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

March 29, 2011

Mr. J. R. Morris
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING REVISION OF THE TECHNICAL SPECIFICATIONS TO RELOCATE SPECIFIC SURVEILLANCE FREQUENCIES TO A LICENSEE-CONTROLLED PROGRAM USING A RISK-INFORMED JUSTIFICATION (TSTF-425) (TAC NOS. ME3722 AND ME3723)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 263 to Renewed Facility Operating License NPF-35 and Amendment No. 259 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 31, 2010, as supplemented by letter dated November 30, 2010.

The amendments revise the Technical Specifications by relocating specific surveillance frequencies to a licensee-controlled document using a risk-informed justification.

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

If you have any questions, please call me at 301-415-1119.

Sincerely,

Jon Thompson

Jon Thompson, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 263 to NPF-35
2. Amendment No. 259 to NPF-52
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 263
Renewed License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-35 filed by the Duke Energy Carolinas, LLC, acting for itself, and North Carolina Electric Membership Corporation (licensees), dated March 31, 2010, as supplemented by letter dated November 30, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page 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-35 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 263 , which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC, shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-35
and the Technical Specifications

Date of Issuance: March 29, 2011

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

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 259
Renewed License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Carolinas, LLC, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated March 31, 2010, as supplemented by letter dated November 30, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 259 , which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC, shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-52
and the Technical Specifications

Date of Issuance: March 29, 2011

ATTACHMENT TO

LICENSE AMENDMENT NO. 263

RENEWED FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND LICENSE AMENDMENT NO. 259

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
Licenses	Licenses
NPF-35, page 4	NPF-35, page 4
NPF-52, page 4	NPF-52, page 4
TSs	TSs
1.1-6	1.1-6
3.1.1-1	3.1.1-1
3.1.2-2	3.1.2-2
3.1.4-3	3.1.4-3
3.1.4-4	3.1.4-4
3.1.5-2	3.1.5-2
3.1.6-3	3.1.6-3
3.1.8-2	3.1.8-2
3.2.1-3	3.2.1-3
3.2.1-4	3.2.1-4
3.2.1-5	3.2.1-5
3.2.2-3	3.2.2-3
3.2.2-4	3.2.2-4
3.2.3-1	3.2.3-1
3.2.4-4	3.2.4-4
3.3.1-9	3.3.1-9
3.3.1.10	3.3.1.10

Remove Pages	Insert Pages
3.3.1-11	3.3.1-11
3.3.1-12	3.3.1-12
3.3.1-13	3.3.1-13
3.3.1-14	3.3.1-14
3.3.1-15	3.3.1-15
3.3.1-16	3.3.1-16
3.3.1-17	3.3.1-17
3.3.1-18	3.3.1-18
3.3.1-19	3.3.1-19
3.3.1-20	3.3.1-20
3.3.1-21	3.3.1-21
-	3.3.1-22
3.3.2-10	3.3.2-10
3.3.2-11	3.3.2-11
3.3.2-12	3.3.2-12
3.3.2-13	3.3.2-13
3.3.2-14	3.3.2-14
3.3.2-15	3.3.2-15
3.3.2-16	3.3.2-16
-	3.3.2-17
3.3.3-3	3.3.3-3
3.3.4-2	3.3.4-2
3.3.5-2	3.3.5-2
3.3.6-2	3.3.6-2
3.3.9-3	3.3.9-3
-	3.3.9-4
3.4.1-3	3.4.1-3
3.4.3-2	3.4.3-2
3.4.4-1	3.4.4-1
3.4.5-3	3.4.5-3
3.4.6-2	3.4.6-2
-	3.4.6-3
3.4.7-2	3.4.7-2
-	3.4.7-3
3.4.8-2	3.4.8-2
3.4.9-2	3.4.9-2
3.4.11-3	3.4.11-3

Remove Pages	Insert Pages
3.4.11-4	3.4.11-4
3.4.12-5	3.4.12-5
3.4.12-6	3.4.12-6
3.4.12-7	3.4.12-7
-	3.4.12-8
3.4.13-2	3.4.13-2
3.4.14-3	3.4.14-3
3.4.14-4	3.4.14-4
3.4.15-4	3.4.15-4
3.4.16-2	3.4.16-2
3.4.16-3	3.4.16-3
3.4.17-1	3.4.17-1
3.5.1-2	3.5.1-2
-	3.5.1-3
3.5.2-2	3.5.2-2
3.5.2-3	3.5.2-3
3.5.4-2	3.5.4-2
3.5.5-2	3.5.5-2
3.6.2-5	3.6.2-5
3.6.3-5	3.6.3-5
3.6.3-6	3.6.3-6
3.6.4-1	3.6.4-1
3.6.5-2	3.6.5-2
3.6.6-1	3.6.6-1
3.6.6-2	3.6.6-2
3.6.8-2	3.6.8-2
3.6.9-2	3.6.9-2
3.6.10-2	3.6.10-2
3.6.11-1	3.6.11-1
3.6.11-2	3.6.11-2
3.6.12-1	3.6.12-1
3.6.12-2	3.6.12-2
3.6.12-3	3.6.12-3
3.6.13-2	3.6.13-2
3.6.13-3	3.6.13-3
3.6.14-2	3.6.14-2
3.6.14-3	3.6.14-3

Remove Pages	Insert Pages
3.6.15-2	3.6.15-2
3.6.16-1	3.6.16-1
3.6.16-2	3.6.16-2
3.7.4-2	3.7.4-2
3.7.5-3	3.7.5-3
3.7.5-4	3.7.5-4
3.7.6-2	3.7.6-2
3.7.7-2	3.7.7-2
3.7.8-3	3.7.8-3
3.7.9-1	3.7.9-1
-	3.7.9-2
3.7.10-3	3.7.10-3
3.7.11-2	3.7.11-2
3.7.12-2	3.7.12-2
3.7.13-2	3.7.13-2
3.7.14-1	3.7.14-1
3.7.15-1	3.7.15-1
3.7.17-1	3.7.17-1
3.8.1-5	3.8.1-5
3.8.1-6	3.8.1-6
3.8.1-7	3.8.1-7
3.8.1-8	3.8.1-8
3.8.1-9	3.8.1-9
3.8.1-10	3.8.1-10
3.8.1-11	3.8.1-11
3.8.1-12	3.8.1-12
3.8.1-13	3.8.1-13
3.8.1-14	3.8.1-14
3.8.1-15	3.8.1-15
3.8.3-2	3.8.3-2
3.8.3-3	3.8.3-3
3.8.4-2	3.8.4-2
3.8.4-3	3.8.4-3
3.8.4-4	3.8.4-4
3.8.6-4	3.8.6-4
3.8.7-2	3.8.7-2
3.8.8-2	3.8.8-2

Remove Pages	Insert Pages
3.8.9-3	3.8.9-3
3.8.10-2	3.8.10-2
3.9.1-1	3.9.1-1
3.9.2-2	3.9.2-2
3.9.3-2	3.9.3-2
3.9.4-2	3.9.4-2
3.9.5-2	3.9.5-2
3.9.6-1	3.9.6-1
3.9.7-1	3.9.7-1
5.5-15	5.5-15
-	5.5-16

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 263 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

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

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

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

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplement wherein this renewed license condition is discussed.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 259 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than February 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

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

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Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

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

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

1.1 Definitions (continued)

THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limit specified in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	---NOTE--- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ------ Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limit specified in the COLR. OR D.1.2 Initiate boration to restore required SDM to within limit. AND D.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify individual rod positions within alignment limit.	In accordance with the Surveillance Frequency Control Program AND Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core $\geq$ 10 steps in either direction.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 2.2 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 551°F$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program AND Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1.	Prior to initiation of PHYSICS TESTS
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 541°F.	In accordance with the Surveillance Frequency Control Program
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.1.8.4	Verify SDM is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

---NOTE---

During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F^M_Q(X,Y,Z)$ is within steady state limit.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified <u>AND</u> In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------------------NOTE------------------------------ 1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}$, and $\dfrac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}} > \dfrac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{OP}}$ then: a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$; or b. Repeat SR 3.2.1.2 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$ is extrapolated to not be met. 2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions. --- Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified <u>AND</u> In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	-------------------------------NOTES------------------------------- 1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}$, and $\dfrac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}} > \dfrac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{RPS}}$ then: a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$; or b. Repeat SR 3.2.1.3 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$ is extrapolated to not be met. 2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions. --- Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified AND In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

--NOTE--

During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify $F^M_{\Delta H}(X,Y)$ is within steady state limit.	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^M_{\Delta H}(X,Y)$ was last verified AND In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 -----------------------------NOTES----------------------------- 1. Extrapolate $F^M_{\Delta H}(X,Y)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_{\Delta H}(X,Y)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_{\Delta H}(X,Y)_{EXTRAPOLATED} \geq F^L_{\Delta H}(X,Y)^{SURV}_{EXTRAPOLATED}$ and $\frac{F^M_{\Delta H}(X,Y)_{EXTRAPOLATED}}{F^L_{\Delta H}(X,Y)^{SURV}_{EXTRAPOLATED}} > \frac{F^M_{\Delta H}(X,Y)}{F^L_{\Delta H}(X,Y)^{SURV}}$ then: a. Increase $F^M_{\Delta H}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$; or b. Repeat SR 3.2.2.2 prior to the time at which $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$ is extrapolated to not be met. 2. Extrapolation of $F^M_{\Delta H}(X,Y)$ is not required for the initial flux map taken after reaching equilibrium conditions. -- Verify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_{\Delta H}(X,Y)$ was last verified <u>AND</u> In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

-----NOTE-----
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-----NOTES----- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER <75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. 3. This SR is not required to be performed until 12 hours after exceeding 50% RTP. ---------- Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable
SR 3.2.4.2	-----NOTES----- Only required to be performed if input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER $\geq$ 75% RTP. ---------- Verify QPTR is within limit using the movable incore detectors.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

--NOTE--
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2	------------------------------NOTES------------------------------ 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is $\geq$ 15% RTP. -- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.3	------------------------------NOTES------------------------------ 1. Adjust NIS channel if absolute difference is $\geq$ 3%. 2. Not required to be performed until 24 hours after THERMAL POWER is $\geq$ 15% RTP. -- Compare results of the incore detector measurements to NIS AFD.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.4	---NOTE--- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.6	---NOTE--- Not required to be performed until 24 hours after THERMAL POWER is $\geq$ 75% RTP. --- Calibrate excore channels to agree with incore detector measurements.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.7	---NOTE--- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. --- Perform COT.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	---------------------------------NOTE------------------------------ This Surveillance shall include verification that interlocks P-6 (for the Intermediate Range channels) and P-10 (for the Power Range channels) are in their required state for existing unit conditions. --- Perform COT.	---------NOTE------- Only required when not performed within the Frequency specified in the Surveillance Frequency Control Program or the previous 184 days ----------------------- Prior to reactor startup AND Four hours after reducing power below P-10 for power and intermediate range instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	-----NOTE----- Verification of setpoint is not required. -----	
	Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.10	-----NOTE----- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -----	
	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.11	-----NOTE----- 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. Power Range Neutron Flux high voltage detector saturation curve verification is not required to be performed prior to entry into MODE 1 or 2. 3. Intermediate Range Neutron Flux detector plateau voltage verification is not required to be performed prior to entry into MODE 1 or 2.* -----	
	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

(continued)

* This Note applies to the Westinghouse-supplied compensated ion chamber neutron detectors. The compensated ion chamber neutron detectors are being replaced with Thermo Scientific-supplied fission chamber neutron detectors which do not require detector plateau voltage verification. Therefore, this Note does not apply to the fission chamber neutron detectors.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.12	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.13	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.14	-------------------------------NOTE------------------------------ Verification of setpoint is not required. --- Perform TADOT.	 In accordance with the Surveillance Frequency Control Program
SR 3.3.1.15	-------------------------------NOTE------------------------------ Verification of setpoint is not required. --- Perform TADOT.	---------NOTE------- Only required when not performed within previous 31 days ---------------------- Prior to reactor startup
SR 3.3.1.16	-------------------------------NOTE------------------------------ Neutron detectors are excluded from response time testing. --- Verify RTS RESPONSE TIME is within limits.	 In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.1.17 Verify RTS RESPONSE TIME for RTDs is within limits.	In accordance with the Surveillance Frequency Control Program

Table 3.3.1-1 (page 1 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110.9% RTP	109% RTP
b. Low	$1^{(b)}$,2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 27.1% RTP	25% RTP
3. Power Range Neutron Flux						
High Positive Rate	1,2	4	D	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec

(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8(l)(m) SR 3.3.1.11(l)(m)	≤ 31% RTP* ≤ 38% RTP	25% RTP
	2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8(l)(m) SR 3.3.1.11(l)(m)	≤ 31% RTP* ≤ 38% RTP	25% RTP
5. Source Range Neutron Flux	2(d)	2	I,J	SR 3.3.1.1 SR 3.3.1.8(l)(m) SR 3.3.1.11(l)(m)	≤ 1.4 E5 cps** ≤ 1.44 E5 cps	1.0 E5 cps
	3(a), 4(a), 5(a)	2	J,K	SR 3.3.1.1 SR 3.3.1.7(l)(m) SR 3.3.1.11(l)(m)	≤ 1.4 E5 cps** ≤ 1.44 E5 cps	1.0 E5 cps
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16 SR 3.3.1.17	Refer to Note 1 (Page 3.3.1-19)	Refer to Note 1 (Page 3.3.1-19)

(continued)

* The ≤ 31% RTP Allowable Value applies to the Westinghouse-supplied compensated ion chamber Intermediate Range neutron detectors. The compensated ion chamber neutron detectors are being replaced with Thermo Scientific-supplied fission chamber neutron detectors. The ≤ 38% RTP Allowable Value applies to the replacement fission chamber Intermediate Range neutron detectors.

** The ≤ 1.4 E5 cps Allowable Value applies to the Westinghouse-supplied boron triflouride (BF_3) Source Range neutron detectors. The BF_3 neutron detectors are being replaced with Thermo Scientific-supplied fission chamber neutron detectors. The ≤ 1.44 E5 cps Allowable Value applies to the replacement fission chamber Source Range neutron detectors.

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(l) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(m) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NOMINAL TRIP SETPOINT (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the UFSAR.

Table 3.3.1-1 (page 3 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16 SR 3.3.1.17	Refer to Note 2 (Page 3.3.1-20)	Refer to Note 2 (Page 3.3.1-20)
8. Pressurizer Pressure						
a. Low	$1^{(e)}$	4	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\geq 1938^{(f)}$ psig	$1945^{(f)}$ psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\leq$ 2399 psig	2385 psig
9. Pressurizer Water Level - High	$1^{(e)}$	3	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	$\leq$ 93.8%	92%
10. Reactor Coolant Flow - Low						
a. Single Loop	$1^{(g)}$	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\geq$ 89.7%	91%
b. Two Loops	$1^{(h)}$	3 per loop	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\geq$ 89.7%	91%

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Time constants utilized in the lead-lag controller for Pressurizer Pressure - Low are 2 seconds for lead and 1 second for lag.

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 4 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
11. Undervoltage RCPs	1(e)	1 per bus	L	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 5016 V	5082 V
12. Underfrequency RCPs	1(e)	1 per bus	L	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 55.9 Hz	56.4 Hz
13. Steam Generator (SG) Water Level - Low Low	1,2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 9% (Unit 1) ≥ 35.1% (Unit 2) of narrow range span	10.7% (Unit 1) 36.8% (Unit 2) of narrow range span
14. Turbine Trip						
a. Stop Valve EH Pressure Low	1(j)	4	N	SR 3.3.1.10 SR 3.3.1.15	≥ 500 psig	550 psig
b. Turbine Stop Valve Closure	1(j)	4	O	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open	NA
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	P	SR 3.3.1.5 SR 3.3.1.14	NA	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(i) Not used.

(j) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 5 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	$2^{(d)}$	2	R	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp*** ≥ 6.6E-6% RTP	1E-10 amp*** 1E-5% RTP
b. Low Power Reactor Trips Block, P-7	1	1 per train	S	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.13	≤ 50.2% RTP	48% RTP
d. Power Range Neutron Flux, P-9	1	4	S	SR 3.3.1.11 SR 3.3.1.13	≤ 70% RTP	69% RTP
e. Power Range Neutron Flux, P-10	1,2	4	R	SR 3.3.1.11 SR 3.3.1.13	≥ 7.8% RTP and ≤ 12.2% RTP	10% RTP
f. Turbine Impulse Pressure, P-13	1	2	S	SR 3.3.1.12 SR 3.3.1.13	≤ 12.2% RTP turbine impulse pressure equivalent	10% RTP turbine impulse pressure equivalent
17. Reactor Trip Breakers(k)	1,2	2 trains	Q,U	SR 3.3.1.4	NA	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2 trains	C	SR 3.3.1.4	NA	NA
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	T	SR 3.3.1.4	NA	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	1 each per RTB	C	SR 3.3.1.4	NA	NA
19. Automatic Trip Logic	1,2	2 trains	P,U	SR 3.3.1.5	NA	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2 trains	C	SR 3.3.1.5	NA	NA

(continued)

*** The ≥ 6E-11 amp Allowable Value and the 1E-10 amp NOMINAL TRIP SETPOINT value apply to the Westinghouse-supplied compensated ion chamber Intermediate Range neutron detectors. The compensated ion chamber neutron detectors are being replaced with Thermo Scientific-supplied fission chamber neutron detectors. The ≥ 6.6E-6% RTP Allowable Value and the 1E-5% RTP NOMINAL TRIP SETPOINT value apply to the replacement fission chamber Intermediate Range neutron detectors.

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 6 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 4.3% (Unit 1) and 4.5% (Unit 2) of RTP.

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

Where: ΔT is the measured RCS ΔT by loop narrow range RTDs, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP (allowed by Safety Analysis), ≤ the values specified in the COLR.
P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, = the value specified in the COLR

K_1 = Overtemperature ΔT reactor NOMINAL TRIP SETPOINT, as presented in the COLR,

K_2 = Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, as presented in the COLR,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, as presented in the COLR,

τ_1, τ_2 = Time constants utilized in the lead-lag compensator for ΔT, as presented in the COLR,

τ_3 = Time constant utilized in the lag compensator for ΔT, as presented in the COLR,

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, as presented in the COLR,

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the COLR, and

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

(i) for $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the COLR; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the COLR; and

(continued)

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

(iii) for each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the COLR.

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 2.6% (Unit 1) and 3.1% (Unit 2) of RTP.

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

Where: ΔT is the measured RCS ΔT by loop narrow range RTDs, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T'' is the nominal T_{avg} at RTP (calibration temperature for ΔT instrumentation), $\leq$ the values specified in the COLR.

K_4 = Overpower ΔT reactor NOMINAL TRIP SETPOINT as presented in the COLR,

K_5 = the value specified in the COLR for increasing average temperature and the value specified in the COLR for decreasing average temperature,

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the COLR for $T > T''$ and K_6 = the value specified in the COLR for $T \leq T''$,

τ_1, τ_2 = Time constants utilized in the lead-lag compensator for ΔT, as presented in the COLR,

τ_3 = Time constant utilized in the lag compensator for ΔT, as presented in the COLR,

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the COLR,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, as presented in the COLR, and

$f_2(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the COLR; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(continued)

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_2(\Delta I)$ "negative" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "negative" slope presented in the COLR; and

(iii) for each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the $f_2(\Delta I)$ "positive" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "positive" slope presented in the COLR.

SURVEILLANCE REQUIREMENTS

--NOTE--
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.3	--------------------------------NOTE------------------------------ Final actuation of pumps or valves not required. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.4	Perform MASTER RELAY TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.5	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.6	Perform SLAVE RELAY TEST.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.7	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.8	-------------------------------NOTE------------------------------ Verification of setpoint not required for manual initiation functions. --- Perform TADOT.	 In accordance with the Surveillance Frequency Control Program
SR 3.3.2.9	-------------------------------NOTE------------------------------ This Surveillance shall include verification that the time constants are adjusted to the prescribed values. --- Perform CHANNEL CALIBRATION.	 In accordance with the Surveillance Frequency Control Program
SR 3.3.2.10	-------------------------------NOTE------------------------------ Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is $\geq$ 600 psig. --- Verify ESFAS RESPONSE TIMES are within limit.	 In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.2.11 Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.12 Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.2-1 (page 1 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection[b]						
a. Manual initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 1.4 psig	1.2 psig
d. Pressurizer Pressure - Low	1,2,3[a]	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 1839 psig	1845 psig
2. Containment Spray*						
a. Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.2 psig	3.0 psig
3. Containment Isolation[b]						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

* The requirements of this Function are not applicable for entry into the applicable MODES following implementation of the modifications associated with ECCS Water Management on the respective unit.

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) The requirements of this Function are not applicable to Containment Purge Ventilation System and Hydrogen Purge System components, since the system containment isolation valves are sealed closed in MODES 1, 2, 3, and 4.

Table 3.3.2-1 (page 2 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
3. Containment Isolation (continued)						
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.2 psig	3.0 psig
4. Steam Line Isolation						
a. Manual Initiation						
(1) System	1,2(b),3(b)	2 trains	F	SR 3.3.2.8	NA	NA
(2) Individual	1,2(b),3(b)	1 per line	G	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(b),3(b)	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High	1,2(b),3(b)	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.2 psig	3.0 psig
d. Steam Line Pressure						
(1) Low	1,2(b),3(a)(b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 744 psig	775 psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Except when all MSIVs are closed and de-activated.

Table 3.3.2-1 (page 3 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
(2) Negative Rate - High	3(b)(c)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 122.8(d) psi	100(d) psi
5. Turbine Trip and Feedwater Isolation						
a. Turbine Trip						
(1) Automatic Actuation Logic and Actuation Relays	1,2	2 trains	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(2) SG Water Level-High-High (P-14)	1,2	4 per SG	J	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	≤ 85.6% (Unit 1) ≤ 78.9% (Unit 2)	83.9% (Unit 1) 77.1% (Unit 2)
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements. See Item 5.a.(1) for Applicable MODES.					
b. Feedwater Isolation						
(1) Automatic Actuation Logic and Actuation Relays	1,2(e),3(e)	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA

(continued)

(b) Except when all MSIVs are closed and de-activated.

(c) Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when Steam Line Isolation Steam Line Pressure - Low is not blocked.

(d) Time constant utilized in the rate/lag controller is ≥ 50 seconds.

(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.2-1 (page 4 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
(2) SG Water Level- High High (P-14)	1,2(e),3(e)	4 per SG	D	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	≤ 85.6% (Unit 1) ≤ 78.9% (Unit 2)	83.9% (Unit 1) 77.1% (Unit 2)
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements. See Item 5.b.(1) for Applicable MODES.					
(4) Tavg-Low	1,2(e)	4	J	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≥ 561°F	564°F
coincident with Reactor Trip, P-4	Refer to Function 8.a (Reactor Trip, P-4) for all initiation functions and requirements.					
(5) Doghouse WaterLevel - High High	1,2(e)	(1/1 logic) 2 per doghouse (2/3 logic) 3 per train per doghouse	L	(1/1 logic) SR 3.3.2.8 (2/3 logic) SR 3.3.2.8 SR 3.3.2.9 SR 3.3.2.12	≤ 12 inches above 577 ft floor level	11 inches above 577 ft floor level
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - Low Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 9% (Unit 1) ≥ 35.1% (Unit 2)	10.7% (Unit 1) 36.8% (Unit 2)
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Loss of Offsite Power	1,2,3	3 per bus	D	SR 3.3.2.3 SR 3.3.2.9 SR 3.3.2.10	≥ 3242 V	3500 V
e. Trip of all Main Feedwater Pumps	1,2	3 per pump	K	SR 3.3.2.8 SR 3.3.2.10	NA	NA
f. Auxiliary Feedwater Pump Train A and Train B Suction Transfer on Suction Pressure - Low	1,2,3	3 per train	M	SR 3.3.2.8 SR 3.3.2.10	A) ≥ 9.5 psig B) ≥ 5.2 psig (Unit 1) ≥ 5.0 psig (Unit 2)	A) 10.5 psig B) 6.2 psig (Unit 1) 6.0 psig (Unit 2)

(continued)

(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. Refueling Water Storage Tank (RWST) Level – Low	1,2,3,4	4	N	SR 3.3.2.1 SR 3.3.2.7(a)(b) SR 3.3.2.9(a)(b) SR 3.3.2.10	≥ 162.4 inches*	177.15 inches*
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.8	NA	NA
b. Pressurizer Pressure, P-11	1,2,3	3	O	SR 3.3.2.5 SR 3.3.2.9	≥ 1944 and ≤ 1966 psig	1955 psig
c. T_{avg} - Low Low, P-12	1,2,3	1 per loop	O	SR 3.3.2.5 SR 3.3.2.9	≥ 550°F	553°F
9. Containment Pressure Control System						
a. Start Permissive	1,2,3,4	4 per train	P	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9	≤ 1.0 psid	0.9 psid
b. Termination	1,2,3,4	4 per train	P	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9	≥ 0.25 psid	0.35 psid
10. Nuclear Service Water Suction Transfer - Low Pit Level	1,2,3,4	3 per pit	Q,R	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.11 SR 3.3.2.12	≥ El. 555.4 ft	El. 557.5 ft

* Following implementation of the modifications associated with ECCS Water Management on the respective unit, the Allowable Value for this Function shall be ≥ 91.9 inches and the Nominal Trip Setpoint for this Function shall be 95 inches.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the UFSAR.

SURVEILLANCE REQUIREMENTS

-----NOTE-----
SR 3.3.3.1 and SR 3.3.3.3 apply to each PAM instrumentation Function in Table 3.3.3-1.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.3.2	Not Used	
SR 3.3.3.3	-----NOTES----- 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. CHANNEL CALIBRATION may consist of an electronic calibration of the Containment Area - High Range Radiation Monitor, 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. ---------- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.2	-------------------------------NOTE------------------------------ Not applicable to Reactor Trip Breaker Position. ---	
	Perform CHANNEL CALIBRATION for each required instrumentation channel.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.5.1	-------------------------------NOTE------------------------------- Testing shall consist of voltage sensor relay testing excluding actuation of load shedding diesel start, and time delay times. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.2	Perform CHANNEL CALIBRATION with NOMINAL TRIP SETPOINT and Allowable Value as follows: a. Loss of voltage Allowable Value $\geq$ 3242 V. Loss of voltage NOMINAL TRIP SETPOINT = 3500 V. b. Degraded voltage Allowable Value $\geq$ 3738 V. Degraded voltage NOMINAL TRIP SETPOINT = 3766 V.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Air Release and Addition Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.2	Perform MASTER RELAY TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.3	Perform SLAVE RELAY TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.4	--------------------------------NOTE------------------------------- Verification of setpoint is not required. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.9.2	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.9.3	Verify each automatic valve moves to the correct position and Reactor Makeup Water pumps stop upon receipt of an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.3.9.4	-------------------------------NOTE------------------------------ Only required to be performed when used to satisfy Required Action A.3 or B.3. --	
	Perform CHANNEL CHECK on the Source Range Neutron Flux Monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.3.9.5	-------------------------------NOTE------------------------------ Only required to be performed when used to satisfy Required Action A.3 or B.3. --	
	Verify combined flowrates from both Reactor Makeup Water Pumps are $\leq$ the value in the COLR.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.9.6	------------------------------NOTE------------------------------ Only required to be performed when used to satisfy Required Action A.3 or B.3. -- Perform COT on the Source Range Neutron Flux Monitors.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2	Verify RCS average temperature is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	Verify RCS total flow rate is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.4	Perform CHANNEL CALIBRATION for each RCS total flow indicator.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE----------- Required Action C.2 shall be completed whenever this Condition is entered. -------------------------------	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within limits.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops — MODES 1 and 2

LCO 3.4.4 Four RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify each RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ 12% narrow range for required RCS loops.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pumps that are not in operation.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One RHR loop OPERABLE. AND ALL RCS loops inoperable.	B.1 Be in MODE 5.	24 hours
C. Both required RCS or RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 and maintain $k_{eff} < 0.99$.	Immediately
	AND	
	C.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.6.1 Verify one RHR or RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2 Verify SG secondary side water levels are $\geq$ 12% narrow range for required RCS loops.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. AND Required SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately Immediately
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.2	Verify SG secondary side water level is ≥ 12% narrow range in required SGs.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND	
	B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one RHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.2	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is $\leq$ 92% (1656 ft^3).	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is $\geq$ 150 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.3	Verify required pressurizer heaters are capable of being powered from an emergency power supply.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.2 Restore one block valve to OPERABLE status if three block valves are inoperable. AND F.3 Restore remaining block valve(s) to OPERABLE status.	2 hours 72 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.11.1 ------------------------------NOTE------------------------------ Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. -- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.2	-----NOTE----- Required to be performed in MODE 3 or MODE 4 when the temperature of all RCS cold legs is > 200°F. ----- Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.3	-----NOTE----- This SR is not applicable to valve NC-36B. ----- Verify the nitrogen supply for each PORV is OPERABLE by: a. Manually transferring motive power from the air supply to the nitrogen supply, b. Isolating and venting the air supply, and c. Operating the PORV through one complete cycle.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of two pumps (charging, safety injection, or charging and safety injection) are capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify each accumulator is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	Verify RHR suction isolation valves are open for each required RHR suction relief valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	Verify PORV block valve is open for each required PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.5	-------------------------------NOTE------------------------------- Not required to be met until 12 hours after decreasing RCS cold leg temperature to $\leq$ 210°F. -- Perform a COT on each required PORV, excluding actuation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.6	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.7	Verify associated RHR suction isolation valves are open, with operator power removed and locked in removed position, for each required RHR suction relief valve.	In accordance with the Surveillance Frequency Control Program

Table 3.4.12-1 (Page 1 of 1)

(UNIT 1 ONLY)

Reactor Coolant Pump Operating Restrictions for Low Temperature Overpressure Protection

Reactor Coolant System Cold Leg Temperature	Maximum Number of Pumps Allowed in Operation
$\geq$ 70°F	2
$\geq$ 126°F	4

Table 3.4.12-1 (Page 1 of 1)

(UNIT 2 ONLY)

Reactor Coolant Pump Operating Restrictions for Low Temperature Overpressure Protection

Reactor Coolant System Cold Leg Temperature	Maximum Number of Pumps Allowed in Operation
≥ 70°F	1
≥ 140°F	4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	------------------------------NOTES------------------------------ 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --	------NOTE------ Only required to be performed during steady state operation ------------------
	Verify RCS Operational LEAKAGE within limits by performance of RCS water inventory balance.	In accordance with the Surveillance Frequency Control Program
SR 3.4.13.2	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after establishment of steady state operation. --	--------NOTE-------- Only required to be performed during steady state operation ----------------------
	Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-----NOTES----- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. ---------- Verify leakage from each RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 psig and $\leq$ 2255 psig.	In accordance with the Inservice Testing Program, and in accordance with the Surveillance Frequency Control Program <u>AND</u> Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months <u>AND</u> Within 24 hours following valve actuation due to automatic or manual action or flow through the valve

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.2	Verify RHR system interlock prevents the valves from being opened with a simulated or actual RCS pressure signal $\geq$ 425 psig.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the containment atmosphere particulate radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform COT of the containment atmosphere particulate radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the containment floor and equipment sump level monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the containment atmosphere particulate radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.5	Perform CHANNEL CALIBRATION of the containment ventilation unit condensate drain tank level monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.6	Perform CHANNEL CALIBRATION of the incore instrument sump level alarm.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.16.1 Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.2	-------------------------------NOTE------------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of $\geq$ 15% RTP within a 1 hour period
SR 3.4.16.3	-------------------------------NOTE------------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 RCS Loops — Test Exceptions

LCO 3.4.17 The requirements of LCO 3.4.4, "RCS Loops — MODES 1 and 2," may be suspended, with THERMAL POWER < P-7.

APPLICABILITY: MODES 1 and 2 during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ P-7.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.17.1 Verify THERMAL POWER is < P-7.	In accordance with the Surveillance Frequency Control Program
SR 3.4.17.2 Perform a COT for each power range neutron flux-low and intermediate range neutron flux channel, P-10, and P-13.	Prior to initiation of startup and PHYSICS TESTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each accumulator is $\geq$ 7630 gallons and $\leq$ 8079 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is $\geq$ 585 psig and $\leq$ 678 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each accumulator is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program <u>AND</u> -------NOTE------ Only required to be performed for affected accumulators --------------------- Once within 6 hours after each solution volume increase of $\geq$ 75 gallons that is not the result of addition from the refueling water storage tank

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is > 1000 psig.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. Number / Position / Function NI162A / Open / SI Cold Leg Injection NI121A / Closed / SI Hot Leg Injection NI152B / Closed / SI Hot Leg Injection NI183B / Closed / RHR Hot Leg Injection NI173A / Open / RHR Cold Leg Injection NI178B / Open / RHR Cold Leg Injection NI100B / Open / SI Pump Suction from RWST NI147B / Open / SI Pump Mini-Flow	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify ECCS piping is full of water.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. Centrifugal Charging Pump Injection Throttle <u>Valve Number</u>: NI14, NI16, NI18, NI20 Safety Injection Pump Throttle <u>Valve Number</u>: NI164, NI166, NI168, NI170	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.8	Verify, by visual inspection, that the ECCS containment sump strainer assembly is not restricted by debris and shows no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is $\geq$ 70°F and $\leq$ 100°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is $\geq$ 363,513 gallons.*	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

* Following implementation of the modifications associated with ECCS Water Management on the respective unit, the RWST borated water volume for this SR shall be $\geq$ 377,537 gallons.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.5.1	---------------------------NOTE-------------------------------- Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at $\geq$ 2215 psig and $\leq$ 2255 psig. -- Verify manual seal injection throttle valves are adjusted to give a flow within limit with centrifugal charging pump operating and the charging flow control valve full open.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	------NOTES------ 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ------ Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Perform a pressure test on each inflatable air lock door seal and verify door seal leakage is < 15 sccm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.3	Verify only one door in the air lock can be opened at a time.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each containment purge supply and exhaust isolation valves for the lower compartment and the upper compartment, instrument room, and the Hydrogen Purge System is sealed closed, except for one purge valve in a penetration flow path while in Condition E of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each Containment Air Release and Addition System isolation valve is closed, except when the valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	-----------------------------NOTE---------------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --- Verify each containment isolation manual valve and blind flange that is located outside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	-----------------------------NOTE---------------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each containment isolation manual valve and blind flange that is located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for Containment Purge System, Hydrogen Purge System, and Containment Air Release and Addition System valves with resilient seals.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be $\geq$ -0.1 psig and $\leq$ +0.3 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment upper compartment average air temperature is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.2	Verify containment lower compartment average air temperature is within limits.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Two containment spray trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic* valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

* Following implementation of the modifications associated with ECCS Water Management on the respective unit, there will be no automatic valves in the Containment Spray System.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.3	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.*	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.*	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.5	Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to manually** start upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6	Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to manually** open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each spray nozzle is unobstructed.	Following activities which could result in nozzle blockage

* Following implementation of the modifications associated with ECCS Water Management on the respective unit, the requirements of SR 3.6.6.3 and SR 3.6.6.4 shall no longer be applicable.

** Following implementation of the modifications associated with ECCS Water Management on the respective unit, spray pump starting and spray pump discharge valve opening are manual functions.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Operate each HSS train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.8.2	Verify the fan motor current is ≤ 69 amps when the fan speed is ≥ 3560 rpm and ≤ 3600 rpm with the hydrogen skimmer fan operating and the motor operated suction valve closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.8.3	Verify the motor operated suction valve opens automatically and the fans receive a start permissive signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.8.4	Verify each HSS train starts on an actual or simulated actuation signal after a delay of ≥ 8 minutes and ≤ 10 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.9.1	Energize each HIS train power supply breaker and verify ≥ 34 ignitors are energized in each train.	In accordance with the Surveillance Frequency Control Program
SR 3.6.9.2	Verify at least one hydrogen ignitor is OPERABLE in each containment region.	In accordance with the Surveillance Frequency Control Program
SR 3.6.9.3	Energize each hydrogen ignitor and verify temperature is ≥ 1700°F.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.1	Operate each AVS train for $\geq$ 10 continuous hours with heaters operating.	In accordance with the Surveillance Frequency Control Program
SR 3.6.10.2	Perform required AVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.10.3	Verify each AVS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.10.4	Verify each AVS filter cooling bypass valve can be opened.	In accordance with the Surveillance Frequency Control Program
SR 3.6.10.5	Verify each AVS train flow rate is $\geq$ 8100 cfm and $\leq$ 9900 cfm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.10.6	Verify each AVS train produces a pressure equal to or more negative than -0.88 inch water gauge when corrected to elevation 564 feet.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.11 Air Return System (ARS)

LCO 3.6.11 Two ARS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARS train inoperable.	A.1 Restore ARS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.11.1 Verify each ARS fan starts on an actual or simulated actuation signal, after a delay of ≥ 8.0 minutes and ≤ 10.0 minutes, and operates for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.11.2	Verify, with the ARS air return fan damper closed and with the bypass dampers open, each ARS fan motor current is $\leq$ 59.0 amps when the fan speed is $\geq$ 1174 rpm and $\leq$ 1200 rpm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.11.3	Verify, with the ARS fan not operating, each ARS motor operated damper opens automatically on an actual or simulated actuation signal after a delay of $\geq$ 9 seconds and $\leq$ 11 seconds.	In accordance with the Surveillance Frequency Control Program
SR 3.6.11.4	Verify the check damper is open with the ARS fan operating.	In accordance with the Surveillance Frequency Control Program
SR 3.6.11.5	Verify the check damper is closed with the ARS fan not operating.	In accordance with the Surveillance Frequency Control Program
SR 3.6.11.6	Verify that each ARS fan is de-energized or is prevented from starting upon receipt of a terminate signal from the Containment Pressure Control System (CPCS) and is allowed to start upon receipt of a start permissive from the CPCS.	In accordance with the Surveillance Frequency Control Program
SR 3.6.11.7	Verify that each ARS fan motor-operated damper is prevented from opening in the absence of a start permissive from the Containment Pressure Control System (CPCS) and is allowed to open upon receipt of a start permissive from the CPCS.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.12 Ice Bed

LCO 3.6.12 The ice bed shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ice bed inoperable.	A.1 Restore ice bed to OPERABLE status.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.12.1 Verify maximum ice bed temperature is $\leq$ 27°F.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.2	-----NOTE----- The chemical analysis may be performed on either the liquid solution or on the resulting ice. ---------- Verify, by chemical analysis, that ice added to the ice condenser meets the boron concentration and pH requirements of SR 3.6.12.7.	Each ice addition
SR 3.6.12.3	Verify, by visual inspection, accumulation of ice on structural members comprising flow channels through the ice bed is ≤ 15 percent blockage of the total flow area for each safety analysis section.	In accordance with the Surveillance Frequency Control Program
SR 3.6.12.4	Verify total mass of stored ice is ≥ 2,132,000 lbs by calculating the mass of stored ice, at a 95 percent confidence, in each of three Radial Zones as defined below, by selecting a random sample of ≥ 30 ice baskets in each Radial Zone, and Verify: 1. Zone A (radial rows 8, 9), has a total mass of ≥ 324,000 lbs 2. Zone B (radial rows 4, 5, 6, 7), has a total mass of ≥ 1,033,100 lbs 3. Zone C (radial rows 1, 2, 3), has a total mass of ≥ 774,900 lbs	In accordance with the Surveillance Frequency Control Program
SR 3.6.12.5	Verify that the ice mass of each basket sampled in SR 3.6.12.4 is ≥ 600 lbs.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.6	Visually inspect, for detrimental structural wear, cracks, corrosion, or other damage, two ice baskets from each group of bays as defined below: a. Group 1 – bays 1 through 8; b. Group 2 – bays 9 through 16; and c. Group 3 – bays 17 through 24.	In accordance with the Surveillance Frequency Control Program
SR 3.6.12.7	------------------------------ NOTE ------------------------------ The requirements of this SR are satisfied if the boron concentration and pH values obtained from averaging the individual sample results are within the limits specified below. -- Verify, by chemical analysis of the stored ice in at least one randomly selected ice basket from each ice condenser bay, that ice bed: a. Boron concentration is $\geq$ 1800 ppm and $\leq$ 2330 ppm; and b. pH is $\geq$ 9.0 and $\leq$ 9.5.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Restore ice condenser door to OPERABLE status and closed positions.	48 hours
D. Required Action and associated Completion Time of Condition A or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.13.1	Verify all lower inlet doors indicate closed by the Inlet Door Position Monitoring System.	In accordance with the Surveillance Frequency Control Program
SR 3.6.13.2	Verify, by visual inspection, each intermediate deck door is closed and not impaired by ice, frost, or debris.	In accordance with the Surveillance Frequency Control Program
SR 3.6.13.3	Verify, by visual inspection, each top deck door: a. Is in place; and b. Has no condensation, frost, or ice formed on the door that would restrict its opening.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.13.4	Verify, by visual inspection, each lower inlet door is not impaired by ice, frost, or debris.	In accordance with the Surveillance Frequency Control Program
SR 3.6.13.5	Verify torque required to cause each lower inlet door to begin to open is ≤ 675 in-lb and verify free movement of the door.	In accordance with the Surveillance Frequency Control Program
SR 3.6.13.6	Deleted.	
SR 3.6.13.7	Verify for each intermediate deck door: a. No visual evidence of structural deterioration; b. Free movement of the vent assemblies; and c. Free movement of the door.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.14.1	Verify, by visual inspection, all personnel access doors and equipment hatches between upper and lower containment compartments are closed.	Prior to entering MODE 4 from MODE 5
SR 3.6.14.2	Verify, by visual inspection, that the seals and sealing surfaces of each personnel access door and equipment hatch have: a. No detrimental misalignments; b. No cracks or defects in the sealing surfaces; and c. No apparent deterioration of the seal material.	Prior to final closure after each opening AND ---------NOTE------- Only required for seals made of resilient materials ----------------------- In accordance with the Surveillance Frequency Control Program
SR 3.6.14.3	Verify, by visual inspection, each personnel access door or equipment hatch that has been opened for personnel transit entry is closed.	After each opening

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.14.4	Remove two divider barrier seal test coupons and verify both test coupons' tensile strength is $\geq$ 39.7 psi.	In accordance with the Surveillance Frequency Control Program
SR 3.6.14.5	Visually inspect $\geq$ 95% of the divider barrier seal length, and verify: a. Seal and seal mounting bolts are properly installed; and b. Seal material shows no evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearance.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE REQUIREMENTS		FREQUENCY
SR 3.6.15.1	Verify, by visual inspection, that: a. Each refueling canal drain valve is locked open; and b. Each refueling canal drain is not obstructed by debris.	Prior to entering MODE 4 from MODE 5 after each partial or complete fill of the canal
SR 3.6.15.2	Verify, by visual inspection that no debris is present in the upper compartment or refueling canal that could obstruct the refueling canal drain.	In accordance with the Surveillance Frequency Control Program
SR 3.6.15.3	Verify for each ice condenser floor drain that the: a. Valve opening is not impaired by ice, frost, or debris; b. Valve seat shows no evidence of damage; c. Valve opening force is $\leq$ 66 lb; and d. Drain line from the ice condenser floor to the lower compartment is unrestricted.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.16 Reactor Building

LCO 3.6.16 The reactor building shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.16.1	Verify the door in each access opening is closed, except when the access opening is being used for normal transit entry and exit.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.16.2	Verify that during the annulus vacuum decay test, the vacuum decay time is $\geq$ 87 seconds.	In accordance with the Surveillance Frequency Control Program
SR 3.6.16.3	Verify reactor building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the reactor building.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one of the nitrogen bottles on each SG PORV is pressurized ≥ 2100 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.7.4.2	Verify one complete cycle of each SG PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.7.4.3	Verify one complete cycle of each SG PORV block valve.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	-------------------------------NOTE------------------------------ Not applicable to automatic valves when THERMAL POWER is ≤ 10% RTP. --- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	-------------------------------NOTE------------------------------ Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. --- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.3	-------------------------------NOTE------------------------------ Not applicable in MODE 4 when steam generator is relied upon for heat removal. --- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	-----NOTES----- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. ---------- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.5	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage system to each steam generator.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CSS inventory is $\geq$ 225,000 gal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	------------------------------NOTE------------------------------ Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	 In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	Verify each CCW automatic valve in the flow path servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	------------------------------NOTE------------------------------ Isolation of NSWS flow to individual components does not render the NSWS inoperable. -- Verify each NSWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	------------------------------NOTE------------------------------ Not required to be met for valves that are maintained in position to support NSWS single supply header operation. -- Verify each NSWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each NSWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.9 Standby Nuclear Service Water Pond (SNSWP)

LCO 3.7.9 The SNSWP shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SNSWP inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of SNSWP is $\geq$ 571 ft mean sea level.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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SURVEILLANCE		FREQUENCY
SR 3.7.9.2	-----------------------------------NOTE----------------------------------- Only required to be performed during the months of July, August, and September. --- Verify average water temperature of SNSWP is ≤ 95°F at an elevation of 568 ft. in SNSWP.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.3	Verify, by visual inspection, no abnormal degradation, erosion, or excessive seepage of the SNSWP dam.	In accordance with the Surveillance Frequency Control Program

REQUIRED ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. One or more CRAVS train(s) heater inoperable.	G.1 Restore CRAVS train(s) heater to OPERABLE status.	7 days
	OR	
	G.2 Initiate action in accordance with Specification 5.6.6.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRAVS train for $\geq$ 10 continuous hours with the heaters operating.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required CRAVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CRAVS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CRACWS trains inoperable in MODE 5 or 6, or during movement of recently irradiated fuel assemblies.	D.1 Suspend movement of recently irradiated fuel assemblies.	Immediately
E. Two CRACWS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify the control room temperature is $\leq$ 90°F.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABFVES train for $\geq$ 10 continuous hours with the heaters operating.	In accordance with the Surveillance Frequency Control Program
SR 3.7.12.2	Perform required ABFVES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABFVES train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.12.4	Verify one ABFVES train can maintain the ECCS pump rooms at negative pressure relative to adjacent areas.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify required FHVES train in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.7.13.2	Operate required FHVES train for $\geq$ 10 continuous hours with the heaters operating.	In accordance with the Surveillance Frequency Control Program
SR 3.7.13.3	Perform required FHVES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.4	Verify one FHVES train can maintain a pressure $\leq$ -0.25 inches water gauge with respect to atmospheric pressure during operation at a flow rate $\leq$ 36,443 cfm.	In accordance with the Surveillance Frequency Control Program
SR 3.7.13.5	Verify each FHVES filter bypass damper can be closed.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Water Level

LCO 3.7.14 The spent fuel pool water level shall be $\geq$ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 -------------NOTE------------- LCO 3.0.3 is not applicable. --------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the spent fuel pool water level is $\geq$ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be within the limit specified in the COLR.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	------------------NOTE------------------ LCO 3.0.3 is not applicable. --	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify the spent fuel pool boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.17 Secondary Specific Activity

LCO 3.7.17 The specific activity of the secondary coolant shall be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.17.1 Verify the specific activity of the secondary coolant is $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	-------------------------------NOTES------------------------------ 1. Performance of SR 3.8.1.7 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage $\geq$ 3950 V and $\leq$ 4580 V, and frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-----NOTES----- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. ----- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 5600 kW and ≤ 5750 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify each day tank contains ≥ 470 gal of fuel oil.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage system to the day tank.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	-----NOTE----- All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts from standby condition and achieves in $\leq$ 11 seconds voltage of $\geq$ 3950 V and frequency of $\geq$ 57 Hz and maintains steady-state voltage $\geq$ 3950 V and $\leq$ 4580 V, and frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.8	Verify automatic and manual transfer of AC power sources from the normal offsite circuit to each alternate offsite circuit.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	-----NOTE----- If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. ---------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 63 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 3950 V and ≤ 4580 V; and c. Within 3 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.10	Verify each DG does not trip and generator speed is maintained ≤ 500 rpm during and following a load rejection of ≥ 5600 kW and ≤ 5750 kW.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	------------NOTES------------ 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --------------------------------	
	Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes the emergency bus in $\leq$ 11 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage $\geq$ 3950 V and $\leq$ 4580 V, 4. maintains steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies auto-connected shutdown loads for $\geq$ 5 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-----NOTE----- All DG starts may be preceded by prelube period. --------------- Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and: a. In $\leq$ 11 seconds after auto-start and during tests, achieves voltage $\geq$ 3950 V and $\leq$ 4580 V; b. In $\leq$ 11 seconds after auto-start and during tests, achieves frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz; c. Operates for $\geq$ 5 minutes; and d. The emergency bus remains energized from the offsite power system.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	Verify each DG's non-emergency automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.14	------------------------------NOTE------------------------------ Momentary transients outside the load and power factor ranges do not invalidate this test. -- Verify each DG operating at a power factor ≤ 0.9 operates for ≥ 24 hours loaded ≥ 5600 kW and ≤ 5750 kW.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	-------------------------------NOTES--------------------------- 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 1 hour loaded ≥ 5600 kW and ≤ 5750 kW or until operating temperature is stabilized. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. -- Verify each DG starts and achieves, in ≤ 11 seconds, voltage ≥ 3950 V, and frequency ≥ 57 Hz and maintains steady state voltage ≥ 3950 V and ≤ 4580 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.16	-------------------------------NOTE------------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to standby operation.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.17	-----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to standby operation; and b. Automatically energizing the emergency load from offsite power.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.18	Verify interval between each sequenced load block is within the design interval for each automatic load sequencer.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	-------------------------------NOTES------------------------------ 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---	
	Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes the emergency bus in $\leq$ 11 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage $\geq$ 3950 V and $\leq$ 4580 V, 4. achieves steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies auto-connected emergency loads for $\geq$ 5 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	-----NOTE----- All DG starts may be preceded by an engine prelube period. ----- Verify when started simultaneously from standby condition, each DG achieves, in ≤ 11 seconds, voltage of ≥ 3950 V and frequency of ≥ 57 Hz and maintains steady state voltage ≥ 3950 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with starting air receiver pressure < 210 psig and ≥ 150 psig.	E.1 Restore starting air receiver pressure to ≥ 210 psig.	48 hours
F. Required Action and associated Completion Time not met. OR One or more DGs diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify the fuel oil storage system contains ≥ 77,100 gal of fuel for each DG.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.2	Verify lubricating oil inventory is $\geq$ 400 gal.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG air start receiver pressure is $\geq$ 210 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. A and/or D channel of DC electrical power subsystem inoperable. AND Associated train of DG DC electrical power subsystem inoperable.	D.1 Enter applicable Condition(s) and Required Action(s) of LCO 3.8.9, "Distribution Systems-Operating", for the associated train of DC electrical power distribution subsystem made inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify DC channel and DG battery terminal voltage is ≥ 125 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Not used.	
SR 3.8.4.3	Verify no visible corrosion at the DC channel and DG battery terminals and connectors. OR Verify battery connection resistance of specific connection(s) meets Table 3.8.4-1 limit.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.4	Verify DC channel and DG battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Remove visible terminal corrosion, verify DC channel and DG battery cell to cell and terminal connections are clean and tight, and are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.6	Verify all DC channel and DG battery connection resistance values meet Table 3.8.4-1 limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.7	Verify each DC channel battery charger supplies ≥ 200 amps and the DG battery charger supplies ≥ 75 amps with each charger at ≥ 125 V for ≥ 8 hours.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.8	-------------------------------NOTES----------------------------- 1. The modified performance discharge test in SR 3.8.4.9 may be performed in lieu of the service test in SR 3.8.4.8. 2. This Surveillance shall not be performed for the DG batteries in MODE 1, 2, 3, or 4. --- Verify DC channel and DG battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.9	-----NOTE----- This Surveillance shall not be performed for the DG batteries in MODE 1, 2, 3, or 4. -----	
	Verify DC channel and DG battery capacity is $\geq$ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program AND 18 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating AND -----NOTE----- Not applicable to DG batteries ----- 24 months when battery has reached 85% of the expected life with capacity $\geq$ 100% of manufacturer's rating

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters of the channels of DC and DG batteries meet Table 3.8.6-1 Category A limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.2	Not used.	
SR 3.8.6.3	Verify battery cell parameters of the channels of DC and DG batteries meet Table 3.8.6-1 Category B limits.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 7 days after a battery discharge < 110 V <u>AND</u> Once within 7 days after a battery overcharge > 150 V
SR 3.8.6.4	Verify average electrolyte temperature for the channels of DC and DG batteries of representative cells is $\geq$ 60°F.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage and alignment to required AC vital buses.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or required boron concentration.	Immediately
	AND	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct voltage and alignment to required AC vital bus.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC channel, DC train, and AC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate actions to restore required AC, channels of DC, DC trains, and AC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2.5 Declare associated required residual heat removal subsystem(s) inoperable and not in operation.	Immediately
	<u>AND</u>	
	A.2.6 Declare affected Low Temperature Overpressure Protection feature(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC channel, DC train, and AC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

--NOTE--
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.9.2.2	-------------------------------NOTE------------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more CPES train(s) heater inoperable.	B.1 Restore CPES train(s) heater to OPERABLE status. OR B.2 Initiate action in accordance with Specification 5.6.6.	7 days 7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.2	Operate each CPES for ≥ 10 continuous hours with the heaters operating.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.3	Perform required CPES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of $\geq$ 1000 gpm and RCS temperature is $\leq$ 140°F.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one RHR loop to operation.	Immediately
	AND	
	B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 1000 gpm and RCS temperature is ≤ 140°F.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LCO 3.9.6 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.7 Unborated Water Source Isolation Valves

LCO 3.9.7 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE------------ Required Action A.3 must be completed whenever Condition A is entered. ------------------------------- One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS. <u>AND</u> A.2 Initiate actions to secure valve in closed position. <u>AND</u> A.3 Perform SR 3.9.1.1.	Immediately Immediately 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.7.1 Verify each valve that isolates unborated water sources is secured in the closed position.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals (continued)

5.5.16 Control Room Envelope Habitability Program

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

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

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

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

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRAVS, operating at a makeup flow rate of $\leq$ 4000 cfm, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

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

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

(continued)

5.5 Programs and Manuals (continued)

5.5.17 Surveillance Frequency Control Program

This Program provides controls for Surveillance Frequencies. The program shall ensure that the Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operations 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 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 263 TO RENEWED FACILITY OPERATING LICENSE NPF-35

AND

AMENDMENT NO. 259 TO RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By application dated March 31, 2010 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML100920160), as supplemented by letter dated November 30, 2010, (ADAMS Accession No. ML103370241), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2). The supplement dated November 30, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published the *Federal Register* on November 16, 2010 (75 FR 70034).

The amendments would revise the TSs by relocating specific surveillance frequencies to a licensee-controlled document using a risk-informed justification. The proposed changes would adopt the Nuclear Regulatory Commission (NRC, the Commission) staff-approved TS Task Force (TSTF) traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF [Risk-Informed TSTF] Initiative 5b" (Reference 1). 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 Inservice Testing Program or the Primary Containment Leakage Rate Testing Program);

- frequencies that are purely event-driven (e.g., "each time the control rod is withdrawn to the 'full out' position");
- frequencies that are event-driven, but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching ≥ 95% RTP [Rated Thermal Power]"); and
- frequencies that are related to specific conditions (e.g., battery degradation, age and capacity) or conditions for the performance of a surveillance requirement (e.g., "drywell to suppression chamber differential pressure decrease").

A new program would be added to the Administrative Controls in TS Section 5.5.17. The new program is called the SFCP and describes the requirements for the program to control changes to the relocated surveillance frequencies. The proposed licensee changes to the Administrative Controls of the TSs to incorporate the SFCP include a specific reference to Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," (Reference 2), as the basis for making any changes to the surveillance frequencies once they are relocated out of the TSs.

By letter dated September 19, 2007, (Reference 3), the NRC staff approved NEI 04-10, Revision 1, as acceptable for referencing by licensees proposing to amend their TSs to establish an SFCP. This acceptance was limited as specified in NEI 04-10, Revision 1, and Reference 3.

The NRC staff issued a "Notice of Availability" for TSTF-425, Revision 3, in the *Federal Register* on July 6, 2009 (74 FR 31996). The notice included a model Safety Evaluation (SE). In its application dated March 31, 2010, the licensee stated that "Duke Energy has concluded that the justifications presented in the TSTF-425 proposal and the safety evaluation prepared by the NRC staff is applicable to Catawba Units 1 and 2, and justify this amendment to incorporate the changes to the Catawba TS." The SE that follows is based, in large part, on the model SE for TSTF-425.

2.0 REGULATORY EVALUATION

In the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," published in the *Federal Register* on July 22, 1993 (58 FR 39132) the NRC staff addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment (PRA)) in determining the content of the TSs. On page 39135 of this *Federal Register* publication, the Commission states, in part, that:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36(c)(2)(ii)] to be deleted from Technical specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed. * * *

> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "* * * probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made * * * about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process of 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." * * *
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> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

Approximately two years later, the NRC provided additional detail concerning the use of PRA in the "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," published in the *Federal Register* on August 16, 1995 (60 FR 42622). On page 42627 of this FR publication, the Commission states, in part, that:

> 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.

On pages 42628 and 42629 of this *Federal Register* publication, the Commission provided its policy on use of PRA which states:

> Although PRA methods and information have thus far been used successfully in nuclear regulatory activities, there have been concerns that PRA methods are not consistently applied throughout the agency, that sufficient agency PRA/statistics expertise is not available, and that the Commission is not deriving full benefit from the large agency and industry investment in the developed risk assessment methods. Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data. Implementation of the policy statement will improve the regulatory process in three areas: Foremost, through safety decision making enhanced by the use of PRA insights; through more efficient use of agency resources; and through a reduction in unnecessary burdens on licensees.
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> 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.

The Commission's regulatory requirements related to the content of the TSs are set forth in 10 CFR 50.36, "Technical specifications." This regulation requires that the TSs include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

As stated in 10 CFR 50.36(c)(3), "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." To meet this requirement, the surveillance requirement must specify an adequate test, calibration, or inspection, and an appropriate frequency of performance. The licensee has proposed to implement changes to surveillance frequencies in the SFCP using the methodology in NEI 04-10, which includes qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, recommended monitoring of structures, systems, and components (SSCs), and documentation of the evaluation. Furthermore, changes to frequencies are subject to regulatory review and oversight of the SFCP implementation through the rigorous NRC review of safety-related SSC performance provided by the reactor oversight program.

The licensee's SFCP is intended to ensure that surveillance requirements specified in the TSs are performed at intervals sufficient to assure the above regulatory requirements are met. Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and Appendix B to 10 CFR Part 50, require licensee monitoring of surveillance test failures and implementation of corrective actions to address such failures. One of these actions may be to consider increasing the frequency at which a surveillance test is performed. In addition, the SFCP implementation guidance in NEI 04-10 requires monitoring the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs. These requirements, and the monitoring required by NEI 04-10, are intended to 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.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 4), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed 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, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Reference 5), describes an acceptable risk-informed approach specifically for assessing proposed TS changes.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 6), 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 Catawba 1 and 2 provides for administrative relocation of applicable surveillance frequencies, and provides for the addition of the SFCP to the Administrative Controls Section of the 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 requirements 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 consistent with the guidance provided in RG 1.174 and RG 1.177.

3.1 RG 1.177, Five Key Safety Principles

RG 1.177 identifies five key safety principles required for risk-informed changes to the TSs. Each of these principles is addressed by the industry methodology document, NEI 04-10, and is evaluated below in SE Sections 3.1.1 through 3.1.5 with respect to the proposed amendment.

3.1.1 The Proposed Change Meets Current Regulations

Paragraph (c)(3) in 10 CFR 50.36 requires that TSs will include surveillance requirements which are "requirements relating to test, calibration, or inspection to assure that necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The proposed amendment would relocate most periodic surveillance requirement frequencies, currently shown in the Catawba 1 and 2 TSs, to a licensee-controlled program (i.e., the SFCP). The surveillance requirements themselves would remain in the TSs, as required by 10 CFR 50.36(c)(3). The requirements for the SFCP would be added to a new subsection in TS Section 5.0. In accordance with TS Section 5.0, any changes to the surveillance requirement frequencies would be made in accordance with NEI 04-10, Revision 1. By letter dated September 19, 2007 (Reference 3), the NRC staff found that the methodology in NEI 04-10, Revision 1, met NRC regulations, specifically 10 CFR 50.36(c)(3), and was an acceptable program for controlling changes to surveillance requirement frequencies.

Based on the above considerations, the NRC staff concludes that the proposed change is consistent with the requirements in 10 CFR 50.36(c)(3). Therefore, the proposed change meets the first key safety principle of RG 1.177.

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

Consistency with the defense-in-depth philosophy, the second key safety principle of RG 1.177, is met 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).

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

- Independence of barriers is not degraded.

- Defenses against human errors are preserved.

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

TSTF-425 requires the application of NEI 04-10 for any changes to surveillance requirement 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 the CDF and the 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. The NRC staff concludes that both the quantitative risk analysis and the qualitative considerations assure that a reasonable balance of defense-in-depth is maintained. Therefore, the proposed change meets 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 surveillance requirement frequencies are revised, will assess the impact of the proposed frequency change in accordance with the principle that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the proposed surveillance test frequency change is not in conflict with approved industry codes and standards or adversely affects any assumptions or inputs to the safety analysis, or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

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

Based on the above considerations, the NRC staff concludes that there is reasonable assurance that safety margins will be maintained through the use of the SFCP methodology. Therefore, the proposed change meets the third key safety principle of RG 1.177.

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

RG 1.177 provides a framework for risk evaluation of proposed changes to surveillance frequencies. This requires 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 requires application of NEI 04-10 in the SFCP. NEI 04-10 satisfies the intent of RG 1.177 requirements 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 Catawba 1 and 2 PRA is compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. That is, the more the potential 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.

The licensee used RG 1.200 to address the Catawba 1 and 2 PRA technical adequacy. RG 1.200 is NRC's developed regulatory guidance which, in Revision 1, endorsed with comments and qualifications the use of "ASME [American Society of Mechanical Engineers] PRA Standard RA-Sb-2005, 'Addenda B to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,'" (Reference 7), NEI 00-02, Revision 1, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," (Reference 8), and NEI 05-04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard" (Reference 9). The licensee has performed an assessment of the PRA models used to support the SFCP against the requirements of RG 1.200 to assure that the PRA models are capable of determining the change in risk due to changes to surveillance requirement frequencies of SSCs, using plant-specific data and models. Capability category II of ASME RA-Sb-2005 was applied as the standard, and any identified deficiencies to those requirements are assessed further to determine any impacts to proposed decreases to surveillance frequencies, including by the use of sensitivity studies where appropriate. The NRC staff notes that in RG 1.200, Revision 2, the NRC staff endorsed with comments and qualifications an updated combined standard which includes requirements for fire, seismic, and other external events PRA models. The existing internal events standard was subsumed into the combined standard, but the technical requirements are essentially unchanged. Since NEI 04-10 specifically identified the use of RG 1.200, Revision 1, to assess the internal events standard, the licensee's approach is reasonable and consistent with the approved methodology.

The NRC staff reviewed the licensee's assessment of the Catawba 1 and 2 PRA and the remaining open deficiencies that do not conform to capability category II of the ASME PRA standard (Table 2-1 of Attachment 2 of the licensee amendment request). The NRC staff's assessment of these open "gaps," to assure that they may be addressed and dispositioned for each surveillance frequency evaluation per the NEI 04-10 methodology, is provided below.

<u>Gap #1</u>: Accident sequence notebooks and system model notebooks should document the phenomenological conditions created by the accident sequence progression. In response to the request for additional information (RAI), the licensee stated that for each surveillance frequency change evaluation, any phenomenological conditions created by the accident sequence progression will be identified, included and documented in the analysis.

<u>Gap #2</u>: SSC boundaries, SSC failure modes and success criteria definitions should be established for failure rates and common cause failure parameters. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will use definitions for SSC boundary, unavailability boundary, failure mode, and success criteria consistently across the systems and data analyses.

<u>Gap #3</u>: Data calculations should be revised to group standby and operating component data. Group components by service condition to the extent supported by the data. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will include sensitivity studies to consider the impact of grouping data into operating vs. standby failure rates and by service condition.

Gap #4: As part of the Bayesian update process, checks are performed to assure that the posterior distribution is reasonable given the prior distribution and plant experience. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will verify that the Bayesian update process produces a reasonable posterior distribution.

Gap #5: Comparisons should be done of the component boundaries assumed for the generic common cause failure (CCF) estimates to those assumed in the PRA to ensure that these boundaries are consistent. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will ensure that CCF probabilities are consistent with component boundaries and plant experience.

Gap #6: Human reliability analysis should consider the potential for calibration errors. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will identify and consider the impact that equipment calibration errors could have on the results and conclusions.

Gap #7: Maintenance and calibration activities that could simultaneously affect equipment in either different trains of a redundant system or diverse systems should be identified. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will identify and consider the impact that equipment calibration errors could have on the results and conclusions.

Gap #8, #12: Mean values should be developed for pre- and post-initiator human error probabilities (HEPs). In response to the RAI, the licensee stated that each surveillance frequency change evaluation will use mean values for pre- and post-initiator HEPs.

Gap #9: When estimating HEPs, the impact of plant-specific and scenario-specific performance shaping factors should be considered and documented. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will use HEP values that have been quantified with consideration of plant-specific and scenario-specific performance shaping factors.

Gap #10, #11, #13: Human reliability analysis documentation should be enhanced to include time available to complete actions, a review of Human Failure Events (HFEs) and their final HEPs relative to each other, and appropriate credit if given for operator recovery actions. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will use HEP events with time available inputs based on plant-specific thermal hydraulic analyses; post-initiator HEPs will be reviewed against each other to check their reasonableness given the scenario context, plant procedures, operating practices and experience; and operator actions will only be credited if they are feasible.

Gap #14: The licensee identified twelve initiating event gaps to the supporting requirements for capability category II of the PRA standard. In response to the RAI, the licensee confirmed that no technical issues were identified for any of these supporting requirements but there remained a need to enhance the documentation. The licensee stated that the Catawba 1 and 2 initiating events analysis is revised with each PRA update to ensure that it remains consistent with industry operating experience as well as current plant design, operation and experience.

Furthermore, the licensee noted that a calculation was performed to address the initiating events supporting requirements. Each surveillance frequency change evaluation will review this calculation for potential impacts on the analysis. In addition, each surveillance frequency change will include a sensitivity analysis to determine the impact of the assumptions and sources of model uncertainty on the 5b analysis result.

Gap #15: Six internal flooding supporting requirements are not met in the Catawba 1 and 2 PRA. In response to the RAI, the licensee stated that a plan and schedule are in place for addressing internal flood issues related to the PRA standard for Catawba 1 and 2. In the interim, for each surveillance frequency change, all supporting requirements not meeting capability category II will be evaluated with sensitivity studies.

Gap #16: In crediting HFEs that support the accident progression analysis, explicitly model reactor coolant system depressurization for small loss-of-coolant accidents (LOCAs) and perform the dependency analysis on the HEPs. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will include a sensitivity study to assess the importance of explicitly modeling RCS depressurization for small LOCAs.

Gap #17, #20, #23, #25, #29: Collectively, these gaps identify deficiencies in the documentation process that do not directly affect the technical adequacy of the PRA model.

Gap #18, #19: Enhancement to the uncertainty analysis by use of a documented, systematic process to identify significant assumptions is recommended. In response to the RAI, the licensee stated that use of this application will include a sensitivity analysis for these gaps per NEI 04-10 if applicable to the specific surveillance test interval evaluation.

Gap #21: Documentation should include thermal hydraulic bases for all safety function success criteria for all initiating events. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will ensure that the success criteria address all initiators.

Gap #22: The acceptability of the results should be shown for the thermal hydraulic, structural, or other supporting engineering bases used to support the success criteria. In response to the RAI, the licensee stated that each surveillance frequency change evaluation will check and ensure the reasonableness and acceptability of the thermal hydraulic analyses result used to support the success criteria.

Gap #24, #27: System documentation should be enhanced to include an up-to-date system walkdown checklist and system engineer review for each system. In response to the RAI, the licensee stated that until each system notebook is updated, the impact of these gaps will be evaluated for each surveillance frequency change.

Gap #26: Quantitative evaluations should be provided for screening criteria associated with system unavailability and unreliability. In response to the RAI, the licensee stated that for each surveillance frequency change, the component and failure mode screening performed in the system analysis will be verified to meet the quantitative requirements provided in SY-A14.

Gap #28: A consideration of potential SSC failures due to adverse environmental conditions should be identified and documented. In response to the RAI, the licensee stated that for each surveillance frequency change, potential SSC failures due to adverse environmental conditions will be identified, included and documented in the analysis.

Based on the licensee's assessment using the applicable PRA standard and RG 1.200, 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, due to impacts from internal events, fires, seismic, other external events, and from shutdown conditions. Consideration is made of both CDF and LERF metrics. In cases where a PRA of sufficient scope or where quantitative risk models were unavailable, the licensee uses bounding analyses, or other conservative quantitative evaluations. A qualitative screening analysis may be used when the surveillance frequency impact on plant risk is shown to be negligible or zero.

Catawba 1 and 2's PRA includes a plant-specific seismic analysis and fire model. The current Catawba 1 and 2 seismic PRA model of record utilizes Seismic Margins Methodology and was recently updated as part of a revision. The fire PRA model is integrated into the overall PRA model, therefore; quantitative fire risk insights can be obtained. Both seismic and fire models use the same analysis and methodology as described in the original Individual Plant Examination for External Events (IPEEE). Furthermore, the licensee is planning to perform a self-assessment against the supporting requirements for both fire and seismic events of ASME/ANS PRA standard RA-Sa-2009, Addendum A to RA-S-2008 for Catawba 1 and 2 fire and seismic PRA. The licensee states that any deviations from ASME Standard Capability Category II requirements for each application of initiative 5b will be addressed.

The Catawba 1 and 2 PRA does not include an approved quantitative shutdown PRA model; therefore the licensee states that it will either 1) utilize the plant shutdown safety assessment tool developed to support implementation of NUMARC 91-06, or 2) perform an alternate qualitative risk evaluation process to assess the proposed surveillance frequency change.

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 will determine whether the SSCs affected by a proposed change to a surveillance frequency are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact may be carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted common cause failure modes, based on the proposed change to the surveillance frequency. Where the SSC is not modeled in the PRA, bounding analyses are performed to characterize the impact of the proposed change to 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.

The licensee will perform quantitative evaluations of the impact of selected testing strategy (i.e., staggered testing or sequential testing) consistent with the guidance of NUREG/CR-6141 and NUREG/CR-5497, as discussed in NEI 04-10.

Thus, through the application of NEI 04-10 the Catawba 1 and 2 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 the Catawba 1 and 2 PRA 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 surveillance frequency. This is consistent with RG 1.177, Section 2.3.3, which permits separation of the failure rate contributions into demand and standby for evaluation of surveillance requirements. If the available data do not support distinguishing between the time-related failures and demand failures, then the change to surveillance frequency is conservatively assumed to impact the total failure probability of the SSC, including both standby and demand contributions. The SSC failure rate (per unit time) is assumed to be unaffected by the change in test frequency, and 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 regard to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus the process is not reliant upon risk analyses as the sole basis for the proposed changes.

The potential beneficial risk impacts of reduced surveillance frequency, including reduced downtime, lesser potential for restoration errors, reduction of potential for test caused transients, and reduced test-caused wear of equipment, are identified qualitatively, but are conservatively not required to be quantitatively assessed. Thus, through the application of NEI 04-10, the licensee has employed reasonable assumptions with regard to extensions of surveillance test intervals, and its approach is consistent with Regulatory Position 2.3.4 of RG 1.177.

3.1.4.5 Sensitivity and Uncertainty Analyses

NEI 04-10 requires sensitivity studies to assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact to the frequency of initiating events, and of any identified deviations from capability category II of ASME PRA Standard ASME RA-Sb-2005. Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered, along with any qualitative considerations that may bear on the results of such sensitivity studies. Required monitoring and feedback of SSC performance, once the revised surveillance frequencies are implemented, will also be performed. Thus, through the application of NEI 04-10, 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 a surveillance frequency must show a risk impact below 1E-6 per year for a change to the CDF, and below 1E-7 per year for a change to the LERF. These criteria are consistent with the limits of RG 1.174 for very small changes in risk. Where the RG 1.174 limits are not met, the process either considers revised surveillance frequencies which are consistent with RG 1.174 or the process terminates without permitting the proposed changes. Where quantitative results are unavailable to permit comparison to acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or zero. Otherwise, bounding quantitative analyses are required which demonstrate the risk impact is at least one order of magnitude lower than the RG 1.174 acceptance guidelines for very small changes in risk. In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact below 1E-5 per year for a change to the CDF, and below 1E-6 per year for a change to the LERF, and the total CDF and the total LERF must be reasonably shown to be less than 1E-4 per year and 1E-5 per year, respectively. These are consistent with the limits of RG 1.174 for acceptable changes in risk, as referenced by RG 1.177 for changes to surveillance frequencies. The NRC staff interprets this assessment of cumulative risk as a requirement to calculate the change in risk from a baseline model utilizing failure probabilities based on the surveillance frequencies prior to implementation of the SFCP, compared to a revised model with failure probabilities based on changed surveillance frequencies. The NRC staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with 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 compared to numerical acceptance guidelines. Post implementation performance monitoring and feedback are also required to assure continued reliability of the components. The NRC staff concludes that the licensee's application of NEI 04-10 provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177. Therefore, the proposed change satisfies the fourth key safety principle of RG 1.177 by assuring that any increase in risk is small and consistent with the intent of "Use of Probabilistic Risk

Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," published in the *Federal Register* on August 16, 1995 (60 FR 42622).

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 requires performance monitoring of SSCs whose surveillance frequency has been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback includes consideration of maintenance rule monitoring of equipment performance. In the event of degradation of SSC performance, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions which may apply as part of the maintenance rule requirements. The NRC staff concludes that 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. Therefore, the proposed change meets the fifth key safety principle of RG 1.177.

3.2 Addition of Surveillance Frequency Control Program to TS Section 5

The proposed amendment would add the SFCP into the Administrative Controls Section of the Catawba 1 and 2 TSs. Specifically, new TS Section 5.5.17, "Surveillance Frequency Control Program," would read as follows:

> This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure that the associated Limiting Conditions for Operation are met.
>
> a. The Surveillance Frequency Control Program shall contain a list of Frequencies of the 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 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

The NRC staff concludes that the proposed addition to the Administrative Controls Section of the TSs adequately identifies the scope of the SFCP and defines the methodology to be used in a revision of surveillance frequencies. Therefore, the proposed TS change is acceptable.

3.3 Technical Evaluation Conclusion

The NRC staff has reviewed the licensee's proposed relocation of some surveillance frequencies to a new licensee-controlled program, the SFCP, and its proposal to control changes to surveillance frequencies in accordance with the new program. Based on the above considerations, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (75 FR 70034). The amendments also relate to changes in record-keeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusions set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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

7.0 REFERENCES

1. TSTF–425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control—RITSTF Initiative 5b," March 18, 2009 (ADAMS Accession No. ML090850642).

2. 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).

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

4. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," NRC, July 1998 (ADAMS Accession No. ML003740133).

5. RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," NRC, August 1998 (ADAMS Accession No. ML003740176).

6. RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," NRC, January 2007 (ADAMS Accession No. ML070240001).

7. ASME PRA Standard ASME RA–Sb–2005, "Addenda B to ASME RA–S–2002, 'Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,'" ASME, New York, New York, December 30, 2005.

8. NEI 00–02, Revision 1, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," May 19, 2006 (ADAMS Accession No. ML061510621).

9. NEI 05–04, Revision 0, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard," NEI, Washington, DC, January 2005.

Principal Contributor: J. Patel, NRR

Date: March 29, 2011

March 29, 2011

Mr. J. R. Morris
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING REVISION OF THE TECHNICAL SPECIFICATIONS TO RELOCATE SPECIFIC SURVEILLANCE FREQUENCIES TO A LICENSEE-CONTROLLED PROGRAM USING A RISK-INFORMED JUSTIFICATION (TSTF-425) (TAC NOS. ME3722 AND ME3723)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 263 to Renewed Facility Operating License NPF-35 and Amendment No. 259 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 31, 2010, as supplemented by letter dated November 30, 2010.

The amendments revise the Technical Specifications by relocating specific surveillance frequencies to a licensee-controlled document using a risk-informed justification.

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

If you have any questions, please call me at 301-415-1119.

Sincerely,

/RA/

Jon Thompson, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 263 to NPF-35
2. Amendment No. 259 to NPF-52
3. Safety Evaluation

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