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

May 22, 2019

ANO Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
N-TSB-58
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - ISSUANCE OF AMENDMENT NO. 264 RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-425, REVISION 3 (EPID L-2018-LLA-0063)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 264 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 12, 2018, as supplemented by letters dated April 26, October 17, and December 11, 2018.

The amendment revises the TSs by relocating certain surveillance frequencies to a licensee-controlled program, consistent with the NRC-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF [Risk-informed TSTF] Initiative 5b."

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

Sincerely,

April L. Pulvirenti for TJW

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 264 to DPR-51
2. Safety Evaluation

cc: Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 264
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (EOI, the licensee), dated March 12, 2018, as supplemented by letters dated April 26, October 17, and December 11, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Renewed Facility Operating License No. DPR-51 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 264, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

    Additionally, paragraph 2.c.(6) of Renewed Facility Operating License No. DPR-51 is amended to read as follows:

    (6) Surveillance Frequency Control Program

    The licensee shall implement the items listed in Table 2 of the enclosure to Entergy letter 1CAN121802, dated December 11, 2018, prior to implementation of the Surveillance Frequency Control Program.

3. This amendment is effective as of its date of issuance and shall be implemented according to License Condition 2.c.(6) of the Renewed Facility Operating License No. DPR-51.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-51 and Technical Specifications

Date of Issuance: May 22, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 264

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

Replace the following pages of the Renewed Facility Operating License No. DPR-51 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Operating License

REMOVE	INSERT
3	3
4	4

Technical Specifications

REMOVE	INSERT
1.1-5	1.1-5
3.1.1-1	3.1.1-1
3.1.2-2	3.1.2-2
3.1.4-2	3.1.4-2
3.1.5-2	3.1.5-2
3.1.6-1	3.1.6-1
3.1.7-1	3.1.7-1
3.1.8-2	3.1.8-2
3.1.8-3	3.1.8-3
3.1.9-2	3.1.9-2
3.2.1-2	3.2.1-2
3.2.2-1	3.2.2-1
3.2.3-1	3.2.3-1
3.2.4-3	3.2.4-3
3.3.1-2	3.3.1-2
3.3.1-3	3.3.1-3
3.3.1-4	3.3.1-4
---	3.3.1-5
3.3.3-2	3.3.3-2
3.3.4-3	3.3.4-3
3.3.5-2	3.3.5-2
3.3.6-2	3.3.6-2
3.3.7-1	3.3.7-1
3.3.8-1	3.3.8-1
3.3.8-2	3.3.8-2
3.3.9-2	3.3.9-2
3.3.10-1	3.3.10-1

Technical Specifications (continued)

REMOVE	INSERT
3.3.10-2	3.3.10-2
3.3.11-2	3.3.11-2
3.3.11-3	3.3.11-3
---	3.3.11-4
3.3.12-2	3.3.12-2
3.3.13-2	3.3.13-2
3.3.14-1	3.3.14-1
3.3.15-2	3.3.15-2
3.3.16-2	3.3.16-2
3.4.1-1	3.4.1-1
3.4.1-2	3.4.1-2
3.4.2-1	3.4.2-1
3.4.3-2	3.4.3-2
3.4.3-3	3.4.3-3
3.4.4-1	3.4.4-1
3.4.5-2	3.4.5-2
3.4.6-2	3.4.6-2
3.4.7-3	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
3.4.12-2	3.4.12-2
3.4.13-2	3.4.13-2
3.4.14-2	3.4.14-2
3.4.15-3	3.4.15-3
3.5.1-1	3.5.1-1
3.5.1-2	3.5.1-2
3.5.2-1	3.5.2-1
3.5.2-2	3.5.2-2
3.5.4-2	3.5.4-2
3.6.2-4	3.6.2-4
3.6.3-4	3.6.3-4
3.6.4-1	3.6.4-1
3.6.5-3	3.6.5-3
3.6.6-1	3.6.6-1
---	3.6.6-2
3.7.2-2	3.7.2-2
3.7.3-3	3.7.3-3
3.7.4-1	3.7.4-1
3.7.5-3	3.7.5-3
3.7.5-4	3.7.5-4
3.7.6-1	3.7.6-1
3.7.7-2	3.7.7-2
3.7.8-1	3.7.8-1
3.7.8-2	3.7.8-2
3.7.9-3	3.7.9-3
3.7.10-2	3.7.10-2

Technical Specifications (continued)

REMOVE	INSERT
3.7.11-1	3.7.11-1
3.7.11-2	3.7.11-2
3.7.13-1	3.7.13-1
3.7.14-1	3.7.14-1
3.8.1-4	3.8.1-4
3.8.1-5	3.8.1-5
3.8.1-6	3.8.1-6
---	3.8.1-7
3.8.2-3	3.8.2-3
3.8.3-2	3.8.3-2
3.8.4-2	3.8.4-2
3.8.6-3	3.8.6-3
---	3.8.6-4
3.8.7-2	3.8.7-2
3.8.8-2	3.8.8-2
3.8.9-2	3.8.9-2
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
5.0-7	5.0-7
5.0-9	5.0-9
5.0-11	5.0-11
5.0-17	5.0-17

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) EOI, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

c. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 264, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

(3) Safety Analysis Report

The licensee's SAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 14, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than May 20, 2014.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Physical Security Plan, Training and Qualifications Plan, and Safeguards Contingency Plan," as submitted on May 4, 2006.

EOI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The EOI CSP was approved by License Amendment No. 244 as supplemented by changes approved by License Amendment Nos. 247, 251, and 255.

(5) Implementation of the Improved Technical Specifications (ITS)

The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee controlled documents, as described in Table R, Relocated Specifications, and Table LA, Removal of Details, attached to the Safety Evaluation for Amendment No. 215. These requirements shall be relocated to the appropriate documents as part of the implementation of the ITS.

The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 215 shall be as follows:

1. For SRs that are new in this amendment, the first performance shall be due at the end of the first surveillance interval, which begins on the date of implementation of this amendment.

2. For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval shall begin upon completion of the first surveillance performed after implementation of this amendment.

3. For SRs that existed prior to this amendment that contained modified acceptance criteria, the performance shall be due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

4. For SRs that existed prior to this amendment whose interval of performance are being extended, the first extended surveillance interval shall begin upon completion of the last surveillance performed prior to the implementation of this amendment.

(6) Surveillance Frequency Control Program

The licensee shall implement the items listed in Table 2 of the enclosure to Entergy letter 1CAN121802, dated December 11, 2018, prior to implementation of the Surveillance Frequency Control Program.

(7) Deleted

1.1 Definition (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and c. There is no change in APSR position.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

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

APPLICABILITY: 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 greater than or equal to the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.2.1	-------------------------------NOTES------------------------- 1. 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. 2. This Surveillance is not required to be performed prior to entry into MODE 2. --- Verify measured core reactivity balance is within $\pm$ 1% $\Delta k/k$ of predicted values.	Once prior to entering MODE 1 after each fuel loading AND ---------NOTE-------- Only required after 60 EFPD ------------------------- In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2.3	--------------NOTE------------- Only required when THERMAL POWER is > 20% RTP. ----------------------------------- Perform SR 3.2.5.1.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours
C. More than one CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both.	C.1.1	Verify SDM to be within the limit provided in the COLR.	1 hour
	<u>OR</u>		
	C.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	C.2	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual CONTROL ROD positions are within 6.5% of their group average height.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.2	Verify CONTROL ROD freedom of movement for each individual CONTROL ROD that is not fully inserted.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each safety rod is fully withdrawn.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

LCO 3.1.6 Each APSR shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One APSR inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1	Perform SR 3.2.5.1.	2 hours <u>AND</u> 2 hours after each APSR movement
B. Require Action and associated Completion Time not met.	B.1	Be in MODE 3	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify position of each APSR is within 6.5% of the group average height.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Position Indicator Channels

LCO 3.1.7 One position indicator channel for each CONTROL ROD and APSR shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTES---

Separate Condition entry is allowed for each CONTROL ROD and APSR.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The required position indicator channel inoperable for one or more rods.	A.1	Declare the rod(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Perform CHANNEL CHECK of required position indicator channel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.2	Perform CHANNEL CALIBRATION of required position indicator channel.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. THERMAL POWER > 85% RTP. <u>OR</u> Nuclear overpower trip setpoint > 10% higher than PHYSICS TESTS power level. <u>OR</u> Nuclear overpower trip setpoint > 90% RTP. <u>OR</u> ------------NOTE------------- Only required when THERMAL POWER is > 20% RTP. --------------------------------- LHR not within limits.	B.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify THERMAL POWER is ≤ 85% RTP.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.1.8.2	-----NOTE----- Only required when THERMAL POWER is > 20% RTP. ----- Perform SR 3.2.5.1.	 In accordance with the Surveillance Frequency Control Program
SR 3.1.8.3	Verify nuclear overpower trip setpoint ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	Within 8 hours prior to performance of PHYSICS TESTS at each test plateau
SR 3.1.8.4	Verify SDM to be within the limits provided in the COLR.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Nuclear overpower trip setpoint is not within limit. <u>OR</u> Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit inoperable.	C.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify THERMAL POWER is ≤ 5% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.1.9.2	Verify nuclear overpower trip setpoint is ≤ 5% RTP.	Within 8 hours prior to performance of PHYSICS TESTS
SR 3.1.9.3	Verify SDM to be within the limit provided in the COLR.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Regulating rod groups inserted in unacceptable operation region.	D.1	Initiate boration to restore SDM to within the limit provided in the COLR.	15 minutes
	AND		
	D.2.1	Restore regulating rod groups to within restricted operation region.	2 hours
	OR		
	D.2.2	Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits.	2 hours
E. Required Actions and associated Completion Times of Conditions C or D not met.	E.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.2.1.2	Verify regulating rod groups meet the insertion limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.2.1.3	Verify SDM ≥ 1% Δk/k.	Within 4 hours prior to achieving criticality

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

LCO 3.2.2 APSRs shall be positioned within the limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. APSRs not within limits.	A.1 -------------NOTE--------------- Only required when THERMAL POWER is > 20% RTP. ------------------------------------ Perform SR 3.2.5.1.	Once per 2 hours
	AND	
	A.2 Restore APSRs to within limits.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify APSRs are within acceptable limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.3 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AXIAL POWER IMBALANCE not within limits.	A.1 Perform SR 3.2.5.1. <u>AND</u> A.2 Reduce AXIAL POWER IMBALANCE to within limits.	Once per 2 hours 24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 40% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify QPT is within limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Open all control rod drive (CRD) trip breakers.	6 hours
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1	Open all CRD trip breakers.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.1-1.	F.1	Reduce THERMAL POWER < 45% RTP.	6 hours
G. As required by Required Action C.1 and referenced in Table 3.3.1-1.	G.1	Reduce THERMAL POWER < 10% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	------------------------------NOTES------------------------------ 1. Adjust power range channel output if the absolute difference is > 2% RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. --- Compare results of calorimetric heat balance calculation to power range channel output.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP
SR 3.3.1.3	------------------------------NOTES------------------------------ 1. Adjust the power range channel imbalance output if the absolute value of the imbalance error is ≥ 2% RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. --- Compare results of out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	------------------------------NOTE------------------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

Table 3.3.1-1
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower –				
a. High Setpoint	1,2[(a)],3[(d)]	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5	≤ 104.9% RTP
b. Low Setpoint	2[(b)],3[(b)] 4[(b)],5[(b)]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 618 °F
3. RCS High Pressure	1,2[(a)],3[(d)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 2355 psig
4. RCS Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 1800 psig
5. RCS Variable Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
6. Reactor Building High Pressure	1,2,3[(c)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 18.7 psia
7. Reactor Coolant Pump to Power	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 55% RTP with one pump operating in each loop.
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
9. Main Turbine Trip (Oil Pressure)	≥ 45% RTP	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 40.5 psig
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ 10% RTP	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 55.5 psig
11. Shutdown Bypass RCS High Pressure	2[(b)],3[(b)] 4[(b)],5[(b)]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 1720 psig

(a) When not in shutdown bypass operation.
(b) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.
(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.
(d) With any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more RTMs inoperable in MODE 4 or 5. OR Required Action and associated Completion Time not met in MODE 4 or 5.	C.1	Open all CRD trip breakers.	6 hours
	OR		
	C.2	Remove power from all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

LCO 3.3.7 The ESAS digital actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each digital actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital actuation logic channels inoperable.	A.1 Place associated component(s) in engineered safeguards configuration.	1 hour
	<u>OR</u>	
	A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.8 Diesel Generator (DG) Loss of Power Start (LOPS)

LCO 3.3.8 Two loss of voltage Function relays and two degraded voltage Function relays DG LOPS instrumentation per DG shall be OPERABLE.

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more relays for one or more DGs inoperable.	A.1 Restore relay(s) to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Declare affected DG(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.8.2	------------------------------NOTE------------------------------ When DG LOPS instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed up to 4 hours for the loss of voltage Function, provided the one remaining relay monitoring the Function for the bus is OPERABLE. ---	
	Perform CHANNEL CALIBRATION with setpoint Allowable Value as follows: a. Degraded voltage ≥ 423.2 V and ≤ 436.0 V with a time delay of 8 seconds ± 1 second; and b. Loss of voltage ≥ 1600 V and ≤ 3000 V with a time delay of ≥ 0.30 seconds and ≤ 0.98 seconds.	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	-----------------------------NOTE---------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	 In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One intermediate range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODE 2
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	-------------------NOTE------------------- Plant temperature changes are allowed provided the temperature change is accounted for in the SDM calculations. ---	
	A.1 Suspend operations involving positive reactivity changes.	Immediately
	AND	
	A.2 Open CRD trip breakers.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.10.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.10.3	-----------------------------NOTE----------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met for Function 1.a or 1.d.	E.1 Reduce THERMAL POWER to ≤ 10% RTP.	6 hours
F. Required Action and associated Completion Time not met for Functions 1.c, 2, or 3.	F.1 Be in MODE 3. AND F.2 Reduce steam generator pressure to < 750 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.

SURVEILLANCE		FREQUENCY
SR 3.3.11.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.11.2	Perform CHANNEL FUNCTIONAL TEST.(Notes 1 & 2)	In accordance with the Surveillance Frequency Control Program
SR 3.3.11.3	Perform CHANNEL CALIBRATION.(Notes 1 & 2)	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

The following notes apply only to the SG Level – Low function:

Note 1: If the as-found channel setpoints are conservative with respect to the Allowable Value but outside their predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoints are not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Note 2: The instrument channel setpoint(s) shall be reset to a value that is equal to or more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint and the predefined as-found acceptance criteria band are specified in the Bases.

Table 3.3.11-1
Emergency Feedwater Initiation and Control System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUES
1. EFW Initiation				
a. Loss of MFW Pumps (Control Oil Pressure)	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 55.5 psig
b. SG Level - Low	1,2,3	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 9.34 inches(c,d)
c. SG Pressure - Low	1,2,3(a)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
d. RCP Status	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2	NA
2. EFW Vector Valve Control				
a. SG Pressure – Low	1,2,3(a)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
b. SG Differential Pressure – High	1,2,3(a)	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≤ 150 psid
3. Main Steam Line Isolation				
a. SG Pressure – Low	1,2,3(a)(b)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig

(a) When SG pressure ≥ 750 psig.

(b) Except when all associated valves are closed and deactivated.

(c) The SG Level – Low "Limiting Trip Setpoint" in accordance with NRC letter dated September 7, 2005, *Technical Specification For Addressing Issues Related To Setpoint Allowable Values*, is ≥ 10.42 inches.

(d) Includes an actuation time delay of ≤ 10.4 seconds.

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met for EFW Initiation Function.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours
E. Required Action and associated Completion Time not met for Main Steam Line Isolation Function.	E.1 Be in MODE 3. <u>AND</u> E.2.1 Reduce steam generator pressure to < 750 psig. <u>OR</u> E.2.2 Close and deactivate all associated valves.	6 hours 12 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.12.1 Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.14 Emergency Feedwater Initiation and Control (EFIC) Vector Logic.

LCO 3.3.14 Four channels of the EFIC vector logic shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 when steam generator pressure is ≥ 750 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vector logic channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Reduce steam generator pressure to < 750 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.14.1 Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------
These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.15.2	---------------NOTE--------------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------------------------------- Perform CHANNEL CALIBRATION.	 In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.16.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.16.2	------------------------------NOTE------------------------------ When the Control Room Isolation – High Radiation instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 3 hours. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.16.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters (loop pressure, hot leg temperature, and RCS total flow rate) shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

--NOTE--------------------------------------
RCS loop pressure limit does not apply during pressure transients due to a THERMAL POWER change > 5% RTP per minute.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	--------------------------------NOTE-------------------------- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. -- Verify RCS loop pressure is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	--------------------------------NOTE-------------------------- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. -- Verify RCS hot leg temperature is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	Verify RCS total flow is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.4	--------------------------------NOTE-------------------------- Only required to be performed when stable thermal conditions are established at ≥ 90% RTP. -- Verify RCS total flow rate is within the limit specified in the COLR by measurement.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 The RCS average temperature (T_{avg}) shall be ≥ 525 °F.

APPLICABILITY: MODE 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} not within limit.	A.1 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} ≥ 525 °F.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. -------------NOTE------------- Required Action D.2 shall be completed whenever this Condition is entered. --------------------------------- Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	D.1	Initiate action to restore parameter(s) to within limit.	Immediately
	AND		
	D.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 operations with fuel in the reactor vessel. --- Verify RCS pressure, RCS temperature, and RCS heatup rates are within the limits specified in Figure 3.4.3-1.	In accordance with the Surveillance Frequency Control Program
SR 3.4.3.2	-----------------------------NOTE---------------------------- Only required to be performed during RCS cooldown operations with fuel in the reactor vessel. --- Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-2.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.4.3.3	---NOTE--- Only required to be performed during RCS heatup and cooldown operations with no fuel in the reactor vessel. --- Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-3.	In accordance with the Surveillance Frequency Control Program
SR 3.4.3.4	---NOTE--- Only required to be performed during PHYSICS TESTS with RCS temperature ≤ 525 °F. --- Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.3-1.	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 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and THERMAL POWER restricted as specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCP not in operation in each loop.	A.1 Restore one non-operating RCP to operation.	18 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two RCS loops inoperable. <u>OR</u> Required RCS loop not in operation.	C.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u>		
	C.2	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.2	-------------------------------NOTE--------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power available to each required pump.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable. OR Required loop not in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. AND	Immediately
	B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required DHR or RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	------------------------------NOTE---------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power available to each required pump.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.2	Verify required SG secondary side water levels are ≥ 20 inches.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.3	-------------------------------NOTE----------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power available to each required DHR pump.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No required DHR loop OPERABLE. <u>OR</u> Required DHR loop not in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u>	
	B.2 Suspend all operations involving reduction in RCS water volume.	Immediately
	<u>AND</u>	
	B.3 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.2	-------------------------------NOTE---------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power available to each required DHR pump.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.9.1	Verify pressurizer water level ≤ 320 inches.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify capacity of ES bus powered pressurizer heaters ≥ 126 kW.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify pressurizer level does not represent a water solid condition.	30 minutes during RCS heatup and cooldown AND In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2	Verify HPI is deactivated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.3	Verify each pressurized CFT is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.4	-------------------------------NOTE---------------------------- Verification of locked, sealed, or otherwise secured open vent path(s) only required to be performed every 31 days. --- Verify OPERABLE pressure relief capability.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.5	Perform CHANNEL CALIBRATION of ERV opening circuitry.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	-------------------------------NOTE---------------------------- Only required to be performed in MODE 1 and 2, MODE 3 with RCS average temperature ≥ 500 °F. -- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 2200 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program

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 at or near operating pressure. 2. Not applicable to primary to secondary LEAKAGE. ---	
	Verify RCS operational LEAKAGE is within limits by performance of an 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. ---	
	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	-----------------------------NOTE----------------------------- Not required to be performed in MODES 3 and 4. -- Verify leakage from each RCS pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent of the Allowable Leakage Limit identified below at a differential test pressure ≥ 150 psid. Pressure Isolation <u>Check Valve(s)</u> — Allowable <u>Leakage Limit</u> DH-14A — ≤ 5 gpm DH-13A and DH-17 — ≤ 5 gpm total DH-14B — ≤ 5 gpm DH-13B and DH-18 — ≤ 5 gpm total	In accordance with the INSERVICE TESTING PROGRAM <u>AND</u> Once 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
SR 3.4.14.2	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal.	In accordance with the Surveillance Frequency Control Program
SR 3.4.14.3	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal: c. ≤ 340 psig for one valve; and d. ≤ 400 psig for the other valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.14.4	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	In accordance with the Surveillance Frequency Control Program
SR 3.4.14.5	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CALIBRATION of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL CALIBRATION of required reactor building sump monitor.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Two CFTs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to $\leq$ 800 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each CFT isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.5.1.2	Verify borated water volume in each CFT is $\geq 970\ ft^3$ and $\leq 1110\ ft^3$.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is ≥ 560 psig and ≤ 640 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each CFT is ≥ 2270 ppm.	In accordance with the Surveillance Frequency Control Program <u>AND</u> ---------NOTE------- Only required to be performed for affected CFT ------------------------ Once within 12 hours after each solution level increase of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration ≥ 2270 ppm
SR 3.5.1.5	Verify power is removed from each CFT isolation valve operator.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Reduce RCS temperature to ≤ 350 °F.	6 hours 12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	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

SURVEILLANCE		FREQUENCY
SR 3.5.2.2	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
SR 3.5.2.3	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 3.5.2.4	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.5	Verify, by visual inspection, each ECCS train reactor building sump suction inlet is not restricted by debris and screens show 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	------------------------------NOTE---------------------------- Only required to be performed when ambient air temperature is < 40 °F or > 110 °F. --- Verify BWST borated water temperature is ≥ 40 °F and ≤ 110 °F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify BWST borated water level is ≥ 38.4 feet and ≤ 42 feet.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify BWST boron concentration is ≥ 2270 ppm and ≤ 2670 ppm.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	-----------------------------NOTE--------------------------- 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 Reactor Building Leakage Rate Testing Program.	In accordance with the Reactor Building Leakage Rate Testing Program
SR 3.6.2.2	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 reactor building purge isolation valve is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	-----------------------------NOTE-------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	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 means. --- Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building 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 3.6.3.4	Verify the isolation time of each automatic power operated reactor building isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic reactor building 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

3.6 REACTOR BUILDING SYSTEMS

3.6.4 Reactor Building Pressure

LCO 3.6.4 Reactor building pressure shall be ≥ -1.0 psig and ≤ +3.0 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building pressure not within limits.	A.1 Restore reactor building pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ---------------NOTE-------------- LCO 3.0.4.a is not applicable when entering Mode 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify reactor building pressure is ≥ -1.0 psig and ≤ +3.0 psig.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify each reactor building spray manual, power operated, and automatic valve in each required 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 3.6.5.2	Operate each required reactor building cooling train fan unit for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.3	Verify each required reactor building cooling train cooling water flow rate is ≥ 1200 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.4	Verify each required reactor building 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 3.6.5.5	Verify each automatic reactor building spray valve in each required 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 3.6.5.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.8	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

3.6 REACTOR BUILDING SYSTEMS

3.6.6 Spray Additive System

LCO 3.6.6 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each Spray Additive System 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 3.6.6.2	Verify sodium hydroxide tank solution volume is $\geq$ 9000 gallons.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.6.6.3	Verify sodium hydroxide tank solution concentration is > 6.0 wt% and < 8.5 wt.% NaOH.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify each Spray Additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify isolation time of each MSIV is within the limits specified in the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	-------------------------------NOTE---------------------------- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.7.3.2	-------------------------------NOTES-------------------------- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. ---	
	Verify that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be ≤ 0.1 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. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant is ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW 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 EFW pump, until 24 hours after reaching ≥ 750 psig in the steam generators. --- Verify the developed head of each EFW 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 required to be met in MODE 4 when steam generator is relied upon for heat removal. --- Verify each EFW 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
SR 3.7.5.4	-----------------------------NOTE---------------------------- Not required to be met in MODE 4 when steam generator is relied upon for heat removal. --- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying manual valve alignment from the "Q" condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days
SR 3.7.5.6	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.6 Q Condensate Storage Tank (QCST)

LCO 3.7.6 The QCST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The QCST inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore QCST to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4 without reliance on steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify QCST volume is ≥ 267,000 gallons when required for both units and ≥ 107,000 gallons when only required for Unit 1.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	-----------------------------NOTE--------------------------- Isolation of SWS flow to individual components does not render the SWS inoperable. --- Verify each SWS 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 SWS 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 3.7.7.3	Verify each required SWS pump starts automatically on an actual or simulated signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.8 Emergency Cooling Pond (ECP)

LCO 3.7.8 The ECP shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Degradation of the ECP noted pursuant to SR 3.7.8.4 below or by other inspection.	A.1 Determine ECP remains acceptable for continued operation.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR LCO not met for reasons other than Condition A.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify that the indicated water level of the ECP is greater than or equal to that required for an ECP volume of 70 acre-ft.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.7.8.2	------------------------------NOTE---------------------------- Only required to be performed from June 1 through September 30. --- Verify average water temperature is ≤ 100 °F.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Perform soundings of the ECP to verify: 1. A contained water volume of ECP ≥ 70 acre-feet, and 2. The minimum indicated water level needed to ensure a volume of 70 acre-feet is maintained.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.4	Perform visual inspection of the ECP to verify conformance with design requirements.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each CREVS train for $\geq$ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify the CREVS automatically isolates the Control Room and switches into a recirculation mode of operation on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.4	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.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify each CREACS train starts, operates for at least 1 hour, and maintains control room air temperature ≤ 84 °F D. B.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Verify system flow rate of 9900 cfm ± 10%.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.11 Penetration Room Ventilation System (PRVS)

LCO 3.7.11 Two PRVS trains shall be OPERABLE.

---NOTE---
The penetration room negative pressure boundary may be opened intermittently under administrative controls.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PRVS train inoperable.	A.1 Restore PRVS train to OPERABLE status.	7 days
B. Two PRVS trains inoperable due to inoperable penetration room negative pressure boundary.	B.1 Restore penetration room negative pressure boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met. <u>OR</u> Both PRVS trains inoperable for reasons other than Condition B.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each PRVS train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.11.2	Perform required PRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.11.3	Verify each PRVS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Water Level.

LCO 3.7.13 The spent fuel pool water level shall be ≥ 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 3.7.13.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Boron Concentration

LCO 3.7.14 The spent fuel pool boron concentration shall be > 2000 ppm.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. 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 3.7.14.1	Verify the spent fuel pool boron concentration is > 2000 ppm.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 ---------------NOTE-------------- LCO 3.0.4.a is not applicable when entering Mode 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	-----------------------------NOTE---------------------------- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. -- Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves "ready-to-load" conditions.	 In accordance with the Surveillance Frequency Control Program

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 follow, without shutdown, a successful performance of SR 3.8.1.2. --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify each day tank contains ≥ 160 gallons 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 tanks to the day tank.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	-----------------------------NOTE---------------------------- This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. --- Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.8	-----------------------------NOTE---------------------------- All DG starts may be preceded by an engine prelube period. --- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in $\leq$ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for $\geq$ 5 minutes.	In accordance with the Surveillance Frequency Control Program

	SURVEILLANCE	FREQUENCY
SR 3.8.1.9	------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. -- 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. achieves "ready-to-load" conditions in $\leq$ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected emergency loads through load sequencing timers, and 4. supplies connected loads for $\geq$ 5 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	-----------------------------NOTES---------------------------- 1. SR 3.8.1.3 is not required to be performed. 2. The 15 second acceptance criteria of SR 3.8.1.2 is not applicable. ---	
	For AC Sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources – Operating," except SR 3.8.1.4, SR 3.8.1.7, SR 3.8.1.8, and SR 3.8.1.9, are applicable.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met. <u>OR</u> One or more DGs with diesel fuel oil or required starting air subsystem not within limits for reasons other than Condition A, B, C, or D.	E.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains $\geq$ 20,000 gallons of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	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 3.8.3.3	Verify each DG required air start receiver pressure is $\geq$ 175 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.4	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. <u>OR</u> Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	-------------------------------NOTE---------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. -- Verify 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 or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	------------------------------NOTE--------------------------- Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. --- Verify each battery float current is ≤ 2 amps.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.2	Verify each battery pilot cell float voltage is ≥ 2.07 V.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.8.6.6	-------------------------------NOTE---------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is ≥ 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 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 ---------------NOTE-------------- LCO 3.0.4.a is not applicable when entering Mode 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours
C. Two or more of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to associated 120 VAC buses RS1, RS2, RS3, and RS4.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.5 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by AC vital bus inverter(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage and alignments to required 120 VAC vital buses.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.4 Initiate actions to restore required AC, DC, and 120 VAC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.5 Declare associated required decay heat removal subsystem(s) inoperable.	Immediately
	AND	
	A.2.6 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by Electrical Power Distribution System.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments to required AC, DC, and 120 VAC 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 and the refueling canal shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

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

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 the 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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.3.1	Verify each required reactor building penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.2	-------------------------------NOTE------------------------------ Not required to be met for reactor building isolation valves and reactor building purge isolation valves in penetrations closed to comply with LCO c.1. --- Verify each required reactor building isolation valve and each reactor building purge isolation valve actuates to the isolation position.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.3	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No DHR loop OPERABLE or in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	B.2 Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately
	AND	
	B.3 Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to each required DHR pump.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling canal water level is ≥ 23 feet above the top of irradiated fuel assemblies seated within the reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program

5.0 ADMINSTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at a frequency in accordance with the Surveillance Frequency Control Program. The provisions of SR 3.0.2 are applicable.

5.5.3 Iodine Monitoring

This program provides controls that ensure the capability to accurately determine the airborne iodine concentration under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for monitoring; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming 10 CFR 20, Appendix B, Table II, Column 2;

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.5 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 Emergency Ventilation System (CREVS), 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 CREVS, operating at the flow rate required by the VFTP, at a Frequency in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the CRE boundary assessment specified in TS 5.5.5.c.

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.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.5.8 Surveillance Frequency Control Program

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. water and sediment within limits;

b. Within 31 days following addition of new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested based on ASTM D-2276, Method A-2 or A-3 at a Frequency in accordance with the Surveillance Frequency Control Program; and

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance Frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

   Proposed changes that do meet these criteria shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the SAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 264 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated March 12, 2018 (Reference 1), as supplemented by letters dated April 26 (Reference 2), October 17 (Reference 3), and December 11 (Reference 4), 2018, Entergy Operations, Inc. (Entergy, the licensee) requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1), which are contained in Appendix A of Renewed Facility Operating License No. DPR-51.

The proposed changes would revise the ANO-1 TSs to adopt the U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications (STS) Change Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk-Informed TSTF] Initiative 5b" (Reference 5).

The supplemental letters dated October 17 and December 11, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on June 5, 2018 (83 FR 26102).

2.0 REGULATORY EVALUATION

2.1 Description of the Proposed Changes

The licensee proposed to modify the ANO-1 TSs by relocating certain surveillance frequencies to a licensee-controlled program (i.e., the Surveillance Frequency Control Program (SFCP)) in accordance with Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies" (Reference 6). The licensee stated that the proposed change is consistent with the adoption of NRC-approved TSTF-425, Revision 3. When implemented, TSTF-425 relocates most periodic frequencies of TS surveillances to the SFCP and provides requirements for the new SFCP in the Administrative Controls section of the TSs. All surveillance frequencies can be relocated except the following:

- Frequencies that reference other approved programs for the specific interval;
- Frequencies that are purely event-driven;
- Frequencies that are event-driven, but have a time component for performing the surveillance on a one-time basis once the event occurs; and
- Frequencies that are related to specific conditions or conditions for the performance of a surveillance requirement.

The licensee proposed to relocate certain surveillance frequencies from the following TS sections to the SFCP:

3.1 Reactivity Control Systems
3.2 Power Distribution Limits
3.3 Instrumentation
3.4 Reactor Coolant System (RCS)
3.5 Emergency Core Cooling Systems (ECCS)
3.6 Reactor Building Systems
3.7 Plant Systems
3.8 Electrical Power Systems
3.9 Refueling Operations
5.5 Programs and Manuals

The licensee proposed to add the SFCP to ANO-1 TS, Section 5.0, "Administrative Controls." Proposed TS 5.5.8, "Surveillance Frequency Control Program," would describe the requirements for the SFCP to control changes to the relocated surveillance frequencies to ensure that surveillances are performed at intervals to ensure that limiting conditions for operation (LCOs) are met. The TS Bases for each affected surveillance would be revised to state that the surveillance frequency is controlled under the SFCP and were included in the application for information only. The proposed changes to the Administrative Controls section of the TSs include a specific reference to NEI 04-10, Revision 1, as the basis for making any changes to the surveillance frequencies when they are relocated out of the TSs.

In a letter dated September 19, 2007 (Reference 7), the NRC staff approved NEI 04-10, Revision 1, as an acceptable methodology for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04-10, Revision 1, and the safety evaluation (SE) providing the basis for NRC acceptance of NEI 04-10, Revision 1.

The licensee proposed other changes and deviations from TSTF-425, which are discussed in Section 3.3 of this SE.

2.2 Applicable Commission Policy Statements

In the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993 (58 FR 39132), the NRC addressed the use of probabilistic safety analysis (PSA, currently referred to as probabilistic risk assessment or PRA) in STS. In this 1993 publication, the NRC states, in part:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36][1] 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][2] 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."...
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

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

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach....
>
> PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner....

---

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

[2] The *Federal Register* Notice 58 FR 39135 (Alteration in Original) explains the brackets.

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

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

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

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

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

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

2.3 Applicable Regulations

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

Paragraph 50.36(c)(3) of 10 CFR states, "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The FR notice published on July 6, 2009 (74 FR 31996), which announced the availability of TSTF-425, Revision 3, states that the addition of the SFCP to the TSs provides the necessary administrative controls to require that surveillance frequencies relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met. The FR notice also states that changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, Revision 1, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and are required to be documented.

Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. Such failures can result in the licensee increasing the frequency of a surveillance test. In addition, by having the TSs require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04-10, Revision 1, the licensee will be required to monitor the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs.

2.4 Applicable NRC Regulatory Guides and Review Plans

Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018 (Reference 8), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated May 2011 (Reference 9), describes an acceptable risk-informed approach specifically for assessing proposed TS changes.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (Reference 10), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors (LWRs).

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (Reference 11), provides general guidance for evaluating the technical basis for proposed risk-informed changes. Guidance on evaluating PRA technical adequacy is provided in NUREG-0800, Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," dated September 2012 (Reference 12). More specific guidance related to risk-informed TS changes is provided in NUREG-0800, Chapter 16, Section 16.1, Revision 1, "Risk-Informed Decision Making: Technical Specifications," dated March 2007 (Reference 13), which includes changes to surveillance test intervals (STIs) (i.e., surveillance frequencies) as part of risk-informed decisionmaking. Section 19.2 of NUREG-0800 references the same criteria as

RG 1.174, Revision 3, and RG 1.177, Revision 1, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

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

The regulatory requirements in 10 CFR 50.65 and 10 CFR Part 50, Appendix B, Criterion XVI, and the performance monitoring required by NEI 04-10, Revision 1, ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified, and appropriate corrective actions taken.

NUREG-1430, Revision 4.0, "Standard Technical Specifications, Babcock and Wilcox Plants," Volume 1, Specifications, and Volume 2, Bases (Reference 14), contains the improved STS for Babcock and Wilcox plants. The improved STS were developed based on the criteria in the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132), which was subsequently codified by changes to 10 CFR 50.36 (60 FR 36953).

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425, Revision 3, provides for administrative relocation of applicable surveillance frequencies to the SFCP and provides for the addition of the SFCP to the Administrative Controls section of the TSs. The changes to the Administrative Controls section of the TSs will also require the application of NEI 04-10, Revision 1, for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes described in TSTF-425, Revision 3, includes documentation regarding the PRA technical adequacy, consistent with the requirements of RG 1.200, Revision 2. NEI 04-10, Revision 1, states that PRA methods are used with plant performance data and other considerations to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is consistent with guidance provided in RG 1.174, Revision 3, and RG 1.177, Revision 1, in support of changes to STIs.

3.1 Key Safety Principles

RG 1.177, Revision 1, identifies five key safety principles required for risk-informed changes to TSs. Each of these principles is addressed by NEI 04-10, Revision 1. Sections 3.1.1 through 3.1.5 of this SE provide a discussion of the five principles, including the NRC staff's evaluation of how the licensee's license amendment request (LAR) satisfies each principle.

3.1.1 The Proposed Change Meets Current Regulations

Paragraph 50.36(c)(3) of 10 CFR requires that TSs include surveillances, which are "requirements relating to test, calibration, or inspection to assure that necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The licensee is required by its TSs to perform surveillance tests, calibration, or inspection on specific safety-related equipment (e.g., reactivity control, power distribution, electrical, and instrumentation) to verify system operability. Surveillance frequencies are based primarily upon deterministic methods, such as engineering judgment, operating experience, and manufacturer's recommendations. The licensee's use of NRC-approved methodologies identified in NEI 04-10, Revision 1, provides a way to establish risk-informed surveillance frequencies that complements the deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

The SRs themselves would remain in the TSs as required by 10 CFR 50.36(c)(3). The requested change is analogous with other NRC-approved TS changes in which the SRs are retained in the TSs, but the related surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the Inservice Testing Program and the Reactor Building Leakage Rate Testing Program. Thus, this proposed change complies with 10 CFR 50.36(c)(3) by retaining the requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components for operation will be met.

The regulatory requirements in 10 CFR 50.65 and 10 CFR Part 50, Appendix B, and the monitoring required by NEI 04-10, Revision 1, ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified, and appropriate corrective actions taken. The licensee's SFCP ensures that SRs specified in the TSs are performed at intervals sufficient to assure that the above regulatory requirements are met. Based on the foregoing, the NRC staff concludes that the proposed change meets the first key safety principle of RG 1.177, Revision 1, by complying with current regulations.

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

The defense-in-depth philosophy (i.e., the second key safety principle of RG 1.177, Revision 1), is maintained if:

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

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

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

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

- independence of barriers is not degraded.
- defenses against human errors are preserved.
- the intent of the General Design Criteria in 10 CFR Part 50, Appendix A, is maintained.

The changes to the Administrative Controls section of the TSs will require the application of NEI 04-10, Revision 1, for any changes to surveillance frequencies within the SFCP. NEI 04-10, Revision 1, uses both the CDF and the LERF metrics to evaluate the impact of proposed changes to surveillance frequencies. In accordance with RG 1.174, Revision 3, and RG 1.177, Revision 1, changes to the CDF and LERF are evaluated using a comprehensive risk analysis, which assesses the impact of proposed changes, including contributions from human errors and CCFs. 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 CCFs. The NRC staff concludes that both the quantitative risk analysis and the qualitative considerations provide reasonable assurance that defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177, Revision 1.

3.1.3 The Proposed Change Maintains Sufficient Safety Margins

The engineering evaluation that will be conducted by the licensee under the SFCP, when frequencies are revised, will assess the impact of the proposed frequency change to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring that 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 bases (including the Safety Analysis Report and the TS Bases), because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. On this basis, the NRC staff concludes that safety margins are maintained by the proposed methodology and that, therefore, the third key safety principle of RG 1.177, Revision 1, is satisfied.

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

The guidance in RG 1.177, Revision 1, provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies, which 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. The changes to the Administrative Controls section of the TSs will require application of NEI 04-10, Revision 1, in the SFCP. The guidance in NEI 04-10, Revision 1, satisfies the intent of RG 1.177, Revision 1, guidance for evaluation of the change in risk and for assuring that such changes are small by providing the technical methodology to support risk-informed TSs for control of surveillance frequencies.

3.1.4.1 Quality of the PRA

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

RG 1.200 provides regulatory guidance for assessing the technical adequacy of a PRA. The current revision (i.e., Revision 2) of this RG endorses, with clarifications and qualifications, the use of the following:

1. American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (hereafter referred to as the ASME/ANS PRA Standard) (Reference 15);

2. NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance" (Reference 16); and

3. NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard" (Reference 17).

The licensee performed an assessment of the PRA models used to support the SFCP using the guidance of RG 1.200, Revision 2, to ensure that the PRA models are capable of determining the change in risk due to changes to surveillance frequencies of SSCs, using plant-specific data and models. Capability Category II of the NRC-endorsed ASME/ANS PRA Standard is the target capability level for supporting requirements for the internal events PRA for this application. Any identified deficiencies to those requirements are further assessed to determine any impacts to proposed decreases to surveillance frequencies, including the use of sensitivity studies where appropriate, in accordance with NEI 04-10, Revision 1.

Internal Events

In Section 3.2.3, "Consistency with Applicable PRA Standards," of Attachment 2 to the LAR (Reference 1), the licensee stated that the latest full-scope peer review for the Internal Events PRA (IEPRA) was conducted in August 2009 by the Pressurized Water Reactor Owners Group using RG 1.200, Revision 2. In 2016, the ANO-1 PRA Model of Record (MOR), Revision 5, was completed, which ensured that all the significant facts and observations (F&Os) from the peer review were addressed. The licensee stated that all the F&Os are captured and documented in its Model Change Requests (MCRs) database and provided them in Tables 2 and 3 of Attachment 2 to the LAR.

The NRC staff reviewed the summary of the peer review findings and the licensee's resolution or assessment of the impact on this application to determine whether any gaps in the PRA model were identified that could impact the application and to ensure that any gaps in meeting CC-II can be addressed for the SFCP per the NEI 04-10, Revision 1, methodology, consistent with RG 1.200, Revision 2.

For F&O MCR A1-3893, the peer review team observed that the licensee's approach produced probabilities rather than frequencies for basic events values within support system initiating event fault trees (IEFTs). The licensee's disposition to F&O MCR A1-3893 states that resolution of the F&O is only a documentation issue. The licensee also stated that the IEFT models were extensively revised and that a very small change to the results is expected. In request for additional information (RAI) PRA-RAI-01.b, by e-mail dated September 19, 2018 (Reference 18), the NRC staff requested the licensee to summarize the revisions that were made to the IEFT models and to explain how the model produced a frequency, or, alternatively, to propose a mechanism to incorporate the appropriate method prior to implementation of the SFCP. In response to RAI-PRA-01.b, by letter dated October 17, 2018 (Reference 3), the licensee stated that the IEFT models have been updated in the MOR, Revision 5, to produce a frequency consistent with the Electric Power Research Institute (EPRI) topical report (TR) 1016741, "Support System Initiating Events: Identification and Quantification Guideline" (Reference 19). The licensee further stated that the MOR, Revision 5, still contains IEFT events with probabilities; however, all the initiators have been verified for the MOR, Revision 6. Because the licensee confirmed that it revised the calculation method to produce more appropriate results in terms of annual failure rate, the NRC staff finds this resolution acceptable for the application.

RG 1.200, Revision 2, provides guidance for determining the technical adequacy of IEPRAs by comparing the PRA to the relevant parts of the ASME/ANS PRA Standard using a peer review process. The NRC staff has reviewed the peer review results, dispositions, and RAI response, and finds that the quality and level of detail of the IEPRA is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1, "Technical Adequacy of the PRA," of RG 1.177, Revision 1. Significant errors and weaknesses with the internal events PRA will be resolved with the completion of the MOR, Revision 6, as required by the license condition (discussed in Section 4.0 of this SE). Therefore, the NRC staff concludes that the quality of the IEPRAs, with the completion of the latest revision to the MOR, meets the standard in TSTF-425, Revision 3.

Internal Flooding

In Section 3.2.2, "Peer Review Facts and Observations (F&Os)," of Attachment 2 to the LAR, the licensee stated that the internal flooding model upgrade was developed in 2016 and a focused-scope peer review was completed in early 2017, which resulted in unresolved F&Os. The licensee provided in Table 2 of Attachment 2 to the LAR, the finding-level F&Os along with their disposition or resolution, if resolved, and the impact on the application.

The NRC staff reviewed the internal flooding F&Os and the summary of the licensee's resolution or assessment of the impact on this application to determine whether any gaps in the PRA model were identified that could impact the application and to ensure that any gaps in meeting CC-II can be addressed for the SFCP per the NEI 04-10, Revision 1, methodology, consistent with RG 1.200, Revision 2. The NRC staff requested additional information from the licensee to clarify its disposition of the findings. The staff's review of unresolved internal flooding F&Os is discussed below.

For F&O MCR A1-5890, the peer review team noted that the IEPRA model "properly considers both flooding impacts and random equipment failures and human errors." However, the peer review team also observed "at least one instance ... in which a modified internal events HEP [Human Error Probability] was not properly included in the integrated model." The peer review team illustrated this observation with an example and proposed a resolution to review the process used to incorporate modified HEPs into the internal flooding model and re-quantify with "corrected HEP values." Similarly, for F&O MCR A1-5891, the peer review team observed one instance where an HEP was identified in flooding scenarios but was not included in the flooding PRA model. The peer review team illustrated with an example and proposed a resolution to review the process used to incorporate new flooding HEPs into the internal flooding model and verify that all intended events were included. In its disposition, the licensee stated that it will update the impacted HEPs, as needed, in the Internal flooding analysis model and documentation. The licensee also stated that the incorrect HEPs are expected to result in a "very small" change to the flooding results. In PRA-RAI-01.a, the NRC staff requested a description of the process that incorporates the modified and new HEPs into the internal flooding model, and an explanation of how the licensee concluded that the impact of such changes would result in a very small change to the flooding results. The NRC staff also requested that the licensee provide a mechanism to incorporate these updates prior to the SFCP implementation. In response to PRA-RAI-01.a, by letter dated October 17, 2018 (Reference 3), as supplemented by letter dated December 11, 2018 (Reference 4), the licensee proposed an implementation item to perform a comprehensive review of post-initiator human failure events (HFEs) for impact of flooding events and to address any HFEs potentially affected by internal flooding events in the PRA logic. This implementation item will be completed, as a condition of the license, prior to implementation of the SFCP (discussed in Section 4.0 of this SE).

F&O MCR A1-5894 stated that the HFE values in the human reliability analysis (HRA) dependency analysis were not seeded with high enough values to ensure the inclusion of all HFE combinations. The licensee's disposition for F&O MCR A1-5894 (related to Supporting Requirement IFQU-A7) stated that the impacted HFEs will be updated as needed in the internal flooding model. In PRA-RAI-01.c, the NRC staff requested confirmation that the appropriate seeding value is incorporated in the PRA model dependency analysis used for STI evaluations. In response to RAI-PRA-01.c, by letter dated October 17, 2018, the licensee updated the HRA seeding values and noted that the values are much higher than the nominal values calculated for individual HFEs. The NRC staff finds this acceptable because the licensee has performed the appropriate analysis to ensure that appropriate values are used for the application.

F&O MCR A1-5886 (related to Supporting Requirements IFPP-B3, IFEV-B3, IFQU-B3, IFSN-B3, IFSO-B3, and IFQU-A7) identified that no documentation of modeling uncertainties is provided in any of the internal flooding notebooks. In Attachment 2 to the LAR, the licensee indicated that there would be uncertainties associated with unresolved items that would be explored via sensitivity studies during the STI evaluation process. In PRA-RAI-01.d (Reference 18), the NRC staff requested a description of the approach used to identify and characterize key assumptions and sources of uncertainty associated with flood area uncertainties. In its response by letter dated October 17, 2018, the licensee stated that that it has developed a report for MOR, Revision 6, for the internal flooding analysis that follows the approach in NUREG-1855 to identify and characterize relevant sources of uncertainty and related assumptions. In a letter dated December 11, 2018 (Reference 4), the licensee further clarified that it used the latest revision (i.e., Revision 1) of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Reference 20). The licensee committed to an implementation item to complete Revision 6 of the PRA MOR prior to the SFCP implementation. The licensee further stated that during the TSTF-425 implementation process, candidate STI changes will be reviewed against the relevant sources of uncertainty identified for the ANO-1 PRA MOR, and that those sources of uncertainty determined to have the potential to challenge the acceptance criteria for the STI change will be identified as key, and sensitivity studies will be performed in accordance with NEI 04-10. This implementation item will be completed, as a condition of the license, prior to implementation of the SFCP (discussed in Section 4.0 of this SE).

The NRC staff finds the licensee's disposition to this F&O acceptable because the licensee has applied an approach for identifying internal flooding sources of uncertainty that is consistent with applicable NRC guidance.

RG 1.200, Revision 2, provides guidance for determining the technical adequacy of internal flooding PRA by comparing the PRAs to the relevant parts of the ASME/ANS PRA Standard using a peer review process. The NRC staff reviewed the peer review results, dispositions to F&Os, and the licensee's RAI responses, and finds that the quality and level of detail of the internal flooding PRA 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, Revision 1. Any discrepancies with the internal flooding PRA will be resolved with the completion of implementation items required by the license condition, as discussed in Section 4.0 of this SE. Therefore, the NRC staff concludes that the quality of the internal flooding PRA, with the completion of the implementation items, is consistent with TSTF-425, Revision 3.

Fire

In Section 3.3, "ANO-1 Fire PRA Model," of Attachment 2 to the LAR, the licensee states that the latest full scope peer review for the ANO-1 fire PRA model was conducted in October 2009 using RG 1.200, Revision 2, which resulted in 41 finding level F&Os. In addition, focused scope peer reviews were performed in 2012 and 2014 on the fire scenario selection and analysis (FSS), fire modeling, and HRA elements of the fire PRA, resulting in a limited number of additional finding-level F&Os. The licensee provided the F&Os in Tables 4 and 5 of Attachment 2 to the LAR. The NRC staff reviewed the summary of the peer review findings and the licensee's resolutions, including the assessment of the impact on this application to determine whether any gaps in the PRA model were identified that could impact the application and to ensure that any gaps in meeting CC-II can be addressed for the SFCP per the NEI 04-10, Revision 1, methodology, and is consistent with the guidance in RG 1.200, Revision 2. The NRC staff requested additional information to clarify the licensee's disposition of the findings, as discussed below.

The licensee's disposition for the F&Os related to Supporting Requirements HRA-A4, HRA-B3, HRA-D1, HRA-G3, and HRA-D6 stated that the HRA methodology was revised to follow the guidance of NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (Reference 21). The NRC staff noted that methods that are new to the subject PRA human error analysis meet the ASME/ANS PRA Standard definition for a PRA upgrade. In PRA-RAI-02.a (Reference 18), the NRC staff requested justification for whether the HRA revision constitutes a PRA upgrade that requires a peer review. In its response to PRA-RAI-02.a by letter dated October 17, 2018 (Reference 3), the licensee clarified that the new HRA methodology was previously identified as a PRA upgrade, and that a focused-scope peer review was performed in June 2014 for this upgrade. The NRC staff finds that the licensee has adequately addressed the resolution of this finding because it conducted a focused-scope peer review in response to a PRA upgrade.

RG 1.200, Revision 2, provides guidance for determining the technical adequacy of fire PRAs by comparing the PRAs to the relevant parts of the ASME/ANS PRA Standard using a peer review process. The NRC staff has reviewed the peer review results, F&O dispositions, and RAI response and finds that the technical quality and level of detail of the fire PRA 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, Revision 1. Therefore, the NRC staff concludes that the technical quality of the fire PRA is consistent with TSTF-425, Revision 3.

3.1.4.2 Scope of the Probabilistic Risk Assessment

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

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

The licensee does not have PRA models for high winds, seismic events, external events, and transportation and nearby facility accidents. These events were assessed in the Individual Plant Examination of External Events, which was a one-time review. In PRA-RAI-03 (Reference 18), the NRC staff requested a description of the process to ensure that the external events data reflected the as-built, as-operated plant and incorporated updated hazard information. In response to PRA-RAI-03 by letter dated October 18, 2018 (Reference 3), the licensee stated that Entergy procedure EN-DC-151, "PSA Maintenance and Update," ensures that the PRA analysts will review all future updates to determine when new external events data should be incorporated into the analysis. Based on this response, the NRC staff finds that the licensee has adequately addressed the RAI.

In accordance with NEI 04-10, Revision 1, the licensee can perform an initial qualitative screening analysis, and if the qualitative information is not sufficient to provide confidence that the net impact of the STI change would be negligible, a bounding analysis will be performed. The bounding analysis will be performed in accordance with Step 10b of NEI 04-10, Revision 1. This is an acceptable approach in accordance with NEI 04-10, Revision 1.

The licensee stated that for assessing the shutdown risk, the shutdown risk management program for implementation of Nuclear Management and Resources Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," dated December 1991 (Reference 22), will be used for the proposed changes to surveillance frequencies under the SFCP. This is an acceptable approach in accordance with NEI 04-10, Revision 1.

Based on the above, the NRC staff concludes that by application of NRC-approved NEI 04-10, Revision 1, the licensee's evaluation methodology is sufficient to ensure that 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, "Scope of the Probabilistic Risk Assessment for Technical Specification Change Evaluations," of RG 1.177, Revision 1.

3.1.4.3 Probabilistic Risk Assessment Modeling

The licensee's methodology includes the determination of whether the SSCs affected by a proposed change to a surveillance frequency are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact may be carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted CCF modes, based on the proposed change to the surveillance frequency. Where the SSC is not modeled in the PRA, bounding analyses are performed to characterize the impact of the proposed change to the surveillance frequency. Potential impacts on the risk analyses due to screening criteria and truncation levels are addressed by the requirements for PRA technical adequacy, consistent with the guidance provided in RG 1.200, Revision 2, and by sensitivity studies identified in NEI 04-10, Revision 1.

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

3.1.4.4 Assumptions for Time-Related Failure Contributions

The failure probabilities of SSCs modeled in PRAs may include a standby time-related contribution and a cyclic demand-related contribution. The licensee identified one source of uncertainty with standby failure rate. The standby time-dependent failure rate evaluation must be performed in accordance with NEI 04-10, Revision 1. The NEI 04-10, Revision 1, criteria adjust the time-related failure contribution of SSCs affected by the proposed change to a surveillance frequency. This is consistent with the guidance in RG 1.177, Revision 1, Section 2.3.3, which permits separation of the failure rate contributions into demand and standby for evaluation of supporting 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, such that the failure probability is assumed to increase linearly with time. This assumption will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The NEI 04-10, Revision 1, 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 NRC staff concludes that the licensee's process is not reliant upon risk analyses as the sole basis for the proposed changes because the licensee has, and will, apply the associated guidance in NRC-approved NEI 04-10, Revision 1.

The potential benefits of a reduced surveillance frequency, including reduced downtime and reduced potential for restoration errors, test-caused transients, and test-caused wear of equipment, are identified qualitatively, but not quantitatively assessed.

The NRC staff concludes that the licensee applied NRC-approved NEI 04-10, Revision 1, to employ reasonable assumptions with regard to extensions of STIs, and that the requested changes are consistent with Regulatory Position 2.3.4, "Assumptions in Completion Time and Surveillance Frequency Evaluations," of RG 1.177, Revision 1.

3.1.4.5 Sensitivity and Uncertainty Analyses

The proposed amended TSs would require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04-10, Revision 1. Therefore, the licensee will be required to have sensitivity studies to assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact on the frequency of initiating events, and any identified deviations from CC-II of the ASME/ANS PRA Standard. Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered along with any qualitative considerations that may bear on the results of such sensitivity studies. Step 5 of the NEI 04-10, Revision 1, process states that the identification of key sources of uncertainty is used to perform the appropriate sensitivity cases required by Step 14.

In Attachment 2 to the LAR (Reference 1), six F&Os, identified as Supporting Requirements IFPP-B3, IFEV-B3, IFQU-B3, IFSN-B3, IFSO-B3, and IFQU-A7, are related to model assumptions and uncertainties. The NRC staff determined that the disposition of these F&Os provided insufficient information to conclude that the F&Os have been sufficiently resolved. In PRA-RAI-01.d (Reference 18), the NRC staff requested details of the process used for the STI evaluation. In response to PRA-RAI-01.d by letter dated October 17, 2018, the licensee explained that the process used for STI evaluations was in accordance with NUREG-1855, Revision 0 (Reference 23). The licensee further explained that the MOR assumptions are reviewed for applicability following the procedures identified in EPRI TR-1013491, "Guideline for Treatment of Uncertainty in Risk-Informed Applications" (Reference 24), and EPRI TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 25). The NRC staff noted that these EPRI TRs are referenced in Revision 0 of NUREG-1855, which has been superseded by Revision 1 (Reference 20). NUREG-1855, Revision 1, uses information available in the updated EPRI TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty" (Reference 26), related to this subject matter. In the letter dated December 11, 2018 (Reference 4), the licensee further clarified that it used the latest revision of NUREG-1855 (i.e., Revision 1). The licensee further stated that during the TSTF-425 implementation process, candidate STI changes will be reviewed against the relevant sources of uncertainty identified for the ANO-1 PRA MOR and that those sources of uncertainty determined to have the potential to challenge the acceptance criteria for the STI change will be identified as key, and sensitivity studies will be performed in accordance with NEI 04-10. This is captured as an implementation item by the licensee and will be completed, as a condition of the license, prior to implementation of the SFCP (discussed in Section 4.0 of this SE). The NRC staff finds the licensee's response acceptable because the licensee has applied an approach for identifying sources of uncertainty that is consistent with acceptable NRC guidance.

In accordance with NEI 04-10, Revision 1, as required by proposed TS 5.5.8, the licensee will also perform monitoring and feedback of SSC performance once the revised surveillance frequencies are implemented. Therefore, the NRC staff concludes that the licensee appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations (with the completion of the implementation items discussed in Section 4.0 of this SE). In addition, the staff concludes that the LAR is consistent with Regulatory Position 2.3.5, "Sensitivity and Uncertainty Analyses Relating to Assumptions in Technical Specification Change Evaluations," of RG 1.177, Revision 1, because the licensee has or will apply the associated guidance in NRC-approved NEI 04-10, Revision 1.

3.1.4.6 Acceptance Guidelines

In accordance with NEI 04-10, Revision 1, as required by proposed TS 5.5.8, 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, Revision 1, in accordance with the TS SFCP. Each individual change to surveillance frequency must show a risk increase below 1E-6 per year for CDF, and below 1E-7 per year for LERF. These changes to CDF and LERF are consistent with the acceptance criteria of RG 1.174, Revision 3, for very small changes in risk. Where the RG 1.174, Revision 3, acceptance criteria are not met, the process in NEI 04-10, Revision 1, either considers revised surveillance frequencies that are consistent with RG 1.174, Revision 3, or the process terminates without permitting the proposed changes. Where quantitative results are unavailable for comparison with the acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or insignificant. Otherwise, bounding quantitative analyses are required to demonstrate that the risk impact is at least one order of magnitude lower than the RG 1.174, Revision 3, acceptance guidelines for very small changes in risk. In addition, in assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk increase less than 1E-5 per year for CDF, and less than 1E-6 per year for LERF. The total CDF and total LERF must be reasonably shown to be less than 1E-4 per year and 1E-5 per year, respectively. These values are consistent with the acceptance criteria of RG 1.174, Revision 3, as referenced by RG 1.177, Revision 1, for changes to surveillance frequencies.

Consistent with the NRC staff's SE dated September 19, 2007 (Reference 7), for NEI 04-10, Revision 1 (Reference 6), the TS SFCP will require the licensee to calculate the total change in risk (i.e., the cumulative risk) by comparing a baseline model that uses failure probabilities based on surveillance frequencies prior to being changed per the SFCP to a revised model that uses failure probabilities based on the changed surveillance frequencies. The 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 (i.e., 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, Revision 3, is supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, and industry standards, or at least bounding, quantitative results of sensitivity studies and SSC performance data and test history. The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results. Post-implementation performance monitoring and feedback are also required to assure continued reliability of the components. The licensee's application of NEI 04-10, Revision 1, provides acceptable methods for evaluating the risk increase associated with proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4, "Acceptance Guidelines for Technical Specification Changes," of RG 1.177, Revision 1. Therefore, the NRC staff concludes that the proposed methodology satisfies the fourth key safety principle of RG 1.177, Revision 1, by assuring that any increase in risk is small, consistent with the intent of the Commission's Safety Goal Policy Statement.

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

The licensee's proposed adoption of TSTF-425, Revision 3, requires application of NEI 04-10, Revision 1, in the SFCP. NEI 04-10, Revision 1, requires performance monitoring of SSCs whose surveillance frequencies have been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback includes consideration of Maintenance Rule (i.e., 10 CFR 50.65) monitoring of equipment performance. In the event of SSC performance degradation, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions that may be required by the Maintenance Rule. The performance monitoring and feedback specified in NEI 04-10, Revision 1, is sufficient to reasonably assure acceptable SSC performance, and is consistent with Regulatory Position 3.2, "Maintenance Rule Control," of RG 1.177, Revision 1. Thus, the NRC staff concludes that the fifth key safety principle of RG 1.177, Revision 1, is satisfied.

3.2 <u>Addition of Surveillance Frequency Control Program to Administrative Controls Section of TSs</u>

The licensee proposed including the SFCP and specific requirements into the ANO-1 TSs, Section 5.5.8, as follows:

<u>Surveillance Frequency Control Program</u>

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

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

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

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

The NRC staff concludes that proposed TS 5.5.8, which requires an acceptable program to control surveillance frequencies to ensure that LCOs are met and includes necessary program and applicability requirements, and its requirements set forth above, are consistent with the model application of TSTF-425, Revision 3, and are, therefore, acceptable.

3.3 <u>Deviations from TSTF-425 and Other Changes</u>

The licensee identified optional changes and variations with the approved TSTF-425, Revision 3, in Section 2.2 of Attachment 1 to the LAR (Reference 1). The NRC staff reviewed the changes and variations and made the following determinations:

1. The ANO-1 TSs contain SRs with numbering that differs from the corresponding SRs in TSTF-425. The NRC staff reviewed each SR and determined that these are administrative deviations only, with no impact on the conclusions reached in the NRC's model SE dated July 6, 2009 (74 FR 31996), and are, therefore, acceptable.

2. The NUREG-1430 TSTF-425 markups add the new SFCP as TS 5.5.18. The licensee is requesting to adopt this new program as TS 5.5.8, which is currently unused. The NRC staff determined that this is an administrative deviation and finds it acceptable.

3. For NUREG-1430 SRs not contained in the ANO-1 TSs, the corresponding markups included in TSTF-425 for these SRs are not applicable to ANO-1. The NRC staff confirmed that this is an administrative deviation from TSTF-425 and finds it acceptable.

4. The licensee included ANO-1 TSs 5.5.2 and 5.5.13 in the scope of this amendment. These TSs have periodic frequencies that were not identified for relocation in TSTF-425, Revision 3. The licensee proposed to revise the first sentence of TS 5.5.2.b. The first sentence of TS 5.5.2.b currently states:

    > Integrated leak test requirements for each system at least once per 18 months.

    The proposed revision to the first sentence of TSs 5.5.2.b would state:

    > Integrated leak test requirements for each system at a frequency in accordance with the Surveillance Frequency Control Program.

    TS 5.5.13.c currently states:

    > Total particulate concentration of the fuel oil is ≤ 10 mg/l [milligrams per liter] when tested every 31 days based on ASTM [American Society for Testing & Materials] D-2276, Method A-2 or A-3; and

    The proposed revision to TS 5.5.13.c would state:

    > Total particulate concentration of the fuel oil is < 10 mg/l when tested based on ASTM D-2276, Method A-2 or A-3 at a frequency in accordance with the Surveillance Frequency Control Program; and

    The NRC staff reviewed the proposed relocation of the subject TS Section 5.5 frequencies and determined that they are periodic frequencies that do not meet the scope exclusion criteria and are, therefore, consistent with the intent of TSTF-425. These frequencies will be controlled under the SFCP, which the staff has found to be acceptable and, thus, the staff concludes that the proposed changes to TS Section 5.5 are acceptable.

5. The licensee proposed a deviation to the TS Bases wording in several places. Specifically, the TSTF-425 TS Bases states the following, in part:

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

The licensee stated that this statement only applies to frequencies that have been changed in accordance with the SFCP and does not apply to frequencies that are relocated but not changed. The licensee proposed variations of this statement in the TS Bases in accordance with NUREG-1430, Revision 4.

The regulation at 10 CFR 50.36(a)(1) states that a summary statement of the bases or reasons for TSs, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TSs. Consistent with 10 CFR 50.36(a)(1), the licensee submitted corresponding TS Bases changes that provide the reasons for the proposed TSs changes. The NRC staff concludes that the proposed TS Bases changes describe the bases for the affected TSs and follow the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132).

6. The licensee stated in the LAR that, "[d]ue to the relocation of SR frequencies and replacing the frequencies with the statement, "In accordance with the Surveillance Frequency Control Program," there are multiple SRs that moved to the next page as identified in the markup of the TS pages...." The following new pages would be added as a result of these changes:

- Page 3.3.1-5 (SR 3.3.1.5 moved to page 3.3.1-4 and Table 3.3.1-1 moved to new page 3.3.1-5)
- Page 3.3.11-4 (SR Notes moved to page 3.3.11-3 and Table 3.3.11-1 moved to new page 3.3.11-4)
- Page 3.6.6-2 (SRs 3.6.6.3 and 3.6.6.4 moved to new page 3.6.6-2)
- Page 3.8.1-7 (SR 3.8.1.9 moved to new page 3.8.1-7)
- Page 3.8.6-4 (SR 3.8.6.6 moved to new page 3.8.6-4)

The NRC staff reviewed the movement of these SRs and has determined that the changes are administrative deviations from TSTF-425 with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). Therefore, the NRC staff has determined that the changes are acceptable.

7. The licensee proposed to revise TS 5.5.5.d. The proposed revision would delete "of 18 months on a STAGGERED TEST BASIS" and replace it with "in accordance with the Surveillance Frequency Control Program." Also, the revision would delete "18 month assessment of the" and add at the end of the sentence "assessment specified in TS 5.5.5.c." TSTF-425 includes the relocation of the frequency for the NUREG-1430 SR 3.7.10.4, associated with verifying that one Control Room Emergency Ventilation System

(CREVS) train can maintain a positive pressure relative to adjacent area(s). This SR was revised under TSTF-448, "Control Room Habitability" (Reference 27), to perform control room envelope unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program. ANO-1 adopted TSTF-448 and designated the Control Room Envelope Habitability Program as TS 5.5.5. The licensee is proposing to adopt the frequency change identified for the NUREG-1430 SR 3.7.10.4 in TSTF-425 as the ANO-1 TS 5.5.5.d frequency. The NRC staff reviewed the proposed changes and confirmed that the frequency for ANO-1 TS SR 3.7.9.4 has been moved to ANO-1 TS 5.5.5.d with the facility's adoption of TSTF-448. The frequency located in TS 5.5.5.d is a periodic frequency, does not meet the scope exclusion criteria, and is consistent with the intent of TSTF-425. These changes are administrative in nature and the frequency will be controlled under the SFCP. Therefore, the proposed changes to TS Section 5.5.5.d are acceptable.

8. The licensee proposes to relocate surveillance frequencies with periodicities different from those in TSTF-425 (e.g., SR 3.3.1.2). These differences have been approved by the NRC in prior amendment requests. The NRC staff reviewed these surveillance frequencies and determined that they do not meet the scope exclusion criteria and are consistent with the intent of TSTF-425. This proposed deviation is administrative in nature and is, therefore, acceptable.

3.4 Technical Evaluation Summary

The NRC staff has reviewed the licensee's proposed relocation of certain surveillance frequencies in ANO-1 TS Sections 3.1 through 3.9 and 5.5 to a licensee-controlled document and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, by the proposed addition of TS 5.5.8 to the Administrative Controls section of TSs. The NRC staff confirmed that this amendment does not relocate surveillance frequencies that reference other approved programs for the specific interval, are purely event-driven, are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs, or are related to specific conditions. The SFCP and TS Section 5.0, Subsection 5.5.8, reference NEI 04-10, Revision 1, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TSs to a licensee-controlled document, provided that those frequencies are changed in accordance with the NRC-approved NEI 04-10, Revision 1.

The licensee's proposed adoption of TSTF-425, Revision 3, and the risk-informed methodology of NRC-approved NEI 04-10, Revision 1, as referenced in the Administrative Controls section of the TSs, satisfies the key principles of risk-informed decisionmaking applied to changes to TSs, as delineated in RG 1.177 and RG 1.174, in that:

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

- The impact of the proposed change is monitored with performance measurement strategies.

In addition, the regulatory requirements in 10 CFR 50.65 and 10 CFR Part 50, Appendix B, Criterion XVI, and the performance monitoring required by NEI 04-10, Revision 1, ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified, and appropriate corrective actions will be taken. The NRC staff concludes that the licensee's SFCP ensures that SRs specified in the TSs are performed at intervals sufficient to assure the above regulatory requirements are met.

The licensee also proposed a license condition, as described in Section 4.0 of this SE, which will ensure that important updates will be incorporated into the PRA model prior to the implementation of the SFCP.

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

4.0 LICENSE CONDITION

With the issuance of this amendment, a condition will be added to the license for ANO-1. The condition will read:

> Surveillance Frequency Control Program
>
> The licensee shall implement the items listed in Table 2 of the enclosure to Entergy letter 1CAN121802, dated December 11, 2018, prior to implementation of the Surveillance Frequency Control Program.

This license condition requires the licensee to complete the following items prior to implementation of the SFCP, as identified in its letter dated December 11, 2018 (Reference 4):

- o A comprehensive review of post-initiator human failure events (HFEs) for impact of flooding events will be performed for the PRA Revision 6 model update. Any HFEs potentially affected by internal flooding events will be appropriately addressed in the PRA logic.

- o The internal flooding update supporting Revision 6 of the ANO-1 PRA model shall be completed.

- o Candidate Surveillance Testing Interval (STI) changes will be reviewed against the relevant sources of uncertainty identified for the ANO-1 PRA Revision 6 model update. Those sources of uncertainty determined to have the potential to challenge the acceptance criteria for the STI change will be identified as key, and sensitivity studies performed in accordance with NEI 04-10.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment on March 8, 2019. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration published in the *Federal Register* on June 5, 2018 (83 FR 26102), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Anderson, R. L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TSTF-425), Arkansas Nuclear One, Unit 1, Docket No. 50-313, License No. DPR-51," dated March 12, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18071A319).

2. Anderson, R. L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplemental Information Supporting the Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TSTF-425), Arkansas Nuclear One, Unit 1, Docket No. 50-313, License No. DPR-51," dated April 26, 2018 (ADAMS Accession No. ML18117A493).

3. Anderson, R. L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Related to the Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TSTF-425), Arkansas Nuclear One, Unit 1, Docket No. 50-313, License No. DPR-51," dated October 17, 2018 (ADAMS Accession No ML18290B060).

4. Anderson, R. L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplement to the Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TSTF-425), Arkansas Nuclear One, Unit 1, Docket No. 50-313, License No. DPR-51," dated December 11, 2018 (ADAMS Accession No. ML18346A539).

5. Technical Specifications Task Force, letter and enclosure to U.S. Nuclear Regulatory Commission, "Transmittal of TSTF-425, Revision 3, 'Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b,'" dated March 18, 2009 (ADAMS Accession No. ML090850642).

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

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

8. U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256).

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

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

11. U.S. Nuclear Regulatory Commission, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," NUREG-0800, Section 19.2, dated June 2007 (ADAMS Accession No. ML071700658).

12. U.S. Nuclear Regulatory Commission, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," NUREG-0800, Section 19.1, Revision 3, dated September 2012 (ADAMS Accession No. ML12193A107).

13. U.S. Nuclear Regulatory Commission, "Risk-Informed Decision Making: Technical Specifications," NUREG-0800, Section 16.1, Revision 1, dated March 2007 (ADAMS Accession No. ML070380228).

14. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Babcock and Wilcox Plants," NUREG-1430, Revision 4, Volume 1, Specifications, and Volume 2, Bases (ADAMS Accession Nos. ML12100A177 and ML12100A178, respectively).

15. American Society of Mechanical Engineers and American Nuclear Society (ASME/ANS) PRA Standard ASME/ANS RA-Sa-2009, "Addenda to ASME RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009, New York, NY.

16. Nuclear Energy Institute, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," NEI 00-02, Revision 1, dated May 2006 (ADAMS Accession No. ML061510619).

17. Nuclear Energy Institute, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," NEI 05-04, Revision 2, dated November 2008 (ADAMS Accession No. ML083430462).

18. Wengert, T., U.S. Nuclear Regulatory Commission, e-mail to Stephenie L. Pyle, Entergy Operations, Inc., "ANO-1 - Final RAI RE: License Amendment Request to Adopt TSTF-425, Revision 3 (EPID L-2018-LLA-0063)," dated September 19, 2018 (ADAMS Accession No. ML18263A302).

19. Electric Power Research Institute, "Support System Initiating Events: Identification and Quantification Guideline," Palo Alto, California, TR-1016741, dated December 2008.

20. U.S. Nuclear Regulatory Commission, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," NUREG-1855, Revision 1, dated March 2017 (ADAMS Accession No. ML17062A466).

21. U.S. Nuclear Regulatory Commission, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," NUREG-1921, dated July 2012 (ADAMS Accession No. ML12216A104).

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

23. U.S. Nuclear Regulatory Commission, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," NUREG-1855, dated March 2009 (ADAMS Accession No. ML090970525).

24. Electric Power Research Institute, "Guideline for Treatment of Uncertainty in Risk-Informed Applications," Palo Alto, California, TR-1013491, dated 2006.

25. Electric Power Research Institute, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," Palo Alto, California, TR-1016737, December 2008.

26. Electric Power Research Institute, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," Pao Alto, California, TR-1026511, dated 2012.

27. Wang, A. B., U.S. Nuclear Regulatory Commission, letter to Entergy Operations, Inc., "Arkansas Nuclear One, Unit No. 1 – Issuance of Amendment Re: Adoption of Technical Specification Task Force (TSTF) Change Traveler TSTF-448, Revision 3, 'Control Room Envelope Habitability' (TAC No. MD7177)," dated October 29, 2009 (ADAMS Accession No. ML082540799).

Principal Contributors: L. Fields, NRR
J. Evans, NRR
T. Sweat, NRR

Date: May 22, 2019

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - ISSUANCE OF AMENDMENT NO. 264 RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-425, REVISION 3 (EPID L-2018-LLA-0063) DATED MAY 22, 2019

DISTRIBUTION:
PUBLIC
PM File Copy
RidsACRS_MailCTR Resource
RidsNrrDeEeob Resource
RidsNrrDeEicb Resource
RidsNrrDeEmib Resource
RidsNrrDmlrMccb Resource
RidsNrrDorlLpl4 Resource
RidsNrrDraApla Resource
RidsNrrDssScpb Resource
RidsNrrDssStsb Resource
RidsNrrLAPBlechman Resource
RidsNrrPMANO Resource
RidsRgn4MailCenter Resource
JEvans, NRR
LFields, NRR
TSweat, NRR

ADAMS Accession No. ML19098A955 *via memorandum **via e-mail

OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DRA/APLA/BC(A)*	NRR/DSS/STSB/BC**
NAME	TWengert	PBlechman	SRosenberg	VCusumano
DATE	05/06/19	05/06/19	04/19/19	04/03/19
OFFICE	NRR/DE/EEOB/BC(A)**	NRR/DE/EICB/BC**	NRR/DSS/SCPB/BC**	NRR/DE/EMIB/BC**
NAME	DWilliams	MWaters	SAnderson	SBailey
DATE	04/29/19	04/29/19	04/29/19	04/29/19
OFFICE	NRR/DMLR/MCCB/BC**	OGC**	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM
NAME	SBloom	JWachutka	RPascarelli	TWengert (APulvirenti for)
DATE	04/24/19	05/21/19	05/22/19	05/22/19

OFFICIAL RECORD COPY