --

*The 'A' Train containment fan coolers and the 'A' Train Containment Spray System are allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each train of containment fan coolers shall be demonstrated OPERABLE:

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

1. Starting each fan train from the control room, and verifying that each fan train operates for at least 15 minutes, and

2. Verifying a cooling water flow rate, after correction to design basis service water conditions, of greater than or equal to 1300 gpm to each cooler.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that each fan train starts automatically on a safety injection test signal.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

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

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

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit specified in the Technical Specification Equipment List Program, plant procedure PLP-106, when tested pursuant to the Inservice Testing Program.

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible. (NOTE: LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. Following restoration of one AFW train, all applicable LCOs apply based on the time the LCOs initially occurred.)

-- NOTE --

*The 'A' Train auxiliary feedwater pump is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or the Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

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

1. Demonstrating that each motor-driven pump satisfies performance requirements by either:

a) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1514 psid at a recirculation flow of greater than or equal to 50 gpm (25 KPPH), or

b) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1259 psid at a flow rate of greater than or equal to 430 gpm (215 KPPH).

SURVEILLANCE REQUIREMENTS (CONTINUED)

2. Demonstrating that the steam turbine - driven pump satisfies performance requirements by either:

****

NOTE: The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

****

a) Verifying the pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1167 psid at a recirculation flow of greater than or equal to 81 gpm (40.5 KPPH) when the secondary steam supply pressure is greater than 210 psig, or

b) Verifying the pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1400 psid at a flow rate of greater than or equal to 430 gpm (215 KPPH) when the secondary steam supply pressure is greater than 280 psig.

b. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying by flow or position check 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 that the isolation valves in the suction line from the CST are locked open.

c. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that each motor-driven auxiliary feedwater pump starts automatically, as designed, upon receipt of a test signal and that the respective pressure control valve for each motor-driven pump and each flow control valve with an auto-open feature respond as required;

2. Verifying that the turbine-driven auxiliary feedwater pump starts automatically, as designed, upon receipt of a test signal. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3; and

3. Verifying that the motor-operated auxiliary feedwater isolation valves and flow control valves close as required upon receipt of an appropriate test signal for steamline differential pressure high coincident with main steam isolation.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained water volume of at least 270,000 gallons of water, which is equivalent to 62% indicated level.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST 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 Emergency Service Water System as a backup supply to the auxiliary feedwater pumps and restore the CST 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 CST shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Emergency Service Water System shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying that each valve, required to permit the Emergency Service Water System to supply water to the auxiliary feedwater pumps, is open whenever the Emergency Service Water System is the supply source for the auxiliary 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 Radioactivity Determination* or Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At the frequency specified in the Surveillance Frequency Control Program.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a. Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b. At the frequency specified in the Surveillance Frequency Control Program, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 15 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators 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 at the frequency specified in the Surveillance Frequency Control Program when the temperature of either the reactor or secondary coolant is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two component cooling water (CCW) pumps*, heat exchangers and essential flow paths shall be OPERABLE.

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

ACTION:

With only one component cooling water flow path OPERABLE, restore at least two flow paths 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 flow paths shall be demonstrated OPERABLE:

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

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

1. Each automatic valve servicing safety-related equipment or isolating non-safety-related components actuates to its correct position on a Safety Injection test signal, and
2. Each Component Cooling Water System pump required to be OPERABLE starts automatically on a Safety Injection test signal.
3. Each automatic valve serving the gross failed fuel detector and sample system heat exchangers actuates to its correct position on a Low Surge Tank Level test signal.

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* The breaker for CCW pump 1C-SAB shall not be racked into either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.

**The 'A' Train component cooling water flow path is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With only one emergency 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.

-- NOTE --

*The 'A' Train emergency service water loop is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two emergency service water loops shall be demonstrated OPERABLE:

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

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

   1. Each automatic valve servicing safety-related equipment or isolating non-safety portions of the system actuates to its correct position on a Safety Injection test signal, and

   2. Each emergency service water pump and each emergency service water booster pump starts automatically on a Safety Injection test signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum auxiliary reservoir water level at or above elevation 250 feet Mean Sea Level, USGS datum, and a minimum main reservoir water level at or above 206 feet Mean Sea Level, USGS datum, and

b. A water temperature as measured at the respective intake structure of less than or equal to 94°F.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITION FOR OPERATION (Continued)

c. During movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

  1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable CREFS train to OPERABLE status within 7 days or immediately initiate and maintain operation of the remaining OPERABLE CREFS train in the recirculation mode; or, immediately suspend movement of irradiated fuel.

  2. With both CREFS trains inoperable for reasons other than an inoperable CRE boundary, or with the OPERABLE CREFS train required to be in recirculation mode by Action c.1., not capable of being powered by an OPERABLE emergency power source, immediately suspend all operations involving movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

  3. With one or more CREFS trains inoperable due to inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

SURVEILLANCE REQUIREMENTS

4.7.6 Each CREFS train shall be demonstrated OPERABLE:

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

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

    1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Position C.5.a. C.5.c. and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980; and

SURVEILLANCE REQUIREMENTS (CONTINUED)

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤0.5% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803 -1989.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤0.5% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803-1989.

d. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.1 inches water gauge while operating the system at a flow rate of 4000 cfm ± 10%;
2. Verifying that, on either a Safety Injection or a High Radiation test signal, the system automatically switches into an isolation with recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks;
3. Deleted.
4. Verifying that the heaters dissipate 14 ± 1.4 kW when tested in accordance with ANSI N510-1980; and
5. Deleted.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

g. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

3/4.7.7 REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent RAB Emergency Exhaust Systems shall be OPERABLE.*

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

ACTION:

a. With one RAB Emergency Exhaust System inoperable, restore the inoperable system 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 RAB Emergency Exhaust Systems inoperable due to an inoperable RAB Emergency Exhaust System boundary, restore the RAB Emergency Exhaust System boundary to OPERABLE status within 24 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each RAB Emergency Exhaust System shall be demonstrated OPERABLE:

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

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

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6800 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980;

   2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodine penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803-1989.

* The RAB Emergency Exhaust Systems boundary may be opened intermittently under administrative controls.

** The ‘A’ Train RAB Emergency Exhaust System is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the ‘A’ Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the ‘A’ Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the ‘A’ Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS (CONTINUED)

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803-1989.

d. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is less than 4.1 inches water gauge while operating the unit at a flow rate of 6800 cfm ± 10%,

   2. Verifying that the system starts on a Safety Injection test signal,

   3. Verifying that the system maintains the areas served by the exhaust system at a negative pressure of greater than or equal to 1/8 inch water gauge relative to the outside atmosphere,

   4. Verifying that the filter cooling bypass valve is locked in the balanced position, and

   5. Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the unit at a flow rate of 6800 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6800 cfm ± 10%.

3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 At least two independent Essential Services Chilled Water System loops shall be OPERABLE.

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

ACTION:

With only one Essential Services Chilled Water System 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.

-- NOTE --

*The 'A' Train Essential Services Chilled Water System loop is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.7.13 The Essential Services Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by the Inservice Testing Program, and

b. At the frequency specified in the Surveillance Frequency Control Program by demonstrating that:

   1. Non-essential portions of the system are automatically isolated upon receipt of a Safety Injection actuation signal, and

   2. The system starts automatically on a Safety Injection actuation signal.

3/4.7.14 FUEL STORAGE POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The boron concentration of spent fuel pools shall be ≥ 2000 ppm.

APPLICABILITY: At ALL TIMES for pools that contain nuclear fuel.

ACTION:

a. With the spent fuel pool boron concentration not within the limits, immediately suspend movement of fuel assemblies.

b. Immediately initiate action to restore pool boron concentration within the limit.

SURVEILLANCE REQUIREMENTS

4.7.14 At the frequency specified in the Surveillance Frequency Control Program verify spent fuel pool boron concentration is within the limit by:

a. Sampling the water volume connected to or in applicable pools.

b. In addition to 4.7.14.a, sampling an individual pool containing nuclear fuel if the pool is isolated from other pools.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

h. With one automatic load sequencer inoperable:

1. Restore the automatic load sequencer to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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:

a. Determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and power availability, and

b. Demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by manually transferring the onsite Class 1E power supply from the unit auxiliary transformer to the startup auxiliary transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the main fuel oil storage tank,

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

4. Verifying the diesel generator can start** and accelerate## to synchronous speed (450 rpm) with generator voltage and frequency 6900 ± 690 volts and 60 ± 1.2 Hz,

5. Verifying the diesel generator is synchronized, gradually loaded** to an indicated 6200-6400 kW*** and operates for at least 60 minutes,

6. Verifying the pressure in at least one air start receiver to be greater than or equal to 190 psig, and

7. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

** 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 for special testing or momentary variations due to changing bus loads shall not invalidate the test.

The voltage and frequency conditions shall be met within 10 seconds or gradual acceleration to no-load conditions per vendor recommendations will be an acceptable alternative.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.8.1.1.2 (Continued)

b. Check for and remove accumulated water:

1. From the day tank, at the frequency specified in the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than 1 hour, and

2. From the main fuel oil storage tank, at the frequency specified in the Surveillance Frequency Control Program .

c. By sampling new fuel oil in accordance with ASTM-D4057-81 prior to addition to storage tanks and:

1. By verifying, in accordance with the tests specified in ASTM-D975-81 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°F, when compared to the supplier's certificate, or an absolute specific gravity at 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 26 degrees but less than or equal to 38 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, if the gravity was not determined by comparison with the supplier's certification;

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

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2. By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

d. At the frequency specified in the Surveillance Frequency Control Program by obtaining a sample of fuel oil from the storage tank, in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A.

e. At the frequency specified in the Surveillance Frequency Control Program, the diesel generators shall be started** and accelerated to at least 450 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz in less than or equal to 10 seconds after the start signal.

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

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.8.1.1.2 (Continued)

The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 6200-6400***kW, and operate for at least 60 minutes. The diesel generator shall be started for this test by using one of the following signals on a rotating basis:

1. Simulated loss of offsite power by itself, and
2. A Safety Injection test signal by itself.

This test, if it is performed so that it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

f. At the frequency specified in the Surveillance Frequency Control Program by:

1. DELETED
2. During shutdown, verifying that, on rejection of a load of greater than or equal to 1078 kW, the voltage and frequency are maintained with 6900 ± 690 volts and 60 ± 6.75 Hz, with frequency stabilizing to 60 ± 1.2 Hz within 10 seconds without any safety-related load tripping out or operating in a degraded condition.
3. During shutdown, verifying that the load sequencing timer is OPERABLE with the interval between each load block within 10% of its design interval.
4. During shutdown, simulating a loss of offsite power by itself, and:

*** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.8.1.1.2 (Continued)

13. During shutdown, verifying that all diesel generator trips, except engine overspeed, loss of generator potential transformer circuits, generator differential, and emergency bus differential are automatically bypassed on a simulated or actual loss of offsite power signal in conjunction with a safety injection signal.

14. During shutdown, verifying that within 5 minutes of shutting down the EDG, after the EDG has operated for at least 2 hours at an indicated load of 6200-6400 kw, the EDG starts and accelerates to 6900 ± 690 volts and 60 ± 1.2 hz in 10 seconds or less.

g. At the frequency specified in the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 450 rpm in less than or equal to 10 seconds.

h. At the frequency specified in the Surveillance Frequency Control Program by:

1) Draining each main fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite solution or other appropriate cleaning solution, and

2) Performing a pressure test, of those isolable portions of the diesel fuel oil piping system designed to Section III, subsection ND of the ASME Code, at a test pressure equal to 110% of the system design pressure.

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

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 Emergency Battery Bank 1A-SA and either full capacity charger, 1A-SA or 1B-SA, and,

b. 125-volt Emergency Battery Bank 1B-SB and either full capacity charger, 1A-SB or 1B-SB.

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

ACTION:

With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt Emergency Battery and charger shall be demonstrated OPERABLE:

a. At the frequency specified in 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. At the frequency specified in 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 connection resistance of these items is less than 150×10^{-6} ohm, and

   3. The average electrolyte temperature of 10 connected cells is above 70° F.

SURVEILLANCE REQUIREMENTS (CONTINUED)

c. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

   1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

   2. The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and

   4. The battery charger will supply at least 150 amperes at greater than or equal to 125 volts for at least 4 hours.

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

e. At the frequency specified in 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. At the frequency specified in the Surveillance Frequency Control Program, this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

LIMITING CONDITION FOR OPERATION

ACTION:

a. With one of the required divisions of A.C. ESF buses not fully energized, reenergize the division within 8 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 118-volt A.C. vital bus not energized from its associated inverter, reenergize the 118-volt A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one 118-volt A.C. vital bus not energized from its associated inverter connected to its associated D.C. bus, re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery, reenergize the D.C. bus from its associated Emergency Battery 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 buses shall be determined energized in the required manner at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the buses.

LIMITING CONDITION FOR OPERATION

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

a. Division A, consisting of:

1. 6900-volt Bus 1A-SA and
2. 480-volt Buses 1A2-SA and 1A3-SA, and
3. 118-volt A.C. Vital Buses 1DP-1A-SI and 1DP-1A-SIII energized from their associated inverter connected to 125-volt D.C. Bus DP-1A-SA, and
4. 125-volt D.C. Bus DP-1A-SA energized from Emergency Battery 1A-SA and chargers 1A-SA or 1B-SA, or

b. Division B, consisting of:

1. 6900-volt Bus 1B-SB and
2. 480-volt Buses 1B2-SB and 1B3-SB, and
3. 118-volt AC Vital Buses 1DP-1B-SII and 1DP-1B-SIV energized from their associated inverter connected to 125-volt D.C. Bus DP-1B-SB, and
4. 125-volt D.C. Bus DP-1B-SB energized from Emergency Battery 1B-SB and chargers 1B-SB or 1A-SB.

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required electrical buses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to energize the required electrical buses in the specified manner as soon as possible; and within 8 hours, depressurize and vent the RCS through a vent of ≥ 2.9 square inches.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified buses shall be determined energized in the required manner at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the buses.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 Each containment penetration conductor overcurrent protective device specified in the Technical Specification Equipment List Program, plant procedure PLP-106, shall be OPERABLE.

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

ACTION:

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

a. Restore the protective device(s) to OPERABLE status or deenergize 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, 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 Each containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program:

1. By verifying that the 6900-volt 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,

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, and

SURVEILLANCE REQUIREMENTS (Continued)

4.8.4.1 (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 with a value equal to 300% of the pickup of the long-time delay trip element and 150% of the pickup of the short time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. 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. At the frequency specified in 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.

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve, specified in the Technical Specification Equipment List Program, plant procedure PLP-106, requiring bypass protection, shall be bypassed only under accident conditions by an OPERABLE bypass device.

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

ACTION:

With the thermal overload protection for one or more of the above required valves not capable of being bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The thermal overload protection for the above required valves shall be verified to be bypassed only under accident conditions by an OPERABLE integral bypass device by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry:

a. At the frequency specified in the Surveillance Frequency Control Program for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions; and

b. Following maintenance on the thermal overload bypass relays and circuitry.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.a The boron concentration of all filled portions of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained uniform and within the limit specified in the COLR.

3.9.1.b The valves listed in Table 3.9-1 shall be in their positions required by Table 3.9-1.

APPLICABILITY: MODE 6.

ACTION:

a. With the requirements of Specification 3.9.1.a not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes, and initiate actions to restore boron concentration to within limits.

b. With the requirements of Specification 3.9.1.b not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes, and initiate action to return the valve(s) to the position required by Table 3.9-1.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be determined by chemical analysis to be within the limits of the COLR at the frequency specified in the Surveillance Frequency Control Program.

4.9.1.2 At the frequency specified in the Surveillance Frequency Control Program, verify that the valves listed in Table 3.9-1 are in their positions required by Table 3.9-1.

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, in addition to Action a. above, immediately initiate actions to restore one source range neutron flux monitor to OPERABLE status and determine the boron concentration of the Reactor Coolant System within 4 hours and once per 12 hours thereafter.

SURVEILLANCE REQUIREMENTS

4.9.2 Each neutron flux monitor shall be demonstrated OPERABLE by performance of:

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

b. A CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.

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

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door is capable of being closed and held in place by a minimum of four bolts*,

b. A minimum of one door in each airlock is capable of being closed*, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Be capable of being* closed by a manual or automatic isolation valve, blind flange or equivalent, or

2. Be capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves*.

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

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition, capable of being closed/isolated*, or capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves at the frequency specified in the Surveillance Frequency Control Program during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are either closed/isolated or capable of being closed/isolated*, or

b. Testing the normal containment purge and containment pre-entry purge makeup and exhaust isolation valves per the applicable portions of Specification 4.6.3.2.

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*Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be opened under administrative controls.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2500 gpm at the frequency specified in the Surveillance Frequency Control Program.

4.9.8.1.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

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*The RHR loop may be removed from operation for up to 1 hour per 2-hour period during the performance of CORE ALTERATIONS and core loading verification in the vicinity of the reactor vessel hot legs.

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.9.8.2.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2500 gpm at the frequency specified in the Surveillance Frequency Control Program whenever the water level is at or above the reactor vessel flange.

4.9.8.2.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 900 gpm at the frequency specified in the Surveillance Frequency Control Program whenever the water level is below the reactor vessel flange.

4.9.8.2.3 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

*The operating RHR loop may be removed from operation for up to 1 hour per 2-hour period during the performance of CORE ALTERATIONS and core loading verification in the vicinity of the reactor vessel hot legs.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

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

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the containment purge makeup and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere.*

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at the frequency specified in the Surveillance Frequency Control Program during CORE ALTERATIONS by verifying that containment ventilation isolation occurs on a two-out-of-four High Radiation test signal from the containment area radiation monitors (Table 3.3-6, item 1.a) and by verifying that isolation occurs for each valve using its control switch in the main control room.

*Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be opened under administrative controls.

3/4.9.10 WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6, during movement of irradiated fuel assemblies within containment, or during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts.

ACTION:

With the requirements of the above specification not satisfied, suspend CORE ALTERATIONS, including operations involving movement of fuel assemblies within containment, and initiate actions to restore refueling cavity water level to within limits.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth at the frequency specified in the Surveillance Frequency Control Program.

3/4.9.11 WATER LEVEL – NEW AND SPENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: Whenever irradiated fuel assemblies are in a pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the affected pool area and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 At the frequency specified in the Surveillance Frequency Control Program, when irradiated fuel assemblies are in a pool, the water level in that pool shall be determined to be at least its minimum required depth.

3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Emergency Exhaust System Trains shall be OPERABLE.*

APPLICABILITY: Whenever irradiated fuel is in a storage pool.

ACTION:

a. With one Fuel Handling Building Emergency Exhaust System Train inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Handling Building Emergency Exhaust System Train is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorber.

b. With no Fuel Handling Building Emergency Exhaust System Trains OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Handling Building Emergency Exhaust System Train is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Emergency Exhaust System trains shall be demonstrated OPERABLE:

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

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

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

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* The Fuel Handling Building Emergency Exhaust System boundary may be opened intermittently under administrative controls.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.9.12 (Continued)

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803-1989.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803-1989.

d. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is not greater than 4.1 inches water gauge while operating the unit at a flow rate of 6600 cfm ± 10%,

2. Verifying that, on a High Radiation test signal, the system automatically starts and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,

3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch water gauge, relative to the outside atmosphere, during system operation at a flow rate of 6600 cfm ± 10%, and

4. Deleted

5. Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the unit at a flow rate of 6600 cfm ± 10%.

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 shutdown and control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated single rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any shutdown and 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 shutdown and 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 shutdown and control rod either partially or fully withdrawn shall be determined at the frequency specified in the Surveillance Frequency Control Program.

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

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

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 1.

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

a. Specification 4.2.2.2 and

b. Specification 4.2.3.2.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

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

SURVEILLANCE REQUIREMENTS

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

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

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at the frequency specified in 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 Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at the frequency specified in the Surveillance Frequency Control Program thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

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 the Digital Rod Position Indication System provided: (1) k_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM downstream of the hydrogen recombiners shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM downstream of the hydrogen recombiners greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM downstream of the hydrogen recombiners greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM shall be determined to be within the above limits by monitoring, at the frequency specified in the Surveillance Frequency Control Program, the waste gases in the GASEOUS RADWASTE TREATMENT SYSTEM.

PROCEDURES AND PROGRAMS (Continued)

p. Surveillance Frequency Control Program

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154 TO RENEWED FACILITY

OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By application dated August 18, 2015 (Reference 1), as supplemented by letters dated September 29, 2015 (Reference 2), February 5, 2016 (Reference 3), April 28, 2016 (Reference 4) and May 19, 2016 (Reference 5), Duke Energy Progress, LLC (Duke Energy) (previously Duke Energy Progress, Inc.) requested changes to the Technical Specifications (TSs) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP), which are contained in Appendix A of Renewed Facility Operating License NPF-63.

The requested change is the adoption of U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF) traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk Informed TSTF] Initiative 5b" (Reference 6) (hereafter "TSTF-425"). When implemented, TSTF-425 relocates most periodic frequencies of TS surveillances to a licensee-controlled program, the Surveillance Frequency Control Program (SFCP), and provides requirements for the new program in the Administrative Controls section of the TSs. All surveillance frequencies can be relocated except:

- Frequencies that reference other approved programs for the specific interval (such as the In-Service Testing Program or the Primary Containment Leakage Rate Testing Program);
- Frequencies that are purely event-driven (e.g., "each time the control rod is withdrawn to the 'full out' position");
- Frequencies that are event-driven, but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching ≥ 95% RTP [Reactor Thermal Power]"); and

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

A new program is added to the Administrative Controls of TS Section 6 as Specification 6.8.4.p. The new program is called the SFCP and describes the requirements for the program to control changes to the relocated surveillance frequencies. The TS Bases for each of the affected surveillance requirements are revised to state that the frequency is set in accordance with the SFCP. The licensee's proposed changes to the Administrative Controls of the TSs to incorporate the SFCP include a specific reference to Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1 (Reference 7) (hereafter "NEI 04-10") as the basis for making any changes to the surveillance frequencies once they are relocated out of the TSs.

In a letter dated September 19, 2007 (Reference 8), the NRC staff approved NEI 04-10 as acceptable for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04-10, and the safety evaluation providing the basis for NRC acceptance of NEI 04-10.

The licensee's supplements dated February 5 and April 28, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration (NSHC) determination[1] as published in the *Federal Register* (FR) on December 8, 2015 (80 FR 76319). By letter dated May 19, 2016, the licensee supplemented its amendment request with a proposed change that expanded the scope of the request. Therefore, the NRC published a second proposed NSHC determination in the FR on July 15, 2016 (81 FR 46118), which superseded the notice dated December 8, 2015 (80 FR 76319).

2.0 REGULATORY EVALUATION

2.1 Applicable Commission Policy Statements

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

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36][2] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the

[1] The original proposed NSHC determination referenced the August 18, 2015, application and the September 29, 2015, supplemental letter.

[2] This clarification is not part of the original policy statement.

> 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][3] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety."...
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

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

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

[3] The FR Notice 58 FR 39135 (Alteration in Original) explains the brackets.

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.

2.2 Applicable Regulations

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

Section 50.36(c)(3) of 10 CFR states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The FR notice published on July 6, 2009 (74 FR 31996), which announced the availability of TSTF-425, states that the addition of the SFCP to the TSs

provides the necessary administrative controls to require that surveillances 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 LCOs will be met. The FR notice also states that changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and are required to be documented.

Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. Such failures can result in the licensee increasing the frequency at which a surveillance test is performed. In addition, the SFCP implementation guidance in NEI 04-10 provides for monitoring the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs.

2.3 Applicable NRC Regulatory Guides

Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 9), 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 10), 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 11), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425 for HNP provides for administrative relocation of applicable surveillance frequencies, and provides for the addition of the SFCP to the Administrative Controls of TSs. TSTF-425 also requires the application of NEI 04-10 for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes proposed in TSTF-425 included documentation regarding the PRA technical adequacy consistent with the guidance of RG 1.200. In accordance with NEI 04-10, PRA methods are used, in combination with plant performance data and other considerations, to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This

is in accordance with guidance provided in RG 1.174 and RG 1.177 in support of changes to surveillance test intervals.

In Reference 1, the licensee provided revised TS Bases pages to be implemented with the associated TS changes. These pages were provided for information only and will be revised in accordance with the HNP TS Bases Control Program.

3.1 RG 1.177 Five Key Safety Principles

RG 1.177 identifies five key safety principles required for risk-informed changes to TSs. Each of these principles is addressed by NEI 04-10.

3.1.1 The Proposed Change Meets Current Regulations

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

This change is consistent with other NRC-approved TS changes in which the surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the In-service Testing Program or the Primary Containment Leakage Rate Testing Program. Thus, this proposed change meets the current regulations in 10 CFR 50.36.

Further the NEI 04-10 guidance provides for monitoring the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs. Thus, this proposed change meets the current regulations for monitoring surveillance test failures and implementing corrective actions to address such failures, as cited in 10 CFR 50.65 and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action."

Thus, this proposed change meets the first key safety principle of RG 1.177 by complying with the current regulations.

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

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

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

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

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

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

- Independence of barriers is not degraded.

- Defenses against human errors are preserved.

- The intent of the plant's design criteria is maintained.

TSTF-425 requires the application of NEI 04-10 for any changes to surveillance frequencies within the SFCP. NEI 04-10 uses both the core damage frequency (CDF) and the large early release frequency (LERF) metrics to evaluate the impact of proposed changes to surveillance frequencies. The guidance of RG 1.174 and RG 1.177 for changes to CDF and LERF is achieved by evaluation using a comprehensive risk analysis, which assesses the impact of proposed changes including contributions from human errors and common cause failures. Defense-in-depth is also included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to an increased likelihood of common cause failures. Both the quantitative risk analysis and the qualitative considerations assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177.

3.1.3 The Proposed Change Maintains Sufficient Safety Margins

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

The design, operation, testing methods, and acceptance criteria for SSCs specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the Updated Final Safety Analysis Report and Bases to TSs), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis.

Thus, safety margins are maintained by the proposed methodology, and the third key safety principle of RG 1.177 is satisfied.

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

RG 1.177 provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies. This provides for the identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. TSTF-425 provides for the application of NEI 04-10 in the SFCP. NEI 04-10 satisfies the intent of RG 1.177 guidance for evaluating the change in risk, and for assuring that such changes are small.

3.1.4.1 Quality of the PRA

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

RG 1.200 provides regulatory guidance for assessing the technical adequacy of a PRA. Revision 2 of this RG endorses (with comments and qualifications) the use of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 12), NEI 00-02, "PRA Peer Review Process Guidelines" (Reference 13) and NEI 05-04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard" (Reference 14). Revision 1 of this RG (Reference 15) had endorsed the internal events PRA standard ASME RA-Sb-2005, "Addenda to ASME RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 16).

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

<u>Internal events PRA</u>

The licensee submitted in the license amendment request (LAR), the history of peer reviews and gap assessments for the internal events PRA. In 2002, a peer review was performed by Westinghouse Owners Group in accordance with the guidance in NEI 00-02. The licensee stated that all findings and observations (F&Os) from this peer review have been resolved. In

2006 the licensee performed a self-assessment against the ASME RA-Sb-2005 version of the PRA standard, as endorsed by RG 1.200, Revision 1. In 2007, a focused scope industry peer review was conducted as a followup to the self-assessment against ASME PRA Standard RA-Sb-2005 and RG 1.200, Revision 1. Further, in its supplement dated September 29, 2015 (Reference 2), the licensee performed a gap assessment against the requirements of ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2 and concluded that all the supporting requirements (SRs) are met at Capability Category II. In the same supplement, the licensee submitted a summary of the changes to the internal events PRA after the 2007 peer review and stated that there were no changes that qualify as PRA upgrades that would require additional peer reviews.

In Table 1 of its supplement dated September 29, 2015 (Reference 2), the licensee submitted the internal events F&Os from the 2007 focused-scope peer review and their disposition. In addition, in Table 10 of the same supplement, the licensee also submitted a summary of closed F&Os from peer reviews prior to 2007. In response to Request for Additional Information (RAI) 1 (Reference 3) the licensee clarified that there were no new or additional F&Os identified as a result of the 2015 gap assessment.

The NRC staff reviewed (1) the summary of the peer review findings, (2) the licensee's resolution to the findings, and (3) the licensee's assessment of the impact on this application, for the internal events F&Os listed in Tables 1 and 10 of the supplement dated September 29, 2015 (Reference 2). The NRC staff assessed these peer review F&Os to ensure that any deficiencies in meeting Capability Category II can be addressed for the SFCP per the NEI 04-10 methodology. The NRC staff's assessment for the licensee's disposition of some of the F&Os is provided below. The NRC staff's review of licensee's resolution to the other F&Os found that they were adequately resolved by the licensee for the application.

F&O DA-C1-01 related to SR DA-C1 was submitted because a value of 0.33 was applied to generic data sources with zero failures, which is not consistent with typically accepted statistical approaches. In resolution to this F&O the licensee stated that it changed the values to 0.5 as part of a plant-specific data update that was incorporated into the PRA working model and that will be used for analyses to be performed under the SFCP. Because the licensee updated the generic parameter data consistent with the current PRA practices the NRC staff finds the licensee adequately dispositioned this F&O for the application.

F&O DA-C8-01 related to SR DA-C8 was submitted because the peer review team found that the licensee performed no estimate of the time that motor-operated valves (MOVs) were configured in standby, as required by the SR. The licensee stated that only 10 infrequently tested MOVs use standby failure rates and for these, the time not in standby is extremely low (fraction less than 0.01%) and, therefore, this F&O has minimal impact on their calculated standby failure rate. The licensee further stated that it also reviewed spurious operation data for MOVs and removed from the data calculation four MOVs that have power removed during plant operations. Because the licensee performed an estimate of the time in standby for the MOVs and updated the standby failure rate data for MOVs, the NRC staff finds the licensee resolution to this F&O acceptable for the application.

F&O DA-C10-01 related to SR DA-C10 was created because the peer review team could not find evidence that partial tests of components are distinguished from complete tests in the surveillance data. Per SR DA-C10, partial tests should not be counted as valid tests if all the elements of the modeled failure mode are not tested. In resolution to this F&O the licensee performed a review of components using plant-specific data and did not identify any partial tests. Further, in response to RAI 3 (Reference 3), the licensee confirmed that after resolution of F&O DA-D6-01, which resulted into modified boundaries for a number of components, all components within the new boundaries are tested with a full surveillance test. Because the licensee performed a review of components using plant-specific data and did not identify any partial tests, the NRC staff finds the licensee's resolution to this F&O acceptable for the application.

F&O DA-D6-03 related to SR DA-D6 found that the generic Common Cause Failure (CCF) Multiple Greek Letters (MGL) parameters used in the HNP's PRA were not assessed to whether they reflect the plant-specific testing practices (i.e., staggered vs. nonstaggered testing). In resolution to this F&O and as clarified in response to RAI 2 (Reference 3) the licensee clarified that it performed a review of all CCF basic events and updated the CCF parameters to reflect the appropriate testing scheme. Because the licensee updated the models to appropriately reflect plant-specific testing practices, the NRC staff finds the licensee's resolution to this F&O acceptable for the application.

F&O HR-D3-01 was submitted because the SR requires each detailed evaluation of a pre-initiator Human Failure Event (HFE) to include assessment of the quality of the written procedures and the quality of the man-machine interface. The peer review team found no justification for the assumption that plant procedures are accurate and consistent with the plant configuration. In resolution to this F&O the licensee updated the PRA documentation with justification on the quality of plant procedures and provided a summary in resolution to the F&O. Because the licensee evaluated the quality of plant procedures as required by the SR, the NRC staff finds the licensee's resolution to this F&O acceptable for the application.

F&O HR-F2-01 was submitted because the peer review team could not find any evidence that sequence specific timing estimates for HFEs were used, as required by the SR. Regarding the cited example of the Feed and Bleed (F&B) timing, the peer review team was concerned that the timing estimate used in the PRA, which is based on transients, may not be limiting for other events. To resolve this F&O the licensee stated that it performed MAAP [Modular Accident Analysis Program] runs for small loss-of-coolant accident and steam generator tube rupture accidents and confirmed that the timing estimate based on transients is limiting. Further, in response to followup RAI 1.a (Reference 4) the licensee stated that the analyses on the timing of initiation of F&B included in the PRA model are consistent with the plant operating procedures. The licensee stated that operator interviews and walk-throughs of the procedures were conducted with an operator to validate the F&B timing. Also, in response to RAI 4 (Reference 3) the licensee provided other examples of operator actions credited in the PRA model where thermal hydraulic analyses are used for deriving sequence specific timing estimates for various Human Error Probabilities (HEPs), consistent with the requirements of the PRA standard. Because the licensee provided sufficient evidence that it evaluated sequence specific timing for HFEs as required by the SR, and that the timing is consistent with plant

operating procedures, the NRC staff finds the licensee's resolution to this F&O acceptable for the application.

F&Os HR D6-01 and HR-G9-01 were submitted because the licensee used median values instead of means for the HEP in their PRA model. In resolution to these F&Os, the licensee stated that it updated the PRA model to use the mean values for pre-initiator and post-initiator HEPs. Therefore the NRC staff finds the licensee's resolution to this F&O acceptable for the application.

Historical F&O 57-HR-H2 provided in Table 10 of supplement dated September 29, 2015 (Reference 2) included operator recovery actions credited in the internal events PRA and discussed a number of recovery actions. In its review of this F&O resolution the NRC staff identified three operator recovery actions that appeared to lack appropriate justification for their inclusion in the PRA model and asked the licensee for additional information. In response to RAI 5 (Reference 3) and further clarified in the response to followup RAIs 1.b, 1.c and 1.d (Reference 4) the licensee provided the necessary justification of why these operator recovery actions are credible and consistent with the operating procedures, as discussed in the assessment of operator recovery actions OPER-70, OPER-14, and OPER-42 below.

OPER-70 models operator action to align the spare Charging Safety Injection Pump (CSIP) when the running CSIP becomes unavailable. The F&O found no basis for assuming 12 hours as the time available for completing the action. The resolution of the F&O stated that the plant has been modified and the original F&O comment was no longer relevant. In response to followup RAI 1.b (Reference 4) the licensee clarified that a plant modification was performed in 2005 to install a local manual transfer switch for alternating current power, which significantly reduced the time required to align and start the spare pump. The licensee provided an overview of the applicable operating procedures and the current timing for this action. The licensee further stated that it recently re-confirmed the procedural guidance and timing for this action with senior reactor operators. The NRC staff finds that the licensee provided reasonable justification for crediting this operator action, consistent with the requirements of the PRA standard.

OPER-Q14 is a post-initiator combination of operator actions that assumes complete loss of both the Emergency Service Water (ESW) and the Normal Service Water (NSW) systems. The F&O challenged the combination of operator actions, which appeared to credit the CSIPs and the Residual Heat Removal (RHR) system without service water cooling. In response to followup RAI 1.c (Reference 4) the licensee clarified that the CSIPs would fail on loss of service water, but Reactor Coolant Pump seal failure is prevented by the Alternate Seal Injection system, which was added through a plant modification and is completely independent and automatically initiated. The licensee further clarified that the RHR system can continue to operate until the Component Cooling Water is too hot to perform its function, or until either the ESW or NSW are restored. Further, the licensee stated that these models are consistent with the operating procedures and reflect the as-built, as-operated plant. The NRC staff finds that the licensee provided reasonable justification for crediting this operator action, consistent with the requirements of the PRA standard.

OPER-42 models operator action to align the CSIP suction for safety injection, when the Refueling Water Storage Tank isolation valves fail to open upon a safety injection signal. The

F&O identified this action as a nonproceduralized action. In response to followup RAI 1.d (Reference 4), the licensee stated that this action is driven by procedures based on the expected annunciators for the scenarios and operator training. The licensee provided an overview of the operating procedures that direct the operator to perform this action. The licensee further stated that a short time is available to perform this action and, therefore, it has a high failure probability of 0.3. Further, in response to followup RAI 1.f (Reference 4) the licensee stated that there are no nonproceduralized operator actions credited in the PRA model. The NRC staff finds that the licensee provided reasonable justification for crediting this operator action, consistent with the requirements of the PRA standard.

Internal flooding PRA

The licensee stated that the internal flooding PRA was updated in 2014 to meet the supporting requirements of the internal flooding PRA portion of the PRA standard, as endorsed by RG 1.200, Revision 2. A focused-scope peer review of the internal flooding PRA against the requirements of the ASME/ANS PRA standard RA-Sa-2009 and RG 1.200, Revision 2 was conducted in 2014. In Table 1 of Enclosure 2 to the LAR (Reference 1) the licensee submitted the F&Os from the 2014 internal flooding peer review. The NRC staff reviewed (1) the summary of the peer review findings, (2) the licensee's resolution to the findings, and (3) the licensee's assessment of the impact on this application, for the internal flooding F&Os, to identify whether any gaps in the PRA model were identified that could impact the application. The NRC staff assessed these peer review F&Os to ensure any deficiencies in meeting Capability Category II can be addressed for the SFCP per the NEI 04-10 methodology. The NRC staff concluded that the F&O dispositions were adequate for this application, but needed additional clarifications for the resolutions to two F&Os, discussed below.

F&O 1-3 was submitted because the licensee did not consider potential effects of splash, dripping or flow along cable trays or other structures as required by the PRA standard. The peer review also found that the licensee considered the effects of spray, but had made some assumptions not supported by engineering evaluations with regards to SSCs susceptible to spray effects. In resolution to this F&O the licensee stated that it reassessed the effects of sprays and other flooding mechanism and provided a summary of this assessment. The licensee stated that all susceptible equipment in small flood compartments and equipment within a zone of influence of the spray location is assumed to fail as a result of a spray event. In RAI 6, the staff requested the licensee clarify whether the MOVs, Air Operated valves and pressure, level and flow transmitters in small flood compartments are assumed failed, since the resolution to F&Os 1-5 and 1-14 appeared to indicate the opposite. In response to RAI 6 (Reference 3), the licensee clarified that these components were tested or assessed to not be susceptible to sprays, as detailed in resolution to F&Os 1-5 and 1-14, and therefore were not failed in a flood compartment for spray scenarios. Because the licensee adequately assessed the effects of sprays and other flooding mechanism as required by the PRA standard, the NRC staff finds the licensee's resolution of this F&O acceptable for the application.

F&O 2-2 was submitted because the peer review team found that the licensee did not consider inter-area flood propagation via backflow through the drain system in a number of places, due to the existence of traps in the drain piping. The peer review team found no documented basis for the ability of those traps to prevent backflow into rooms in which a flooding event did not

originate. In resolution to this F&O the licensee performed a sensitivity analysis on drain backflow and found no incremental impact to the inter-area flood propagation. In response to RAI 7 (Reference 3) the licensee explained why the sensitivity study showed no impact. Small floods have negligible impact on any core damage sequence because the rooms at HNP are large, with capacity to receive a large volume of water. For large floods, the drain backflow is overwhelmed by normal propagation of water from the break. Because the licensee adequately evaluated the potential for drain backflow as required by the PRA standard, the NRC staff finds the licensee's resolution of this F&O acceptable for the application.

Fire PRA

The licensee stated that the fire PRA was developed in support of the National Fire Protection Association (NFPA) Standard 805 fire protection program. In 2008 the fire PRA was subject to an NRC staff review and a followup partial-scope industry peer review. In response to RAI 8 (Reference 3) the licensee specified that the industry peer review was performed against the requirements of the American National Standards Institute (ANSI)/ANS standard 58.23-2007 (which precedes Part 4, Requirements for fires at-power PRA, of the currently endorsed PRA standard ASME/ANS RA-Sa–2009). In its supplement dated September 29, 2015 (Reference 2), the licensee provided a summary of the PRA revisions since 2007 and stated that an update of the fire PRA was performed in 2013, which did not require a peer review. Furthermore, in response to RAI 8 (Reference 3) the licensee performed a gap assessment of the fire PRA against the SRs of the current PRA standard, ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2.

The NRC staff reviewed the results of the fire PRA gap assessment and the fire PRA F&Os and their dispositions provided in response to RAI 8 (Enclosures 3 and 4 of Reference 3) to determine whether any gaps in the PRA model were identified that could impact the application and to ensure that any gaps in meeting Capability Category II can be addressed for the SFCP. The staff's assessment of the SRs that were not met at Capability Category II and the related F&Os is provided below. The staff found the licensee can address and disposition these gaps for each surveillance frequency evaluation per the NRC-approved NEI 04-10 methodology, including performing appropriate sensitivity analyses and reviews of the PRA model results.

F&Os FSS-F3-01 and FSS-F-1 were submitted because the peer review team found that the impact of fires on exposed structural steel was not addressed, and that SR FSS-F2 was met at Capability Category I and that SR FSS-F3 was not met. As discussed in the safety evaluation for transition to NFPA-805 (Reference 17, Table 3.4-2 of Attachment C2, "Fire PRA F&O Resolution"), the licensee identified and assessed qualitatively that the only structure containing significant exposure of structural steel is the turbine building. The licensee's resolution to these F&Os was previously reviewed by the NRC staff and found acceptable for transition to NFPA-805 (Reference 17). The licensee provided additional information on the resolution to FSS-F3-01 in Reference 3, which included a qualitative discussion of the F&O issue and indicated that no fire scenarios were developed based on insurance-related analyses. While the NRC staff has not endorsed the insurance-related analyses, the licensee has satisfactorily addressed these F&Os for their transition to NFPA-805, which the NRC staff also finds to be adequate for the TSTF-425 program. In addition, NEI 04-10 provides guidance on the treatment

of model incompleteness, and the NRC staff finds that the resolution of this F&O can be treated using this guidance.

F&O IGN-A4-1 related to SR IGN-A4 was submitted because the licensee did not consider plant-specific experience for fire events as required by the SR. In resolution to this F&O the licensee states that a review of plant-specific fire events has been performed and concluded that the events experienced at HNP were typical of what would be seen throughout the industry. Based on the licensee's evaluation of plant-specific experience for fire events and the determination that no outlier experience existed, the NRC staff finds this F&O has been appropriately resolved.

F&O FSS-E3 identified that no uncertainty analysis was performed and F&Os FQ-A4-02 and FQ-D1-01 identified that correlated uncertainty estimates for CDF and LERF were not performed. Because of this, the licensee also identified that SR FSS-H5 and FSS-H6 were met only at Capability Category I. In resolution to F&O FSS-E3 the licensee stated that uncertainty parameters and analysis have been documented. However in resolution to F&Os FQ-A4-02 and FQ-D1-01 the licensee stated that correlated numerical uncertainty for CDF or LERF have not been performed. The licensee stated that these parametric uncertainties are not expected to be significant compared to the fire modeling factors (such as fire growth, heat release rates, severity factors, nonsuppression probability factors, and associated fire brigade response time), which are identified in the HNP uncertainty. This is acceptable because the methodology in NEI 04-10 requires assessing the impact of uncertainties for the SFCP through sensitivity studies as discussed in Section 3.1.4.5 of this safety evaluation.

Additionally, the licensee identified that SR FSS-D7 and FSS-D9 were met at Capability Category I, but stated that there were no F&Os related to these SRs. These are discussed below.

SR FSS-D7 at Capability Category II requires that, when crediting fire detection and suppression systems, generic estimates of total system unavailability should be used when these systems have not experienced outlier behavior with regards to unavailability. As discussed in the safety evaluation for transition to NFPA-805 (Reference 17, Table 3.4-2 of Attachment C2, "Fire PRA F&O Resolution"), the treatment of detection system unavailability is expected to have a minor impact based on the conservative treatment of unavailability in the generic data when compared to the actual system performance, as well as ongoing monitoring of unavailability. Therefore, the staff finds that Capability Category I for this SR is expected to have minor impact for the application, and evaluation of uncertainty due to not including outliers can be performed when NEI 04-10 guidance is followed, and is acceptable for the application.

SR FSS-D9 at Capability Category II requires an evaluation of the potential for smoke damage to the fire PRA equipment. As discussed in the safety evaluation for transition to NFPA-805 (Reference 17, Table 3.4-2 of Attachment C2, "Fire PRA F&O Resolution"), the treatment of smoke effects is expected to have minor impact based on the plant-specific design for smoke mitigation, as well as administrative controls for management of smoke. Therefore, the staff finds that Capability Category I is expected to have minor impact for the application and finds it acceptable for the application.

In addition to the above gaps, the NRC staff found documentation missing related to F&O FSS-B2-01. In followup RAI 2 (Reference 3) the staff requested the licensee to ensure this documentation is available to support decision making for the SFCP. In response to the RAI (Reference 3) the licensee provided a summary of the documentation available to assess for use in the SFCP. Therefore the NRC staff finds the licensee's resolution of this F&O acceptable for the application.

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

3.1.4.2 Scope of the PRA

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

HNP has full-power internal events, internal floods, as well as internal fire PRA models. These models received peer reviews, self-assessments, and focused scope peer reviews as discussed above in Section 3.1.4.1 of this safety evaluation. In accordance with NEI 04-10, the licensee will use these models to perform quantitative evaluations to support the development of changes to surveillance frequencies in the SFCP. This is acceptable because the NRC-approved methodology in NEI 04-10 allows for more refined analyses to be performed to support changes to surveillance frequencies in the SFCP.

HNP does not have PRA models for seismic events, external flooding, high winds, and transportation and nearby facility accidents. For these external hazards, the licensee stated in the LAR that it will use the hazard screening performed for the Individual Plant Examination of External Events (IPEEE). The IPEEE seismic evaluation used the EPRI Seismic Margin Analysis. The IPEEE external flooding evaluation utilized a screening approach. High winds and potential accidents associated with nearby air traffic, runways, roads, railways, and fixed facilities are not considered a significant hazard, based on the IPEEE studies. In response to RAI 9 (Reference 3), the licensee confirmed that qualitative bounding analyses for these external events will reflect the current plant configuration, operating, risk insights and operating experience, including the insights from the external flooding and seismic assessments performed in 2013 and 2014 in response to the Near-Term Task Force review of insights from the Fukushima Dai-ichi accident. In accordance with NEI 04-10, the licensee will perform an initial qualitative screening analysis, and, if the qualitative information is not sufficient, a bounding analysis will be performed. The bounding analysis will be performed in accordance with NEI 04-10, Step 10b. This is an acceptable approach in accordance with NEI 04-10.

The licensee stated that for assessing the shutdown risk, the shutdown risk management program for implementation of NUMARC 91-06, "Guidelines for Industry Actions to Address Shutdown Management," dated December 1991 (Reference 19), will be used for the proposed changes to surveillance frequencies under the SFCP. This is an acceptable approach in accordance with NEI 04-10.

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

3.1.4.3 PRA Modeling

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

Thus, through the application of NEI 04-10, the HNP 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 of RG 1.177.

3.1.4.4 Assumptions for Time Related Failure Contributions

The failure probabilities of SSCs modeled in PRAs may include a standby time-related contribution and a cyclic demand-related contribution. NEI 04-10 criteria adjust the time-related failure contribution of SSCs affected by the proposed change to a surveillance frequency. This is consistent with RG 1.177, Section 2.3.3, which permits separation of the failure rate contributions into demand and standby for evaluation of surveillance requirements. Further, consistent with the guidance, the licensee states in Enclosure 2 to the LAR that if the breakdown between the standby time-dependent failure rate and the demand-related failure rate probability for affected SSCs is unknown, then the total failure probability will be assumed to be time-related to obtain the maximum test-limited risk condition. The SSC failure rate (per unit time) is assumed to be unaffected by the change in test frequency, such that the failure probability is assumed to increase linearly with time, and will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The process requires consideration of qualitative sources of information with regards to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus, the process is not reliant upon risk analyses as the sole basis for the proposed changes.

The potential benefits of 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 not quantitatively assessed. NEI 04-10 provides for performance monitoring of SSCs whose surveillance frequencies have been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback includes consideration of Maintenance Rule monitoring of equipment performance. In the event of SSC performance degradation, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions that may be required by the Maintenance Rule.

Thus, through the application of NEI 04-10, the licensee has employed reasonable assumptions with regard to extensions of surveillance test intervals, and is consistent with Regulatory Position 2.3.4 of RG 1.177.

3.1.4.5 Sensitivity and Uncertainty Analyses

NEI 04-10 provides for sensitivity studies to assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact on the frequency of initiating events, and any identified deviations from Capability Category II of the PRA standard. Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered, along with any qualitative considerations that may bear on the results of such sensitivity studies. Required monitoring and feedback of SSC performance once the revised surveillance frequencies are implemented will also be performed. Thus, through the application of NEI 04-10, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations, and is consistent with Regulatory Position 2.3.5 of RG 1.177.

3.1.4.6 Acceptance Guidelines

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

change to CDF, and less than 1E-6 per year for change to LERF. The total CDF and total LERF must be reasonably shown to be less than 1E-4 per year and 1E-5 per year, respectively. These are consistent with the acceptance criteria of RG 1.174, as referenced by RG 1.177 for changes to surveillance frequencies. The NRC staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with insignificant risk increases (less than 5E-8 CDF and 5E-9 LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

The quantitative acceptance guidance of RG 1.174 is supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history.

The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results. Post-implementation performance monitoring and feedback are also required to assure continued reliability of the SSCs. The licensee's application of NEI 04-10 provides acceptable methods for evaluating the risk increase associated with proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177. Therefore, the proposed methodology satisfies the fourth key safety principle of RG 1.177 by assuring any increase in risk is small consistent with the intent of the Commission's Safety Goal Policy Statement.

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

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

3.2 Addition of Surveillance Frequency Control Program to Administrative Controls

The licensee proposed including the SFCP and specific requirements into the HNP TSs, Section 6, as follows:

> Surveillance Frequency Control Program
>
> This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications

are performed at intervals sufficient to assure that the associated Limiting Conditions for Operation are met.

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

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

This is proposed as new page 6-19j to be added to TS 6.8.4. In the application dated August 18, 2015 (Reference 1) a lettered list was used as is reflected above. Alternately a numbered list may be used. Both as a lettered or numbered list, the proposed program is consistent with the model application of TSTF-425 and, therefore, the NRC staff concludes that it is acceptable.

3.3 Deviations from TSTF-425 and Other Changes

The LAR is consistent with TSTF-425 except for variations that were identified by the licensee. These variations are described and evaluated below.

Clean TS pages were not included due to the large overall number of TS pages affected and that providing only mark-ups satisfies the requirements of 10 CFR 50.90. Since the LAR still met the requirements and this administrative difference had no impact on the technical content of the amendment, this variation is acceptable.

The definition of STAGGERED TEST BASIS is being retained in the HNP TSs due to its continued use in Administrative TS Section 6.8.4.o, "Control Room Envelope Habitability Program." Since this defined term is still used elsewhere in the TSs it is required to remain in the TSs according to the NUREG-1431 guidance; therefore, this deviation is acceptable.

The textual inserts for the relocated surveillance frequencies were changed from "in accordance with the surveillance frequency control program" to "at the frequency specified in the Surveillance Frequency Control Program." Additionally other textual changes were made to fit the HNP plant-specific implementation of NRC approved TSTF-425. The licensee states that these changes were made to the existing wording to fit the HNP TS format. The NRC staff reviewed these changes and determined that they are consistent with NRC approved TSTF-425 and the Commission's final policy statement on TSs as published in the *Federal Register* on July 22, 1993 (55 FR 39132). The NRC staff agrees that the textual changes fit the HNP current TS format and, therefore, the TS surveillance requirements will continue to meet 10 CFR 50.36(c)(3).

There are surveillances included in TSTF-425 that are not included in the HNP TSs. Additionally HNP has surveillance requirements that are not included in the NUREG-1431 guidance. TSTF-425 transfers control of frequencies for existing surveillances to the SFCP, but it does not add, delete, or modify the content of the surveillance actions themselves. Based on this, the amendment, which represents a plant-specific adoption of TSTF-425, relocates only existing fixed periodic surveillance frequencies for existing surveillances in the HNP current TSs. This is consistent with the Commission's final policy statement on TSs as published in the *Federal Register* on July 22, 1993 (55 FR 39132) and will ensure that the HNP surveillance requirements will continue to meet 10 CFR 50.36(c)(3). Therefore the NRC staff finds these changes acceptable.

3.4 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of specific surveillance frequencies to a licensee-controlled document, and controlling changes to these surveillance frequencies in accordance with a new program, the SFCP, identified in the Administrative Controls of TSs. The NRC staff confirmed that this amendment does not relocate surveillance frequencies that reference other approved programs for the specific interval, are purely event-driven, are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs, or are related to specific conditions. The SFCP and TS Section 6.0 references NEI 04-10, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TSs to a licensee-controlled document, provided those frequencies are changed in accordance with the NEI 04-10, which is specified in the Administrative Controls section of the TSs.

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

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

Paragraph 50.36(c) of 10 CFR discusses the categories that will be included in TSs. Paragraph 50.36(c)(3) of 10 CFR discusses the specific category of SRs and states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure

that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

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

Based on the above evaluation, the NRC staff concludes that, with the proposed relocation of surveillance frequencies to a licensee-controlled document and administratively controlled in accordance with the TS SFCP, the licensee continues to meet the requirements in 10 CFR 50.36(c)(3), 10 CFR 50.65, and 10 CFR 50, Appendix B, Criterion XVI.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and an inspection or surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (81 FR 46118). Accordingly, the amendment meets 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 amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Waldrep, B. C., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Application for Technical Specification Change Regarding Risk-informed

Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," August 18, 2015 (ADAMS Accession No. ML15236A265).

2. Waldrep, B. C., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplement to Harris Nuclear Plant Application for Technical Specification Change Regarding Risk-informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," September 29, 2015 (ADAMS Accession No. ML15272A443).

3. Waldrep, B. C., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Harris Nuclear Plant License Amendment Request to Revise Technical Specifications by Relocating Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TAC MF6583)," February 5, 2016 (ADAMS Accession No. ML16036A091).

4. Waldrep, B. C., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Response to Second Request for Additional Information Regarding Harris Nuclear Plant License Amendment Request to Revise Technical Specifications by Relocating Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TAC MF6583)," April 28, 2016 (ADAMS Accession No. ML16119A326).

5. Waldrep, B. C., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplement to the Harris Nuclear Plant License Amendment Request to Revise Technical Specifications by Relocating Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TAC MF6583)," May 19, 2016 (ADAMS Accession No. ML16141A048).

6. Technical Specifications Task Force, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control—RITSTF Initiative 5b," March 18, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090850642).

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

8. Nieh, H. K., U.S. Nuclear Regulatory Commission, letter to Bill Bradley, Nuclear Energy Institute, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 04-10, Revision 1, 'Risk-Informed Technical Specifications Initiative 5b Risk-Informed Method for Control of Surveillance Frequencies' (TAC No. MD6111)," September 19, 2007 (ADAMS Accession No. ML072570267).

9. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 (ADAMS Accession No. ML100910006).

10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 1, May 2011 (ADAMS Accession No. ML100910008).

11. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 (ADAMS Accession No. ML090410014).

12. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

13. Nuclear Energy Institute, NEI 00-02, Revision 1 "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Revision 1, May 2006 (ADAMS Accession Nos. ML061510621 and ML061510619).

14. Nuclear Energy Institute, NEI 05-04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard," Revision 2, November 2008 (ADAMS Accession No. ML083430450).

15. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007 (ADAMS Accession No. ML070240001).

16. ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum B to ASME RA-S-2002, ASME, New York, NY, December 30, 2005.

17. Vaaler, M., U.S. Nuclear Regulatory Commission, letter to Christopher L. Burton, Carolina Power & Light Company, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Regarding Adoption of National Fire Protection Association Standard 805, 'Performance-based Standard for Fire Protection for Light Water Reactor Electric Generating Plants' (TAC No. MD8807)," June 28, 2010 (ADAMS Accession Nos. ML101750602 and ML101750604).

18. Nuclear Management and Resources Council, Inc., NUMARC 91-06, "Guidelines for Industry Actions to Address Shutdown Management," December 1991 (ADAMS Accession No. ML14365A203)

Principal Contributors: Mihaela Biro
Pete Snyder

Date: November 29, 2016

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

Sincerely,

/RA/

Martha Barillas, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 154 to NPF-63
2. Safety Evaluation

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